VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

November 3, 2004

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No. 04-494A
NL&OS/ETS R0
Docket Nos. 50-338
50-339
License Nos. NPF-4
NPF-7

VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)
NORTH ANNA POWER STATION UNITS 1 AND 2
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
PROPOSED TECHNICAL SPECIFICATION CHANGES
IMPLEMENTATION OF ALTERNATE SOURCE TERM
REVISED DOSE ANALYSIS AND TECHNICAL SPECIFICATION CHANGES

In a letter dated September 12, 2003 (Serial No. 03-464) Dominion requested amendments in the form of changes to the Technical Specifications to Facility Operating Licenses Numbers NPF-4 and NPF-7 for North Anna Power Station Units 1 and 2, respectively. The proposed changes were requested based on the radiological dose analysis margins obtained by using an alternate source term consistent with 10 CFR 50.67. In an August 3, 2004 telephone conference call, the NRC Staff requested additional information regarding the dose analysis methods/assumptions, charcoal filter testing efficiencies, and the proposed Technical Specifications changes. Information regarding the dose analysis, including some corrected analysis input data (i.e., pages 50 and 68 of the original dose assessment) was provided in an August 18, 2004 letter (Serial No. 04-494). This information has been incorporated into the revised technical report discussed below.

During the August 3, 2004 phone call, the NRC clarified their position on the assumptions to be used for evaluating the radiological consequences of a fuel handling accident consistent with RG 1.183. This position specifically addresses the use of effective decontamination factors (DFs). Based on the NRC's position on the use of effective DFs instead of using DFs for elemental and organic species, an additional dose assessment was required to support the requested Technical Specification changes.

Attachment 1 to this letter provides the requested information, which includes the revised analytical bases to support the proposed Technical Specifications changes. The information is provided as a revision to the original technical report provided in the September 12, 2003 letter (Serial No. 03-464) and the associated changes are identified by either underlining inserted text and/or revision lines in the margins. In addition, the proposed Technical Specification were revised to address the changes in assumptions for DFs and filter efficiencies and also the mode of applicability for the

A001

ECCS Pump Room Exhaust Air Cleanup System. In each of these instances the proposed Technical Specifications changes originally submitted have now been deleted.

Revised Technical Specification marked-up pages and the proposed Technical Specification pages, which incorporate the changes necessitated by the revisions to the DFs, filters efficiencies, and mode applicability are provided in Attachment 2 and 3, respectively.

Revised Technical Specification Bases changes are included for information only. In accordance with the Technical Specification Bases Control Program identified in Technical Specification 5.5.13, the Technical Specification Bases will be revised, following NRC approval of the license amendment.

We have evaluated the TS change request previously submitted with respect to the requested changes provided herein and have determined that the additional changes do not require any revision of the No Significant Hazards Consideration or environmental assessment provided in our original September 12, 2003 submittal. The No Significant Hazards Considerations and environmental assessments remain bounding.

Due to the number of program and procedure changes necessary to implement these changes, we continue to request ninety days from the issuance date of the amendments to implement the Technical Specifications changes.

If you have any further questions or require additional information, please contact Mr. Thomas Shaub at (804) 273-2763.

Very truly yours,

Leslie N. Hartz
Vice President – Nuclear Engineering

Attachments

Commitments made in this letter: None

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SN: 04-494A
Docket Nos.: 50-338/339
Subject: AST RAI – Revised Dose Analysis and Technical Specification Changes

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Leslie N. Hartz who is Vice President – Nuclear Engineering of Virginia Electric and Power Company. She has affirmed before me that she is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of her knowledge and belief.

Acknowledged before me this 3rd day of November, 2004.

My Commission Expires: May 31, 2006.

Vicki L. Hull
Notary Public

(SEAL)

Attachment 1
(Letter Serial No. 04-494A)

Proposed Technical Specification Changes
Implementation of Alternate Source Term

Revised Technical Report, Which Includes the
Response to Request for Additional Information

Virginia Electric and Power Company
(Dominion)
North Anna Power Station Units 1 and 2

DISCUSSION OF CHANGE
PROPOSED ALTERNATE SOURCE TERM LICENSE AMENDMENT

ASSESSMENT OF ACCIDENT RADIOLOGICAL CONSEQUENCES
USING NUREG-1465 METHODOLOGY

VIRGINIA ELECTRIC AND POWER COMPANY
(DOMINION)

NORTH ANNA POWER STATION UNITS 1 AND 2

OCTOBER 2004

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1.0 Introduction & Background

1.1 Introduction

This report describes the evaluations conducted to assess the radiological consequences of implementing the NUREG-1465 [Reference 1] accident source term methodology for North Anna Units 1 and 2. The accident source term documented in Reference [1] is herein referred to as the Alternative Source Term (AST). This convention is adopted following that originated by the NRC staff in the rulemaking proceeding associated with application of AST technology. The NRC, in Reference [2], issued the final rule and draft regulatory guidance associated with use of alternative source terms at operating reactors. The discussion in this report provides justification for the license amendment request, per the provisions of 10 CFR 50.67, "Accident Source Term".

The evaluations documented herein have employed the detailed methodology contained in Regulatory Guide (RG) 1.183 [Reference 3] for use in design basis accident (DBA) analyses for alternative source terms. The results have been compared with the acceptance criteria contained either in 10 CFR 50.67 or supplemental guidance in RG-1.183.

This amendment would: permit implementation of NUREG-1465 as the design basis source term for North Anna; allow North Anna to achieve a consistent design basis for all accident dose assessments; increase operational flexibility by allowing increased ECCS leakage and unfiltered control room inleakage; and eliminate the surveillance requirement to test the bottled air flow rate.

All the radiological dose analyses for the above accidents were performed with version 1.0p3 (QA) of the computer code RADTRAD-NAI [Reference 25]. The RADTRAD computer code calculates the control room and offsite doses resulting from releases of radioactive isotopes based on user supplied atmospheric dispersion factors, breathing rates, occupancy factors and dose conversion factors. Innovative Technology Solutions (ITS) of Albuquerque, New Mexico developed the RADTRAD code for the NRC. The original version of the NRC RADTRAD code was documented in NUREG/CR-6604 [Reference 4]. The NAI version of RADTRAD was originally derived from version 3.01 of the NRC/ITS RADTRAD. Subsequently, RADTRAD-NAI was revised to conform to the 3.02 version of the NRC/ITS RADTRAD. The RADTRAD-NAI code is maintained under NAI's QA program, which conforms to the requirements of 10CFR50, Appendix B.

1.2 Current Licensing Basis Summary

The current design basis radiological analyses that appear in the North Anna Updated Final Safety Analysis Report (UFSAR) consist of assessments of the following events:

1) Loss of Coolant Accident
2) Main Steam Line Break
3) Steam Generator Tube Rupture
4) Locked Rotor Accident
5) Fuel Handling Accident
6) Waste Gas Decay Tank Rupture
7) Volume Control Tank Rupture

The existing analyses for these events were performed at various times using different codes and in some cases, manual calculations. The common element for these events is the assumption of the radiological source term documented in TID-14844 [Reference 5].

1.3 Analysis Assumptions & Key Parameter Values

1.3.1 Selection of Events Requiring Reanalysis

A full implementation of the AST (as defined in Section 1.2.1 of Reference [3]) is proposed for North Anna Units 1 and 2. To support the licensing and plant operation changes discussed in Section 2.0, the Loss of Coolant Accident (LOCA), Fuel Handling Accident (FHA), Main Steam Line Break (MSLB) accident, Steam Generator Tube Rupture (SGTR) accident, and the Locked Rotor Accident (LRA) were reanalyzed employing the NUREG-1465 source term. The analysis methodology generally applied the guidance of RG-1.183, in conjunction with the total effective dose equivalent (TEDE) methodology. If this request is granted, the source term documented in NUREG-1465, as implemented in this plant-specific application, will become the source term employed in design basis radiological analyses for North Anna Units 1 and 2. The Waste Gas Decay Tank Rupture and the Volume Control Tank Rupture accidents were not reanalyzed using the AST methodology since these two events are not affected by the proposed licensing and plant operation changes and the existing analyses are considered bounding.

The proposed licensing and plant operational changes are discussed in Section 2.0. These changes require appropriate changes to the North Anna Technical Specifications, which are also described in Section 2.0 of this report. The key licensing basis changes considered are listed below:

a. Provide the dose analysis margin to allow positive containment pressure for up to four hours after the DBA (versus the current limit of one hour) in the Bases of Specifications 3.6.4, 3.6.6 and 3.6.7.

b. Define recently irradiated fuel as fuel that has occupied part of a critical reactor core within the previous 100 hours in the Bases of Specifications 3.9.4 and 3.7.15.

c. Require two Main Control Room and Emergency Switchgear Room Emergency Ventilation System (MCR/ESGR EVS) trains to be operable during Modes 1, 2, 3, and 4 (LCO 3.7.10) rather than 2 trains from the affected unit and one train from the other unit.

d. Delete the requirement to measure the bottled air flow rate during the 18 month surveillance (LCO 3.7.13).

e. Change the Technical Specification definition of DOSE EQUIVALENT I-131 to include an allowance to use dose conversion factors from RG 1.109 Revision 1 [Reference 16] in the calculation of DOSE EQUIVALENT I-131.

As indicated in Section 1.2.1 of Reference [3], the design basis LOCA must be reanalyzed to support an application for full implementation of the AST. The positive containment pressure (Item a) change mentioned above would also impact the LOCA accident dose results. A discussion of the DBA analyses affected by changes a through e (above) is presented below.

Item a – The current subatmospheric containment design basis requires that the engineered safeguards systems act to depressurize containment to less than atmospheric pressure within one hour and to maintain subatmospheric conditions thereafter. The proposed change would allow the calculation of pressures slightly above atmospheric pressure for a limited duration (1 – 4 hours) after the design basis event. This change could potentially impact the radiological consequences of either the design basis LOCA or main steam line break events. Since only the LOCA event has significant radiological releases into containment, it is the only analyzed event impacted by this change.

Item b – By defining recently irradiated fuel as fuel that has been part of a critical reactor core within the previous 100 hours, Specifications 3.9.4 and 3.7.15 will not be applicable for movement of fuel that is conducted more than 100 hours after shutdown. These specifications will thus not be applicable for core offloads at North Anna that begin more than 100 hours after core shutdown. This change impacts the radiological consequences of the design basis Fuel Handling Accident. No other DBAs are impacted by these changes.

Item c – By requiring two MCR/ESGR EVS trains to be available instead of three during Modes 1 through 4 (LCO 3.7.10) the design basis LOCA, SGTR, MSLB and Locked Rotor accidents are affected. Since two trains instead of three would be available, pressurization of the MCR or recirculation of the air in the MCR could be modeled, but not both. The reanalysis of each event takes this requirement into consideration.

Item d – The accident analyses take credit for operation of the air bottle system to pressurize the control room to ≥ 0.05 inches of water, consistent with the existing SR 3.7.13.4. However, the flow surveillance in SR 3.7.13.4 is considered redundant and is being deleted, since no credit is taken in the analyses for the cleansing effect of the bottled air. The requirement to pressurize the control room to ≥ 0.05 inches of water is retained because isolation and initiation of the bottled air system is credited in the LOCA and FHA analyses. Therefore, the requirement to measure 340 cfm of bottle airflow is deleted from SR 3.7.13.4.

Item e – The Technical Specification coolant activity calculations of DOSE EQUIVALENT I-131 and the appearance rates calculated for the MSLB and SGTR accidents are based on dose conversion factors from Regulatory Guide 1.109. Therefore, the definition of DOSE EQUIVALENT I-131 is revised to allow the calculation of DOSE EQUIVALENT I-131 with dose conversion factors from Regulatory Guide 1.109.

It can be concluded from the evaluation summarized above that implementing the AST, in conjunction with the proposed plant operational changes, requires reanalysis of the LOCA, FHA, SGTR, MSLB, and LRA. Sections 3.1 through 3.5, respectively, provide the detailed description of the reanalyses for these events.

1.3.2 Analysis Assumptions & Key Parameter Values

This section describes the general analysis approach and presents analysis assumptions and key parameter values that are common to the accident analyses performed to implement the NUREG-1465 source term. Sections 3.1 through 3.5 provide specific assumptions that were employed for the LOCA, FHA, SGTR, MSLB and LRA, respectively.

The dose analyses documented in this application employ the Total Effective Dose Equivalent (TEDE) calculation method, consistent with the radiation protection standards in 10 CFR Part 20 and as specified in RG-1.183 for AST applications. The total effective dose equivalent (TEDE) doses are determined at the exclusion area boundary (EAB) for the worst 2-hour interval. The TEDE doses for individuals at the low population zone (LPZ) and for the main control room (MCR) personnel are calculated for the assumed 30-day duration of the event.

The TEDE concept is defined to be the deep dose equivalent, DDE, (from external exposure) plus the committed effective dose equivalent, CEDE, (from internal exposure). In this manner, the TEDE dose assesses the impact of all relevant nuclides upon all body organs, in contrast with the previous single, critical organ (thyroid) concept for assessing internal exposure. The DDE is nominally equivalent to the effective dose equivalent (EDE) from external exposure if the whole body is irradiated uniformly. Since this is a reasonable assumption for submergence exposure situations, EDE is used in lieu of DDE in determining the contribution of external dose to the TEDE. EDE dose conversion factors were taken from Table III.1 of Federal Guidance Report 12 [Reference 8] per Section 4.1.4 of Reference [3].

There are a number of analysis assumptions and plant features that are common to the analysis of all of the events. These items are presented in Table 1.3-1.

The onsite atmospheric dispersion factors were calculated for each of the accident scenarios by Dominion using the ARCON96 code [Reference 17] and guidance from Draft Guide 1111 [Reference 9]. More detailed descriptions of the onsite atmospheric dispersion factor development for each accident are contained in Section 3. In June 2003 Regulatory Guide 1.194 [Reference 15] was issued, superceding Draft Guide 1111. It has been subsequently verified that the incremental changes from Draft Guide 1111 to Regulatory Guide 1.194 have no impact on the calculated atmospheric dispersion coefficients.

Table 1.3-1: Analysis Assumptions & Key Parameter Values Employed in All Analyses

NSSS Parameters	
Core Power	2958 MWt*
Number of Fuel Assemblies	157
Containment Free Volume	1.84E6 ft^3
MCR/ESGR Parameters	
Effective Volume	77,000 ft^3 (MCR) or 230,000 ft^3 (MCR/ESGR)**
Offsite Atmospheric Dispersion Factors	
Exclusion Area Boundary, EAB (0 – 2 hours)	3.10E-4 sec/m^3
Low Population Zone, LPZ	
0 – 8 hours	1.10E-5 sec/m^3
8 – 24 hours	7.30E-6 sec/m^3
24 – 96 hours	3.00E-6 sec/m^3
96 – 720 hours	8.20E-7 sec/m^3
Breathing Rates	
Control Room	3.5E-4 m^3/sec
Offsite (EAB & LPZ)	
0 – 8 hours	3.5E-4 m^3/sec
8 – 24 hours	1.8E-4 m^3/sec
24 – 720 hours	2.3E-4 m^3/sec
Control Room Occupancy Factors	
0 – 24 hours	1.0
24 – 96 hours	0.6
96 – 720 hours	0.6***
Pressurization Flow Rate	1000 +/- 10% cfm ****

* Rated Thermal Power for North Anna is 2893 MWt. This value is slightly greater than 102% of Rated Thermal Power.

** The volume of the control room envelope, which includes the MCR plus the ESGR, is 230,000 ft^3. The value used, 77,000 ft^3 or 230,000 ft^3, depends on the accident sensitivity and modeling considerations to account for the extent of mixing between floors.

*** Note: the North Anna Operations' shift is based on a 12 hour workday plus turnover. Therefore, a value of 0.6 was used for this time period.

**** The values used in the analyses were 0, 900, or 1100 depending on the accident sensitivity and modeling considerations. Pressurization flow starts 1 hour after isolation of the MCR.

2.0 Proposed Licensing Basis Changes

This section provides a summary description of the key proposed licensing basis changes that are justified with the North Anna AST analyses accompanying this license amendment request.

2.1 Implementation of NUREG-1465 Methodology as Design Basis Source Term

This report supports a request to revise the design basis accident source term for North Anna Units 1 and 2. Subsequent to approval of this license amendment, the design basis source term for use in evaluating the consequences of design basis accidents will become the source term documented in NUREG-1465 [Reference 1], including any deviations approved by the NRC staff. This license amendment application is made pursuant to the requirements of 10 CFR 50.67(b)(1), which specifies that any licensee seeking to revise its current accident source term used in design basis radiological consequences analysis shall apply for a license amendment.

2.2 Open Equipment Hatch & Penetrations during Movement of Recently Irradiated Fuel

Currently, Technical Specification 3.9.4, Containment Penetrations, of the North Anna ITS requires the equipment hatch to be closed, and at least one door in the personnel airlock to be capable of being closed during the movement of recently irradiated fuel. In addition, each penetration providing a direct path from containment atmosphere to the outside atmosphere must have operable isolation valves or be closed during movement of recently irradiated fuel. Current requirements allow these direct access paths to be unisolated under administrative control. These requirements are consistent with the existing analysis for a fuel handling accident inside containment, which only models radioactive releases through the personnel airlock. Based on the AST analyses, recently irradiated fuel is defined as fuel that has been part of a critical reactor core within the previous 100 hours. The term recently irradiated fuel movement is currently applied to all irradiated fuel.

Within 100 hours of reactor shutdown, Technical Specification 3.9.4 will remain applicable. In addition, for consistency with TSTF-51, a proposed modification to LCO 3.9.4.b requires the personnel airlock to be closed, instead of capable of being closed, during movement of recently irradiated fuel.

Beyond 100 hours when Technical Specification 3.9.4 is no longer applicable, the AST analysis of FHA accommodates releases from an open containment equipment hatch, personnel airlock and other penetrations providing a direct path to the outside atmosphere. Changes to procedures will be implemented to ensure that the capability to close the equipment hatch is maintained during refueling operations, and that the required actions can be accomplished in accordance with Regulatory Guide 1.183. (See Section 5.)

Closure of the equipment hatch is the duty of a team trained for that task and controlled in accordance with station procedures. Equipment hatch closure will be accomplished as allowed by containment dose rates following a fuel handling accident, since hatch closure requires actions from inside containment. Since the revised radiological analysis does not take credit for the containment closure actions, no commitment is proposed concerning the required timeframe for achieving containment closure. To preclude creating a personnel radiological hazard, closure will only be accomplished as allowed by containment dose rates. This represents an exception to the guidance proposed in RG-1.183, which recommends an assumed 30 minute closure time.

2.3 Fuel Building Ventilation System

No changes are proposed to LCO 3.7.15, which requires the Fuel Building Ventilation System to be operable and in operation during the movement of recently irradiated fuel to limit the consequences of a fuel handling accident. However, the Bases Section for 3.7.15 is changed to define recently irradiated fuel as fuel that has occupied part of a critical reactor core within the previous 100 hours. An administrative change is also proposed to the Bases section to indicate that the 10 CFR 50.67 limit is for the AST.

The analysis of the fuel handling accident with the AST did not credit filtration of the fuel building ventilation exhaust. The flow rate of the fuel building exhaust was varied over a wide range to accommodate the effects of forced or natural circulation flow. A flow rate of 80,000 cfm, which bounds the capacity of the fuel building and containment ventilation systems, maximized the doses. Doses at the EAB, LPZ and in the Control Room decreased with decreasing exhaust flow.

2.4 Redefinition of Subatmospheric Containment Depressurization Criteria

This change proposes a relaxation of the current containment design basis acceptance criteria concerning achieving and maintaining subatmospheric conditions following a loss of coolant accident. North Anna Units 1 and 2 have a subatmospheric containment design that has the following acceptance criteria for the design basis LOCA containment integrity analyses:

1. calculated peak pressure must be less than 45 psig
2. containment must be depressurized to less than atmospheric within 1 hour
3. calculated peak pressure after one hour must be less than 0.0 psig

The second and third criteria are relaxed as part of the present application. The proposed acceptance criteria for design basis LOCA containment integrity analyses are as follows (the first item remains unchanged):

1. calculated peak pressure must be less than 45 psig
2. containment must be depressurized to 0.5 psig within 1 hour and to subatmospheric pressure within 4 hours
3. calculated peak pressure after 4 hours must be less than 0.0 psig

The current criteria require that following the initial containment depressurization to less than atmospheric pressure, operation of the Recirculation Spray subsystems indefinitely maintains pressure less than atmospheric. These criteria are currently reflected in the Bases of the following North Anna Technical Specifications: 3.6.4, Containment Pressure; 3.6.6 Quench Spray System; 3.6.7, Recirculation Spray System. The AST license amendment proposes changes to the bases for each of these three Technical Specifications to indicate the relaxed pressure criterion at 1 hour and the extension of the requirement to achieve subatmospheric pressure until 4 hours. The radiological analyses have accommodated greater than atmospheric pressure and the associated period of additional leakage for an interval of up to 4 hours after the DBA. The analyses for implementation of the AST for North Anna have assumed a containment leakage rate that corresponds to a maximum containment pressure of 0.5 psig for the timeframe of 1 to 4 hours following the loss of coolant accident and zero leakage thereafter. Section 3.1 provides the justification for the leak rate assumed in the LOCA analysis.

There are no proposed changes to the existing containment structure, heat removal systems, containment integrity accident analyses or Technical Specifications associated with these items as part of this application. The proposed changes are intended to provide potential future flexibility by utilizing a portion of the margin that was made available by application of the AST analysis methodology.

2.5 Control Room Habitability Systems

2.5.1 Main Control Room/Emergency Switchgear Room (MCR/ESGR) Emergency Ventilation System (EVS) – Modes 1, 2, 3, and 4

Based on the analyses of the accidents that could occur during Modes 1 through 4, a change is proposed to LCO 3.7.10 to require only two MCR/ESGR EVS trains to be operable in Modes 1 through 4. The analyses of the Steam Generator Tube Rupture, Main Line Steam Break, and Locked Rotor accidents do not take credit for either Main Control Room isolation or recirculation of the air within the Main Control Room. The Loss of Coolant Accident analysis credits pressurization and isolation of the control room but does not credit recirculation of the air within the control room. Therefore, only two trains are required to be operable to ensure pressurization of the control room in the event of a single failure. Additionally, SR 3.7.10.3, which verifies automatic actuation of each MCR/ESGR EVS train, is no longer needed because fan operation is not credited in the first hour of the LOCA. Since the MCR/ESGR EVS train associated with 1-HV-F-41 cannot be used to provide outside air for filtered pressurization due to the location of its air intake with respect to vent Stack B, 1-HV-F-41 is restricted from use as a pressurization fan.

2.5.2 Main Control Room/Emergency Switchgear Room (MCR/ESGR) Emergency Ventilation System (EVS) – During Movement of Recently Irradiated Fuel

Changes are proposed to the Bases section of 3.7.14 to reflect the Fuel Handling Accident analysis. Currently LCO 3.7.14 requires two MCR/ESGR EVS trains to be operable to supply filtered air to pressurize the MCR/ESGR envelope during handling of recently irradiated fuel and approximately 60 minutes after actuation of the bottled air system. The proposed bases changes are related to the use of 1-HV-F-41 and the definition of recently irradiated fuel. As discussed above, since the MCR/ESGR EVS train associated with 1-HV-F-41 cannot be used to provide outside air for filtered pressurization due to the location of its air intake with respect to vent Stack B, 1-HV-F-41 is restricted from use as a pressurization fan.

2.5.3 Main Control Room/ Emergency Switchgear Room (MCR/ESGR) Bottled Air System

No changes are proposed to any LCOs for the Technical Specification governing the MCR/ESGR Bottled Air System. A change is proposed to SR 3.7.13.4 to eliminate the specification of minimum bottle makeup flow rate. Specification of the makeup flow rate in the surveillance is unnecessary and is deleted since no credit is taken in the analyses for the cleansing effect of the bottled air in the accident analysis. Changes are proposed to the BASES section to indicate that control room operator dose limits are for the AST, and to define recently irradiated fuel.

2.6 Miscellaneous Bases Only Changes

Bases-only changes will be made primarily to change "10 CFR 100" to "10 CFR 50.67" or "Regulatory Guide 1.183", and to delete phrases such as "well within" and "small fraction of" that will not have regulatory significance with the AST design Basis.

The Bases changes are provided for information only and will be implemented in accordance with the Technical Specification Bases Control Program and 50.59.

3.0 Radiological Event Re-analyses & Evaluation

As documented in Section 1.3.1, this application involves the reanalysis of the design basis radiological analyses for the LOCA, Fuel Handling Accident (FHA), Steam Generator Tube Rupture (SGTR) accident, Main Steam Line Break (MSLB) accident, and Locked Rotor accident. These analyses have incorporated the features of the AST, including the TEDE analysis methodology and modeling of plant systems and equipment operation that influence the events. The calculated radiological consequences are compared with the revised limits provided in 10 CFR 50.67(b)(2), as clarified per the additional guidance in RG-1.183 for events with a higher probability of occurrence. Dose calculations are performed at the exclusion area boundary (EAB) for the worst 2 hour period, and for the low population zone (LPZ) and control room for the duration of the accident (30 days). Virginia Electric and Power Company (Dominion) performed all the radiological consequence calculations for the AST with the RADTRAD-NAI computer code system as discussed above. The dose acceptance criteria that apply for implementing the AST are provided in Table 3.0-1.

Table 3.0-1: Accident Dose Acceptance Criteria

Accident or Case	Control Room	EAB & LPZ
Design Basis LOCA	5 rem TEDE	25 rem TEDE
Steam Generator Tube Rupture		
Fuel Damage or Pre-incident Spike	5 rem TEDE	25 rem TEDE
Coincident Iodine Spike	5 rem TEDE	2.5 rem TEDE
Main Steam Line Break		
Fuel Damage or Pre-incident Spike	5 rem TEDE	25 rem TEDE
Coincident Iodine Spike	5 rem TEDE	2.5 rem TEDE
Locked Rotor Accident	5 rem TEDE	2.5 rem TEDE
Rod Ejection Accident	5 rem TEDE	6.3 rem TEDE
Fuel Handling Accident	5 rem TEDE	6.3 rem TEDE

3.1 Design Basis Loss of Coolant Accident (LOCA) Reanalysis

This section describes the methods employed in, and results obtained from, the LOCA design basis radiological analysis. The analysis includes dose from several sources: the containment leakage plume, leakage from Emergency Core Cooling System (ECCS) components and the RWST, that persists throughout the assumed 30 day duration of the accident, and from control room filter loading that also persists for 30 days. Doses were calculated at the exclusion area boundary (EAB), at the low population zone boundary (LPZ), and in the control room. The methodology used to evaluate the control room and offsite doses resulting from a LOCA was consistent with RG-1.183.

3.1.1 LOCA Scenario Description

The design basis LOCA scenario for radiological calculations is initiated assuming a major rupture of the primary reactor coolant system piping. In order to result in radioactive releases of the magnitude specified in NUREG-1465, it is also assumed that the ECCS does not provide adequate core cooling, such that significant core melting occurs. This general scenario does not represent any specific accident sequence, but is representative of a class of severe damage incidents that were evaluated in the development of the NUREG-1465 source term characteristics. Such a scenario would be expected to require multiple failures of systems and equipment and lies beyond the severity of incidents evaluated for design basis transient analysis.

3.1.2 LOCA Source Term Definition

NUREG-1465 [Reference 1] provides explicit description of the key AST characteristics recommended for use in design basis radiological analyses. There are significant differences between the source term in Reference [1] and the existing design basis source term documented in TID-14844 [Reference 5]. The primary differences between the key characteristics of the two source terms are shown in Table 3.1-1 below.

Table 3.1-1: Comparison of TID-14844 and NUREG-1465 Source Terms

Characteristic	TID Source Term	NUREG-1465 Source Term
Core Fractions Released To Containment	Noble gases – 100% Iodine – 50% (half of this plates out) Solids – 1% Iodine – 50% to sump	Noble gases – 100% Iodine – 40% Cesium – 30% Tellurium – 5% Barium – 2% Others – 0.02% to 0.25%
Timing of Release	Instantaneous	Released in Two Phases Over 1.8 hour Interval
Iodine Chemical and Physical Form	91% inorganic vapor 4% organic vapor 5% aerosol	4.85% inorganic vapor 0.15% organic vapor 95% aerosol
Solids	Ignored in analysis	Treated as an aerosol

NUREG-1465 divides the releases from the core into two phases: 1) the fuel gap release phase during the first 30 minutes, and 2) the early in-vessel release phase in the subsequent 1.3 hours. The later release phases documented in NUREG-1465 are not considered for design basis accidents, consistent with the guidance from RG-1.183. Table 3.1-2 shows the fractions of the total core inventory of various isotope groups assumed to be released in each of the two phases of the LOCA analysis. Table 3.1-2 also shows the rate of release or production for each isotope group, assuming that the releases are linear with respect to time.

Table 3.1-2: NUREG-1465 Release Phases

	Core Release Fractions		Production Rate (Fractions/hr)[a]	
Isotope Group	Gap	Early In-Vessel	Gap	Early In-Vessel
Noble Gases[b]	0.05	0.95	0.1	7.31E-01
Halogens	0.05	0.35	0.1	2.69E-01
Alkali Metals	0.05	0.25	0.1	1.92E-01
Tellurium	0	0.05	0	3.85E-02
Barium, Strontium	0	0.02	0	1.54E-02
Noble Metals	0	0.0025	0	1.92E-03
Cerium	0	0.0005	0	3.85E-04
Lanthanides	0	0.0002	0	1.54E-04
Duration (hr)[a]	0.5	1.3		

a. Release duration and production rates apply only to the Containment release. The ECCS leakage portion of the analysis conservatively assumes that the entire core release fraction is in the containment sump from the start of the LOCA.

b. Noble Gases are not scrubbed from the containment atmosphere and therefore are not found in either the sump or ECCS fluid.

The core radionuclide inventory for use in determining source term releases was generated using the ORIGENS code. ORIGENS is part of the SCALE computer code system [Reference 6]. The ORIGENS output was converted to Ci/MWt for input to the RADTRAD-NAI code. Table 3.1-3 lists the 94 isotopes and the associated curies per megawatt at the end of a fuel cycle that were input to RADTRAD-NAI. Eight of the isotopes included as input to RADTRAD-NAI did not come from ORIGENS but were input into RADTRAD-NAI so that their daughters could be tracked. These eight isotopes were Te-131, Nd-97m, Te-125m, Tc-99, Y-91m, Cs-138, Rb-88 and Rb-89. Also shown in Table 3.1-3 are the CEDE and EDE dose conversion factors for each of the isotopes. These dose conversion factors were taken from Federal Guidance Reports 11 and 12 [References 7 and 8].

Table 3.1-3: Core Inventory and Dose Conversion Factors by Isotope

Isotope	Isotope Group	Ci/MWt	EDE Sv-m^3/Bq-sec	CEDE Sv/Bq
Kr83m	Noble gas	3.2280E+03	1.500E-18	0.000E+00
Kr85	Noble gas	2.6620E+02	1.190E-16	0.000E+00
Kr85m	Noble gas	6.7055E+03	7.480E-15	0.000E+00
Kr87	Noble gas	1.3462E+04	4.120E-14	0.000E+00
Kr88	Noble gas	1.8683E+04	1.020E-13	0.000E+00
Kr89	Noble gas	2.3231E+04	0.000E+00	0.000E+00
Xe131m	Noble gas	3.5840E+02	3.890E-16	0.000E+00
Xe133	Noble gas	5.5165E+04	1.560E-15	0.000E+00
Xe133m	Noble gas	1.7325E+03	1.370E-15	0.000E+00
Xe135	Noble gas	1.4624E+04	1.190E-14	0.000E+00
Xe135m	Noble gas	1.1392E+04	2.040E-14	0.000E+00
Xe137	Noble gas	5.0197E+04	0.000E+00	0.000E+00
Xe138	Noble gas	4.6983E+04	5.770E-14	0.000E+00
Br82	Halogen	8.8527E+01	1.300E-13	4.130E-10
Br83	Halogen	3.2278E+03	3.820E-16	2.410E-11
Br84	Halogen	6.0158E+03	9.410E-14	2.270E-11
I130	Halogen	5.6201E+02	1.040E-13	7.140E-10
I131	Halogen	2.6636E+04	1.820E-14	8.890E-09
I132	Halogen	3.9028E+04	1.120E-13	1.030E-10
I133	Halogen	5.5087E+04	2.940E-14	1.580E-09
I134	Halogen	6.1036E+04	1.300E-13	3.550E-11
I135	Halogen	5.2506E+04	7.980E-14	3.320E-10
Rb86	Alkali Metal	4.7846E+01	4.810E-15	1.790E-09
Rb88	Alkali Metal	0.0000E+00	3.360E-14	2.260E-11
Rb89	Alkali Metal	0.0000E+00	1.060E-13	1.160E-11
Cs134	Alkali Metal	4.3425E+03	7.570E-14	1.250E-08
Cs135	Alkali Metal	1.2444E-02	5.650E-19	1.230E-09
Cs136	Alkali Metal	1.2974E+03	1.060E-13	1.980E-09
Cs137	Alkali Metal	3.1180E+03	7.740E-18	8.630E-09
Cs138	Alkali Metal	0.0000E+00	1.210E-13	2.740E-11
Sb125	Tellurium	2.1059E+02	2.020E-14	3.300E-09
Sb127	Tellurium	2.2615E+03	3.330E-14	1.630E-09
Sb129	Tellurium	8.5060E+03	7.140E-14	1.740E-10
Te125m	Tellurium	0.0000E+00	4.530E-16	1.970E-09
Te127	Tellurium	2.2216E+03	2.420E-16	8.600E-11
Te127m	Tellurium	3.6323E+02	1.470E-16	5.810E-09
Te129	Tellurium	8.0728E+03	2.750E-15	2.090E-11
Te129m	Tellurium	1.6331E+03	1.550E-15	6.470E-09
Te131	Tellurium	0.0000E+00	2.040E-14	1.290E-10
Te131m	Tellurium	5.2148E+03	7.010E-14	1.730E-09
Te132	Tellurium	3.8349E+04	1.030E-14	2.550E-09
Te133	Tellurium	2.9863E+04	4.600E-14	2.490E-11
Sr89	barium-strontium	2.6485E+04	7.730E-17	1.120E-08

Table 3.1-3: Core Inventory and Dose Conversion Factors by Isotope

Isotope	Isotope Group	Ci/MWt	EDE Sv-m^3/Bq-sec	CEDE Sv/Bq
Sr90	barium-strontium	2.3043E+03	7.530E-18	3.510E-07
Sr91	barium-strontium	3.2692E+04	3.450E-14	4.490E-10
Sr92	barium-strontium	3.4755E+04	6.790E-14	2.180E-10
Ba137m	barium-strontium	2.9637E+03	2.880E-14	0.000E+00
Ba139	barium-strontium	4.8610E+04	2.170E-15	4.640E-11
Ba140	barium-strontium	4.8872E+04	8.580E-15	1.010E-09
Ag110m	Noble metal	1.0045E+02	1.360E-13	2.170E-08
Ag111	Noble metal	1.5613E+03	1.290E-15	9.120E-10
Mo99	Noble Metal	5.0097E+04	7.280E-15	1.070E-09
Rh103m	Noble Metal	4.1569E+04	8.800E-18	1.380E-12
Rh105	Noble Metal	2.6006E+04	3.720E-15	2.580E-10
Rh106	Noble Metal	1.5233E+04	1.040E-14	0.000E+00
Ru103	Noble Metal	4.1623E+04	2.250E-14	2.420E-09
Ru105	Noble Metal	2.8402E+04	3.810E-14	1.230E-10
Ru106	Noble Metal	1.3581E+04	0.000E+00	1.290E-07
Tc99	Noble Metal	4.4372E+04	1.620E-18	2.250E-09
Tc99m	Noble Metal	0.0000E+00	5.890E-15	8.800E-12
Ce141	Cerium	4.4932E+04	3.430E-15	2.420E-09
Ce143	Cerium	4.1552E+04	1.290E-14	9.160E-10
Ce144	Cerium	3.3729E+04	8.530E-16	1.010E-07
Np239	Cerium	5.2542E+05	7.690E-15	6.780E-10
Pu238	Cerium	8.9111E+01	4.880E-18	7.790E-05
Pu239	Cerium	9.0388E+00	4.240E-18	8.330E-05
Pu240	Cerium	1.1970E+01	4.750E-18	8.330E-05
Pu241	Cerium	3.7076E+03	7.250E-20	1.340E-06
Am241	Lanthanides	4.1215E+00	8.180E-16	1.200E-04
Cm242	Lanthanides	1.0133E+03	5.690E-18	4.670E-06
Cm244	Lanthanides	1.0030E+02	4.910E-18	6.700E-05
Eu156	Lanthanides	5.7941E+03	6.750E-14	3.820E-09
La140	Lanthanides	5.0568E+04	1.170E-13	1.310E-09
La141	Lanthanides	4.4400E+04	2.390E-15	1.570E-10
La142	Lanthanides	4.3390E+04	1.440E-13	6.840E-11
Nb95	Lanthanides	4.6384E+04	3.740E-14	1.570E-09
Nb95m	Lanthanides	5.2889E+02	2.930E-15	6.590E-10
Nb97	Lanthanides	4.3654E+04	3.180E-14	2.240E-11
Nb97m	Lanthanides	0.0000E+00	3.550E-14	0.000E+00
Nd147	Lanthanides	1.7984E+04	6.190E-15	1.850E-09
Pm147	Lanthanides	4.5942E+03	6.930E-19	1.060E-08
Pm148	Lanthanides	4.6964E+03	2.890E-14	2.950E-09
Pm148m	Lanthanides	7.8497E+02	9.680E-14	6.100E-09
Pm149	Lanthanides	1.6674E+04	5.410E-16	7.930E-10
Pr143	Lanthanides	4.0599E+04	2.100E-17	2.190E-09
Pr144	Lanthanides	3.4016E+04	1.950E-15	1.170E-11

Table 3.1-3: Core Inventory and Dose Conversion Factors by Isotope

Isotope	Isotope Group	Ci/MWt	EDE Sv-m^3/Bq-sec	CEDE Sv/Bq
Sm153	Lanthanides	1.2169E+04	2.280E-15	5.310E-10
Y90	Lanthanides	2.3923E+03	1.900E-16	2.280E-09
Y91	Lanthanides	3.4332E+04	2.600E-16	1.320E-08
Y91m	Lanthanides	0.0000E+00	2.550E-14	9.820E-12
Y92	Lanthanides	3.5017E+04	1.300E-14	2.110E-10
Y93	Lanthanides	2.6640E+04	4.800E-15	5.820E-10
Zr95	Lanthanides	4.6031E+04	3.600E-14	6.390E-09
Zr97	Lanthanides	4.3352E+04	9.020E-15	1.170E-09

3.1.3 Determination of Atmospheric Dispersion Factors (X/Q)

3.1.3.1 Onsite (Main Control Room) X/Q

The onsite atmospheric dispersion factors were calculated by Dominion using the ARCON96 code [Reference 17] and guidance from Draft Guide 1111 [Reference 9]. Site meteorological data taken over the years 1997-2001 were used in the calculations. For the Main Control Room, X/Qs were calculated for the LOCA for these source points: Unit 1 and Unit 2 Containment buildings, Auxiliary Building louvers, Refueling Water Storage Tanks (RWST's), Equipment Hatches, Primary Ventilation Blowout Panels, and Vent Stacks A and B. The receptor points modeled were the four emergency control room intakes as well as the normal control room air intake.

Only ground level releases were modeled with ARCON96 for the above release points because none of the sources were at an elevation (two and half times the height of the buildings on site) sufficient to be considered an elevated release. Vent releases are not to be modeled with ARCON96 per the guidance of Draft Guide 1111.

The Auxiliary Building louvers, the Equipment Hatches, the RWST vents, Vent Stacks A and B, and the Primary Ventilation Blowout Panels were modeled as point sources. The Containment Buildings were treated as diffuse sources.

For all of the ARCON96 runs, the cross sectional area of one of the Containment Buildings above grade was used to model a wake effect, since all of the receptor points modeled are expected to be in the wake of one of the Containment Buildings.

The various source point to receptor point geometries were analyzed and the combination which resulted in the largest set of X/Q values for the LOCA control room dose analysis are reported in Table 3.1-4. For example, X/Q values were calculated for both the Unit 1 and the Unit 2 RWST to all of the Emergency Intakes. The largest X/Q values were calculated for the Unit 1 RWST source to one of the four Emergency Intakes. These values were then reported in Table 3.1-4 for the RWST to Emergency Intake. In some cases, such as for the Unit 1 and Unit 2 Containments, neither Containment source point resulted in a higher X/Q value for all time steps. Therefore, the values presented in Table 3.1-4 are a composite with the highest value for either Containment for each time step. The X/Q values reported in Table 3.1-4 are the values used in the LOCA control room dose analyses.

3.1.3.2 Offsite (EAB & LPZ) X/Q

The Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) atmospheric dispersion factors were developed by Stone and Webster and are part of the existing design basis offsite dose calculations. These X/Q values, which were not revised for the AST analysis, are listed in Table 1.3-1.

Table 3.1-4: Control Room Atmospheric Dispersion Factors for the LOCA

SOURCE	RECEPTOR POINT	ATMOS. DISPERSION FACTOR (sec/m^3)	
Containment	Normal Control Room Intake	2.61E-3	0-2 hr
		1.83E-3	2-8 hr
		7.72E-4	8-24 hr
		5.69E-4	24-96 hr
		4.35E-4	96-720 hr
Vent Stack	Emergency Control Room Intake	3.75E-3	0-2 hr
		2.65E-3	2-8 hr
		1.03E-3	8-24 hr
		7.77E-4	24-96 hr
		5.70E-4	96-720 hr
RWST Vent	Emergency Control Room Intake	2.18E-3	0-2 hr
		1.42E-3	2-8 hr
		4.89E-4	8-24 hr
		3.84E-4	24-96 hr
		2.72E-4	96-720 hr
Vent Stack Blowout Panel	Emergency Control Room Intake	2.12E-3	0-2 hr
		1.38E-3	2-8 hr
		5.29E-4	8-24 hr
		3.76E-4	24-96 hr
		2.93E-4	96-720 hr
Auxiliary Building Louver	Emergency Control Room Intake	3.66E-3	0-2 hr
		2.46E-3	2-8 hr
		9.87E-4	8-24 hr
		6.80E-4	24-96 hr
		5.02E-4	96-720 hr
Equip Hatch	Emergency Control Room Intake	8.47E-4	0-2 hr
		6.41E-4	2-8 hr
		2.66E-4	8-24 hr
		1.84E-4	24-96 hr
		1.36E-4	96-720 hr
Containment	Emergency Control Room Intake	1.23E-3	0-2 hr
		9.02E-4	2-8 hr
		3.57E-4	8-24 hr
		2.55E-4	24-96 hr
		1.91E-4	96-720 hr

3.1.4 Determination of Containment Spray Iodine Removal Coefficients

There are six different spray headers belonging to two different systems (Quench Spray and Recirculation Spray) inside the North Anna containment. The Quench Spray system has two separate pump trains. Each Quench Spray pump train supplies a separate circular dome header at two different elevations near the top of the containment. The Recirculation Spray system is also redundant, consisting of four pumps and heat exchangers and 4 semi-circular spray headers. Two of the Recirculation Spray pumps are located inside containment (Inside Recirculation Spray, IRS) and two of the Recirculation Spray pumps are located outside containment (Outside Recirculation Spray, ORS) in the safeguards area. Each of the IRS or ORS pumps supplies one semi-circular spray header. It is conservative for the analysis of spray removal during LOCA to assume a single failure of one train of engineered safeguards equipment, resulting in the analysis assuming that one Quench Spray train and one train each of the Inside and Outside Recirculation Spray subsystems are operating.

The containment spray removal rates for aerosol fission products are calculated using the methodology of NUREG/CR-5966 [Reference 10], which presents removal equations at 10, 50, and 90 percentile levels. Only the 10 percentile (most conservative) equations are used. No credit is taken for iodine plateout. The removal rates were calculated separately as a function of time for each of the spray subsystem headers, and combined to yield the following effective aerosol removal coefficients for all the sprays:

Table 3.1-5: Combined Quench and Recirculation Spray Aerosol Iodine Removal Coefficients (λ_{mf})

Aerosol Removal Constant		
Time (hr)		**λ_{mf}**
From	**To**	**(hr-1)**
2.50E-02	8.01E-02	3.7267E+00
8.01E-02	1.33E-01	1.0799E+01
1.33E-01	1.56E+00	1.6672E+01
1.56E+00	1.80E+00	1.2528E+01
1.80E+00	1.87E+00	7.9863E+00
1.87E+00	1.97E+00	5.5782E+00
1.97E+00	2.33E+00	2.9768E+00
2.33E+00	3.76E+00	1.6191E+00
3.76E+00	5.35E+00	1.4460E+00
5.35E+00	6.97E+00	1.4239E+00
6.97E+00	8.59E+00	1.4211E+00
8.59E+00	1.61E+02	1.4207E+00

The removal of elemental iodine by sprays continues at a rate of 10 hr^{-1} until a decontamination factor (DF) of 200 is reached, as specified in Section 6.5.2 of NUREG-0800 [Reference 11]. This DF is reached when the elemental iodine activity in the containment at the end of the early in-vessel release phase is reduced by a factor of 200. The time it takes to achieve this reduction in activity is determined as follows:

$$A = A_0 e^{-\lambda t}$$

$$DF = A_0/A = e^{\lambda t}$$

$$t = \ln(DF)/\lambda = \ln(200)/10 = 0.53 \text{ hr}$$

This is the duration required starting at the end of early in-vessel phase at 1.8 hr. Hence, the post accident time at which elemental iodine removal stops is 2.33 hours (1.80 hr + 0.53 hr). Spray removal of organic iodine is not modeled.

3.1.5 LOCA Analysis Assumptions & Key Parameter Values

Offsite TEDE is calculated for the LOCA at the Exclusion Area Boundary (EAB) and, Low Population Zone (LPZ) from dose contributions from the following sources: 1) Containment leakage, 2) ECCS leakage and 3) leakage from the Refueling Water Storage Tank (RWST). To determine the TEDE in the Control Room, dose contributions resulting from control room filter loading were added to the contributions from the above three sources.

To provide flexibility for plant operations, the dose impact of ECCS leakage on the Control Room and offsite doses was assessed assuming that the iodine that evolves from the ECCS leakage was either filtered by the auxiliary building charcoal filters or unfiltered. See Section 3.1.5.2 "Model on ECCS Leakage" below. This resulted in two values of allowable ECCS leakage that will be apportioned within station procedures based upon where in the plant the ECCS leakage occurs. The Containment, ECCS, and RWST leakage cases each assumed 250 cfm of unfiltered control room inleakage while the cases that modeled control room filter loading assumed 500 cfm of unfiltered control room inleakage. Each of the two ECCS leakage cases (filtered or unfiltered) produces a control room dose of 5 REM TEDE when summed with the dose contributions from the containment leakage, the RWST leakage and the control room filter loading. Separate RADTRAD-NAI models were set up to determine the dose contributions for each of the above three sources and the filter loading. These models warrant further discussion below.

3.1.5.1 Containment Leakage Model

The following acceptance criteria, replicated from the Section 2.4 discussion, are proposed for this application in modeling the North Anna subatmospheric containment design:

- calculated peak pressure must be less than 45 psig
- containment must be depressurized to 0.5 psig within 1 hour and to subatmospheric pressure within 4 hours
- calculated peak pressure after 4 hours must be less than 0.0 psig

The LOCA analysis for implementation of the AST has been performed to conform to these revised acceptance criteria. The LOCA analysis has assumed continued leakage during the 1-4 hour interval after the DBA, but at a diminished rate corresponding to a containment pressure of 0.5 psig. Beyond 4 hours, the pressure is assumed to be less than 0.0 psig, terminating leakage from containment. This section describes the model details for determination of the appropriate leak rate associated with a pressure that is slightly above atmospheric.

The containment is modeled with a volumetric leak rate of 0.1% per day for the first hour and 0.021% per day for the next three hours. The leak rate of 0.021% per day corresponded to the revised maximum allowable containment pressure of 0.5 psig for hours 1 through 4. It was confirmed that the calculated leak rate of 0.021% per day in the Surry AST [References 12 and

13] was applicable to North Anna. During the NRC review of the Surry AST submittal, the NRC confirmed that the assumed leak rate of 0.021% per day was acceptable to use since it was conservative [Reference 14].

The RADTRAD-NAI computer code model represents the containment airborne leakage to the environment with four compartments: (1) the control room environment, (2) the portion of the containment covered by the containment spray systems, (3) the portion of the containment not covered by the containment spray systems and, (4) the environment. RADTRAD-NAI modeled seven pathways between the four compartments. The pathways are as follows:

P1 – containment sprayed region to the environment

P2 – containment unsprayed region to the environment

P3 – containment sprayed region to the containment unsprayed region

P4 – containment unsprayed region to the containment sprayed region

P5 – environment to the control room through the emergency intakes by fans (filtered)

P6 – environment to the control room by unfiltered inleakage

P7 – control room to the environment

The transfer of radionuclides is modeled in RADTRAD-NAI by specifying flow rates along the pathways modeled. The input for these flow rates is described in Table 3.1-6 below.

Table 3.1-6: Flow Rates Used in the RADTRAD-NAI Containment Model

Pathway	Time (hr.)		Flow Rate (CFM)*
	From	To	
P1	0	1	0.894
	1	4	0.188
P2	0	1	0.383
	1	4	0.0805
P3	0	4	1.84E+4
P4	0	4	1.84E+4
P5	1	720	900
P6	0	720	250
P7	0	1	250
	1	720	1150

* Flow rates are based upon the following input

Total Containment Volume = 1.84 E+6 ft^3

Sprayed Volume = 1.288 E+6 ft^3

Unsprayed Volume = 5.52 E+5 ft^3

Mixing rate between sprayed and unsprayed volumes is 2 unsprayed volumes per hour

Containment leak rate during first hour is 0.1% volume per day

Containment leak rate from hour one to hour four is 0.021% volume per day

3.1.5.2 Model of ECCS Leakage

The Emergency Core Cooling System (ECCS) fluid consists of the contaminated water in the sump of the containment. This water contains, according to the alternate source term methodology, 40% of the core inventory of iodine isotopes and the released core isotopes which are assumed to be in particulate form. During a LOCA the highly radioactive ECCS fluid is pumped from the containment sump to the recirculation spray headers and sprayed back into the containment sump. This is done to keep the atmosphere clean and cool the containment atmosphere after a LOCA. Since one set of recirculation spray pumps is located in the safeguards building there is a potential for ECCS fluid leakage in the safeguards building after a LOCA. Any iodine which evolves from the ECCS leakage in the safeguards building would be drawn by ventilation fans and exhausted out vent stack B on top of the service building next to the turbine building. Also, following a design basis LOCA, valve realignment occurs to switch the suction water source for the ECCS from the RWST to the containment sump. This automatic action is taken upon the level in the RWST reaching a defined setpoint. At this point, both the low head safety injection (SI) pumps (located in the safeguards building) and the high head charging pumps (located in the charging pump cubicle in the auxiliary building) would begin pumping ECCS fluid and would be another potential source of ECCS fluid leakage. Any iodine which evolves from the ECCS fluid leakage in the charging pump cubicle would be drawn by ventilation fans into the auxiliary building central exhaust flow and be exhausted out of vent stack A on top of the service building next to the turbine building. Finally, there are portions of the ECCS piping located in the quench spray basement and auxiliary building that are not treated by the auxiliary building charcoal filters and ECCS leakage in these areas is treated as unfiltered leakage.

The pumping of the ECCS fluid by the outside recirculation spray pumps is assumed to start at 288.5 seconds after the LOCA. The charging pumps can start flowing ECCS fluid as early as 32 minutes after the initiation of the LOCA. The RADTRAD-NAI model conservatively assumes that all ECCS leakage starts at 288.5 seconds after the LOCA. The auxiliary building charcoal filters are conservatively modeled as not being available for filtering ECCS iodine gases for 60 minutes after the onset of the LOCA.

The RADTRAD-NAI source term used to model the ECCS leakage contains only the iodine isotopes. This is because iodine is the only element in the containment sump water that was modeled as coming out of solution and becoming airborne. All other isotopes in the ECCS fluid remain in solution or suspension in accordance with RG-1.183. Forty percent of the core inventory of iodine isotopes that are released to the containment atmosphere were conservatively modeled as being instantaneously transported from the core to the containment sump. Ten percent of the iodine isotopes in the ECCS fluid were modeled as coming out of solution. As required by RG-1.183, ECCS leakage was modeled at two times the allowable ECCS leakage.

The RADTRAD-NAI computer code model represents the ECCS leakage to the environment with three compartments: (1) the control room, (2) the environment and, (3) the containment sump. Four pathways are used in RADTRAD-NAI to model the liquid and gaseous flow between the three compartments. The physical pathways of the ECCS leakage to the safeguards and auxiliary buildings are not modeled in RADTRAD-NAI since no credit is taken for dilution or holdup in either of these buildings. The gaseous and liquid pathways used in the model of ECCS leakage are as follows:

P1 – environment to control room by fans (filtered pressurization flow)

P2 – environment to control room by unfiltered inleakage

P3 – control room to environment

P4 – containment sump to environment

Pathway P1 represents the flow by fans of contaminated air from the environment into the control room. Pathway P2 represents the unfiltered inleakage of contaminated air from the environment into the control room. Pathway P3 represents the flow of control room air back into the environment so that the control room pressure remains constant. Pathway P4 represents the iodine release from the ECCS fluid to the environment.

The modeled ECCS leakage was increased until the dose to the control room from ECCS leakage equaled the difference between the 5 REM TEDE limit and the dose from the other sources of control room dose - containment leakage, RWST leakage and filter shine. A total control room dose of 5 REM TEDE with 250 cfm of unfiltered control room inleakage was obtained with two ECCS leakage scenario's; either 1700 cc/hr unfiltered (modeled as 3400 cc/hr) or 17200 cc/hr filtered (modeled as 34400 cc/hr). Two additional ECCS leakage rates were conservatively used to calculate the offsite doses, 3070 cc/hr unfiltered (modeled as 6140 cc/hr) and 31500 cc/hr filtered (modeled as 63000 cc/hr). The allowable ECCS leakage will be limited based on the Control Room analysis to the 1700 cc/hr unfiltered or 17200 cc/hr filtered values.

The transfer of radionuclides is modeled in RADTRAD-NAI by specifying flow rates along the pathways modeled. As indicated in Table 3.1-7, Pathway P4 was modeled with the four different ECCS leakage rates discussed above.

Table 3.1-7: Flow Rates Used in the RADTRAD-NAI ECCS Leakage Model

Pathway	Time (hr. unless noted)		Flow Rate (CFM unless noted)
	From	To	
P1	1	720	900
P2	0	720	250
P3	0	1	250
	1	720	1150
P4 [1] (Control Room Dose)	288.5 sec	720	3400 cc/hr unfiltered or 34400 cc/hr filtered
P4 [2] (EAB & LPZ Dose)	288.5 sec	720	6140 cc/hr unfiltered or 63000 cc/hr filtered

1. As required by RG 1.183, the ECCS leakage (pathway P4) for the control room analysis was modeled as two times the allowable values of 1700 cc/hr unfiltered or 17200 cc/hr filtered.
2. ECCS leakage (pathway P4) for the EAB & LPZ analysis was conservatively modeled as two times the values of 3070 cc/hr unfiltered or 31500 cc/hr filtered.

ECCS Leakage Control Room Modeling Parameters

Control Room Volume : 77,000 ft^3

Containment Sump Volume: 5.2970E+4 ft^3

Filter Efficiencies for Pathway P1 and recirculation filters in control room compartment (recirculation only considered for filter loading):

- 98% particulate iodine (HEPA efficiency that is applied to all particulates),
- 95% elemental iodine, and
- 70% organic iodine

Chemical form of Iodine in Sump compartment: 97% elemental and 3% organic

Containment Sump Inventory is 40 % of the core iodines

Iodine evolution is 10% of iodine in the ECCS fluid

Filter Efficiencies for Pathway P4 to model filtration by auxiliary building filters

- 0% particulate iodine (there are no particulates in this release pathway),
- 95% elemental iodine, and
- 70% organic iodine

Filter Efficiencies for Pathway P4 to model no filtration by auxiliary building filters

- 0% particulate iodine
- 0% elemental iodine
- 0% organic iodine

3.1.5.3 Model of ECCS Back Leakage to RWST

Following a design basis LOCA, valve realignment occurs to switch the suction water source for the ECCS from the Refueling Water Storage Tank (RWST) to the containment sump. This action is taken upon the level in the RWST reaching a defined setpoint and is conservatively modeled in RADTRAD-NAI as occurring at 30 minutes following the initiation of the LOCA. In this configuration, MOV's and check valves in the normal suction line from the RWST and MOV's in the recirculation line provide isolation between this contaminated flow stream and the RWST. This RADTRAD-NAI analysis of the LOCA models leakage of ECCS fluid through these valves back into the RWST with subsequent leakage of the evolved iodine through the gooseneck vent at the top of the North Anna RWST to the environment.

The RADTRAD-NAI source term used to model the ECCS leakage into the RWST contains only the iodine isotopes. This is because iodine is the only element in the containment sump water, which was modeled as coming out of solution and becoming airborne. Forty percent of the core inventory of iodine isotopes were conservatively modeled as being instantaneously transported from the core to the containment sump. This iodine is modeled to be 97% in the elemental chemical form and 3% in the organic chemical form in accordance with RG-1.183.

The RWST leakage model was created with the RADTRAD-NAI computer code and consisted of 4 compartments and 7 Pathways. The compartments used to model the RWST leakage dose consequences were 1) control room, 2) environment, 3) containment sump, and 4) RWST air volume. The gaseous and liquid pathways used in the model are as follows:

P1 - environment to the control room through the emergency intakes by fans (filtered)
P2 – environment to the control room by unfiltered inleakage
P3 – control room to environment
P4 – containment sump ECCS liquid to the bottom of the RWST (HHSI Suction)
P5 – RWST to environment
P6 – environment to RWST
P7 – containment sump ECCS liquid to the top of the RWST (LHSI Suction)

The flow rates for pathways P1, P2, and P3 are the same ones used to model the ECCS leakage shown in Table 3.1-7. As required by RG-1.183, ECCS leakage into the RWST was modeled at two times the allowable ECCS leakage. The RWST was modeled as having 480 cc/hr of ECCS fluid leaking through the recirculation lines from the discharge side of the LHSI pumps to the top of the RWST (Pathway P7). An additional 1920 cc/hr of ECCS fluid was modeled as leaking back into the RWST through the 16" diameter LHSI suction line (Pathway P4). The evolution of iodine from the leakage from P4 and P7 was modeled as 10%. Pathways P5 and P6 were each modeled with a flow rate of 4 cfm. Since the only motive forces for moving air into or out of the RWST would be thermal expansion or contraction of the air and gases inside the RWST, displacement of the air and gases by ECCS liquid, and wind-induced pressure variations at the mouth of the gooseneck vent pipe, the resultant average discharge rate from the tank was estimated at 4 cfm. Table 3.1-8 summarizes the flow rates for the 7 pathways.

Table 3.1-8: Flow Rates Used in the RADTRAD-NAI RWST Leakage Model

Pathway	Time (hr. unless noted)		Flow Rate (CFM unless noted)
	From	To	
P1	1	720	900
P2	0	720	250
P3	0	1	250
	1	720	1150
P4 *	0.5	720	1920 cc/hr
P5	0.5	720	4
P6	0.5	720	4
P7 *	0.5	720	480 cc/hr

* As required by RG 1.183, the ECCS leakage into the RWST for pathways P4 and P7 was modeled as two times the allowable values of 960 cc/hr and 240 cc/hr.

RWST Back Leakage Control Room Modeling Parameters

Control Room Volume : 77,000 ft^3

Containment Sump Volume: 5.2970E+4 ft^3

RWST Volume: 68,159 ft^3

Filter Efficiencies for Pathway P1 and recirculation filters in control room compartment (recirculation only considered for filter loading):

- 98% particulate iodine (HEPA efficiency that is applied to all particulates),
- 95% elemental iodine, and
- 70% organic iodine

Chemical form of Iodine in Sump compartment: 97% elemental and 3% organic

Containment Sump Inventory is 40 % of the core iodines

Iodine evolution is 10% of the iodine in the ECCS leakage

3.1.5.4 Dose from Filter Loading

The MCR/ESGR emergency ventilation system consists of four trains of fans with filters on two elevations. At least one control room emergency fan draws air from the environment into the control room and keeps the control room pressurized. This air passes through a set of HEPA and charcoal filters. Another control room emergency fan may be aligned as a recirculation fan. A recirculation fan draws air from the control room and passes it through another set of HEPA and charcoal filters before returning it to the control room. As a consequence of passing air containing radioactive contaminants through the pressurization filter there will be a gradual buildup of radioactive material in the pressurization filter media. Similarly, the recirculation filter media would also become loaded with radioactive isotopes from the containment, ECCS, and RWST leakage to the atmosphere. Although the benefit from recirculation flow is not credited in the LOCA analysis, recirculation flow was considered conservatively in the filter loading portion of the analysis to maximize the dose from filter loading.

The isotopes inside the filter housings would emit gamma radiation that would create a shine dose to the operators. For conservatism, it was assumed that both the recirculation filter and the pressurization/intake filter are on the MCR level of the control room envelope where the control room operators would be performing their duties for the assumed accident duration (30 days) following a LOCA.

3.1.5.4.1 Filter Loading from Containment Leakage

To determine the isotopic loading on the pressurization/intake filter, two additional RADTRAD-NAI runs were made using the Containment leakage model. In the first run, control room recirculation, control room inleakage, out flow from the control room, and filtration of the pressurization flow were turned off. The second run was the same as the first with the exception that the pressurization flow was filtered (98% particulate, 95% elemental iodine, and 70% organic iodine). The difference in the control room isotopic inventory between the two runs represents the isotopes residing on the filter media due to the containment leakage source. The contribution from noble gases is not credited since they are not normally held up in the filter media.

To determine the isotopic loading on the recirculation filter resulting from the containment leakage source, two additional RADTRAD-NAI runs were made using the Containment leakage model with an additional compartment modeled to accumulate the isotopes. In the first of these runs the control room was modeled with filtered intake, inleakage, and outleakage. Instead of 1100 cfm of recirculation through filters, the recirculation flow (1100 cfm) was modeled as flowing into the special compartment with no outflow for a period of 30 days. This compartment accumulated all of the isotopes that would normally flow through the recirculation filters. The second run was identical to the first with the exception that the flow into the special compartment was filtered (98% particulate, 95% elemental iodine, and 70% organic iodine). The difference in the isotopic inventory in the special compartment between the two runs represents the isotopes residing on the filter media due to the containment leakage source.

Once the loading of the pressurization and recirculation filters was determined, the gamma spectrum was determined for each filter using the ORIGENS code. Once the spectrum was determined, a conservative dose point was selected to maximize the shine dose the operator would receive from the filters and the shine dose was calculated with the QADS code. QADS like ORIGENS is part of the SCALE computer code system [Reference 6].

3.1.5.4.2 Filter Loading from ECCS and RWST Leakage

The methodology for calculating the filter loading caused by the ECCS leakage was identical to the procedure described in the Containment leakage section above with the exception that the RADTRAD-NAI model for ECCS leakage was used. The source term in this case consisted only of the iodine isotopes in the ECCS liquid. The ECCS leakage was modeled as 8000 cc/hr unfiltered, which exceeds two times the maximum allowed unfiltered ECCS leakage of 1700 cc/hr (modeled as 3400 cc/hr.) that is indicated in Table 3.1-7.

No model was set up for calculating the filter loading resulting from RWST leakage since this leakage resulted in such a small control room dose. However, a dose was estimated for the RWST leakage contribution to the filter loading by determining the ratio of I-131 inventory released from the RWST to that from the ECCS leakage and multiplying it by the ECCS filter loading dose.

3.1.6 Results

The worst 2-hour dose to the EAB and the total dose to the LPZ were calculated by adding the dose contributions from the containment leakage, the RWST leakage and the ECCS leakage. The contribution to the EAB dose from ECCS leakage was based on the ECCS leakage scenario that produced the largest EAB dose. This corresponded to an allowable ECCS leakage of 31500 cc/hr filtered (modeled as 63000 cc/hr) for the EAB dose. Similarly, the largest ECCS contribution to the LPZ dose was with an allowable ECCS leakage of 3070 cc/hr unfiltered (modeled as 6140 cc/hr). However, ECCS leakage will be limited based on the Control Room analysis to allowable leakage of 1700 cc/hr unfiltered (modeled as 3400 cc/hr.) or 17200 cc/hr filtered (modeled as 34400 cc/hr). The total doses to the EAB and LPZ are summarized in Table 3.1-9.

Table 3.1-9: Offsite Dose Results for the LOCA (REM TEDE)

Release Pathway	EAB Dose	LPZ Dose
Containment Leakage Contribution	1.09	0.05
RWST Leakage Contribution	0.004	0.005
ECCS Leakage Contribution	0.76	0.06
Total Dose	1.85	0.12

The control room doses resulting from containment leakage, RWST leakage, ECCS leakage, and filter loading shine are summed below in Table 3.1-10. The results shown in Table 3.1-10 assume 250 cfm of unfiltered control room inleakage for all release pathways. The filter loading shine includes contributions from the containment leakage, the ECCS leakage and the RWST leakage with an assumed unfiltered control room inleakage of 500 cfm and a bounding ECCS leak rate (modeled as 8000 cc/hr unfiltered). RADTRAD-NAI runs were made with combinations of filtered and unfiltered ECCS leakage rates that resulted in total control room doses of 5 rem TEDE when added to doses from containment leakage, RWST leakage, and filter shine. ECCS leakage will be limited, based on the Control Room analysis, to allowable leakage of 1700 cc/hr unfiltered (modeled as 3400 cc/hr.) or 17200 cc/hr filtered (modeled as 34400 cc/hr).

Table 3.1-10: Control Room Dose Results for the LOCA (REM TEDE)

Source or Release Pathway	Control Room Dose
Containment Leakage Contribution	1.07
RWST Leakage Contribution	0.32
ECCS Leakage Contribution	3.02
Filter Loading Contribution	0.58
Total Dose	5.0

3.2 Fuel Handling Accident (FHA)

This section describes the methods and results employed in the Fuel Handling Accident (FHA) design basis radiological analysis. The analysis includes doses associated with release of gap activity from a fuel assembly either inside containment or in the Fuel Building. Doses were calculated at the exclusion area boundary (EAB), at the low population zone boundary (LPZ), and in the control room. The methodology used to evaluate the control room and offsite doses resulting from the FHA was generally consistent with RG-1.183 in conjunction with TEDE radiological units and limits, ARCON96 based onsite atmospheric dispersion factors, and Federal Guidance Reports No. 11 and 12 dose conversion factors.

3.2.1 FHA Scenario Description

The design basis scenario for the radiological analysis of the FHA assumes that cladding damage has occurred to all of the fuel rods in one fuel assembly. This scenario is unchanged from the assumption in the existing UFSAR analysis. The rods are assumed to instantaneously release their fission gas contents to the water surrounding the fuel assemblies. The analyses include the evaluation of FHA cases that occur in both containment and the Fuel Building, with appropriate modeling for the influence of the different release pathways and operation of ventilation systems.

3.2.2 FHA Source Term Definition

In accordance with Regulatory Position 3 of RG 1.183 the core source was determined with ORIGIN-ARP at the End of Cycle (EOC) using a bounding cycle design (based on the Dominion fuel management scheme). The ORIGEN-ARP editor was used to create an ORIGENS input deck. The core inventory from ORIGEN-ARP was input into the ORIGENS deck in order to determine the decay of the isotopes at various times after shutdown. Table 3.2-1 contains the ORIGENS decay results at 100 hours after shutdown for the 14 Noble Gas and Iodine isotopes that have significant activity remaining, and would be gaseous and water insoluble, i.e., have the potential to become airborne and contribute to the dose consequences.

For the FHA analyses, the core inventory Table 3.2-1 was used to calculate the gap activity of 1 fuel assembly for input to RADTRAD-NAI as shown in Table 3.2-2.

RADTRAD-NAI makes use of dose conversion factors from Table 2.1 of Federal Guidance Report 11 (CEDE), and Table III.1 of Federal Guidance Report 12 (EDE). The dose conversion factors for the 14 isotopes listed in Table 3.2-2 and for RB-88 and Cs-135, which were used to track daughter product decay and dose contribution in RADTRAD-NAI cases, can be found in Table 3.1-3.

Table 3.2-1: Noble Gas and Iodine Core Inventory after 100 Hours Decay
(ORIGENS Output for 2981 MWt)

Nuclide	Inventory (Ci)
Kr-83m	1.160E-05
Kr-85	7.932E+05
Kr-85m	3.821E+00
Kr-88	1.387E-03
I-130	6.148E+03
I-131	5.715E+07
I-132	4.854E+07
I-133	5.895E+06
I-135	4.095E+03
Xe-131m	9.959E+05
Xe-133	1.134E+08
Xe-133m	2.103E+06
Xe-135	2.158E+05
Xe-135m	6.688E+02

Table 3.2-2: Noble Gas and Iodine Gap Inventory for a Fuel Assembly After 100 hours of Decay
(RADTRAD-NAI Input)

Isotope	Core Activity (Ci/MWt)[1]	RG 1.183 Non-LOCA Gap Fraction	Limiting FA Gap Activity @ 2958 MWt (Curies)[2]
	A	B	C
Kr-83m	3.891E-09	0.05	6.049E-09
Kr-85	2.661E+02	0.10	8.272E+02
Kr-85m	1.282E-03	0.05	1.992E-03
Kr-88	4.653E-07	0.05	7.232E-07
I-130	2.062E+00	0.05	3.206E+00
I-131	1.917E+04	0.08	4.768E+04
I-132	1.628E+04	0.05	2.531E+04
I-133	1.978E+03	0.05	3.074E+03
I-135	1.374E+00	0.05	2.135E+00
Xe-131m	3.341E+02	0.05	5.193E+02
Xe-133	3.804E+04	0.05	5.913E+04
Xe-133m	7.055E+02	0.05	1.097E+03
Xe-135	7.239E+01	0.05	1.125E+02
Xe-135m	2.244E-01	0.05	3.487E-01

1) Ci from Table 3.2.1 divided by 2981 MWt.
2) The Fuel Assembly Gap activity in Column C is determined by dividing the Core Activity by 157 assemblies and by multiplying by the core power (2958 MWt), the non-LOCA gap fraction and the radial peaking factor of 1.65.

3.2.3 Release Transport

The limiting fuel assembly gap activity shown in Table 3.2-2 is released instantaneously into the spent fuel pool or reactor cavity. The chemical form of the radioiodine released from the fuel to the spent fuel pool is 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. The CsI released from the fuel completely dissociates in the pool water. Because of the low pH of the pool water, the iodine instantaneously re-evolves as elemental iodine. The resulting chemical form of the radioiodine is 99.85% elemental iodine and 0.15% organic iodide.

Since the spent fuel pool or reactor cavity water is at least 23 feet above the top of the fuel, RG 1.183 indicates that an effective DF of 200 should be used to model iodine retention in the pool. In the RADTRAD-NAI model, the effective DF of 200 is modeled by dividing the iodine release fraction by 200 in the Release Fraction Timing input file. The speciation of the iodine above the water is entered as 57% elemental iodine and 43% organic iodine in the Plant Scenario File file.

The flow rates from the fuel building and containment were varied over a wide range to accommodate the effects of forced or natural circulation flow. A flow rate of 80,000 cfm, which bounds the capacity of fuel building and containment ventilation systems, maximized the doses. Doses calculated with little or no exhaust flow were much lower. The 80,000 cfm flow rate assured that greater than 99.99% of the activity in the buildings was released within 2 hours. No credit was taken for filtration of the release from either the fuel building or containment. Due to the high exhaust flow rate, no credit was taken for dilution or mixing of the activity released to the fuel building or containment air volumes. The release points for the releases from the fuel building and containment are discussed below in Section 3.2.4.

3.2.4 Determination of Atmospheric Dispersion Factors (X/Q)

3.2.4.1 Onsite Atmospheric Dispersion Factors

The onsite atmospheric dispersion factors were calculated by Dominion using the ARCON96 code and guidance from DG-1111. Site meteorological data taken over the years 1997-2001 were used in the calculations. Control room ground level X/Q values were calculated at the release elevation of the Auxiliary Building 291 ft. elevation louvers, Vent Stack A, Vent Stack B, and the Equipment Hatch for the Fuel Handling Accident. Only ground level releases were modeled with ARCON96 for these release points because none of the sources were at an elevation sufficient to be considered an elevated release (two and half times the height of the buildings on site). Since control room isolation is modeled during the fuel handling accident, the only receptor points used in the analysis were the control room emergency intakes.

The Containment Personnel Airlock release considered source locations of Vent Stack A from the general area exhaust portion of the Auxiliary Building ventilation subsystem and the East and West Auxiliary Building 291 ft. elevation louvers. No credit is taken for closure of the personnel airlock. Due to the site layout the Containment Equipment Hatch release is bounded by the release from the personnel airlock. The Fuel Building release occurs in the Fuel Building and to maximize Control Room doses is released through Vent Stack B. Based on these considerations the Personnel Airlock and Fuel Building releases are independent of the Unit that has had the FHA. Again, due to the site layout, the most limiting Atmospheric Dispersion Factors for the control room emergency intakes were from Vent Stack A for the Personnel Airlock release. The assumed values for control room X/Qs are reported in Table 3.2-3. These values are conservative for potential releases from the fuel building, containment equipment hatch, personnel airlock and other penetrations with a direct path to the outside atmosphere.

3.2.4.2 Offsite Atmospheric Dispersion Factors (X/Q)

The offsite atmospheric dispersion factors (EAB and LPZ) used for the FHA analysis are the same as those used for LOCA. They are reported in Table 1.3-1.

3.2.5 FHA Analysis Assumptions & Key Parameter Values

The Fuel Handling Accident model was created with the RADTRAD-NAI computer code and consisted of 3 compartments and 4 Pathways. The compartments used to model the FHA dose consequences were 1) Environment, 2) Fuel Building or 50% Containment Air Volume, and 3) Main Control Room and Emergency Switchgear Room or Main Control Room. The gaseous pathways used in the model are as follows:

P1 – Fuel Building or Containment Air Volume Release to the Environment

P2 – Unfiltered Environment flow to the Control Room

P3 – Filtered Environment flow to the Control Room

P4 – Control Room Flow to the Environment

Table 3.2-3 contains the flow rates for these pathways, along with a list of analysis assumptions and key parameter values used in the FHA analysis.

Table 3.2-3: Analysis Assumptions & Key Parameter Values Employed in the FHA Analysis

Containment Parameters			
Release Flow Rate (0 – 720 hours)			80,000 cfm *
Free Volume (for holdup; 50% of total)			920,000 ft^3
Core and Fuel Assembly Characteristics			
Number of Fuel Assemblies in Core			157
Maximum Fuel Assembly Radial Peaking Factor			1.65
Assumed Iodine Physical Form in Gap			99.85% elemental 0.15% organic
Control Room Atmospheric Dispersion Factors (sec/m^3)			
Source	Receptor		
Personnel Airlock	Emergency Control Room Intakes	0-2hr	3.75E-03
		2-8hr	2.60E-03
		8-24hr	1.03E-03
		24-96hr	7.03E-04
		96-720hr	5.52E-04
MCR/ESGR Filter Efficiencies			
Particulate	98%		
Elemental Iodine	95%		
Organic Iodine	70%		
Miscellaneous			
Effective Iodine Decontamination Factor			200
Minimum Depth of Water Over Fuel			23 feet
Fuel Building Release Flow Rate (0 – 720 hours)			80,000 cfm*
Fuel Building Volume			160,000 ft^3
Control Room Volume (isolated)			77,000 ft^3
Unfiltered Environment flow to the Control Room			400 cfm
Control Room Flow to the Environment			400 cfm
Bottled Air Flow to the Control Room			None assumed
Pressurization Flow			900 cfm**

* The modeled flow rate of 80,000 cfm exceeds the capacity of fuel building and containment ventilation systems. This flow rate assured that greater than 99.99% of the activity in the buildings was released within 2 hours.

** Pressurization flow starts 1 hour after MCR isolation.

3.2.6 FHA Analysis Results

The FHA analysis for the control room credits isolation of the MCR coincident with the FHA release, 400 cfm of unfiltered inleakage, and 900 cfm of filtered pressurization that begins 1 hour after isolation. Either automatic (signal from fuel building radiation monitors) or manual actuation of the bottled air banks causes isolation of the control room. The FHA analysis results in a dose of 4.9 rem TEDE, which is less than the limit of 5.0 rem stipulated in 10 CFR 50.67.

The offsite dose for the above configuration is 1.0 rem TEDE for the worst-case 2-hour EAB and 0.1 rem TEDE for the 30-day LPZ. These results are less than the offsite limit of 6.3 rem TEDE from RG 1.183.

3.3 Steam Generator Tube Rupture Accident

This section describes the methods employed in the Steam Generator Tube Rupture (SGTR) design basis radiological analysis and the results. This analysis included doses associated with the releases of the radioactive material initially present in primary liquid, secondary liquid, secondary steam, and releases from fuel rods that had failed before the transient. Doses were calculated at the exclusion area boundary (EAB), at the low population zone (LPZ), and in the control room. The methodology used to evaluate the control room and offsite doses resulting from the SGTR accident was consistent with Regulatory Guide 1.183 in conjunction with TEDE radiological units and limits, ARCON96 based onsite atmospheric dispersion factors, and Federal Guidance Reports (FGR) No. 11 and 12 dose conversion factors.

3.3.1 SGTR Scenario Description

A steam generator tube rupture (SGTR) is a break in a tube carrying primary coolant through the steam generator. This postulated break allows primary liquid to leak to the secondary side of one of the steam generators (denoted as the affected generator) with an assumed release to the environment through the steam generator Power Operated Relief Valves (PORVs) and the steam generator safety valves. The affected generator discharges steam to the environment for 30 minutes until the generator is isolated. The unaffected generator (2 generators modeled as one) discharges steam for a period of 8 hours until the primary system has cooled sufficiently to allow a switchover to the residual heat removal system. Consistent with current licensing analysis basis, the SGTR analysis was performed assuming both a pre-accident iodine spike and a concurrent accident iodine spike. In addition, both loss-of-offsite power (LOOP) and no loss-of-offsite power conditions were considered.

3.3.2 SGTR Source Term Definition

Initial radionuclide concentrations in the primary and secondary systems for the SGTR accident must be determined. The T/H analysis of the SGTR accident indicates that no additional fuel rod failures occur as a result of these transients. Thus, radioactive material releases were determined by the radionuclide concentrations initially present in primary liquid, secondary liquid, secondary steam, and any releases from fuel rods that have failed before the transient.

These radionuclide inventories and concentrations for the primary and secondary system liquid had been previously determined for 1% failed fuel and are documented in Tables 11.1-6 and 11.1-7 of the North Anna UFSAR, respectively. These initial values are the starting point for determining the initial curie input for the RADTRAD-NAI code runs.

Regulatory Guide 1.183 indicates that the released activities should be the maximum allowed by the Technical Specifications. Regulatory Guide 1.183 also dictates that SGTR accidents consider iodine spiking above the value allowed for normal operations based both on a pre-accident iodine spike and a concurrent accident spike. For North Anna, the maximum iodine concentration allowed as the result of an iodine spike is 60 μCi/gm dose equivalent I-131. Regulatory Guide 1.183 defines a concurrent iodine spike as an accident initiated value 335 times the release rate corresponding to the Technical Specification limit for normal operation (1 μCi/gm DE-I-131 RCS TS limit) for a period of 8 hours.

To limit the releases to the maximum allowed by the Technical Specifications, the quantity of each radionuclide in the primary system resulting from 1% failed fuel (UFSAR Table 11.1-6) was normalized to 1 μCi/gm dose equivalent I-131. This was accomplished by density correcting the quantity of each radionuclide from UFSAR Table 11.1-6 that is required to be used by Regulatory Guide 1.183 and then dividing by the total quantity of dose equivalent I-131 (calculated using RG 1.109 dose conversion factors [Reference 16]) resulting from 1% failed fuel. Similarly the quantity of each radionuclide in the secondary side liquid resulting from 1% failed fuel (UFSAR Table 11.1-7) was density corrected and divided by the quantity of dose equivalent I-131 in the secondary side liquid resulting from 1% failed fuel.

The initial secondary steam noble gas inventory is found by multiplying the primary system noble gas inventory by the dilution ratio. This dilution ratio is the ratio of the primary to secondary leak rate divided by the steam flow rate. This assumes that all noble gases are carried through the steam generator with steam flow and pass out the PORVs or safety valves and do not build up in the secondary steam.

Table 3.3-1 lists all the primary and secondary liquid radionuclide inventories resulting from 1% failed fuel that are required to be used in the analysis by Regulatory Guide 1.183. Table 3.3-2 lists the normalized primary and secondary inventories used as a starting point for the concurrent and pre-accident iodine spike cases. The concurrent iodine spike appearance rates are listed following Table 3.3-2.

The dose conversion factors used to calculate the TEDE doses for the Steam Generator Tube Rupture accident were taken from Table 3.1-3 for the isotopes required by Regulatory Guide 1.183 for the SGTR analysis.

Table 3.3-1: Primary Coolant and Secondary Side Liquid Nuclide Inventories for 1% Failed Fuel

Isotope	Primary Coolant Concentration (µCi/gm)	Secondary Liquid Concentration (µCi/gm)
Column A	Column B	Column C
Br-84	4.437E-02	4.80E-06
Rb-88	3.865E+00	2.33E-04
Rb-89	1.053E-01	5.31E-06
I-131	2.581E+00	5.83E-03
I-132	9.404E-01	8.50E-04
I-133	4.172E+00	6.43E-03
I-134	5.818E-01	1.02E-04
I-135	2.246E+00	1.51E-05
Cs-134	2.763E-01	6.61E-04
Cs-136	1.535E-01	2.68E-04
Cs-137	1.384E+00	3.31E-03
Cs-138	9.753E-01	1.02E-04
Kr-85	5.386E+00	
Kr-85m	2.219E+00	
Kr-87	1.284E+00	
Kr-88	3.879E+00	
Xe-133	2.972E+02	
Xe-133m	3.293E+00	
Xe-135	6.446E+00	
Xe-135m	1.995E-01	
Xe-138	7.088E-01	

Column B – Concentrations from North Anna UFSAR Table 11.1-6 were divided by primary coolant density to yield concentration in µCi/gm.

Column C - Concentrations from North Anna UFSAR Table 11.1-7 were divided by steam generator liquid density to yield concentration in µCi/gm.

Table 3.3-2:
Tech. Spec Weighted Iodine-Equivalent Primary and Secondary Side Nuclide Inventory For Use in the Concurrent and Pre-accident Iodine Spike Cases

Isotope	Primary Coolant Activity Ci	Secondary Liquid Activity (All 3 SG's) Ci **	Secondary Steam Activity (All 3 SG's) Ci
Column A	Column B	Column C	Column D
Br-84	2.57E+00	9.102E-03	
Rb-88	2.238E+02	4.414E-01	
Rb-89	6.101E+00	1.007E-02	
I-131	1.495E+02*	1.107E+01	7.754E-03
I-132	5.446E+01*	1.613E+00	9.410E-04
I-133	2.416E+02*	1.219E+01	1.043E-02
I-134	3.370E+01*	1.928E-01	2.784E-04
I-135	1.301E+02*	2.868E-02	4.018E-03
Cs-134	1.600E+01	1.254E+00	
Cs-136	8.888E+00	5.087E-01	
Cs-137	8.016E+01	6.284E+00	
Cs-138	5.648E+01	1.943E-01	
Kr-85	3.119E+02		7.771E-03
Kr-85m	1.285E+02		3.201E-03
Kr-87	7.434E+01		1.852E-03
Kr-88	2.246E+02		5.597E-03
Xe-133	1.721E+04		4.288E-01
Xe-133m	1.907E+02		4.751E-03
Xe-135	3.733E+02		9.302E-03
Xe-135m	1.155E+01		2.879E-04
Xe-138	4.105E+01		1.023E-03

* Iodine values must be multiplied by a value of 60 for the Pre-Accident Iodine Spike Case

** Value for the affected generator is 1/3 of this value and value for the unaffected generator is 2/3 of this value.

Column B = Primary Coolant Activity generated by dividing Column B of Table 3.3-1 by 3.4294 μCi/gm and multiplying by the total mass of the primary coolant in grams.

Column C = Secondary Coolant Activity generated by dividing Column C of Table 3.3-1 by 0.007 μCi/gm, multiplying by 0.1 (Tech Spec Limit for Secondary Side) and by multiplying by the total mass of the secondary liquid in grams.

Concurrent Spike Iodine Appearance Rates (Ci/hr) are:

I-131 7.74E+03
I-132 8.25E+03
I-133 1.49E+04
I-134 1.06E+04
I-135 1.11E+04

These appearance rates are based on a 120 gpm letdown flow rate to address the issues raised by NSAL-00-04 [Reference 24], and TS primary-to-secondary leakage.

3.3.3 Release Transport

The source term resulting from the radionuclides in the primary system coolant and from the iodine spiking in the primary system is transported to the affected steam generator by the break flow. A fraction of the break flow is assumed to flash to steam in the affected generator and to pass directly into the steam space of the affected generator with no credit taken for scrubbing by the steam generator liquid. The radionuclides initially in the steam space and those entering the steam space as the result of flashing pass directly to the environment through the Steam Generator PORVs or safety valves. The remainder of the break flow enters the steam generator liquid. Releases of radionuclides initially in the steam generator liquid and those entering the steam generator from the break flow are released as a result of secondary liquid boiling including an allowance for a partition factor of 100 for all non-noble gas isotopes. Thus 1% of the iodines and particulates are released from the steam generator liquid to the environment along with the steam flow. (Moisture carryover is not actually modeled but is instead bounded by application of the partitioning factor.) All noble gases are released from the primary system to the environment without reduction or mitigation. Releases were assumed to continue from the affected generator for 30 minutes until the affected generator was isolated. The transport model utilized for iodine and particulates was consistent with Appendix E of Regulatory Guide 1.183.

The source term resulting from the radionuclides in the primary system coolant and from the iodine spiking in the primary system is assumed to be transported to the unaffected generators by the 1_gpm_primary-to-secondary leakage specified in the Technical Specifications. All radionuclides in the primary coolant leaking into the unaffected generator are assumed to enter the steam generator liquid. Releases of radionuclides initially in the steam generator liquid and those entering the steam generator from the leakage flow are released as a result of secondary liquid boiling including an allowance for a partition factor of 100 for all non-noble gas isotopes. Thus 1% of the iodines and particulates are assumed to pass directly to the environment. No flow is assumed to pass from the liquid to the steam space to minimize holdup and decay. Radionuclides initially in the steam space are modeled to pass quickly to the environment. Again, all noble gases that are released from the primary system to the unaffected generator are released to the environment without reduction or mitigation. Releases were assumed to continue from the unaffected generator for a period of 8 hours until the primary system had cooled sufficiently to allow a switchover to the residual heat removal system.

3.3.4 Determination of Atmospheric Dispersion Factors (X/Q)

3.3.4.1 Control Room Atmospheric Dispersion Factors

The control room X/Q values are new values that were calculated by using the ARCON96 and guidance from DG-1111. The control room X/Q values are ground level X/Q values calculated at the PORV release elevation that have been reduced by a factor of 5 to credit plume rise. This reduction was taken after verifying that (1) the release point is uncapped and vertically oriented and (2) the time-dependent vertical velocity exceeds the 95th-percentile highest wind speed (at the release point height) by a factor of 5. The normal control room ventilation intake was used as the intake point for the control room since no credit was taken for control room isolation during the accident and because the normal intake is closer to the release point than the emergency ventilation intakes. The control room X/Q values used in the SGTR analysis are listed in Table 3.3-3. Table 3.3-3 shows control room X/Q values for both the normal and emergency ventilation intakes for comparative purposes. The X/Q values for the PORV release point bound releases from the steam generator safety valves.

Table 3.3-3: Control Room Atmospheric Dispersion Factors

Release Point	Receptor Point	Time Interval	Atmospheric Dispersion Factors (seconds/cubic meter)
PORV	Normal CR Intake	0 – 2 hours	2.08E-03
		2 – 8 hours	1.64E-03
		8 – 24 hours	6.46E-04
		24 – 96 hours	4.50E-04
		96 – 720 hours	3.36E-04
PORV	Emergency CR Intake	0 – 2 hours	6.28E-04
		2 – 8 hours	4.36E-04
		8 – 24 hours	1.67E-04
		24 – 96 hours	1.18E-04
		96 – 720 hours	8.72E-05

3.3.4.2 Offsite Atmospheric Dispersion Factors

The EAB and LPZ values are the same as those used in the LOCA analysis and are discussed in Section 3.1.3.2 and are listed in Table 1.3-1.

Table 3.3-4: Analysis Assumptions and Key Parameter Values Employed in the SGTR Analysis

Primary and Secondary Side Parameters

Primary system volume (cubic feet)	9786
SG Steam volume (cubic feet/SG)	3838
SG Liquid volume (cubic feet/SG)	2054
SG liquid mass (gm/SG)	4.43E+07
Control room volume (cubic feet)	2.30E+05
Primary System Mass (lb or gm)	4.37845E+05 lb or 1.986E+08 gm.
Secondary Steam Mass (lb or gm per generator)	7200 lb/SG or 3.266E+06 gm/SG
Steam Mass Dilution	2.81E+05

Full Power Properties	Steam Generator	RCS Coolant Liquid
Temperature (degrees F)	525.24	580.8
Pressure (psia)	850	2250
Density (gm/cc)	0.76096	0.71669

SGTR Flow Rates (all flow rates are in cubic feet per minute)

LOOP - Affected SG

Time	RCS to SG Liquid	RCS to SG Steam	SG Liquid to Steam	SG Steam to Environment
0 - 103 sec	9.096E+01	1.164E+01	1.494E+03	0.000E+00
103 - 232 sec	7.659E+01	3.470E+00	1.972E+02	4.994E+03
232 - 1800 sec	7.878E+01	4.860E+00	1.308E+02	3.313E+03

LOOP - Unaffected SG

Time	RCS to SG Liquid	SG Liquid to Environment
0 – 103 sec	0.1337	0
103 - 232 sec	0.1337	356.09
232 – 1800 sec	0.1337	100.81
1800 – 2 hrs	0.1337	53.80
2 hrs – 8 hrs	0.1337	38.71

NO-LOOP - Affected SG

Time	RCS to SG Liquid	RCS to SG Steam	SG Liquid to Steam	SG Steam to Environment
0 - 107 sec	8.021E+01	1.838E+01	1.494E+03	0.000E+00
107 - 196 sec	8.505E+01	3.190E+00	2.726E+02	6.904E+03
196 - 1800 sec	8.228E+01	2.460E+00	1.521E+02	3.850E+03

NO-LOOP - Unaffected SG

Time	RCS to SG Liquid	SG Liquid to Environment
0 – 107 sec	0.1337	0
107 - 196 sec	0.1337	484.85
196 – 1800 sec	0.1337	184.30
1800 – 2 hrs	0.1337	65.49
2 hrs – 8 hrs	0.1337	50.35

3.3.5 SGTR Key Analysis Assumptions and Inputs

The primary and secondary volumes along with the primary and secondary water and steam properties used in the analyses are provided in Table 3.3-4. The steam generator flow rates used in the analyses are presented in Table 3.3-4 for both the LOOP and No-LOOP cases.

In order to determine the set of control room parameters that would lead to the most conservative control room dose, a series of sensitivity cases were run in which the control room parameters such as in-leakage flow rates, normal flow rates, and recirculation flow rates were varied to determine the most limiting case. The limiting case for the control room dose assumes the control room does not isolate and unfiltered intake airflow at 3300 cfm, which includes 500 cfm of inleakage. The RADTRAD-NAI code was used to analyze cases for the pre-accident and concurrent iodine spikes with LOOP and with No-LOOP conditions.

3.3.6 SGTR Analysis Results

The results of the RADTRAD-NAI cases for the Concurrent Spike and for the Pre-Accident Iodine Spike cases are presented in Table 3.3-5 along with the applicable limits for both the LOOP and No-LOOP scenarios.

Table 3.3-5: RADTRAD-NAI-Code SGTR Results

SGTR Concurrent Iodine Spike			
	EAB 2 hour TEDE (rem)	LPZ EDE (rem)	Control Room TEDE (rem)
LOOP	1.71E-01	6.31E-03	1.02E+00
No-LOOP	1.05E-01	3.96E-03	6.16E-01
Dose Limit	2.5	2.5	5.0

SGTR Pre-accident Iodine Spike			
	EAB 2 hour TEDE (rem)	LPZ EDE (rem)	Control Room TEDE (rem)
LOOP	6.67E-01	2.38E-02	4.19E+00
No-LOOP	4.90E-01	1.75E-02	3.06E+00
Dose Limit	25	25	5.0

3.4 Main Steam Line Break Analysis

This section describes the methods employed in the Main Steam Line Break (MSLB) design basis radiological analysis and the results. This analysis included doses associated with the releases of the radioactive material initially present in primary liquid, secondary liquid, and secondary steam, plus releases from fuel rods that had failed before the transient. Doses were calculated at the exclusion area boundary (EAB), at the low population zone (LPZ), and in the control room. The methodology used to evaluate the control room and offsite doses resulting from the MSLB accident was consistent with Regulatory Guide 1.183 in conjunction with TEDE radiological units and limits, ARCON96 based onsite atmospheric dispersion factors, and FGR No. 11 and 12 dose conversion factors.

3.4.1 MSLB Scenario Description

The Main Steam Line Break (MSLB) accident begins with a break in one of the main steam lines leading from a steam generator (affected generator) to the turbine. In order to maximize control room dose, the break is assumed to occur in the turbine building. The affected steam generator releases steam for 30 minutes, at which time it is isolated. Also, it is expected that the generator will dry out in 30 minutes. Loss of off-site power is assumed. As a result, the condenser is lost and cool-down of the primary system is through the release of steam from the unaffected generators. The release from the unaffected generators continues for 8 hours through the PORV's. To maximize both the control room and the off-site doses, air exhaust from the turbine building is modeled in two separate ways. Since the emergency intake for the control room takes suction from the turbine building, slow air exhaust from the turbine building is modeled (0.2 vol./hr.) to maximize the control room dose. To maximize the offsite dose, rapid air exhaust is modeled (10 vol./hr). In accordance with RG 1.183, Appendix E, two independent cases are evaluated. Case one assumes a pre-accident iodine spike above the value allowed for normal operation, while the second case assumes a concurrent iodine spike.

3.4.2 MSLB Source Term Definition

As with the SGTR accident, the analysis of the MSLB accident indicates that no additional fuel rod failures occur as a result of the transient. Thus, radioactive material releases are determined by the radionuclide concentrations initially present in primary liquid, secondary liquid, and secondary steam, plus any releases from fuel rods that have failed before the transient.

The Main Steam Line Break analysis uses the SGTR analysis source term discussed in Section 3.3.2. The only exception is that the MSLB accident assumes a concurrent accident iodine spike

500 times the release rate corresponding to the Technical Specification limit for normal operation (1 μCi/gm DE-I-131) for a period of 8 hours, consistent with RG 1.183. The concurrent iodine spike appearance rates used in the MSLB accident analysis are shown in Table 3.4-1.

Table 3.4-1: Appearance Rates
MSLB Concurrent Accident Spike

Isotope	Appearance Rate Curies/hour
I-131	1.155E+04
I-132	1.231E+04
I-133	2.226E+04
I-134	1.583E+04
I-135	1.659E+04

3.4.3 Release Transport

The source term resulting from the radionuclides in the primary system coolant and from the iodine spiking in the primary system is transported to the steam generators by the leak-rate limiting condition for operation (1 gpm) specified in the Technical Specifications. The maximum amount of primary to secondary leakage allowed by the Technical Specifications to any one steam generator is 500 gallon per day. This leakage (500 gpd) was assigned to the affected generator. The remainder of the 1 gpm primary side to secondary side leakage was assigned to the two unaffected generators (modeled as one generator). For the affected generator, all of the leakage flow is assumed to flash to steam and to pass directly into the turbine building with no credit taken for scrubbing by the steam generator liquid. The radionuclides initially in the steam generator liquid and steam pass directly to turbine building through the broken steam line. From the turbine building it passes to the control room and to the environment. Releases were assumed to continue from the affected generator for 30 minutes until the affected generator was isolated. The transport model utilized for iodine and particulates was consistent with Appendix E of Regulatory Guide 1.183.

All radionuclides in the primary coolant leaking (940 gpd) into the unaffected generator are assumed to enter the steam generator liquid. Releases of radionuclides initially in the steam generator liquid and those entering the steam generator from the leakage flow are released as a result of secondary liquid boiling including an allowance for a partition factor of 100 for all non-noble gas isotopes. Thus 1% of the iodines and particulates are assumed to pass directly to the environment through the steam generator PORV's. No flow is assumed to pass from the liquid to the steam space to minimize holdup and decay. Radionuclides initially in the steam space are modeled as passing quickly to the environment through the PORV's. Again, all noble gases that are released from the primary system to the unaffected generator are released to the environment through the PORV's without reduction or mitigation. Releases were assumed to continue from the unaffected generator for a period of 8 hours until the primary system had cooled sufficiently to allow a switchover to the residual heat removal system.

3.4.4 Determination of Atmospheric Dispersion Factors

3.4.4.1 Control Room Atmospheric Dispersion Factors

Table 3.4-2 contains the control room X/Q values used in the MSLB dose consequence analysis. The control room X/Q values are new values that were calculated by using the ARCON96 code and guidance from DG-1111. The control room X/Q values are ground level X/Q values calculated at the PORV release elevation. They have not been reduced by a factor of 5 to credit plume rise since the releases through the unaffected generator PORV's could not always be shown to be five times the 95th percentile wind speed at the elevation of release. The normal control room ventilation intake was used as the intake point for the control room since no credit was taken for control room isolation during the accident.

Table 3.4-2: Control Room and Offsite Atmospheric Dispersion Factors

Control Room Atmospheric Dispersion Factors			
Release Point	**Receptor Point**	**Time Interval**	**Atmospheric Dispersion Factors (seconds/cubic meter)**
PORV	Normal CR Intake	0 – 2 hours	1.04E-02
		2 – 8 hours	8.20E-03
		8 – 24 hours	3.23E-03
		24 – 96 hours	2.25E-03
		96 – 720 hours	1.68E-03

3.4.4.2 Offsite Atmospheric Dispersion Factors

The EAB and LPZ values used in the MSLB analysis are the same as those used in the LOCA analysis and are discussed in Section 3.1.3.2 and are listed in Table 1.3-1.

3.4.5 MSLB Key Analysis Assumptions and Inputs

The primary and secondary volumes along with the primary and secondary water and steam properties used in the MSLB analyses are the same as those used in the SGTR analyses and are provided in Table 3.3-4. The flow rates used in the MSLB analyses are presented in Table 3.4-3. These flow rates were obtained from.

In order to determine the set of control room parameters that would lead to the most conservative control room dose, a series of sensitivity cases were run in which the control room parameters such as in-leakage flow rates, normal flow rates, and recirculation flow rates were varied to determine the most limiting case. The limiting case for the control room dose assumes the control room does not isolate and unfiltered intake airflow at 3300 cfm, which includes 500 cfm of inleakage. The RADTRAD-NAI code was used to analyze cases for the pre-accident and concurrent iodine spikes with turbine building air exchange rates of 0.2 volumes/hour and 10 volumes/hour.

Table 3.4-3: Flow rates used in MSLB Analyses
(All flow rates are in cubic feet per minute (cfm) and are based on liquid density)

From /To	**Flow Rate**	**Time**
From Primary Coolant to Unaffected SG Liquid	0.0872 cfm (940 gpd)	0 to 720 hours
From Primary Coolant to Affected SG Liquid	0.0464 cfm (500 gpd)	0 to 720 hours
From Unaffected SG Liquid to Environment	120.0 cfm (900 gpd Feed flow)	0 to 8 hours
From Affected SG Steam to Turbine Building	0.334 cfm	0 to 0.5 hours
From Unaffected SG Steam to Environment	0.666 cfm	0 to 8 hours

Additional Flow Rates

Time	**Affected SG Liquid to Turbine Building**	**Turbine Building to Environment**
0 - 10 sec	1.01E+04	1.14E+07
10 - 180 sec	5.72E+03	1.36E+06
180 - 1800 sec	1.57E+03	1.83E+05
1800 –8 hours	0.00E+00	0.00E+00

3.4.6 MSLB Analysis Results

The results of the RADTRAD-NAI cases for the Concurrent Spike and for the Pre-Accident Iodine Spike cases are presented in Table 3.4-4 along with the applicable dose limits. The control room doses are based on a turbine building air exchange rate of 0.2 volumes per hour while the offsite doses are based upon a turbine building exchange rate of 10 volumes per hour.

Table 3.4-4: RADTRAD-NAI Code MSLB Results

MSLB Concurrent Iodine Spike			
	EAB 2 hour TEDE (rem)	LPZ EDE (rem)	Control Room TEDE (rem)
Concurrent Iodine Spike	3.00E-02	1.31E-03	3.33E+00
Dose Limits	2.5	2.5	5.0

MSLB Pre-Accident Iodine Spike			
	EAB 2 hour TEDE (rem)	LPZ EDE (rem)	Control Room TEDE (rem)
Pre-Accident Iodine Spike	3.25E-02	1.21E-03	3.96E+00
Dose Limits	25.0	25.0	5.0

3.5 Locked Rotor Analysis

This section describes the methods employed in, and results of the Locked Rotor design basis radiological analysis. This analysis included doses associated with the failure of 13% of the fuel rods, which are assumed to enter DNB during the accident. Doses were calculated at the exclusion area boundary (EAB), at the low population zone (LPZ), and in the control room. The methodology used to evaluate the control room and offsite doses resulting from the Locked Rotor accident included Regulatory Guide 1.183 TEDE radiological units and limits, ARCON96-based onsite atmospheric dispersion factors, and Federal Guidance Reports (FGR) No. 11 and 12 dose conversion factors.

3.5.1 Locked Rotor Scenario Description

The Locked Rotor Accident (LRA) begins with instantaneous seizure of the rotor or the break of the shaft of a reactor coolant pump. The sudden decrease in core coolant flow while the reactor is at power results in a degradation of core heat transfer, which could result in fuel damage. It is assumed in this analysis that the reactor is operating at 102% power level (2958 MWt). A turbine trip and coincident Loss-Of-Offsite Power (LOOP) are incorporated into the analysis, resulting in a release of the accident source term through the steam generator PORVs and safety valves.

The release is modeled as starting immediately, at time t=0.0 seconds, and continuing for 8 hours, by which time the RCS temperature has reached 350°F. At this point the RHR system is activated, and the release to the atmosphere through the steam generator PORVs is terminated.

3.5.2 Locked Rotor Source Term Definition

The core source term (Curies/MWt) used in the Locked Rotor Analysis are taken from Table 3.1-3. These values were adjusted for use within RADTRAD-NAI to account for the assumed core power (2958MWt), the fraction of the fuel rods assumed to fail during the accident, and finally by the fractions of the core inventory assumed to be in the pellet-to-clad gap. In order to account for differences in power level across the core, a radial peaking factor was applied to the source term by adjusting the power level within the RADTRAD-NAI NIF file.

The LRA analysis is based on the assumption that 13% of the fuel in the core enters into DNB during the accident, and is therefore assumed to fail. Currently, the analysis for the core DNBR response confirms that no (0%) fuel failures, defined as the minimum DNBR less than the limit, occur. However, the 13% failed fuel assumption has been retained for future core design changes that may result in a core DNBR response greater than 0% fuel failures.

For non-LOCA events (including the Locked Rotor Accident) the fractions of the core inventory assumed to be in the pellet-to-clad gap for the various groups of isotopes are given in Table 3.5-1. These values (taken from RG 1.183, Table 3) were used as indicated above to adjust the core inventory data for use in the RADTRAD-NAI input.

Regulatory Guide 1.183 indicates that a radial peaking factor from the COLR should be used to account for differences in the power level across the core. For North Anna Power Station, the COLR value of radial peaking is 1.49. However, the appropriate peaking factor, including uncertainties for events that do not employ a statistical DNBR evaluation methodology (e.g. LRA), is 1.55. In order to accommodate the transition to a new fuel vendor and future design changes, a radial peaking factor of 1.65 was used. This factor was applied to the source term by adjusting the power level within the RADTRAD-NAI NIF file.

The chemical form of radioiodine released from the fuel was assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine releases from the steam generators to the environment were assumed to be 97% elemental and 3% organic.

Table 3.5-1: Non-LOCA Fraction of Fission Product Inventory in Gap

Group	Fraction
I-131	0.08
Kr-85	0.10
Other Noble Gases	0.05
Other Halogens	0.05
Alkali Metals	0.12

3.5.3 Release Transport

The source term resulting from the fuel failures discussed above is transported by the primary coolant through the leaks in the steam generator to the secondary coolant and is released to the environment through the steam generator PORVs and safety valves. The primary to secondary leak rate was assumed to be the leak-rate limiting condition for operation specified in the Technical Specifications. The leakage was assumed to continue until shutdown cooling was in operation and releases from the steam generators were terminated. The transport model utilized for iodine and particulates was consistent with Appendix E of Regulatory Guide 1.183. All noble gases that are released from the primary system were assumed to be released to the environment without reduction or mitigation.

3.5.4 Determination of Atmospheric Dispersion Factors (X/Q)

3.5.4.1 Control Room Atmospheric Dispersion Factors (X/Q)

The control room X/Q values are new values that were calculated by using the ARCON96 code and guidance from DG-1111. The control room X/Q values are ground level X/Q values calculated at the PORV release elevation that have been reduced by a factor of 5 to credit plume rise. This reduction was taken after verifying that (1) the release point is uncapped and vertically oriented and (2) the time-dependent vertical velocity exceeds the 95th-percentile highest wind speed (at the release point height) by a factor of 5. The control room X/Q values used in the LRA analysis are the normal control room intake values listed in Table 3.5-2. The discussion in Section 3:3.4.1 for the SGTR control room X/Q values is applicable to the LRA control room X/Q values.

3.5.4.2 Offsite X/Q

The EAB and LPZ values used in the Locked Rotor analysis are the same as those used in the LOCA analysis and are discussed in Section 3.1.3.2 and are listed in Table 1.3-1

3.5.5 Locked Rotor Analysis Assumptions and Key Parameters

Previous North Anna LRA analyses credited Control Room isolation at the start of the release. The current control room design basis calculation for the LRA contains no explanations for the assumption of the control room being isolated at the start of the release, but a stuck open PORV would cause an SI signal on the low pressurizer pressure early in the release. In order to eliminate the requirement for either manual or automatic action to isolate the control room, this calculation assumes no control room isolation. It also does not credit the function of the MCR/ESGR Bottled Air System and Emergency Ventilation System. It is assumed that, throughout the duration of the LRA, air is supplied to the control room through the normal intake, which has a nominal flow rate of 2800 cfm, has no filters, and is closer to the release point than the emergency ventilation intakes. The normal ventilation flow rate was varied from 1000 cfm to 4000 cfm to determine the control room dose calculation sensitivity to this parameter and to account for unfiltered inleakage. A value of 3500 cfm, which includes approximately 500 cfm of unfiltered inleakage, was assumed in the analysis.

Table 3.5- 2 summarizes the analysis assumptions and key input parameter values that are used in the Locked Rotor analysis.

Table 3.5-2: Analysis Assumptions & Key Parameter Values Employed in the LRA

NSSS Parameters	
Core Power	2958 MWt
Number of Fuel Assemblies	157
Primary System (RCS) Volume	9786 ft^3
Steam Generator Liquid Volume	6162 ft^3
Steam Generator Steam Volume	11,514 ft^3
Radial Peaking Factor	1.65
Fuel Failure During Event	13%
RG-1.183 Non-LOCA Gap Fractions	See Table 3.5-1
Main Control Room (MCR) Parameters	
Free Volume	2.30E5 ft^3
Normal Intake and Exhaust Flow Rate	3500 CFM
No Isolation of the Control Room	
Onsite Atmospheric Dispersion Factors	
Main Control Room Normal Intake	
0 – 2 hours	2.08E-3 sec/m^3
2 – 8 hours	1.64E-3 sec/m^3
8 – 24 hours	6.46E-4 sec/m^3
24 – 96 hours	4.50E-4 sec/m^3
96 – 720 hours	3.36E-4 sec/m^3

Leak Rates

Description		Flow rate (cfm)
Primary to Secondary Leakage		0.1337
SG liquid to steam	0 hr	441.6
	0.25 hr	249.9
	0.5 hr	65.5
	2 hr	50.3
	8 hr	0
SG steam to Environment	0 hr	11183
	0.25 hr	6328
	0.5 hr	1658
	2 hr	1275
	8 hr	0

3.5.6 Locked Rotor Results

The results of the design basis Locked Rotor analysis are presented in Table 3.5-3. These results report the calculated dose for the worst 2-hour interval (EAB), and for the assumed 30 day duration of the event for the control room and the LPZ. The doses are calculated with the TEDE methodology, and are compared with the applicable acceptance criteria specified in 10 CFR 50.67 and RG 1.183. As indicated in table 3.5-3, each of the results meets the acceptance criteria.

Table 3.5-3: Locked Rotor Analysis Results

	Control Room Dose (rem TEDE)	EAB Dose (rem TEDE)	LPZ Dose (rem TEDE)
Total Dose	2.33	0.24	0.03
Acceptance Criteria	5.0	2.5	2.5

4.0 Additional Design Basis Considerations

In addition to the explicit evaluation of radiological consequences that had direct impact from the changes associated with implementing the AST, other areas of plant design were also considered for potential impacts. The evaluation of these additional design areas is documented below.

4.1 Impact Upon Equipment Environmental Qualification

The NRC, in its rebaselining study of AST impact [Reference 19], considered the effects of the AST on analyses of the postulated integrated radiation doses for plant components exposed to containment atmosphere radiation sources and those exposed to containment sump radiation sources. The NRC study concluded that the increased concentration of cesium in the containment sump water could result in an increase in the postulated integrated doses for certain plant components subject to equipment qualification. The increased cesium concentration in the source term causes (beyond a specific timeframe) the calculated integrated sump doses for the NUREG-1465 source term to exceed the doses based upon the TID-14844 source term. The Reference [19] analyses indicated that the timeframe at which the doses based upon the TID-14844 source term may be exceeded and become non-conservative is from approximately 7 to 30 days after the postulated LOCA, depending upon plant-specific assumptions and features.

In the Federal Register notice issuing the final rule for use of alternative source terms at operating reactors [Reference 2], the NRC stated that it will evaluate this issue as a generic safety issue to determine whether further regulatory actions are justified. This issue was subsequently designated as Issue 187: The Potential Impact of Postulated Cesium Concentration on Equipment Qualification. Further guidance is provided in SECY-99-240 [Reference 21], which transmitted the final AST rule changes for the Commission's approval. The following is stated in the 'Discussion' section, regarding evaluation of the equipment qualification issue before its final resolution:

"In the interim period before final resolution of this issue, the staff will consider the TID-14844 source term to be acceptable in reanalyses of the impact of proposed plant modifications on previously analyzed integrated component doses regardless of the accident source term used to evaluate offsite and control room doses."

In NUREG-0933, Supplement 25 [Reference 20], the NRC staff reported its conclusions concerning the assessment of Issue 187. The staff concluded that there was no clear basis to require that the equipment qualification design basis be modified to adopt the AST. It was stated that there would be no discernable risk reduction from such a requirement. This issue was thus dropped from further pursuit. Consistent with this guidance, no further evaluation of this issue is presented in support of implementing the AST for North Anna Units 1 and 2. The existing equipment qualification analyses, which are based upon the TID-14844 source term, are considered acceptable.

4.2 Risk Impact of Proposed Changes Associated with AST Implementation

Implementation of ASTs is of benefit to licensees because of the potential to obtain relaxation in specific safeguards systems operability or surveillance requirements, since such changes can reduce regulatory burden and streamline operations. Such changes are warranted if they can be pursued without creating an unacceptable impact upon plant risk characteristics as compared with the existing system licensing and operational basis. The proposed changes associated with implementation of the AST for North Anna Units 1 and 2 have been considered for their risk effects. A discussion of these considerations is presented below.

The proposed changes are presented here for convenience; these changes are described in report sections 2.2 through 2.6:

a. Provide the dose analysis margin to allow positive containment pressure for up to four hours after the DBA (versus the current limit of one hour) in the Bases of Specifications 3.6.4, 3.6.6 and 3.6.7.

b. Define recently irradiated fuel as fuel that has occupied part of a critical reactor core within the previous 100 hours in the Bases of Specifications 3.9.4 and 3.7.15.

c. Require two MCR/ESGR EVS trains to be operable during Modes 1, 2, 3, and 4 (LCO 3.7.10) rather than 2 trains from the affected unit and one train from the other unit.

d. Delete the requirement to measure the bottled air flow rate during the 18 month surveillance (LCO 3.7.13).

e. Change the Technical Specification definition of DOSE EQUIVALENT I-131 to include an allowance to use dose conversion factors from RG 1.109 Revision 1 in the calculation of DOSE EQUIVALENT I-131.

Item a - The proposed change to allow a short duration of slightly super-atmospheric containment conditions beyond the current one hour timeframe following the design basis LOCA is in effect an increase in the containment leak rate. Reference [19] evaluated the impact of a change in containment leak rate upon plant risk. It was concluded that plant risk was not very sensitive to such a change since risk is dominated by accident sequences that result in early containment failure or bypass of containment. The same conclusion is reached, in which there is negligible effect upon overall plant risk from the proposed operation.

Item b – The changes associated with defining recently irradiated fuel are not applicable during power operation. These changes thus have no impact upon plant risk and mitigation of events occurring during power operation. These changes only provide a definition that is consistent with the decay period assumed in the Fuel Handling Accident.

Item c – The changes relating to the required number of OPERABLE MCR/ESGR EVS trains in effect removes the requirement for an MCR/ESGR EVS train to provide filtered/recirculated air during modes 1 through 4 to the control room. This relaxation has been factored into the assumptions of the LOCA analysis. These changes are accomplished while maintaining calculated doses to control room operators within the 5 rem TEDE limit of 10CFR50.67, which has been judged to be an acceptable consequence. This change is expected to have negligible impact upon plant risk associated with severe accident sequences.

Item d – The change to eliminate the flow rate measurement for the control room air bottle system relaxes a surveillance requirement that is redundant, and thus is not needed to ensure system performance is consistent with analysis assumptions. The revised AST analyses have taken credit for the pressurization function, but do not credit the cleansing effect of clean airflow from the air bottles. The air bottles are designed to provide flow during the first hour of the DBA, after which the Emergency Ventilation System provides ventilation functions. The air bottles do not perform a risk-significant function since accident scenarios that involve large radioactive releases typically develop over a timeframe that exceeds the 1 hour air bottle design. The proposed change does not have an impact upon plant risk for such accident sequences.

Item e – This change merely allows use of dose conversion factors from RG 1.109, Revision 1, which have been previously found to be acceptable for use in dose calculations. This change has no impact upon plant risk from severe accident scenarios.

It is concluded that the proposed changes associated with AST implementation for North Anna Units 1 and 2 will have insignificant effect upon the risk associated with severe accidents. This is primarily due to the fact that the risk significant accident sequences involve the failure of systems or structures (e.g., containment) that are not impacted by the relatively minor operational changes proposed herein.

4.3 Impact Upon Emergency Planning Radiological Assessment Methodology

This application of the AST for North Anna replaces the existing design basis source term with the source term defined in NUREG-1465. The MIDAS model that is employed for emergency planning radiological assessments includes definitions of source terms for various design basis accidents. Calculated results from MIDAS are used in various emergency preparedness processes. The basis of the existing source term definitions in the MIDAS calculations will be evaluated to determine: 1) the manner in which the source terms used in emergency preparedness activities rely upon the design basis event source term definition and 2) what specific changes may be warranted in the emergency preparedness source terms and their detailed usage. This assessment of potential impact will also include radiation monitor setpoint calculations for accident high range monitors, which use data input similar to MIDAS.

5.0 Conclusions

The alternative source term defined in NUREG-1465 and associated analysis guidance provided in RG-1.183 has been incorporated into the reanalysis of radiological effects from five key accidents for North Anna Units 1 and 2. This amendment request represents a full implementation of the alternative source term, making NUREG-1465 the licensing basis source term for assessment of design basis events. The analysis results from the reanalyzed events meet all of the acceptance criteria as specified in 10 CFR 50.67 and RG-1.183.

6.0 References

1. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants", U.S. Nuclear Regulatory Commission, February 1995.

2. "Use of Alternative Source Terms at Operating Reactors", Final Rule, in Federal Register No. 64, p. 71990, December 23, 1999.

3. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors", USNRC, Office of Nuclear Regulatory Research", July 2000.

4. NUREG/CR-6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation", USNRC, June 1997, S.L. Humphreys et al.

5. Technical Information Document, TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites", USAEC, 1962.

6. Computer code Scale 4.4a Version 1 Mod 0.

7. Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion", EPA 520/1-88-020, Environment Protection Agency, 1988.

8. Federal Guidance Report No. 12, "External Exposures to Radionuclides in Air, Water and Soil", EPA 420-r-93-081, Environmental Protection Agency, 1993.

9. Draft Regulatory Guide DG-1111, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants", U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Steve LaVie, December 2001.

10. NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays", USNRC, June 1993.

11. NUREG-0800, "Standard Review Plan", Section 6.5.2, "Containment Spray as a Fission Product Cleanup System", U.S. Nuclear Regulatory Commission, Rev. 2, December 1988.

12. Letter from D. A. Christian, Virginia Electric and Power Company, "Surry Power Station Units 1 and 2 – Proposed Technical Specifications and Bases Change – Alternate Source Term Implementation", April 11, 2000, Serial No. 00-123.

13. Letter from D. A. Christian, Virginia Electric and Power Company, "Surry Power Station Units 1 and 2 – Response to Request for Additional Information Alternate Source Term – Proposed Technical Specification Change", July 31, 2001, Serial No. 01-037A.

6.0 References (continued)

14. Letter from Gordon E. Edison, Office of Nuclear Reactor Regulation to David A. Christian, Virginia Electric and Power Company, "SURRY UNITS 1 AND 2 – ISSUANCE OF AMENDMENTS RE: ALTERNATIVE SOURCE TERM (TAC NOS. MA8649 AND MA8650)", March 8, 2002, Serial No. 02-170.

15. Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants", June 2003.

16. Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Compliance with 10 CFR Part 50, Appendix I", Revision 1, October 1977.

17. NUREG/CR-6331, Rev. 1, "Atmospheric Relative Concentrations in Building Wakes, ARCON96", USNRC, 1997.

18. NUREG/CR-5009, PNL-6258, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors", February 1988.

19. SECY-98-154, "Results of the Revised (NUREG-1465) Source Term Rebaselining for Operating Reactors", June 30, 1998.

20. NUREG-0933, "A Prioritization of Generic Safety Issues", Supplement 25, June 2001.

21. SECY-99-240, "Final Amendment to 10 CFR Parts 21, 50, and 54 and Availability for Public Comment of Draft Regulatory Guide DG-1081 and Draft Standard Review Plan Section 15.0.1 Regarding Use of Alternative Source Terms at Operating Reactors", October 5, 1999.

22. ASTM D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon", Reapproved 1995.

23. Generic Letter 99-02, "Laboratory Testing Of Nuclear-Grade Activated Charcoal", June 3, 1999.

24. NSAL-00-004, "Nonconservatisms in Iodine Spiking Calculations", March 7, 2000.

25. NAI 9912-04 Rev 2, "RADTRAD-NAI Version 1.0p3(QA) Documentation", July 2002.

7.0 TECHNICAL SPECIFICATION AND BASES SPECIFIC CHANGES

The following Technical Specifications for North Anna Units 1 and 2 are revised as noted below to reflect implementation of the NUREG-1465 alternative source term (AST) as the Design Basis Source Term. The AST implementation analyses provide justification for the following changes to the North Anna Technical Specifications:

a. Provide the dose analysis margin to allow positive containment pressure for up to four hours after the DBA (versus the current limit of one hour) in the Bases of Specifications 3.6.4, 3.6.6 and 3.6.7.

b. Define recently irradiated fuel as fuel that has occupied part of a critical reactor core within the previous 100 hours in the Bases of Specifications 3.9.4 and 3.7.15.

c. Require two Main Control Room and Emergency Switchgear Room Emergency Ventilation System (MCR/ESGR EVS) trains to be operable during Modes 1, 2, 3, and 4 (LCO 3.7.10) rather than 2 trains from the affected unit and one train from the other unit.

d. Delete the requirement to measure the bottled air flow rate during the 18 month surveillance (LCO 3.7.13).

e. Change the Technical Specification definition of DOSE EQUIVALENT I-131 to include an allowance to use dose conversion factors from RG 1.109 Revision 1 in the calculation of DOSE EQUIVALENT I-131.

The associated Bases changes are provided for information only. The Technical Specification Bases will be revised in accordance with the Technical Specification Bases Control Program (TS 5.5.13), following approval of the AST license amendment.

7.1 Specific Changes

In this section deleted text is omitted and inserted text is underlined in the <u>To</u> portion of each revision. The Bases changes are included with each TS that is changed. The Bases only changes are discussed after the Technical Specification changes.

Definitions

<u>Revise the current Definition of DOSE EQUIVALENT I-131 from:</u>

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites."

<u>To</u>

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites" <u>or in NRC Regulatory Guide 1.109, Revision 1, October 1977.</u>

3.7.10 Main Control Room / Emergency Switchgear Room (MCR/ESGR) Emergency Ventilation System (EVS) Modes 1-4

Revise LCO 3.7.10 from:

The following MCR/ESGR EVS trains shall be OPERABLE:
a. Two MCR/ESGR Emergency Ventilation System (EVS) trains;
and
b. One MCR/ESGR EVS train on the other unit.

To

Two MCR/ESGR EVS trains shall be OPERABLE.

Revise 3.7.10 Action A from:

One required LCO 3.7.10.a or LCO 3.7.10.b MCR/ESGR EVS train inoperable.

To

One required MCR/ESGR EVS train inoperable.

Revise 3.7.10 Action B from:

Two or more required LCO 3.7.10.a or LCO 3.7.10.b MCR/ESGR EVS trains inoperable due to inoperable MCR/ESGR boundary.

To

Two required MCR/ESGR EVS trains inoperable due to inoperable MCR/ESGR boundary.

Revise 3.7.10 Action D from:

Two or more required LCO 3.7.10.a or LCO 3.7.10.b MCR/ESGR EVS trains inoperable for reasons other than CONDITION B.

To

Two required MCR/ESGR EVS trains inoperable for reasons other than CONDITION B.

Revise SR 3.7.10.3 from:

SR 3.7.10.3 Verify each LCO 3.7.10.a MCR/ESGR EVS train actuates on an actual or simulated actuation signal.

To

Not Used

Revise Bases 3.7.10 "Background" from:

The MCR/ESGR EVS consists of four redundant trains that can filter and recirculate air inside the MCR/ESGR envelope, or supply filtered air to the MCR/ESGR envelope. The two independent and redundant unit MCR/ESGR EVS trains can actuate automatically in recirculation. Either of these trains can also be aligned to provide filtered outside air for pressurization when appropriate. One train from the other unit is required for redundancy, and can be manually actuated to provide filtered outside air or to recirculate and filter air approximately 60 minutes after the event. Each train consists of a heater, demister filter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves, dampers, and instrumentation also form part of the system. Two EVS trains are capable of performing the safety function, one supplying outside filtered air for pressurization, one filtering recirculated air. Two LCO 3.7.10.a trains and one LCO 3.7.10.b train are required for independence and redundancy.

Upon receipt of the actuating signal(s), normal air supply to and exhaust from the MCR/ESGR envelope is isolated, the two LCO 3.7.10.a trains of MCR/ESGR EVS actuate to recirculate air, and airflow from the bottled air banks maintains a positive pressure in the MCR/ESGR envelope. The MCR/ESGR envelope consists of the MCR, ESGRs, computer rooms, logic rooms, instrument rack rooms, air conditioning rooms, battery rooms, the MCR toilet, and the stairwell behind the MCR. Approximately 60 minutes after actuation of the MCR/ESGR bottled air system, a single MCR/ESGR EVS train is manually actuated to provide filtered outside air to the MCR/ESGR envelope through HEPA filters and charcoal

To

The MCR/ESGR EVS was designed as four redundant trains that can filter and recirculate air inside the MCR/ESGR envelope, or supply filtered air to the MCR/ESGR envelope. The two independent and redundant unit MCR/ESGR EVS trains on the accident unit can actuate automatically in recirculation. Either of these trains can also be aligned to provide filtered outside air for pressurization when appropriate. One train from the other unit can be manually actuated to provide filtered outside air approximately 60 minutes after the event. Each train consists of a heater, demister filter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves, dampers, and instrumentation also form part of the system. One EVS train is capable of performing the safety function of supplying outside filtered air for pressurization. Due to the location of the air intake for 1-HV-F-41, it can not be used to satisfy the requirements of LCO 3.7.10. Two of the three remaining trains (1-HV-F-42, 2-HV-F-41, and 2-HV-F-42) are required for independence and redundancy.

Upon receipt of the actuating signal(s), normal air supply to and exhaust from the MCR/ESGR envelope is isolated, two trains of MCR/ESGR EVS actuate to recirculate air, and airflow from the bottled air banks maintains a positive pressure in the MCR/ESGR envelope. The MCR/ESGR envelope consists of the MCR, ESGRs, computer rooms, logic rooms, instrument rack rooms, air conditioning rooms, battery rooms, the MCR toilet, and the stairwell behind the MCR. Approximately 60 minutes after actuation of the MCR/ESGR bottled air system, a single MCR/ESGR EVS train is manually actuated to provide filtered outside air to the MCR/ESGR envelope through HEPA filters and charcoal

Revise Bases 3.7.10 "Background" from:

Pressurization of the MCR/ESGR envelope prevents infiltration of unfiltered air from the surrounding areas of the envelope.

To

Pressurization of the MCR/ESGR envelope prevents infiltration of unfiltered air from the areas adjacent to the envelope.

Revise Bases 3.7.10 "Background" from:

Category I requirements. The actuation signal will only start the LCO 3.7.10.a MCR/ESGR EVS trains. Requiring both LCO 3.7.10.a MCR/ESGR EVS trains provides redundancy, assuring that at least one train starts in recirculation when the actuation signal is received.

The MCR/ESGR EHS is designed to maintain the control room environment for 30 days of continuous occupancy after a DBA without exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 3), and NUREG-0800, Section 6.4 (Ref. 4).

To

Category I requirements. The actuation signal will only start the MCR/ESGR EVS trains for the affected unit. Requiring two of the three MCR/ESGR EVS trains provides redundancy, assuring that at least one train is available to be realigned to provide filtered outside air for pressurization.

The MCR/ESGR EHS is designed to maintain the control room environment for 30 days of continuous occupancy after a DBA without exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 3) for alternative source terms.

Revise Bases 3.7.10 "Applicable Safety Analyses" from:

by the control room accident dose analyses for the most limiting design basis accident fission product release presented in the UFSAR, Chapter 15 (Ref. 2).

To

by the control room accident dose analyses for the most limiting design basis accident fission product release presented in the UFSAR, Chapter 15 (Ref. 2). This accident analysis assumes that at least one train is aligned for control room pressurization approximately 60 minutes after actuation of bottled air, but does not take any credit for filtration of recirculated air. Since, the MCR/ESGR EVS train associated with 1-HV-F-41 can not be used to provide outside air for filtered pressurization (due to the location of its air intake with respect to Vent Stack B) it can not be used to satisfy the requirements of LCO 3.7.10.

Revise Bases 3.7.10 "LCO" from:

Two independent and redundant MCR/ESGR EVS trains and one other unit independent and redundant MCR/ESGR EVS train are required to be OPERABLE to ensure that at least one train automatically actuates to filter recirculated air in the MCR/ESGR envelope, and at least one train is available to pressurize and to provide filtered air to the MCR/ESGR envelope, assuming a single failure disables one of the two required OPERABLE trains that automatically actuate, or disables the other unit train. Total system failure could result in exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 3), and NUREG-0800, Section 6.4 (Ref. 4), in the event of a large radioactive release.

The MCR/ESGR EVS—MODES 1, 2, 3, and 4 is considered OPERABLE when the individual components necessary to limit operator exposure are OPERABLE in the three required trains of the MCR/ESGR EVS—MODES 1, 2, 3, and 4, which include one other unit train.

To

Two independent and redundant MCR/ESGR EVS trains are required to be OPERABLE to ensure that at least one train is available to be manually aligned to pressurize and to provide filtered air to the MCR/ESGR envelope, assuming a single failure disables one of the two required OPERABLE trains. Total system failure could result in exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 3) for alternative source terms, in the event of a large radioactive release.

The MCR/ESGR EVS—MODES 1, 2, 3, and 4 is considered OPERABLE when the individual components necessary to limit operator exposure are OPERABLE in the two required trains of the MCR/ESGR EVS—MODES 1, 2, 3, and 4. 1-HV-F-41 can not be used to satisfy the requirements of LCO 3.7.10.

Revise Bases 3.7.10 "Actions A.1" from:

When one required LCO 3.7.10.a or LCO 3.7.10.b MCR/ESGR EVS train is inoperable, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining required OPERABLE MCR/ESGR EVS trains are adequate to perform the MCR/ESGR envelope protection function. However, the overall reliability is reduced because a single failure in the required OPERABLE EVS trains could result in loss of MCR/ESGR EVS function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining trains to provide the required capability.

To

When one required MCR/ESGR EVS train is inoperable, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining required OPERABLE MCR/ESGR EVS train is adequate to perform the MCR/ESGR envelope protection function. However, the overall reliability is reduced because a single failure in the required OPERABLE EVS train could result in loss of MCR/ESGR EVS function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

Revise Bases 3.7.10 "Actions B.1" from:

If the MCR/ESGR boundary is inoperable in MODE 1, 2, 3, or 4, the MCR/ESGR EVS cannot perform its intended function.

To

If the MCR/ESGR boundary is inoperable in MODE 1, 2, 3, or 4 (e.g., excessive control room inleakage or excessive ECCS leakage), the MCR/ESGR EVS cannot perform its intended function.

Revise Bases 3.7.10 "Actions D.1" from:

When two or more required LCO 3.7.10.a or LCO 3.7.10.b

To

When two required

Revise Bases 3.7.10 "SR 3.7.10.1" from:

heaters energized. The 31 day Frequency is based on the reliability of the equipment and the two train redundancy availability.

To

heaters energized. The 31 day Frequency is based on the reliability of the equipment and the one train redundancy availability.

Revise Bases 3.7.10 "SR 3.7.10.3" from:

This SR verifies that each LCO 3.7.10.a MCR/ESGR EVS train starts and operates on an actual or simulated actuation signal. The Frequency of 18 months is consistent with performing this test on a refueling interval basis.

To

Not Used

3.7.13 Main Control Room / Emergency Switchgear Room (MCR/ESGR) Bottled Air System

Revise Required Action 3.7.13.C.1 from:

Restore at least two MCR/ESGR bottled air system train to OPERABLE status.

To

Restore at least two MCR/ESGR bottled air system trains to OPERABLE status.

Revise SR 3.7.13.4 from:

Verify two required MCR/ESGR bottled air system trains can maintain a positive pressure of ≥0.05 inches water gauge, relative to the adjacent areas at a makeup flow rate of ≥340 cfm for at least 60 minutes.

To

Verify two required MCR/ESGR bottled air system trains can maintain a positive pressure of ≥0.05 inches water gauge, relative to the adjacent areas for at least 60 minutes.

Revise Bases 3.7.13 "Background" from:

In MODES 1, 2, 3, or 4, upon receipt of the actuating signal(s), normal air supply to and exhaust from the MCR/ESGR envelope is isolated, the two LCO 3.7.10.a trains of MCR/ESGR EVS actuate to recirculate air, and airflow from the bottled air banks maintains a positive pressure in the MCR/ESGR envelope. In case of a Fuel Handling Accident (FHA) during movement of recently irradiated fuel assemblies, automatic actuation of bottled air is not required, and no train of MCR/ESGR EVS is required to recirculate air.

To

In MODES 1, 2, 3, or 4, upon receipt of the actuating signal(s), normal air supply to and exhaust from the MCR/ESGR envelope is isolated, two trains of MCR/ESGR EVS actuate to recirculate air, and airflow from the bottled air banks maintains a positive pressure in the MCR/ESGR envelope. In case of a fuel handling accident (FHA) in the fuel building automatic actuation of bottled air is possible. A FHA in containment can not cause an automatic actuation of bottled air, but manual actuation is possible. After 300 hours of decay, actuation of bottled air is not required.

Revise Bases 3.7.13 "Background" from:

Pressurization of the MCR/ESGR envelope prevents infiltration of unfiltered air from the surrounding areas of the envelope.

To

Pressurization of the MCR/ESGR envelope prevents infiltration of unfiltered air from the areas adjacent to the envelope.

Revise Bases 3.7.13 "Background" from:

The MCR/ESGR EHS is designed to maintain the MCR/ESGR envelope environment for 30 days of continuous occupancy after a DBA without exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 2), and NUREG-0800, Section 6.4 (Ref. 3).

To

The MCR/ESGR EHS is designed to maintain the MCR/ESGR envelope environment for 30 days of continuous occupancy after a DBA without exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 2) for alternative source terms.

Revise Bases 3.7.13 "LCO" from:

train. Total system failure could result in exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 2), and NUREG-0800, Section 6.4 (Ref. 3), in the event of a large radioactive release.

To

train. Total system failure could result in exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 2) for alternative source terms, in the event of a large radioactive release.

Revise Bases 3.7.13 "APPLICABILITY" from:

During movement of recently irradiated fuel assemblies, the MCR/ESGR bottled air system must be OPERABLE to respond to the release from a fuel handling accident involving handling recently irradiated fuel. The MCR/ESGR bottled air system is only required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within a time frame established by analysis. The term recently is defined as all irradiated fuel assemblies, until analysis is performed to determine a specific time), due to radioactive decay.

To

During movement of recently irradiated fuel assemblies, the MCR/ESGR bottled air system must be OPERABLE to respond to the release from a fuel handling accident involving handling recently irradiated fuel. The MCR/ESGR bottled air system is only required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 300 hours), due to radioactive decay.

Revise Bases 3.7.13 "ACTIONS B.1" from:

If the MCR/ESGR boundary is inoperable in MODE 1, 2, 3, or 4, the MCR/ESGR bottled air system cannot perform its intended function.

To

If the MCR/ESGR boundary is inoperable in MODE 1, 2, 3, or 4 (e.g., excessive control room inleakage or excessive ECCS leakage), the MCR/ESGR bottled air system cannot perform its intended function.

Revise Bases 3.7.13 "SR 3.7.13.4" from:

unfiltered inleakage. The MCR/ESGR bottled air system is designed to maintain this positive pressure with two trains for at least 60 minutes at a makeup flow rate of ≥340 cfm. Testing two trains at a time at the Frequency of 18 months on a STAGGERED TEST BASIS is consistent with the guidance provided in NUREG-0800 (Ref. 3).

To

unfiltered inleakage. The MCR/ESGR bottled air system is designed to maintain this positive pressure with two trains for at least 60 minutes. Testing two trains at a time at the Frequency of 18 months on a STAGGERED TEST BASIS is consistent with the guidance provided in NUREG-0800 (Ref. 3).

3.9.4 Containment Penetrations

Revise LCO 3.9.4.b from:

b. One door in each air lock is capable of being closed; and

To

b. One door in each air lock is closed; and

Revise Bases 3.9.4 "Background" from:

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.

To

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 within the requirements of Regulatory Guide 1.183 (Ref. 2).

Revise Bases 3.9.4 "Background" from:

During movement of recently 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.

To

During movement of recently irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain closed.

Revise Bases 3.9.4 "Background" from:

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

To

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted from escaping to the environment. The closure restrictions are sufficient to restrict fission product radioactivity release from the containment due to a fuel handling accident involving handling of recently irradiated fuel.

Revise Bases 3.9.4 "APPLICABLE SAFETY ANALYSES" from:

(Ref. 1). Fuel handling accidents, analyzed in Reference 2, involve dropping a single irradiated fuel assembly and handling tool. The requirements of LCO 3.9.7, "Refueling Cavity Water Level," in conjunction with a minimum decay time of 100 hours prior to movement of recently irradiated fuel with containment closure capability or movement of fuel that has not been recently irradiated without 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. 2), 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).

To

(Ref. 1). Fuel handling accidents, analyzed in Reference 2, involve dropping a single irradiated fuel assembly and handling tool. The requirements of LCO 3.9.7, "Refueling Cavity Water Level," in conjunction with a minimum decay time of 100 hours prior to movement of irradiated fuel (i.e., fuel that has not been recently irradiated) without containment closure capability ensures that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are within the guideline values specified in Regulatory Guide 1.183 (Ref. 2).

Revise Bases 3.9.4 "LCO" from:

The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge and exhaust penetrations and containment personnel air locks.

To

The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge and exhaust penetrations.

Delete the following from Bases 3.9.4 "LCO":

The containment personnel air lock doors may be open during movement of recently irradiated fuel in the containment provided that one door is capable of being closed in the event of a fuel handling accident. Should a fuel handling accident occur inside containment, one personnel air lock door will be closed following an evacuation of the containment.

Revise Bases 3.9.4 "APPLICABILITY" from:

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. Additionally, due to radioactive decay, a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within a time frame established by analysis. The term recently is defined as all irradiated fuel assemblies, until analysis is performed to determine a specific time.) will result in doses that are well within the guideline values specified in 10 CFR 100 even without containment closure capability. Therefore, under these conditions no requirements are placed on containment penetration status.

To

In MODES 5 and 6, when movement of irradiated fuel assemblies within containment is not being conducted, the potential for a design basis fuel handling accident does not exist.

Additionally, due to radioactive decay, containment closure capability is only required during a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours). A fuel handling accident involving fuel with a minimum decay time of 100 hours prior to movement will result in doses that are within the guideline values specified in Regulatory Guide 1.183 even without containment closure capability. Therefore, under these conditions no requirements are placed on containment penetration status.

Revise Bases 3.9.4 "SR 3.9.4.1" from:

As such, this Surveillance ensures that a postulated fuel handling accident involving handling recently irradiated fuel 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 15.7.4 (Ref. 2).

To

As such, this Surveillance ensures that a postulated fuel handling accident involving handling recently irradiated fuel 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 Regulatory Guide 1.183 (Ref. 2).

Revise Bases 3.9.4 "References" from:

2. NUREG-0800, Rev. 2, July 1981.

To

2. Regulatory Guide 1.183, July 2000.

BASES ONLY CHANGES

Revise Bases 2.1.2 "Background" and "Safety Limit Violations" from:

10 CFR 100, "Reactor Site Criteria"

To

10 CFR 50.67

Revise Bases 2.1.2 "References" from:

10 CFR 100

To

10 CFR 50.67

Revise Bases 3.1.1 "LCO" from:

10 CFR 100, "Reactor Site Criteria,"

To

Regulatory Guide 1.183,

Revise Bases 3.1.1 "References" from:

10 CFR 100.

To

Regulatory Guide 1.183, July 2000.

Revise Bases 3.3.1 "Background" from:

10 CFR 50 and 10 CFR 100

To

10 CFR 50

Revise Bases 3.3.1 "Background" from:

10 CFR 100

To

10 CFR 50.67

Revise Bases 3.4.13 "APLICABLE SAFETY ANALYSES" from:

The dose consequences resulting from the SLB accident are well within the limits defined in the staff approved licensing basis.

To

The dose consequences resulting from the SLB accident are within the limits defined in the staff approved licensing basis.

Revise Bases 3.4.16 "Background" from:

The maximum dose to the whole body and the thyroid that an individual at the site boundary can receive for 2 hours during an accident is specified in 10 CFR 100 (Ref. 1). The limits on specific activity ensure that the doses are held to a small fraction of the 10 CFR 100 limits during analyzed transients and accidents.

To

The maximum dose that an individual at the site boundary can receive for 2 hours during an accident is specified in 10 CFR 50.67 (Ref. 1). The limits on specific activity ensure that the doses are held to the limits specified in Regulatory Guide 1.183 (Ref. 2) during analyzed transients and accidents.

Revise Bases 3.4.16 "Background" from:

The allowable levels are intended to limit the 2 hour dose at the site boundary to a small fraction of the 10 CFR 100 dose guideline limits. The limits in the LCO are standardized, based on parametric evaluations of offsite radioactivity dose consequences for typical site locations.

The parametric evaluations showed the potential offsite dose levels for a SGTR accident were an appropriately small fraction of the 10 CFR 100 dose guideline limits.

To

The allowable levels are intended to limit the 2 hour dose at the site boundary to the dose guideline limits specified in Regulatory Guide 1.183. The limits in the LCO are standardized, based on parametric evaluations of offsite radioactivity dose consequences for typical site locations.

The parametric evaluations showed the potential offsite dose levels for a SGTR accident were less than the dose guideline limits specified in Regulatory Guide 1.183.

Revise Bases 3.4.16 "APLICABLE SAFETY ANALYSES" from:

The LCO limits on the specific activity of the reactor coolant ensures that the resulting 2 hour doses at the site boundary will not exceed a small fraction of the 10 CFR 100 dose guideline limits following a SGTR accident. The SGTR safety analysis (Ref. 2) assumes the specific activity of the reactor coolant at the LCO limit and an existing reactor coolant steam generator (SG) tube leakage rate of 1 gpm.

To

The LCO limits on the specific activity of the reactor coolant ensures that the resulting 2 hour doses at the site boundary will not exceed the dose guideline limits specified in Regulatory Guide 1.183 following a SGTR accident. The SGTR safety analysis (Ref. 3) assumes the specific activity of the reactor coolant at the LCO limit and an existing reactor coolant steam generator (SG) tube leakage rate of 1 gpm.

Revise Bases 3.4.16 "APLICABLE SAFETY ANALYSES" from:

by a factor of 500 immediately after the accident. The second

To

by a factor of 335 immediately after the accident. The second

Revise Bases 3.4.16 "APLICABLE SAFETY ANALYSES" from:

In both cases, the noble gas activity in the reactor coolant assumes 1% failed fuel, which closely equals the LCO limit of $100/\bar{E}$ μCi/gm for gross specific activity.

To

In both cases, the noble gas activity in the reactor coolant is determined by normalizing the 1% failed fuel inventory from the UFSAR to the amount of failed fuel associated with 1.0 μCi/gm DOSE EQUIVALENT I-131.

Revise Bases 3.4.16 "APLICABLE SAFETY ANALYSES" from:

The safety analysis shows the radiological consequences of an SGTR accident are within a small fraction of the Reference 1 dose guideline limits.

To

The safety analysis shows the radiological consequences of an SGTR accident are within the Reference 2 dose guideline limits.

Revise Bases 3.4.16 "APLICABLE SAFETY ANALYSES" from:

The occurrence of an SGTR accident at these permissible levels could increase the site boundary dose levels, but still be within 10 CFR 100 dose guideline limits.

To

The occurrence of an SGTR accident at these permissible levels could increase the site boundary dose levels, but still be within 10 CFR 50.67 dose guideline limits.

Revise Bases 3.4.16 "LCO" from:

The limit on DOSE EQUIVALENT I-131 ensures the 2 hour thyroid dose to an individual at the site boundary during the Design Basis Accident (DBA) will be a small fraction of the allowed thyroid dose. The limit on gross specific activity ensures the 2 hour whole body dose to an individual at the site boundary during the DBA will be a small fraction of the allowed whole body dose.

To

The limit on DOSE EQUIVALENT I-131 ensures the 2 hour dose to an individual at the site boundary during the Design Basis Accident (DBA) will be within the limits specified in Regulatory Guide 1.183.

Revise Bases 3.4.16 "LCO" from:

The SGTR accident analysis (Ref. 2) shows that the 2 hour site boundary dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SGTR, lead to site boundary doses that exceed the 10 CFR 100 dose guideline limits.

To

The SGTR accident analysis (Ref. 3) shows that the 2 hour site boundary dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SGTR, lead to site boundary doses that exceed the Regulatory Guide 1.183 dose guideline limits.

Revise Bases 3.4.16 "REFERENCES" from:

1. 10 CFR 100.11.
2. UFSAR, Section 15.4.3.

To

1. 10 CFR 50.67.
2. Regulatory Guide 1.183, July 2000.
3. UFSAR, Section 15.4.3

Revise Bases 3.6.4 "Applicable Safety Analyses" from:

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

Containment pressure satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

To

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

The radiological consequences analysis demonstrates acceptable results provided the containment pressure decreases to 0.5 psig in 1 hour and does not exceed 0.5 psig for the interval from 1 to 4 hours following the Design Basis Accident (Ref. 3).

Containment pressure satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

Revise Bases 3.6.4 References from:

1. UFSAR, Section 6.2.
2. 10 CFR 50, Appendix K.

To

1. UFSAR, Section 6.2.
2. 10 CFR 50, Appendix K.
3. UFSAR, Section 15.4.1.7.

Revise Bases 3.6.6 "Applicable Safety Analyses" from:

Inadvertent actuation of the QS System is evaluated in the analysis, and the resultant reduction in containment pressure is calculated. The maximum calculated reduction in containment pressure results in containment pressures within the design containment minimum pressure.

The QS System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

To

Inadvertent actuation of the QS System is evaluated in the analysis, and the resultant reduction in containment pressure is calculated. The maximum calculated reduction in containment pressure results in containment pressures within the design containment minimum pressure.

The radiological consequences analysis demonstrates acceptable results provided the containment pressure decreases to 0.5 psig in 1 hour and does not exceed 0.5 psig for the interval from 1 to 4 hours following the Design Basis Accident (Ref. 4).

The QS System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

Revise Bases 3.6.6 "SR 3.6.6.2" due to of the addition of a reference from:

Flow and differential head are normal tests of centrifugal pump performance required by the ASME Code (Ref. 4).

To

Flow and differential head are normal tests of centrifugal pump performance required by the ASME Code (Ref. 5).

Revise Bases 3.6.6 "References" from:

1. UFSAR, Section 6.2.
2. 10 CFR 50.49.
3. 10 CFR 50, Appendix K.
4. ASME Code for Operation and Maintenance of Nuclear Power Plants.

To

1. UFSAR, Section 6.2.
2. 10 CFR 50.49.
3. 10 CFR 50, Appendix K.
4. UFSAR, Section 15.4.1.7.
5. ASME Code for Operation and Maintenance of Nuclear Power Plants.

Revise Bases 3.6.7 "Applicable Safety Analyses" from:

For certain aspects of accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 3).

The RS System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

To

For certain aspects of accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 3).

The radiological consequences analysis demonstrates acceptable results provided the containment pressure decreases to 0.5 psig in 1 hour and does not exceed 0.5 psig for the interval from 1 to 4 hours following the Design Basis Accident (Ref. 4).

The RS System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

Revise Bases 3.6.7 "SR 3.6.7.5" due to of the addition of a reference from:

Flow and differential head are normal tests of centrifugal pump performance required by the ASME Code (Ref. 4).

To

Flow and differential head are normal tests of centrifugal pump performance required by the ASME Code (Ref. 5).

Revise Bases 3.6.7 "References" from:

2. 10 CFR 50.49.
3. 10 CFR 50, Appendix K.
4. ASME Code for Operation and Maintenance of Nuclear Power Plants.

To

2. 10 CFR 50.49.
3. 10 CFR 50, Appendix K.
4. UFSAR, Section 15.4.1.7.
5. ASME Code for Operation and Maintenance of Nuclear Power Plants.

Revise Bases 3.7.2 "LCO" from:

10 CFR 100

To

10 CFR 50.67

Revise Bases 3.7.2 "References" from:

10 CFR 100.
To

10 CFR 50.67

Revise Bases 3.7.7 "BACKGROUND" from:

With the specified activity limit, the resultant 2 hour thyroid dose to a person at the exclusion area boundary (EAB) would be about 0.58 rem if the main steam safety valves (MSSVs) open for 2 hours following a trip from full power.

Operating a unit at the allowable limits could result in a 2 hour EAB exposure of a small fraction of the 10 CFR 100 (Ref. 1) limits, or the limits established as the NRC staff approved licensing basis.

To

If the main steam safety valves (MSSVs) open for 2 hours following a trip from full power with the specified activity limit, the resultant 2 hour dose to a person at the exclusion area boundary (EAB) would be less than 0.033 rem TEDE (the consequences of the design basis main steam line break accident).

Operating a unit at the allowable limits could result in a 2 hour EAB exposure at the Regulatory Guide 1.183 (Ref. 1) limits, or the limits established as the NRC staff approved licensing basis.

Revise Bases 3.7.7 "APPLICABLE SAFETY ANALYSES" from:

The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed a small fraction of the unit EAB limits (Ref. 1) for whole body and thyroid dose rates.

To

The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed the limits specified in Regulatory Guide 1.183 (Ref. 1).

Revise Bases 3.7.7 "LCO" from:

As indicated in the Applicable Safety Analyses, the specific activity of the secondary coolant is required to be $\leq$0.10 μCi/gm DOSE EQUIVALENT I-131 to limit the radiological consequences of a Design Basis Accident (DBA) to a small fraction of the required limit (Ref. 1).

To

As indicated in the Applicable Safety Analyses, the specific activity of the secondary coolant is required to be $\leq$0.10 μCi/gm DOSE EQUIVALENT I-131 to limit the radiological consequences of a Design Basis Accident (DBA) to the required limit (Ref. 1).

Revise Bases 3.7.7 "References" from:

10 CFR 100.11.

To

Regulatory Guide 1.183, July 2000.

Revise Bases 3.7.11 "Applicability" from:

The MCR/ESGR ACS is only required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within a time frame established by analysis. The term recently is defined as all irradiated fuel assemblies, until analysis is performed to determine a specific time), due to radioactive decay.

To

The MCR/ESGR EVS is only required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 300 hours), due to radioactive decay.

Revise Bases 3.7.12 "Applicable Safety Analyses" from:

during the recirculation mode. In such a case, the system limits radioactive release to within the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 4), and NUREG-0800, Section 6.4 (Ref. 5). The analysis

To

during the recirculation mode. In such a case, the system limits radioactive release to within the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 4) for alternative source terms. The analysis

Revise Bases 3.7.12 "LCO" from:

Two redundant trains of the ECCS PREACS are required to be OPERABLE to ensure that at least one is available. Total system failure could result in exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 4), and NUREG-0800, Section 6.4 (Ref. 5).

To

Two redundant trains of the ECCS PREACS are required to be OPERABLE to ensure that at least one is available. Total system failure could result in exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 4) for alternative source terms.

Revise Bases 3.7.14 "Background" from:

The MCR/ESGR EVS consists of four independent, redundant trains that can filter and recirculate air inside the MCR/ESGR envelope, or supply filtered air to the MCR/ESGR envelope. Each train consists of a heater, demister filter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves and dampers, and instrumentation also form part of the system. One EVS train is capable of performing the safety function, supplying filtered air for pressurization. Two of the four EVS trains are required for independence and redundancy.

In case of a Design Basis Accident (DBA) during movement of recently irradiated fuel assemblies, normal air supply to and exhaust from the MCR/ESGR envelope is manually isolated, and airflow from the bottled air banks is manually actuated to maintain a positive pressure in the MCR/ESGR envelope. The MCR/ESGR envelope consists of the MCR, ESGRs, computer rooms, logic rooms, instrument rack rooms, air conditioning rooms, battery rooms, the MCR toilet, and the stairwell behind the MCR. Approximately 60 minutes after actuation of the MCR/ESGR bottled air system, a single MCR/ESGR EVS train is manually actuated to provide filtered outside air to the MCR/ESGR envelope through HEPA filters and charcoal adsorbers for pressurization. The demisters remove any

To

The MCR/ESGR EVS was designed as four independent, redundant trains that can filter and recirculate air inside the MCR/ESGR envelope, or supply filtered air to the MCR/ESGR envelope. Each train consists of a heater, demister filter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves and dampers, and instrumentation also form part of the system. One EVS train is capable of performing the safety function of supplying filtered air for pressurization. Due to the location of the air intake for 1-HV-F-41, it can not be used to satisfy the requirements of LCO 3.7.14. Two of the three remaining trains (1-HV-F-42, 2-HV-F-41, and 2-HV-F-42) are required for independence and redundancy.

In case of a Design Basis Accident (DBA) during movement of recently irradiated fuel assemblies, an automatic (signal from the fuel building radiation monitors) or manual actuation of airflow from the bottled air banks is required. Actuation of airflow from the bottled air banks also automatically isolates the MCR/ESGR envelope to maintain positive pressure in the envelope and automatically starts all available EVS trains in recirculation mode.

The MCR/ESGR envelope consists of the MCR, ESGRs, computer rooms, logic rooms, instrument rack rooms, air conditioning rooms, battery rooms, the MCR toilet, and the stairwell behind the MCR. Approximately 60 minutes after actuation of the MCR/ESGR bottled air system, a single MCR/ESGR EVS train is manually actuated or aligned to provide filtered outside air to the MCR/ESGR envelope through HEPA filters and charcoal adsorbers for pressurization. Due to the location of the air intake for 1-HV-F-41, it should not be used in pressurization mode during a design basis fuel handling accident. There is no restriction on the use of 1-HV-F-41 in the recirculation mode. The demisters remove any

Revise Bases 3.7.14 "Background" from:

Pressurization of the MCR/ESGR envelope prevents infiltration of unfiltered air from the surrounding areas of the envelope.

To

Pressurization of the MCR/ESGR envelope prevents infiltration of unfiltered air from the areas adjacent to the envelope.

Revise Bases 3.7.14 "Background" from:

Redundant MCR/ESGR EVS supply trains provide the required pressurization and filtration should an excessive pressure drop develop across the other filter train.

To

Redundant MCR/ESGR EVS trains provide the required pressurization and filtration should an excessive pressure drop develop across the other filter train.

Revise Bases 3.7.14 "Background" from:

The MCR/ESGR EHS is designed to maintain the control room environment for 30 days of continuous occupancy after a DBA without exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 2), and NUREG-0800, Section 6.4 (Ref. 3).

To

The MCR/ESGR EHS is designed to maintain the control room environment for 30 days of continuous occupancy after a DBA without exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 2) for alternative source terms.

Revise Bases 3.7.14 "LCO" from:

Total system failure could result in exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 2), and NUREG-0800, Section 6.4 (Ref. 3), in the event of a large radioactive release.

To

Total system failure could result in exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 2) for alternative source terms, in the event of a large radioactive release.

Revise Bases 3.7.14 "Applicability" from:

The MCR/ESGR EVS is only required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within a time frame established by analysis. The term "recently irradiated fuel" is defined as all irradiated fuel assemblies, until analysis is performed to determine a specific time), due to radioactive decay.

To

The MCR/ESGR EVS is only required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 300 hours), due to radioactive decay.

Revise Bases 3.7.14 "ACTIONS B.1 and B.2" from:

B.1 and B.2

To

B.1

Revise Bases 3.7.14 "SR 3.7.14.1" from:

The 31 day Frequency is based on the reliability of the equipment and the two train redundancy availability.

To

The 31 day Frequency is based on the reliability of the equipment and the one train redundancy availability.

Revise Bases 3.7.15 "Applicable Safety Analyses" from:

Due to radioactive decay, FBVS is only required to be OPERABLE during fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within a time frame established by analysis. The term recently is defined as all irradiated fuel assemblies, until analysis is performed to determine a specific time). These assumptions and the analysis follow the guidance provided in Regulatory Guide 1.25 (Ref. 3).

To

Due to radioactive decay, FBVS is only required to be OPERABLE during fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours). These assumptions and the analysis follow the guidance provided in Regulatory Guide 1.183 (Ref. 3).

Revise Bases 3.7.15 "LCO" from:

Total system failure could result in the atmospheric release from the fuel building exceeding the 10 CFR 50, Appendix A, GDC-19 (Ref. 4) limits in the event of a fuel handling accident involving handling recently irradiated fuel.

To

Total system failure could result in the atmospheric release from the fuel building exceeding the 10 CFR 50, Appendix A, GDC-19 (Ref. 4) limits for alternative source terms, in the event of a fuel handling accident involving handling recently irradiated fuel.

Revise Bases 3.7.15 "REFERENCES" from:

3. Regulatory Guide 1.25.

To

3. Regulatory Guide 1.183, July 2000.

Revise Bases 3.7.16 "APPLICABLE SAFETY ANALYSES" from:

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 within the 10 CFR 100 (Ref. 5) limits.

To

The minimum water level in the fuel storage pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.183 (Ref. 4). The resultant 2 hour dose per person at the exclusion area boundary is within the Regulatory Guide 1.183 limits.

Revise Bases 3.7.16 "References" from:

4. Regulatory Guide 1.25.
5. 10 CFR 100.11.

To

4. Regulatory Guide 1.183, July 2000.

Revise Bases 3.8.2 "Applicability" from:

b. Systems needed to mitigate a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within a time frame established by analysis. The term recently is defined as all irradiated fuel assemblies, until analysis is performed to determine a specific time frame.) are available;

To

b. Systems needed to mitigate a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 300 hours), are available;

Revise Bases 3.8.5 "Applicability" from:

b. Required features needed to mitigate a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within a time frame established by analysis. The term recently is defined as all irradiated fuel assemblies, until analysis is performed to determine a specific time frame.) are available;

To

b. Required features needed to mitigate a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 300 hours) are available;

Revise Bases 3.8.8 "Applicability" from:

b. Systems needed to mitigate a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical core within a time frame established by analysis. The term recently is defined as all irradiated fuel assemblies, until analysis is performed to determine a specific time frame.) are available;

To

b. Systems needed to mitigate a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical core within the previous 300 hours) are available;

Revise Bases 3.8.10 "Applicability" from:

b. Systems needed to mitigate a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical core within a time frame established by analysis. The term recently is defined as all irradiated fuel assemblies, until analysis is performed to determine a specific time frame.) are available;

To

b. Systems needed to mitigate a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical core within the previous 300 hours) are available;

Revise Bases 3.9.7 "BACKGROUND" from:

Sufficient iodine activity would be retained to limit offsite doses from the accident to well below 10 CFR 100 limits.

To

Sufficient iodine activity would be retained to limit offsite doses from the accident to the limits of Regulatory Guide 1.183.

Revise Bases 3.9.7 "APPLICABLE SAFETY ANALYSES" from:

During movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment, as postulated by Regulatory Guide 1.25 (Ref. 1). A minimum water level of 23 ft (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1).

To

During movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment, as postulated by Regulatory Guide 1.183 (Ref. 1). A minimum water level of 23 ft allows an effective decontamination factor of 200 (Appendix B Assumption 2 of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99.5% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 8% of the fuel rod I-131 inventory and 5% of all other iodine isotopes, which are included as other halogens (Ref. 1).

Revise Bases 3.9.7 "APPLICABLE SAFETY ANALYSES" from:

maintained within allowable limits (Ref. 3).

To

maintained within allowable limits (Ref. 1).

Revise Bases 3.9.7 "References" from:

1. Regulatory Guide 1.25, March 23, 1972.
2. UFSAR, Section 15.4.7.
3. 10 CFR 100.10.

To

1. Regulatory Guide 1.183, July 2000.
2. UFSAR, Section 15.4.7.

Attachment 2
(Letter Serial No. 04-494A)

Proposed Technical Specification Changes
Implementation of Alternate Source Term

Revised Marked-up Technical Specifications

Virginia Electric and Power Company
(Dominion)
North Anna Power Station Units 1 and 2

The Technical Specifications proposed in the September 13, 2003 submittal have been revised to address the NRC's questions and concerns associated with the decontamination factors, filter efficiencies, and the Applicability for ECCS PREACS. The following provides the basis for the change to the original Technical Specifications pages:

TS Page	Action	Bases for Action
3.7-10-2	Revise	Editorial change to prevent renumbering of SRs
3.7.12-1	Deleted	Eliminated proposed Applicability change
3.7.14-1	Deleted	Change in DF assumptions
5.5-2	Deleted	Eliminated proposed Applicability change
5.5-13	Deleted	Unnecessary administrative change
5.5-14	Deleted	Eliminated proposed changes in filter efficiency

Bases Page		
B3.7-10-6	Revise	Editorial change to prevent renumbering of SRs
B3.7.11-2	Revised	Address recently irradiate fuel
B3.7.12-4	Deleted	Eliminated proposed Applicability change
B3.7.13-3	Revised	Change in DF assumptions
B3.7.14-1	Revised	Eliminated proposed Applicability change
B3.7.14-2	Revised	Eliminated proposed Applicability change
B3.7.14-3	Revised	Eliminated proposed Applicability change
B3.7.14-4	Revised	Eliminated proposed Applicability change
B3.8.2-4	Revised	Address recently irradiate fuel
B3.8.5-2	Revised	Address recently irradiate fuel
B3.8.8-2	Revised	Address recently irradiate fuel
B3.8.10-2	Revised	Address recently irradiate fuel
B3.9.7-1	Revised	Change in DF assumptions

1.1 Definitions

CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. The COT may be performed by means of any series of sequential, overlapping, or total channel steps.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites" or in NRC Regulatory Guide 1.109, Revision 1, October 1977.

3.7 PLANT SYSTEMS

3.7.10 Main Control Room/Emergency Switchgear Room (MCR/ESGR) Emergency Ventilation System (EVS)—MODES 1, 2, 3, and 4

LCO 3.7.10 ~~The following~~ Two MCR/ESGR EVS trains shall be OPERABLE~~:~~.

~~a. Two MCR/ESGR Emergency Ventilation System (EVS) trains; and~~

~~b. One MCR/ESGR EVS train on the other unit.~~

NOTE

The MCR/ESGR boundary may be opened intermittently under administrative control.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required ~~LCO 3.7.10.a or LCO 3.7.10.b~~ MCR/ESGR EVS train inoperable.	A.1 Restore MCR/ESGR EVS train to OPERABLE status.	7 days
B. Two ~~or more~~ required ~~LCO 3.7.10.a or LCO 3.7.10.b~~ MCR/ESGR EVS trains inoperable due to inoperable MCR/ESGR boundary.	B.1 Restore MCR/ESGR boundary to OPERABLE status.	24 hours
C. Required Action and associated Completion Time of 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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two or more required LCO 3.7.10.a or LCO 3.7.10.b MCR/ESGR EVS trains inoperable for reasons other than Condition B.	D.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Operate each required MCR/ESGR EVS train for ≥ 10 continuous hours with the heaters operating.	31 days
SR 3.7.10.2	Perform required MCR/ESGR EVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with VFTP
SR 3.7.10.3	~~Verify each LCO 3.7.10.a MCR/ESGR EVS train actuates on an actual or simulated actuation signal.~~ Not Used	~~18 months.~~
SR 3.7.10.4	Verify each required MCR/ESGR EVS train can maintain a positive pressure of ≥ 0.04 inches water gauge, relative to the adjacent areas, during the pressurization mode of operation at a makeup flow rate of ≥ 900 cfm and ≤ 1100 cfm.	18 months on a STAGGERED TEST BASIS

3.7 PLANT SYSTEMS

3.7.13 Main Control Room/Emergency Switchgear Room (MCR/ESGR) Bottled Air System

LCO 3.7.13 Three MCR/ESGR bottled air system trains shall be OPERABLE.

---------------------------- NOTE ----------------------------
The MCR/ESGR boundary may be opened intermittently under administrative control.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required MCR/ESGR bottled air system train inoperable.	A.1 Restore MCR/ESGR bottled air system train to OPERABLE status.	7 days
B. Two or more required MCR/ESGR bottled air system trains inoperable due to inoperable MCR/ESGR boundary in MODE 1, 2, 3, or 4.	B.1 Restore MCR/ESGR boundary to OPERABLE status.	24 hours
C. Two or more required MCR/ESGR bottled air system trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B.	C.1 Restore at least two MCR/ESGR bottled air system trains to OPERABLE status.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.4	Verify two required MCR/ESGR bottled air system trains can maintain a positive pressure of ≥ 0.05 inches water gauge, relative to the adjacent areas ~~at a makeup flow rate of ≥ 340 cfm~~ for at least 60 minutes.	18 months on a STAGGERED TEST BASIS

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

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

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

b. One door in each air lock is ~~capable of being~~ closed; 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 purge and exhaust isolation valve.

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

Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

APPLICABILITY: During movement of recently 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 recently irradiated fuel assemblies within containment.	Immediately

B 2.1 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure during operating conditions, the continued integrity of the RCS is ensured. According to GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, in accordance with GDC 28, "Reactivity Limits" (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2500 psia. During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR ~~100, "Reactor Site Criteria~~" 50.67 (Ref. 4).

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the reactor high pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.

(continued)

BASES

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized.

SAFETY LIMIT VIOLATIONS

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR ~~100, "Reactor Site Criteria,"~~ 50.67 limits (Ref. 4).

The allowable Completion Time of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized.

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

REFERENCES

1. UFSAR, Sections 3.1.10, 3.1.11, and 3.1.24.

2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.

3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWX-5000.

4. 10 CFR ~~100~~ 50.67.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The ejection of a control rod rapidly adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The ejection of a rod also produces a time dependent redistribution of core power.

SDM satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). Even though it is not directly observed from the control room, SDM is considered an initial condition process variable because it is periodically monitored to ensure that the unit is operating within the bounds of accident analysis assumptions.

LCO

SDM is a core design condition that can be ensured during operation through control rod positioning (control and shutdown banks) and through the soluble boron concentration.

The MSLB (Ref. 2) accident is the most limiting analysis that establishes the SDM value of the LCO. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed ~~10 CFR 100, "Reactor Site Criteria,"~~ Regulatory Guide 1.183 limits (Ref. 3).

APPLICABILITY

In MODE 2 with $k_{eff} < 1.0$ and in MODES 3, 4, and 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration." In MODES 1 and 2 with $k_{eff} > 1.0$, SDM is ensured by complying with LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits."

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 (continued)

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and the low probability of an accident occurring without the required SDM. This allows time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.

REFERENCES

1. UFSAR, Section 3.1.22.

2. UFSAR, Chapter 15.

3. ~~10 CFR 100~~. Regulatory Guide 1.183, July 2000

BASES

BACKGROUND (continued)

The Allowable Value specified in Table 3.3.1-1 serves as the LSSS such that a channel is OPERABLE if the trip setpoint is found not to exceed the Allowable Value during the CHANNEL OPERATIONAL TEST (COT). As such, the Allowable Value differs from the Trip Setpoint by an amount primarily equal to the expected instrument loop uncertainties, such as drift, during the surveillance interval. In this manner, the actual setting of the device will still meet the LSSS definition and ensure that a Safety Limit is not exceeded at any given point of time as long as the device has not drifted beyond that expected during the surveillance interval. If the actual setting of the device is found to have exceeded the Allowable Value the device would be considered inoperable for a technical specification perspective. This requires corrective action including those actions required by 10 CFR 50.36 when automatic protective devices do not function as required. Note that, although the channel is "OPERABLE" under these circumstances, the trip setpoint should be left adjusted to a value within the established trip setpoint calibration tolerance band, in accordance with uncertainty assumptions stated in the referenced set point methodology (as-left criteria), and confirmed to be operating within the statistical allowances of the uncertainty terms assigned.

During AOOs, which are those events expected to occur one or more times during the unit life, the acceptable limits are:

1. The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling (DNB);

2. Fuel centerline melt shall not occur; and

3. The RCS pressure SL of 2750 psia shall not be exceeded.

Operation within the SLs of Specification 2.0, "Safety Limits (SLs)," also maintains the above values and assures that offsite dose will be within the 10 CFR 50 ~~and 10 CFR 100~~ criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the unit life. The acceptable limit during accidents is that offsite dose shall be maintained within an acceptable fraction of 10 CFR ~~100~~ 50.67 limits. Different accident categories are allowed a different fraction of these limits, based on probability of

(continued)

BASES

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes a 1 gpm primary to secondary LEAKAGE as the initial condition.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The UFSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is only briefly released via safety valves and the majority is steamed to the condenser if offsite power is available. The 1 gpm primary to secondary LEAKAGE is relatively inconsequential in this case. If offsite power is not available, releases continue through the unaffected steam generators until the Residual Heat Removal System is placed in service. In this case, the 1 gpm primary to secondary LEAKAGE is more significant.

The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes primary to secondary LEAKAGE as an initial condition. The dose consequences resulting from the SLB accident are ~~well~~ within the limits defined in the staff approved licensing basis.

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

LCO

RCS operational LEAKAGE shall be limited to:

a. <u>Pressure Boundary LEAKAGE</u>

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND

The maximum dose ~~to the whole body and the thyroid~~ that an individual at the site boundary can receive ~~for 2 hours~~ during an accident is specified in 10 CFR ~~100~~ (50.67) (Ref. 1). The limits on specific activity ensure that the doses are held to ~~a small fraction of~~ the ~~10 CFR 100~~ limits specified in Regulatory Guide 1.183 (Ref. 2) during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The allowable levels are intended to limit the 2 hour dose at the site boundary to ~~a small fraction of~~ the ~~10 CFR 100~~ dose guideline limits specified in Regulatory Guide 1.183. The limits in the LCO are standardized, based on parametric evaluations of offsite radioactivity dose consequences for typical site locations.

The parametric evaluations showed the potential offsite dose levels for a SGTR accident were ~~an appropriately small~~ less than ~~fraction of~~ the ~~10 CFR 100~~ dose guideline limits. Each evaluation assumes a broad range of site applicable atmospheric dispersion factors in a parametric evaluation.

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensures that the resulting 2 hour doses at the site boundary will not exceed ~~a small fraction of~~ the ~~10 CFR 100~~ dose guideline limits specified in Regulatory Guide 1.183 following a SGTR accident. The SGTR safety analysis (Ref. ~~2~~ 3) assumes the specific activity of the reactor coolant at the LCO limit and an existing reactor coolant steam generator (SG) tube leakage rate of 1 gpm. The safety analysis assumes the specific activity of the secondary coolant at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.7, "Secondary Specific Activity."

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The analysis for the SGTR accident establishes the acceptance limits for RCS specific activity. Reference to this analysis is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The analysis is for two cases of reactor coolant specific activity. One case assumes specific activity at 1.0 µCi/gm DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the I-131 release rate to the reactor coolant by a factor of ~~500~~ 335 immediately after the accident. The second case assumes the initial reactor coolant iodine activity at 60.0 µCi/gm DOSE EQUIVALENT I-131 due to a pre-accident iodine spike caused by an RCS transient. In both cases, the noble gas activity in the reactor coolant ~~assumes~~ is determined by normalizing the 1% failed fuel, ~~which closely equals the LCO limit of 100/E µCi/gm for gross specific activity~~. inventory from the USFSAR to the amount of failed fuel associated with 1.0 µCi/gm DOSE EQUIVALENT I-131.

The radiologically limiting SGTR analysis also assumes a loss of offsite power at the same time as the SGTR event. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature ΔT signal.

The coincident loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG power operated relief valves and the main steam safety valves. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends.

The safety analysis shows the radiological consequences of an SGTR accident are within ~~a small fraction of~~ the Reference 2 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed the limits shown in Figure 3.4.16-1, in the applicable specification, for more than 48 hours.

The remainder of the above LCO limit permissible iodine levels shown in Figure 3.4.16-1 are acceptable because of the low probability of a SGTR accident occurring during the established 48 hour time limit. The occurrence of an SGTR accident at these permissible levels could increase the site boundary dose levels, but still be within 10 CFR ~~100~~ 50.67 dose guideline limits.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The specific iodine activity is limited to 1.0 μCi/gm DOSE EQUIVALENT I-131, and the gross specific activity in the reactor coolant is limited to the number of μCi/gm equal to 100 divided by $\bar{E}$ (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on DOSE EQUIVALENT I-131 ensures the 2 hour ~~thyroid~~ dose to an individual at the site boundary during the Design Basis Accident (DBA) will be ~~a small fraction of the allowed thyroid dose. The limit on gross specific activity ensures the 2 hour whole body dose to an individual at the site boundary during the DBA will be a small fraction of the allowed whole body dose.~~

within the limits specified in Regulatory Guide 1.183 (Ref. 2).

The SGTR accident analysis (Ref. ~~2~~ 3) shows that the 2 hour site boundary dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SGTR, lead to site boundary doses that exceed the ~~10 CFR 100~~ Regulatory Guide 1.183 dose guideline limits.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS average temperature ≥ 500°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and gross specific activity are necessary to contain the potential consequences of an SGTR to within the acceptable site boundary dose values.

For operation in MODE 3 with RCS average temperature < 500°F, and in MODES 4 and 5, the release of radioactivity in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves and SG power operated relief valves.

BASES

REFERENCES

1. 10 CFR ~~100.11~~ 50.67.

2. Regulatory Guide 1.183, July 2000.

3 ~~2~~. UFSAR, Section 15.4.3.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The maximum design internal pressure for the containment is 45.0 psig. The LOCA analyses establish the limits for the containment air partial pressure operating range. The initial conditions used in the containment design basis LOCA analyses were an air partial pressure of 11.7 psia and an air temperature of 120°F. This resulted in a maximum peak containment internal pressure of 44.1 psig, which is less than the maximum design internal pressure for the containment.

The containment was also designed for an external pressure load of 9.2 psid (i.e., a design minimum pressure of 5.5 psia). The inadvertent actuation of the QS System was analyzed to determine the reduction in containment pressure (Ref. 1). The initial conditions used in the analysis were 8.43 psia and 120°F. This resulted in a minimum pressure inside containment of 7.07 psia, which is considerably above the design minimum of 5.5 psia.

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

INSERT 1 →

Containment pressure satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

Maintaining containment pressure within the limits shown in Figure 3.6.4-1 of the LCO ensures that in the event of a DBA the resultant peak containment accident pressure will be maintained below the containment design pressure. These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of inadvertent actuation of the QS System. The LCO limits also ensure the return to subatmospheric conditions within 60 minutes following a DBA.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1

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

REFERENCES

1. UFSAR, Section 6.2.

2. 10 CFR 50, ~~Appendix K.~~

3. UFSAR, Section 15.4.1.7

BASES

APPLICABLE SAFETY ANALYSES (continued)

Therefore, it is concluded that the calculated transient containment atmosphere temperatures are acceptable for the SLB.

The modeled QS System actuation from the containment analysis is based upon a response time associated with exceeding the containment High-High pressure signal setpoint to achieving full flow through the spray nozzles. A delayed response time initiation provides conservative analyses of peak calculated containment temperature and pressure responses. The QS System total response time of 71.1 seconds comprises the signal delay, diesel generator startup time, and system startup time, including pipe fill time.

For certain aspects of accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 3).

Inadvertent actuation of the QS System is evaluated in the analysis, and the resultant reduction in containment pressure is calculated. The maximum calculated reduction in containment pressure results in containment pressures within the design containment minimum pressure.

Insert 3 →

The QS System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

During a DBA, one train of the QS System is required to provide the heat removal capability assumed in the safety analyses for containment. In addition, one QS System train, with spray pH adjusted by the contents of the chemical addition tank, is required to scavenge iodine fission products from the containment atmosphere and ensure their retention in the containment sump water. To ensure that these requirements are met, two QS System trains must be OPERABLE with power from two safety related, independent power supplies. Therefore, in the event of an accident, at least one train of QS will operate, assuming that the worst case single active failure occurs.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1 (continued)

since they were verified to be in the correct position prior to being secured. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment and capable of potentially being mispositioned are in the correct position.

SR 3.6.6.2

Verifying that each QS pump's developed head at the flow test point is greater than or equal to the required developed head ensures that QS pump performance is consistent with the safety analysis assumptions. Flow and differential head are normal tests of centrifugal pump performance required by the ASME Code (Ref. 5). Since the QS System pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.6.3 and SR 3.6.6.4

These SRs ensure that each QS automatic valve actuates to its correct position and each QS pump starts upon receipt of an actual or simulated Containment Pressure high-high signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillances when performed at an 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.6.5

With the quench spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections or an inspection of the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.5 (continued)

nozzles can be performed. This SR ensures that each spray nozzle is unobstructed and that spray coverage of the containment during an accident is not degraded. Due to the passive nature of the design of the nozzle and the non-corrosive design of the system, a test performed following maintenance which could result in nozzle blockage is considered adequate to detect obstruction of the nozzles.

REFERENCES

1. UFSAR, Section 6.2.

2. 10 CFR 50.49.

3. 10 CFR 50, Appendix K.

~~4.~~ 5. ASME Code for Operation and Maintenance of Nuclear Power Plants.

4. UFSAR, Section 15.4.1.7

BASES

APPLICABLE SAFETY ANALYSES (continued)

the containment atmosphere temperature exceeds the containment design temperature is short enough that there would be no adverse effect on equipment inside containment. Therefore, it is concluded that the calculated transient containment atmosphere temperatures are acceptable for the SLB and LOCA.

The RS System actuation model from the containment analysis is based upon a response time associated with exceeding the High-High containment pressure signal setpoint to achieving full flow through the RS System spray nozzles. A delay in response time initiation provides conservative analyses of peak calculated containment temperature and pressure. The RS System's total response time is determined by the delay timers and system startup time.

For certain aspects of accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 3).

INSERT 3

The RS System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

During a DBA, one train (one inside and one outside RS subsystem in the same train) or two outside RS subsystems of the RS System are required to provide the minimum heat removal capability assumed in the safety analysis. To ensure that this requirement is met, four RS subsystems and the casing cooling tank must be OPERABLE. This will ensure that at least one train will operate assuming the worst case single failure occurs, which is no offsite power and the loss of one emergency diesel generator. Inoperability of the casing cooling tank, the casing cooling pumps, the casing cooling valves, piping, instrumentation, or controls, or of the QS System requires an assessment of the effect on RS subsystem OPERABILITY.

Each RS train consists of one RS subsystem outside containment and one RS subsystem inside containment. Each RS subsystem includes one spray pump, one spray cooler, one

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.7.5

Verifying that each RS and casing cooling pump's developed head at the flow test point is greater than or equal to the required developed head ensures that these pumps' performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal ~~pump~~ performance required by the ASME Code (Ref. ~~1~~). Since the RS System pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program. 5

SR 3.6.7.6

These SRs ensure that each automatic valve actuates and that the RS System and casing cooling pumps start upon receipt of an actual or simulated High-High containment pressure signal. Start delay times are also verified for the RS System pumps. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was considered to be acceptable from a reliability standpoint.

SR 3.6.7.7

This SR ensures that each spray nozzle is unobstructed and that spray coverage of the containment will meet its design bases objective. Either an inspection of the nozzles or an air or smoke test is performed through each spray header. Due to the passive design of the spray header and its normally dry state, a test performed following maintenance which could result in nozzle blockage is considered adequate for detecting obstruction of the nozzles.

REFERENCES

1. UFSAR, Section 6.2.

BASES

REFERENCES (continued)

2. 10 CFR 50.49.

3. 10 CFR 50, Appendix K.

~~4.~~ 5. ASME Code for Operation and Maintenance of Nuclear Power Plants.

4. UFSAR, SECTION 15.4.1.7

Insert for Containment Systems (Bases 3.6.4, 3.6.6, and 3.6.7)

1) The radiological consequences analysis demonstrates acceptable results provided the containment pressure decreases to 0.5 psig in 1 hour and does not exceed 0.5 psig for the interval from 1 to 4 hours following the Design Basis Accident (Ref. 3). Beyond 4 hours the containment pressure is assumed to be less than 0.0 psig, terminating leakage from containment.

2) <u>This insert not used</u>

3) The radiological consequences analysis demonstrates acceptable results provided the containment pressure decreases to 0.5 psig in 1 hour and does not exceed 0.5 psig for the interval from 1 to 4 hours following the Design Basis Accident (Ref. 4). Beyond 4 hours the containment pressure is assumed to be less than 0.0 psig, terminating leakage from containment.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

d. Following a steam generator tube rupture, closure of the MSTVs isolates the ruptured steam generator from the intact steam generators to minimize radiological releases.

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

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

LCO

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

This LCO provides assurance that the MSTVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR ~~100~~ (Ref. 4) limits or the NRC staff approved licensing basis. R 50,67

APPLICABILITY

The MSTVs must be OPERABLE in MODE 1, and in MODES 2 and 3 except when closed and de-activated, when there is significant mass and energy in the RCS and steam generators. When the MSTVs are closed, they are already performing the safety function.

In MODE 4, the steam generator energy is low and the MSTVs are not required to support the safety analyses due to the low probability of a design basis accident.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.2.2

This SR verifies that each MSTV closes on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. The Frequency of MSTV testing is every 18 months. The 18 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 10.3.
2. UFSAR, Section 6.2.
3. UFSAR, Section 15.4.2.
4. 10 CFR ~~100.11.~~ 50.67
5. ASME Code for Operation and Maintenance of Nuclear Power Plants.

B 3.7 PLANT SYSTEMS

B 3.7.7 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicates current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

This limit is lower than the activity value that might be expected from a 1 gpm tube leak (LCO 3.4.13, "RCS Operational LEAKAGE") of primary coolant at the limit of 1.0 μCi/gm (LCO 3.4.16, "RCS Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives, (i.e., < 20 hours).

Insert 1 ~~With the specified activity limit, the resultant 2 hour thyroid dose to a person at the exclusion area boundary (EAB) would be about 0.58 rem if the main steam safety valves (MSSVs) open for 2 hours following a trip from full power.~~

Operating a unit at of the allowable limits could result in a 2 hour EAB exposure ~~of a small fraction of~~ the ~~10 CFR 100~~ (Ref. 1) limits, or the limits established ~~as the NRC staff~~ Regulatory Guide 1.183 approved licensing basis.

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB), as discussed in the UFSAR, Chapter 15 (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 μCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed ~~a small fraction of the unit EAB limits (Ref. 1) for whole body and thyroid dose rates~~ the limits specified in Regulatory Guide 1.183 (Ref. 1).

With the loss of offsite power, the remaining steam generators are available for core decay heat dissipation by venting steam to the atmosphere through the MSSVs and steam generator power operated relief valves (SG PORVs). The Auxiliary Feedwater System supplies the necessary makeup to the steam generators. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Residual Heat Removal System to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generator is assumed to discharge steam and any entrained activity through the MSSVs and SG PORV during the event. Since no credit is taken in the analysis for activity plateout or retention, the resultant radiological consequences represent a conservative estimate of the potential integrated dose due to the postulated steam line failure.

Secondary specific activity limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

As indicated in the Applicable Safety Analyses, the specific activity of the secondary coolant is required to be ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131 to limit the radiological consequences of a Design Basis Accident (DBA) to ~~a small fraction of~~ the required limit (Ref. 1).

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

In MODES 5 and 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS and contributes to increased post accident doses. If the secondary specific activity cannot be restored to within limits within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

This SR verifies that the secondary specific activity is within the limits of the accident analysis. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.

REFERENCES

1. ~~10 CFR 100.11~~. REGULATORY GUIDE 1.183, JULY 2000.

2. UFSAR, Chapter 15.

Inserts for Bases 3.7.7

1. If the main steam safety valves (MSSVs) open for 2 hours following a trip from full power with the specified activity limit, the resultant 2 hour dose to a person at the exclusion area boundary (EAB) would be less than 0.033 rem TEDE (the consequences of the design basis main steam line break accident).

B 3.7 PLANT SYSTEMS

B 3.7.10 Main Control Room/Emergency Switchgear Room (MCR/ESGR) Emergency Ventilation System (EVS)—MODES 1, 2, 3, and 4

BASES

BACKGROUND

The MCR/ESGR Emergency Habitability System (EHS) provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity. The MCR/ESGR EHS consists of the MCR/ESGR bottled air system (LCO 3.7.13) and the MCR/ESGR EVS (LCO 3.7.10 and LCO 3.7.14).

The MCR/ESGR EVS was designed as ~~consists of~~ four redundant trains that can filter and recirculate air inside the MCR/ESGR envelope, or supply filtered air to the MCR/ESGR envelope. The two independent and redundant unit MCR/ESGR EVS trains on the accident unit can actuate automatically in recirculation. Either of these trains can also be aligned to provide filtered outside air for pressurization when appropriate. One train from the other unit ~~is required for redundancy, and~~ can be manually actuated to provide filtered outside air ~~or to recirculate and filter air~~ approximately 60 minutes after the event. Each train consists of a heater, demister filter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves, dampers, and instrumentation also form part of the system. One ~~Two~~ EVS train is ~~are~~ capable of performing the safety function of ~~one~~ supplying outside filtered air for pressurization. ~~one filtering recirculated air. Two LCO 3.7.10.a trains and one LCO 3.7.10.b train~~ Insert 1 are required for independence and redundancy.

Upon receipt of the actuating signal(s), normal air supply to and exhaust from the MCR/ESGR envelope is isolated, ~~the~~ two ~~LCO 3.7.10.a~~ trains of MCR/ESGR EVS actuate to recirculate air, and airflow from the bottled air banks maintains a positive pressure in the MCR/ESGR envelope. The MCR/ESGR envelope consists of the MCR, ESGRs, computer rooms, logic rooms, instrument rack rooms, air conditioning rooms, battery rooms, the MCR toilet, and the stairwell behind the MCR. Approximately 60 minutes after actuation of the MCR/ESGR bottled air system, a single MCR/ESGR EVS train is manually actuated to provide filtered outside air to the MCR/ESGR envelope through HEPA filters and charcoal

(continued)

BASES

BACKGROUND (continued)

adsorbers for pressurization. The demisters remove any entrained water droplets present, to prevent excessive moisture loading of the HEPA filters and charcoal adsorbers. Continuous operation of each train for at least 10 hours per month, with the heaters on, reduces moisture buildup on the HEPA filters and adsorbers. Both the demister and heater are important to the effectiveness of the HEPA filters and charcoal adsorbers.

Pressurization of the MCR/ESGR envelope prevents infiltration of unfiltered air from the ~~surrounding~~ areas adjacent to ~~of~~ the envelope.

A single train of the MCR/ESGR EVS will pressurize the MCR/ESGR envelope to ≥ 0.04 inches water gauge. The MCR/ESGR EHS operation in maintaining the MCR/ESGR envelope habitable is discussed in the UFSAR, Section 6.4 (Ref. 1).

Redundant MCR/ESGR EVS supply and recirculation trains provide the required pressurization and filtration should an excessive pressure drop develop across the other filter train. Normally closed isolation dampers are arranged in series pairs so that the failure of one damper to open will not result in an inability of the system to perform the function based on the presence of the redundant train. The MCR/ESGR EHS is designed in accordance with Seismic Category I requirements. The actuation signal will only start the ~~LCO 3.7.10.a~~ MCR/ESGR EVS trains for the affected unit. Requiring ~~both~~ two of the three ~~LCO 3.7.10.a~~ MCR/ESGR EVS trains provides redundancy, assuring that at least one train is available to be realigned to provide filtered outside air for pressurization ~~starts in recirculation when the actuation signal is received~~.

The MCR/ESGR EHS is designed to maintain the control room environment for 30 days of continuous occupancy after a DBA without exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 3) for alternative source terms. ~~and NUREG-0800, Section 6.4 (Ref. 4)~~

APPLICABLE SAFETY ANALYSES

The MCR/ESGR EVS components are arranged in redundant, safety related ventilation trains. The location of most components and ducting within the MCR/ESGR envelope ensures an adequate supply of filtered air to all areas requiring access. The MCR/ESGR EHS provides airborne radiological protection for the control room operators, as demonstrated

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

by the control room accident dose analyses for the most limiting design basis accident fission product release presented in the UFSAR, Chapter 15 (Ref. 2). Insert (2)

The worst case single active failure of a component of the MCR/ESGR EVS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

The MCR/ESGR EVS—MODES 1, 2, 3, and 4 satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two independent and redundant MCR/ESGR EVS trains ~~and one other unit independent and redundant MCR/ESGR EVS train~~ are required to be OPERABLE to ensure that ~~at least one train automatically actuates to filter recirculated air in the MCR/ESGR envelope, and~~ at least one train is available to be manually aligned to pressurize and to provide filtered air to the MCR/ESGR envelope, assuming a single failure disables one of the two required OPERABLE trains. ~~that automatically actuate, or disables the other unit train,~~ Total system failure could result in exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 3)~~, and NUREG-0800, Section 6.4 (Ref. 4)~~, in the event of for alternative source terms, a large radioactive release.

The MCR/ESGR EVS—MODES 1, 2, 3, and 4 is considered OPERABLE when the individual components necessary to limit operator exposure are OPERABLE in the ~~three~~ two required trains of the MCR/ESGR EVS—MODES 1, 2, 3, and 4. ~~which include one other unit train.~~

Insert (3)

An MCR/ESGR EVS train is OPERABLE when the associated:

a. Fan is OPERABLE;

b. Demister filters, HEPA filters and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions; and

c. Heater, ductwork, valves, and dampers are OPERABLE, and air flow can be maintained.

The MCR/ESGR EVS is shared by Unit 1 and Unit 2.

(continued)

BASES

LCO (continued)

In addition, the MCR/ESGR boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.

The LCO is modified by a Note allowing the MCR/ESGR boundary to be opened intermittently under administrative controls. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for MCR/ESGR isolation is indicated.

APPLICABILITY

In MODES 1, 2, 3, and 4, MCR/ESGR EVS must be OPERABLE to control operator exposure during and following a DBA.

ACTIONS

A.1

When one required ~~LCO 3.7.10.a or LCO 3.7.10.b~~ MCR/ESGR EVS train is inoperable, action must be taken to restore OPERABLE status within 7 days. In this Condition, the is remaining required OPERABLE MCR/ESGR EVS train~~s are~~ adequate to perform the MCR/ESGR envelope protection function. However, the overall reliability is reduced because a single failure in the required OPERABLE EVS trains could result in loss of MCR/ESGR EVS function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining trains to provide the required capability.

B.1

If the MCR/ESGR boundary is inoperable in MODE 1, 2, 3, or 4, INSERT 4 the MCR/ESGR EVS cannot perform its intended function. Actions must be taken to restore an OPERABLE MCR/ESGR boundary within 24 hours. During the period that the MCR/ESGR boundary is inoperable, appropriate compensatory measures (consistent with the intent of GDC 19) should be utilized to protect control room operators from potential hazards such as radioactive contamination. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and

(continued)

Inserts for Bases 3.7.10

1. Due to the location of the air intake for 1-HV-F-41, it can not be used to satisfy the requirements of LCO 3.7.10. Two of the three remaining trains (1-HV-F-42, 2-HV-F-41, and 2-HV-F-42)

2. This accident analysis assumes that at least one train is aligned for control room pressurization approximately 60 minutes after actuation of bottled air, but does not take any credit for filtration of recirculated air. Since, the MCR/ESGR EVS train associated with 1-HV-F-41 can not be used to provide outside air for filtered pressurization (due to the location of its air intake with respect to Vent Stack B) it can not be used to satisfy the requirements of LCO 3.7.10.

3. 1-HV-F-41 can not be used to satisfy the requirements of LCO 3.7.10.

4. (e.g., excessive control room inleakage or excessive Emergency Core Cooling System leakage)

BASES

ACTIONS

B.1 (continued)

the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan, and possibly repair, and test most problems with the MCR/ESGR boundary.

C.1 and C.2

In MODE 1, 2, 3, or 4, if the inoperable required MCR/ESGR EVS train or the inoperable MCR/ESGR boundary cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE that minimizes accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

D.1

When two ~~or more~~ required ~~LCO 3.7.10.a or LCO 3.7.10.b~~ MCR/ESGR EVS trains are inoperable in MODE 1, 2, 3, or 4 for reasons other than an inoperable MCR/ESGR boundary (i.e., Condition B), the MCR/ESGR EVS may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on the MCR/ESGR EVS are not too severe, testing each required train once every month provides an adequate check of this system. Monthly heater operations dry out any moisture accumulated in the charcoal and HEPA filters from humidity in the ambient air. Each required train must be operated for ≥ 10 continuous hours with the heaters energized. The 31 day Frequency is based on the reliability of the equipment and the ~~two~~ one train redundancy availability.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.10.2

This SR verifies that the required MCR/ESGR EVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing the performance of the demister filter, HEPA filter, charcoal adsorber efficiency, minimum and maximum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.10.3

Not Used

~~This SR verifies that each LCO 3.7.10.a MCR/ESGR EVS train starts and operates on an actual or simulated actuation signal. The Frequency of 18 months is consistent with performing this test on a refueling interval basis.~~

SR 3.7.10.4

This SR verifies, by pressurizing the MCR/ESGR envelope, the integrity of the MCR/ESGR envelope, and the assumed inleakage rates of the potentially contaminated air. The MCR/ESGR envelope positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper functioning of the MCR/ESGR EVS. During the emergency mode of operation, the MCR/ESGR EVS is designed to pressurize the MCR/ESGR envelope ≥ 0.04 inches water gauge positive pressure with respect to adjacent areas in order to prevent unfiltered inleakage. The MCR/ESGR EVS is designed to maintain this positive pressure with one train at a makeup flow rate of ≥ 900 cfm and ≤ 1100 cfm. The Frequency of 18 months on a STAGGERED TEST BASIS is consistent with the guidance provided in NUREG-0800 (Ref. 4).

REFERENCES

1. UFSAR, Section 6.4.

2. UFSAR, Chapter 15.

3. 10 CFR 50, Appendix A.

4. NUREG-0800, Rev. 2, July 1981.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The MCR/ESGR ACS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two independent and redundant subsystems of the MCR/ESGR ACS, providing cooling to the unit ESGR and associated portion of the MCR, are required to be OPERABLE to ensure that at least one is available, assuming a single failure disabling the other subsystem. Total system failure could result in the equipment operating temperature exceeding limits in the event of an accident.

The MCR/ESGR ACS is considered to be OPERABLE when the individual components necessary to cool the MCR/ESGR envelope air are OPERABLE in both required subsystems. Each subsystem consists of two air handling units (one for the MCR and one for the ESGR), one chiller, valves, piping, instrumentation and controls. The two subsystems provide air temperature cooling to the portion of the MCR/ESGR envelope associated with the unit. In addition, an OPERABLE MCR/ESGR ACS must be capable of maintaining air circulation. An MCR/ESGR ACS subsystem does not have to be in operation to be considered OPERABLE. The MCR/ESGR ACS is considered OPERABLE when it is capable of being started by manual actions within 10 minutes. The time of 10 minutes is based on the time required to start the system manually following required testing.

APPLICABILITY

In MODES 1, 2, 3, and 4, and during movement of recently irradiated fuel assemblies, the MCR/ESGR ACS must be OPERABLE to ensure that the MCR/ESGR envelope temperature will not exceed equipment operational requirements following isolation of the MCR/ESGR envelope. The MCR/ESGR ACS is only required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within ~~a time frame established by analysis. The term recently is defined as all irradiated fuel assemblies, until analysis is performed to determine a specific time~~ the previous 300 hours), due to radioactive decay.

BASES

BACKGROUND (continued)

Building General area exhaust, fuel building exhaust, decontamination building exhaust, and containment purge exhaust.

One Safeguards Area exhaust fan is normally operating and dampers are aligned to bypass the HEPA filters and charcoal adsorbers. During emergency operations, the ECCS PREACS dampers are realigned to begin filtration. Upon receipt of the actuating Engineered Safety Feature Actuation System signal(s), normal air discharges from the Safeguards Area room are diverted through the filter banks. Two Auxiliary Building Central Exhaust fans are normally operating. Air discharges from the Auxiliary Building Central Exhaust area are manually diverted through the filter banks. Required Safeguards Area and Auxiliary Building Central Exhaust area fans are manually actuated if they are not already operating. The prefilters remove any large particles in the air to prevent excessive loading of the HEPA filters and charcoal adsorbers.

The ECCS PREACS is discussed in the UFSAR, Section 9.4 (Ref. 1) and it may be used for normal, as well as post accident, atmospheric cleanup functions. The primary purpose of the heaters is to maintain the relative humidity at an acceptable level during normal operations, generally consistent with iodine removal efficiencies per Regulatory Guide 1.52 (Ref. 3). The heaters are not required for post-accident conditions.

APPLICABLE SAFETY ANALYSES

The design basis of the ECCS PREACS is established by the large break LOCA. The system evaluation assumes ECCS leakage outside containment, such as safety injection pump leakage, during the recirculation mode. In such a case, the system limits radioactive release to within the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 4) for alternative source terms, ~~and NUREG-0800, Section 6.4 (Ref. 5)~~. The analysis of the effects and consequences of a large break LOCA is presented in Reference 2. The ECCS PREACS also may actuate following a small break LOCA, in those cases where the ECCS goes into the recirculation mode of long term cooling, to clean up releases of smaller leaks, such as from valve stem packing. The analyses assume the filtration by the ECCS PREACS does not begin for 60 minutes following an accident.

The ECCS PREACS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

Two redundant trains of the ECCS PREACS are required to be OPERABLE to ensure that at least one is available. Total system failure could result in exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 4) for alternative source terms, ~~and NUREG-0800, Section 6.4 (Ref. 5)~~.

ECCS PREACS is considered OPERABLE when the individual components necessary to maintain the ECCS pump room filtration are OPERABLE in both trains.

An ECCS PREACS train is considered OPERABLE when its associated:

a. Safeguards Area exhaust fan is OPERABLE;

b. One Auxiliary Building HEPA filter and charcoal adsorber assembly (shared with the other unit) is OPERABLE;

c. One Auxiliary Building Central exhaust system fan (shared with other unit) is OPERABLE;

d. Controls for the Auxiliary Building Central exhaust system filter and bypass dampers (shared with the other unit) are OPERABLE;

e. HEPA filter and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions; and

f. Ductwork, valves, and dampers are OPERABLE.

The Auxiliary Building Central Exhaust subsystem may be removed from service (e.g., tag out fans, open ductwork, etc.), in order to perform required testing and maintenance. The Auxiliary Building Central Exhaust subsystem is OPERABLE in this condition if it can be restored to service and perform its function within 60 minutes following an accident.

In addition, the required Safeguards Area and charging pump cubicle boundaries for charging pumps not isolated from the Reactor Coolant System must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors, except for those openings which are left open by design, including charging pump ladder wells.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.13 Main Control Room/Emergency Switchgear Room (MCR/ESGR) Bottled Air System

BASES

BACKGROUND

The MCR/ESGR Emergency Habitability System (EHS) provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity. The MCR/ESGR EHS consists of the MCR/ESGR bottled air system (LCO 3.7.13) and the MCR/ESGR Emergency Ventilation System (EVS) (LCO 3.7.10 and LCO 3.7.14).

The MCR/ESGR bottled air system consists of four trains of bottled air lined up to provide air to the MCR/ESGR envelope when the system actuates. The air is provided via four trains which feed a common header, supplying air to the Unit 1 and Unit 2 ESGRs. The header is also capable of being aligned to supply air directly to the MCR. Each train is provided air by one of the bottled air banks. Unit 1 and Unit 2 each provide two trains of bottled air. Two bottled air trains are capable of providing dry air of breathing quality to maintain a positive interior pressure in the MCR/ESGR envelope for Unit 1 and Unit 2 for a period of one hour following a Design Basis Accident (DBA).

In MODES 1, 2, 3, or 4, upon receipt of the actuating signal(s), normal air supply to and exhaust from the MCR/ESGR envelope is isolated, ~~the~~ two ~~LCO 3.7.10.a~~ trains of MCR/ESGR EVS actuate to recirculate air, and airflow from the bottled air banks maintains a positive pressure in the MCR/ESGR envelope. Insert 1 ~~In case of a Fuel Handling Accident (FHA) during movement of recently irradiated fuel assemblies, automatic actuation of bottled air is not required, and no train of MCR/ESGR EVS is required to recirculate air.~~ The MCR/ESGR envelope consists of the MCR, ESGRs, computer rooms, logic rooms, instrument rack rooms, air conditioning rooms, battery rooms, the MCR toilet, and the stairwell behind the MCR. Approximately 60 minutes after actuation of the MCR/ESGR bottled air system, a single MCR/ESGR EVS train is manually actuated to provide filtered outside air to the MCR/ESGR envelope through high efficiency particulate air (HEPA) filters and charcoal adsorbers for pressurization.

(continued)

BASES

BACKGROUND (continued)

Pressurization of the MCR/ESGR envelope prevents infiltration of unfiltered air from the ~~surrounding~~ areas ~~of~~ adjacent to the envelope.

Two trains of the MCR/ESGR bottled air system will pressurize the MCR/ESGR envelope to ≥ 0.05 inches water gauge. The MCR/ESGR EHS operation in maintaining the MCR/ESGR envelope habitable is discussed in the UFSAR, Section 6.4 (Ref. 1).

The MCR/ESGR EHS is designed in accordance with Seismic Category I requirements.

The MCR/ESGR EHS is designed to maintain the MCR/ESGR envelope environment for 30 days of continuous occupancy after a DBA without exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 2)~~, and NUREG-0800, Section 6.4 (Ref. 3)~~ for alternative source terms.

APPLICABLE SAFETY ANALYSES

The MCR/ESGR bottled air system is arranged in redundant, safety related trains providing pressurized air from the required bottled air banks to maintain a habitable environment in the MCR/ESGR envelope.

The MCR/ESGR EHS provides airborne radiological protection for the control room operators, as demonstrated by the control room accident dose analyses for the most limiting design basis accident fission product release presented in the UFSAR, Chapter 15 (Ref. 4).

The worst case single active failure of a component of the MCR/ESGR bottled air system, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

The MCR/ESGR bottled air system satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Three independent and redundant MCR/ESGR bottled air system trains are required to be OPERABLE to ensure that at least two are available assuming a single failure disables one train. Total system failure could result in exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 2) for alternative source terms~~, and NUREG-0800, Section 6.4 (Ref. 3)~~, in the event of a large radioactive release.

(continued)

BASES

LCO (continued)

The MCR/ESGR bottled air system is considered OPERABLE when the individual components necessary to limit operator exposure are OPERABLE in the three required trains of the MCR/ESGR bottled air system.

A MCR/ESGR bottled air system train is OPERABLE when:

a. One OPERABLE bottled air bank of 69 bottles is in service;

b. A flow path, including associated valves and piping, is OPERABLE; and

c. The common exhaust header is OPERABLE.

The MCR/ESGR bottled air system trains are shared by Unit 1 and Unit 2.

In addition, the MCR/ESGR boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.

The LCO is modified by a Note allowing the MCR/ESGR boundary to be opened intermittently under administrative controls. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for MCR/ESGR isolation is indicated.

APPLICABILITY

In MODES 1, 2, 3, and 4, and during movement of recently irradiated fuel assemblies, MCR/ESGR bottled air system must be OPERABLE to control operator exposure during and following a DBA.

During movement of recently irradiated fuel assemblies, the MCR/ESGR bottled air system must be OPERABLE to respond to the release from a fuel handling accident involving handling recently irradiated fuel. The MCR/ESGR bottled air system is only required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within ~~a time frame established by analysis. The term recently is defined as all irradiated fuel assemblies, until analysis is performed to determine a specific time)~~ the previous 300 hours) due to radioactive decay.

BASES

ACTIONS

A.1

When one required MCR/ESGR bottled air system train is inoperable, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining required OPERABLE MCR/ESGR bottled air system trains are adequate to perform the MCR/ESGR envelope protection function. However, the overall reliability is reduced because a single failure in one of the remaining required OPERABLE trains could result in loss of MCR/ESGR bottled air system function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining trains to provide the required capability.

B.1 (e.g., excessive control room inleakage or excessive Emergency Core Cooling System leakage)

If the MCR/ESGR boundary is inoperable in MODE 1, 2, 3, or 4, the MCR/ESGR bottled air system cannot perform its intended function. Actions must be taken to restore an OPERABLE MCR/ESGR boundary within 24 hours. During the period that the MCR/ESGR boundary is inoperable, appropriate compensatory measures (consistent with the intent of GDC 19) should be utilized to protect control room operators from potential hazards such as radioactive contamination. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan, and possibly repair, and test most problems with the MCR/ESGR boundary.

C.1

When two or more required trains of the MCR/ESGR bottled air system are inoperable in MODE 1, 2, 3, or 4 for reasons other than an inoperable MCR/ESGR boundary (i.e., Condition B), action must be taken to restore at least two of the required MCR/ESGR bottled air system trains to OPERABLE status within 24 hours. During the period that two or more required trains of the MCR/ESGR bottled air system are inoperable, appropriate compensatory measures (consistent with the intent of GDC 19) should be utilized to protect control room operators from potential hazards such as radioactive contamination. Preplanned measures should be available to address these concerns for intentional and unintentional

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.13.2

This SR verifies that the proper number of MCR/ESGR air bottles are in service, with one bank of 69 air bottles in each required train. This SR requires verification that each bottled air bank manual valve not locked, sealed, or otherwise secured and required to be open during accident conditions is open. This SR helps to ensure that the bottled air banks required to be OPERABLE to pressurize the MCR/ESGR boundary are in service. The 31 day Frequency is based on engineering judgment and was chosen to provide added assurance of the correct positions. This SR does not apply to valves that are locked, sealed, or otherwise secured in the open position, since these were verified to be in the correct position prior to locking, sealing, or securing. 2

SR 3.7.13.3

This SR verifies that each required MCR/ESGR bottled air system train actuates by verifying the flow path is opened and that the normal air supply to and exhaust from the MCR/ESGR envelope is isolated on an actual or simulated actuation signal. The Frequency of 18 months is consistent with performing this test on a refueling interval basis.

SR 3.7.13.4

This SR verifies, by pressurizing the MCR/ESGR envelope, the integrity of the MCR/ESGR envelope, and the assumed inleakage rates of the potentially contaminated air. The MCR/ESGR envelope positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper functioning of the MCR/ESGR bottled air system. During the emergency mode of operation, the MCR/ESGR bottled air system is designed to pressurize the MCR/ESGR envelope to ≥ 0.05 inches water gauge positive pressure with respect to adjacent areas in order to prevent unfiltered inleakage. The MCR/ESGR bottled air system is designed to maintain this positive pressure with two trains for at least 60 minutes~~, at a makeup flow rate of ≥ 340 cfm~~. Testing two trains at a time at the Frequency of 18 months on a STAGGERED TEST BASIS is consistent with the guidance provided in NUREG-0800 (Ref. 3).

Inserts for Bases 3.7.13

1. In case of a fuel handling accident (FHA) in the fuel building automatic actuation of bottled air is possible. A FHA in containment can not cause an automatic actuation of bottled air, but manual actuation is possible. After 300 hours of decay, actuation of bottled air is not required.

B 3.7 PLANT SYSTEMS

B 3.7.14 Main Control Room/Emergency Switchgear Room (MCR/ESGR) Emergency Ventilation System (EVS)—During Movement of Recently Irradiated Fuel Assemblies

BASES

BACKGROUND

The MCR/ESGR Emergency Habitability System (EHS) provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity. The MCR/ESGR EHS consists of the MCR/ESGR bottled air system (LCO 3.7.13) and the MCR/ESGR EVS (LCO 3.7.10 and LCO 3.7.14).

The MCR/ESGR EVS ~~consists of~~ was designed as four independent, redundant trains that can filter and recirculate air inside the MCR/ESGR envelope, or supply filtered air to the MCR/ESGR envelope. Each train consists of a heater, demister filter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves and dampers, and instrumentation also form part of the system. One EVS train is capable of performing the safety function of supplying filtered air for pressurization. ~~Two of the four EVS trains~~ Insert 1 are required for independence and redundancy.

In case of a Design Basis Accident (DBA) during movement of recently irradiated fuel assemblies, Insert 2 ~~normal air supply to and exhaust from the MCR/ESGR envelope is manually isolated, and airflow from the bottled air banks is manually actuated to maintain a positive pressure in the MCR/ESGR envelope.~~

¶ The MCR/ESGR envelope consists of the MCR, ESGRs, computer rooms, logic rooms, instrument rack rooms, air conditioning rooms, battery rooms, the MCR toilet, and the stairwell behind the MCR. Approximately 60 minutes after actuation of the MCR/ESGR bottled air system, a single MCR/ESGR EVS train is manually actuated or aligned to provide filtered outside air to the MCR/ESGR envelope through HEPA filters and charcoal adsorbers for pressurization. Insert 3 The demisters remove any entrained water droplets present in the air, to prevent excessive moisture loading of the HEPA filters and charcoal adsorbers. Continuous operation of each train for at least 10 hours per month, with the heaters on, reduces moisture

(continued)

BASES

BACKGROUND (continued)

buildup on the HEPA filters and adsorbers. Both the demister and heater are important to the effectiveness of the HEPA filters and charcoal adsorbers.

Pressurization of the MCR/ESGR envelope prevents infiltration of unfiltered air from the ~~surrounding~~ areas ~~of~~ adjacent to the envelope.

A single train of the MCR/ESGR EVS will pressurize the MCR/ESGR envelope to ≥ 0.04 inches water gauge. The MCR/ESGR EHS operation in maintaining the MCR/ESGR envelope habitable is discussed in the UFSAR, Section 6.4 (Ref. 1).

Redundant MCR/ESGR EVS supply trains provide the required pressurization and filtration should an excessive pressure drop develop across the other filter train. Normally closed isolation dampers are arranged in series pairs so that the failure of one damper to open will not result in an inability of the system to perform the function based on the presence of the redundant train. The MCR/ESGR EHS is designed in accordance with Seismic Category I requirements.

The MCR/ESGR EHS is designed to maintain the control room environment for 30 days of continuous occupancy after a DBA without exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 2) for alternative source terms ~~, and NUREG-0800, Section 6.4 (Ref. 3)~~.

APPLICABLE SAFETY ANALYSES

The MCR/ESGR EVS components are arranged in redundant, safety related ventilation trains. The location of most components and ducting within the MCR/ESGR envelope ensures an adequate supply of filtered air to all areas requiring access. The MCR/ESGR EHS provides airborne radiological protection for the control room operators, as demonstrated by the control room accident dose analyses for the most limiting design basis accident fission product release presented in the UFSAR, Chapter 15 (Ref. 4).

The worst case single active failure of a component of the MCR/ESGR EVS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

The MCR/ESGR EVS–During Movement of Recently Irradiated Fuel Assemblies satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

Two independent and redundant MCR/ESGR EVS trains are required to be OPERABLE to ensure that at least one is available assuming a single failure disables the other train. Total system failure could result in exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 2), ~~and NUREG-0800, Section 6.4 (Ref. 3)~~ for alternative source terms, in the event of a large radioactive release.

The MCR/ESGR EVS–During Movement of Recently Irradiated Fuel Assemblies is considered OPERABLE when the individual components necessary to limit operator exposure are OPERABLE in the two required trains of the MCR/ESGR EVS–During Movement of Recently Irradiated Fuel Assemblies.

An MCR/ESGR EVS train is OPERABLE when the associated:

a. Fan is OPERABLE;

b. Demister filters, HEPA filters and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions; and

c. Heater, ductwork, valves, and dampers are OPERABLE, and air flow can be maintained.

The MCR/ESGR EVS is shared by Unit 1 and Unit 2.

In addition, the MCR/ESGR boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.

The LCO is modified by a Note allowing the MCR/ESGR boundary to be opened intermittently under administrative controls. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for MCR/ESGR isolation is indicated.

APPLICABILITY

During movement of recently irradiated fuel assemblies, MCR/ESGR EVS–During Movement of Recently Irradiated Fuel Assemblies must be OPERABLE to control operator exposure during and following a DBA.

(continued)

BASES

APPLICABILITY (continued)

During movement of recently irradiated fuel assemblies, the MCR/ESGR EVS must be OPERABLE to respond to the release from a fuel handling accident involving handling recently irradiated fuel. The MCR/ESGR EVS is only required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within ~~a time frame established by analysis. The term "recently irradiated fuel" is defined as all irradiated fuel assemblies, until analysis is performed to determine a specific time~~ the previous 300 hours), due to radioactive decay.

Regarding the MCR/ESGR EVS, it ~~should be noted that they~~ are required to be OPERABLE by other LCOs in other MODES.

ACTIONS

A.1

When one required MCR/ESGR EVS train is inoperable, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining required OPERABLE MCR/ESGR EVS train is adequate to perform the MCR/ESGR envelope protection function. However, the overall reliability is reduced because a single failure in the required OPERABLE MCR/ESGR EVS train could result in loss of MCR/ESGR EVS function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining trains to provide the required capability.

B.1 ~~and B.2~~

During movement of recently irradiated fuel assemblies, if the required inoperable MCR/ESGR EVS train cannot be restored to OPERABLE status within the required Completion Time or two required MCR/ESGR EVS trains are inoperable, action must be taken to immediately suspend activities that could result in a release of radioactivity that might require isolation of the MCR/ESGR envelope. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on the MCR/ESGR EVS are not too severe,

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1 (continued)

filters from humidity in the ambient air. Each required train must be operated for ≥ 10 continuous hours with the heaters energized. The 31 day Frequency is based on the reliability of the equipment and the ~~two~~ one train redundancy availability.

SR 3.7.14.2

This SR verifies that the required MCR/ESGR EVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing the performance of the demister filter, HEPA filter, charcoal adsorber efficiency, minimum and maximum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.14.3

This SR verifies, by pressurizing the MCR/ESGR envelope, the integrity of the MCR/ESGR envelope, and the assumed inleakage rates of the potentially contaminated air. The MCR/ESGR envelope positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper functioning of the MCR/ESGR EVS. During the emergency mode of operation, the MCR/ESGR EVS is designed to pressurize the MCR/ESGR envelope ≥ 0.04 inches water gauge positive pressure with respect to adjacent areas in order to prevent unfiltered inleakage. The MCR/ESGR EVS is designed to maintain this positive pressure with one train at a makeup flow rate of ≥ 900 cfm and ≤ 1100 cfm. The Frequency of 18 months on a STAGGERED TEST BASIS is consistent with the guidance provided in NUREG-0800 (Ref. 3).

REFERENCES

1. UFSAR, Section 6.4.
2. 10 CFR 50, Appendix A.
3. NUREG-0800, Rev. 2, July 1981.
4. UFSAR, Chapter 15.

Inserts for Bases 3.7.14

1. Due to the location of the air intake for 1-HV-F-41, it can not be used to satisfy the requirements of LCO 3.7.14. Two of the three remaining trains (1-HV-F-42, 2-HV-F-41, and 2-HV-F-42)

2. an automatic (signal from the fuel building radiation monitors) or manual actuation of airflow from the bottled air banks is required. Actuation of airflow from the bottled air banks also automatically isolates the MCR/ESGR envelope to maintain positive pressure in the envelope and automatically starts all available EVS trains in recirculation mode.

3. Due to the location of the air intake for 1-HV-F-41, it should not be used in pressurization mode during a design basis fuel handling accident. There is no restriction on the use of 1-HV-F-41 in the recirculation mode.

B 3.7 PLANT SYSTEMS

B 3.7.15 Fuel Building Ventilation System (FBVS)

BASES

BACKGROUND

The FBVS discharges airborne radioactive particulates from the area of the fuel pool following a fuel handling accident. The FBVS, in conjunction with other normally operating systems, also provides environmental control of temperature and humidity in the fuel pool area.

The FBVS consists of ductwork, valves and dampers, instrumentation, and two fans.

The FBVS, which may also be operated during normal plant operations, discharges air from the fuel building.

The FBVS is discussed in the UFSAR, Sections 9.4.5 and 15.4.5 (Refs. 1 and 2, respectively) because it may be used for normal, as well as post accident functions.

APPLICABLE SAFETY ANALYSES

The FBVS design basis is established by the consequences of the limiting Design Basis Accident (DBA), which is a fuel handling accident involving handling recently irradiated fuel. The analysis of the fuel handling accident, given in Reference 2, assumes that all fuel rods in an assembly are damaged. The DBA analysis of the fuel handling accident assumes that the FBVS is functional with at least one fan operating. The amount of fission products available for release from the fuel building is determined for a fuel handling accident. Due to radioactive decay, FBVS is only required to be OPERABLE during fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within ~~a time frame established by analysis. The term recently is defined as all irradiated fuel assemblies, until analysis is performed to determine a specific time)~~ the previous 100 hours). These assumptions and the analysis follow the guidance provided in Regulatory Guide ~~1.25~~ 1.183 (Ref. 3).

The fuel handling accident analysis for the fuel building assumes all of the radioactive material available for release is discharged from the fuel building by the FBVS.

The FBVS satisfies Criterion 3 of the 10 CFR 50.36(c)(2)(ii).

BASES

LCO

for alternative source terms,

The FBVS is required to be OPERABLE and in operation. Total system failure could result in the atmospheric release from the fuel building exceeding the 10 CFR 50, Appendix A, GDC-19 (Ref. 4) limits in the event of a fuel handling accident involving handling recently irradiated fuel.

The FBVS is considered OPERABLE when the individual components are OPERABLE. The FBVS is considered OPERABLE when at least one fan is OPERABLE and in operation, the associated FBVS ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained. In addition, an OPERABLE FBVS must maintain a pressure in the fuel building pressure envelope ≤ -0.125 inches water gauge with respect to atmospheric pressure.

The LCO is modified by a Note allowing the fuel building boundary to be opened intermittently under administrative controls. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for fuel building isolation is indicated.

APPLICABILITY

During movement of recently irradiated fuel in the fuel handling area, the FBVS is required to be OPERABLE to alleviate the consequences of a fuel handling accident.

ACTIONS

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

BASES

ACTIONS (continued)

A.1

When the FBVS is inoperable or not in operation during movement of recently irradiated fuel assemblies in the fuel building, 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 recently irradiated fuel assemblies in the fuel building. This does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

This SR verifies the integrity of the fuel building pressure envelope. The ability of the fuel building to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the FBVS. The FBVS is designed to maintain a slight negative pressure in the fuel building, to prevent unfiltered LEAKAGE. The FBVS is designed to maintain a ≤ -0.125 inches water gauge with respect to atmospheric pressure. The Frequency of 18 months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 5).

REFERENCES

1. UFSAR, Section 9.4.5.
2. UFSAR, Section 15.4.5.
3. Regulatory Guide ~~1.25,~~ 1.183, July 2000.
4. 10 CFR 50, Appendix A, GDC-19.
5. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.

B 3.7 PLANT SYSTEMS

B 3.7.16 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 UFSAR, Section 9.1.2 (Ref. 1). A description of the Spent Fuel Pool Cooling and Cleanup System is given in the UFSAR, Section 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in the UFSAR, Section 15.4.5 (Ref. 3).

APPLICABLE SAFETY ANALYSES

1.183

Regulatory Guide 1.183

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 within the ~~10 CFR 100 (Ref. 5)~~ limits.

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

The fuel storage pool water level satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1 (continued)

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. UFSAR, Section 9.1.2.
2. UFSAR, Section 9.1.3.
3. UFSAR, Section 15.4.5.
4. Regulatory Guide ~~1.25.~~ 1.183, JULY 2000,
~~5. 10 CFR 100.11.~~

BASES

LCO (continued)

It is acceptable for trains to be cross tied during shutdown conditions, allowing a single offsite power circuit to supply all required trains.

APPLICABILITY

The AC sources required to be OPERABLE in MODES 5 and 6 and during movement of recently irradiated fuel assemblies provide assurance that:

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

b. Systems needed to mitigate a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within ~~a time frame established by analysis. The term recently is defined as all irradiated fuel assemblies, until analysis is performed to determine a specific time frame.~~ the previous 300 hours) are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The AC power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.1.

ACTIONS

A.1

An offsite circuit would be considered inoperable if it were not available to the necessary portions of the electrical power distribution subsystem(s). One train with offsite power available may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS and recently irradiated fuel movement. By the allowance of the option to declare required features inoperable, with no offsite power available, appropriate restrictions will be implemented in accordance with the affected required features LCO's ACTIONS.

BASES

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

LCO

The DC electrical power subsystem(s), each subsystem consisting of two batteries, one battery charger per battery, and the corresponding control equipment and interconnecting cabling within the train, are required to be OPERABLE to support required trains of the distribution systems required OPERABLE by LCO 3.8.10, "Distribution Systems–Shutdown." The EDG DC system, consisting of a battery, battery charger, and the corresponding control equipment and interconnection cabling for the EDG, are required to be OPERABLE to support the EDG required by LCO 3.8.2, "AC Sources–Shutdown." This ensures the availability of sufficient DC electrical power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents involving handling recently irradiated fuel).

APPLICABILITY

The DC electrical power sources and EDG DC system required to be OPERABLE in MODES 5 and 6, and during movement of recently irradiated fuel assemblies, provide assurance that:

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

b. Required features needed to mitigate a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within ~~a time frame established by analysis. The term recently is defined as all irradiated fuel assemblies, until analysis is performed to determine a specific time frame.~~ the previous 300 hours) are available;

c. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The inverters were previously identified as part of the distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The required inverter(s) ensure the availability of electrical power for the instrumentation for systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. The battery powered inverters provide uninterruptible supply of AC electrical power to the AC vital buses even if the 4.16 kV safety buses are de-energized. OPERABILITY of the inverters requires that the AC vital bus be powered by the inverter. This ensures the availability of sufficient inverter power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents involving handling recently irradiated fuel). Supported system(s) that do not provide automatic function(s) may be connected to a vital bus that is powered by a constant voltage transformer (example: Low Temperature Overpressure Protection, when not in automatic).

APPLICABILITY

The inverters required to be OPERABLE in MODES 5 and 6 and during movement of recently irradiated fuel assemblies provide assurance that:

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

b. Systems needed to mitigate a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical core within ~~a time frame established by analysis. The term recently is defined as all irradiated fuel assemblies, until analysis is performed to determine a specific time frame.~~) are available;

the previous 300 hours

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The AC and DC electrical power distribution systems satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Various combinations of subsystems, equipment, and components are required OPERABLE by other LCOs, depending on the specific unit condition. Implicit in those requirements is the required OPERABILITY of necessary support required features. This LCO explicitly requires energization of the portions of the electrical distribution system necessary to support OPERABILITY of required systems, equipment, and components–all specifically addressed in each LCO and implicitly required via the definition of OPERABILITY.

Maintaining these portions of the distribution system energized ensures the availability of sufficient power to operate the unit in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents involving handling recently irradiated fuel).

APPLICABILITY

The AC and DC electrical power distribution subsystems required to be OPERABLE in MODES 5 and 6, and during movement of recently irradiated fuel assemblies, provide assurance that:

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

b. Systems needed to mitigate a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical core within ~~a time frame established by analysis. The term recently is defined as all irradiated fuel assemblies, until analysis is performed to determine a specific time frame.~~ the previous 300 hours) are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition and refueling condition.

B 3.9 REFUELING OPERATIONS

B 3.9.4 Containment Penetrations

BASES

BACKGROUND

During movement of recently 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~~ Regulatory Guide 1.183 (Ref. 2). 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 recently 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." One of the containment air locks is an integral part of the containment equipment hatch. During refueling the air lock that is part of the containment equipment hatch is typically

(continued)

BASES

BACKGROUND (continued)

replaced by a temporary hatch plate. While the temporary hatch plate is installed, there is only one air lock by which to enter containment. The LCO only applies to containment air locks that are installed. 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 recently 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.

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

Insert 1

The Containment Purge and Exhaust System includes a 36 inch purge penetration and a 36 inch exhaust penetration. During MODES 1, 2, 3, and 4, the two valves in each of the purge and exhaust flow paths are secured in the closed position. The Containment Purge and Exhaust System is not subject to a Specification in MODE 5.

In MODE 6, large air exchanges are necessary to conduct refueling operations. The 36 inch purge system is used for this purpose.

The 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 recently irradiated fuel movements.

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident involving handling recently irradiated fuel. The fuel handling accident is a postulated event that involves damage to irradiated fuel

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

(Ref. 1). Fuel handling accidents, analyzed in Reference 2, involve dropping a single irradiated fuel assembly and handling tool. The requirements of LCO 3.9.7, "Refueling Cavity Water Level," in conjunction with a minimum decay time of 100 hours prior to movement of ~~recently~~ irradiated fuel ~~with containment closure capability or movement of~~ (i.e., fuel that has not been recently irradiated) without 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 Regulatory Guide 1.183 (Ref. 2). ~~10 CFR 100. Standard Review Plan, Section 15.7.4, Rev. 1 (Ref. 2), 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)~~.

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

LCO

This LCO limits the consequences of a fuel handling accident involving handling recently irradiated fuel in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge and exhaust penetrations. ~~and containment personnel air locks~~. For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that these penetrations are isolable by a containment purge and exhaust isolation valve.

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 recently irradiated fuel assemblies within containment, and 2) specified individuals are designated and readily available to isolate the flow path in the event of a fuel handling accident.

~~The containment personnel air lock doors may be open during movement of recently irradiated fuel in the containment provided that one door is capable of being closed in the event of a fuel handling accident. Should a fuel handling~~

(continued)

BASES

LCO (continued)

~~accident occur inside containment, one personnel air lock door will be closed following an evacuation of the containment~~.

APPLICABILITY

The containment penetration requirements are applicable during movement of recently 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 design basis fuel handling accident does not exist. INSERT (2) Additionally, due to radioactive decay, a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within INSERT (3) ~~a time frame established by analysis. The term recently is defined as all irradiated fuel assemblies, until analysis is performed to determine a specific time.)~~ will result in doses that are ~~well~~ within the guideline values specified in ~~10 CFR 100~~ Regulatory Guide 1.183 (Ref. 2), even without containment closure capability. 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 Purge and Exhaust Isolation System not capable of manual 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 recently 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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1 (continued)

closing. Also the Surveillance will demonstrate that each valve operator has motive power, which will ensure that each valve is capable of being manually closed.

The Surveillance is performed every 7 days during movement of recently 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 involving handling recently irradiated fuel 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 15.7.4~~ Regulatory Guide 1.183 (Ref. 2).

SR 3.9.4.2

This Surveillance demonstrates that each containment purge and exhaust valve actuates to its isolation position on manual initiation. The 18 month Frequency maintains consistency with other similar valve testing requirements. This Surveillance performed during MODE 6 will ensure that the valves are capable of being closed after a postulated fuel handling accident involving handling recently irradiated fuel to limit a release of fission product radioactivity from the containment. The SR is modified by a Note stating that this Surveillance is not required to be met for valves in isolated penetrations. The LCO provides the option to close penetrations in lieu of requiring manual initiation capability.

REFERENCES

1. UFSAR, Section 15.4.7.

2. ~~NUREG-0800, Rev. 2, July 1981.~~ Regulatory Guide 1.183, July 2000

Inserts for Bases 3.9.4

1. from escaping to the environment. The closure restrictions are sufficient to restrict fission product radioactivity release from the containment due to a fuel handling accident involving handling of recently irradiated fuel.

2. containment closure capability is only required during

3. the previous 100 hours). A fuel handling accident involving fuel with a minimum decay time of 100 hours prior to movement

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. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to ~~well below 10 CFR 100~~ the limits of Regulatory Guide 1.183.

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment, as postulated by Regulatory Guide ~~1.25~~ 1.183 (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)~~ Insert 1 to be used in the accident analysis for iodine. This relates to the assumption that ~~99%~~ 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 Insert 2 ~~10% of the total fuel rod iodine inventory (Ref. 1)~~.

The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained within allowable limits (Ref. ~~3~~ 1).

Refueling cavity water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits.

BASES

APPLICABILITY

LCO 3.9.7 is applicable when moving irradiated fuel assemblies within containment. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel assemblies are not present in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.16, "Fuel Storage Pool Water Level."

ACTIONS

A.1

With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving movement of irradiated fuel assemblies within the containment shall be suspended immediately to ensure that a fuel handling accident cannot occur.

The suspension of fuel movement shall not preclude completion of movement of a component to a safe position.

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. Regulatory Guide ~~1.25, March 23, 1972~~. 1.183, July 2000.

2. UFSAR, Section 15.4.7.

~~3. 10 CFR 100.10.~~

Inserts for Bases 3.9.7

1) A minimum water level of 23 ft allows an <u>effective</u> iodine decontamination factor of <u>200</u> (Appendix B Assumption 2 of Ref. 1)

2) 8% of the fuel rod I-131 inventory and 5% of all other iodine isotopes, which are included as other halogens

Attachment 3
(Letter Serial No. 04-494A)

Proposed Technical Specification Changes
Implementation of Alternate Source Term

Proposed Technical Specifications

Virginia Electric and Power Company
(Dominion)
North Anna Power Station Units 1 and 2

1.1 Definitions

CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. The COT may be performed by means of any series of sequential, overlapping, or total channel steps.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites" or in NRC Regulatory Guide 1.109, Revision 1, October 1977.

3.7 PLANT SYSTEMS

3.7.10 Main Control Room/Emergency Switchgear Room (MCR/ESGR) Emergency Ventilation System (EVS)—MODES 1, 2, 3, and 4

LCO 3.7.10 Two MCR/ESGR EVS trains shall be OPERABLE.

--NOTE--
The MCR/ESGR boundary may be opened intermittently under administrative control.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required MCR/ESGR EVS train inoperable.	A.1 Restore MCR/ESGR EVS train to OPERABLE status.	7 days
B. Two required MCR/ESGR EVS trains inoperable due to inoperable MCR/ESGR boundary.	B.1 Restore MCR/ESGR boundary to OPERABLE status.	24 hours
C. Required Action and associated Completion Time of 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
D. Two required MCR/ESGR EVS trains inoperable for reasons other than Condition B.	D.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Operate each required MCR/ESGR EVS train for ≥ 10 continuous hours with the heaters operating.	31 days
SR 3.7.10.2	Perform required MCR/ESGR EVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with VFTP
SR 3.7.10.3	Not Used	
SR 3.7.10.4	Verify each required MCR/ESGR EVS train can maintain a positive pressure of ≥ 0.04 inches water gauge, relative to the adjacent areas, during the pressurization mode of operation at a makeup flow rate of ≥ 900 cfm and ≤ 1100 cfm.	18 months on a STAGGERED TEST BASIS

3.7 PLANT SYSTEMS

3.7.13 Main Control Room/Emergency Switchgear Room (MCR/ESGR) Bottled Air System

LCO 3.7.13 Three MCR/ESGR bottled air system trains shall be OPERABLE.

------------------------------NOTE------------------------------
The MCR/ESGR boundary may be opened intermittently under administrative control.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required MCR/ESGR bottled air system train inoperable.	A.1 Restore MCR/ESGR bottled air system train to OPERABLE status.	7 days
B. Two or more required MCR/ESGR bottled air system trains inoperable due to inoperable MCR/ESGR boundary in MODE 1, 2, 3, or 4.	B.1 Restore MCR/ESGR boundary to OPERABLE status.	24 hours
C. Two or more required MCR/ESGR bottled air system trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B.	C.1 Restore at least two MCR/ESGR bottled air system trains to OPERABLE status.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.4	Verify two required MCR/ESGR bottled air system trains can maintain a positive pressure of ≥ 0.05 inches water gauge, relative to the adjacent areas for at least 60 minutes.	18 months on a STAGGERED TEST BASIS

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

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

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

b. One door in each air lock is closed; 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 purge and exhaust isolation valve.

NOTE

Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

APPLICABILITY: During movement of recently 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 recently irradiated fuel assemblies within containment.	Immediately

B 2.1 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure during operating conditions, the continued integrity of the RCS is ensured. According to GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, in accordance with GDC 28, "Reactivity Limits" (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2500 psia. During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 50.67 (Ref. 4). |

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the reactor high pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.

(continued)

BASES

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized.

SAFETY LIMIT VIOLATIONS

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 50.67 limits (Ref. 4).

The allowable Completion Time of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized.

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

REFERENCES

1. UFSAR, Sections 3.1.10, 3.1.11, and 3.1.24.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.
3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWX-5000.
4. 10 CFR 50.67.
5. UFSAR, Section 7.2.
6. USAS B31.1, Standard Code for Pressure Piping, American Society of Mechanical Engineers, 1967.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The ejection of a control rod rapidly adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The ejection of a rod also produces a time dependent redistribution of core power.

SDM satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). Even though it is not directly observed from the control room, SDM is considered an initial condition process variable because it is periodically monitored to ensure that the unit is operating within the bounds of accident analysis assumptions.

LCO

SDM is a core design condition that can be ensured during operation through control rod positioning (control and shutdown banks) and through the soluble boron concentration.

The MSLB (Ref. 2) accident is the most limiting analysis that establishes the SDM value of the LCO. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed Regulatory Guide 1.183 limits (Ref. 3).

APPLICABILITY

In MODE 2 with $k_{eff} < 1.0$ and in MODES 3, 4, and 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration." In MODES 1 and 2 with $k_{eff} > 1.0$, SDM is ensured by complying with LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits."

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 (continued)

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and the low probability of an accident occurring without the required SDM. This allows time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.

REFERENCES

1. UFSAR, Section 3.1.22.

2. UFSAR, Chapter 15.

3. Regulatory Guide 1.183, July 2000.

BASES

BACKGROUND (continued)

The Allowable Value specified in Table 3.3.1-1 serves as the LSSS such that a channel is OPERABLE if the trip setpoint is found not to exceed the Allowable Value during the CHANNEL OPERATIONAL TEST (COT). As such, the Allowable Value differs from the Trip Setpoint by an amount primarily equal to the expected instrument loop uncertainties, such as drift, during the surveillance interval. In this manner, the actual setting of the device will still meet the LSSS definition and ensure that a Safety Limit is not exceeded at any given point of time as long as the device has not drifted beyond that expected during the surveillance interval. If the actual setting of the device is found to have exceeded the Allowable Value the device would be considered inoperable for a technical specification perspective. This requires corrective action including those actions required by 10 CFR 50.36 when automatic protective devices do not function as required. Note that, although the channel is "OPERABLE" under these circumstances, the trip setpoint should be left adjusted to a value within the established trip setpoint calibration tolerance band, in accordance with uncertainty assumptions stated in the referenced set point methodology (as-left criteria), and confirmed to be operating within the statistical allowances of the uncertainty terms assigned.

During AOOs, which are those events expected to occur one or more times during the unit life, the acceptable limits are:

1. The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling (DNB);

2. Fuel centerline melt shall not occur; and

3. The RCS pressure SL of 2750 psia shall not be exceeded.

Operation within the SLs of Specification 2.0, "Safety Limits (SLs)," also maintains the above values and assures that offsite dose will be within the 10 CFR 50 criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the unit life. The acceptable limit during accidents is that offsite dose shall be maintained within an acceptable fraction of 10 CFR 50.67 limits. Different accident categories are allowed a different fraction of these limits, based on probability of

(continued)

BASES

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes a 1 gpm primary to secondary LEAKAGE as the initial condition.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The UFSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is only briefly released via safety valves and the majority is steamed to the condenser if offsite power is available. The 1 gpm primary to secondary LEAKAGE is relatively inconsequential in this case. If offsite power is not available, releases continue through the unaffected steam generators until the Residual Heat Removal System is placed in service. In this case, the 1 gpm primary to secondary LEAKAGE is more significant.

The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes primary to secondary LEAKAGE as an initial condition. The dose consequences resulting from the SLB accident are within the limits defined in the staff approved licensing basis.

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

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND

The maximum dose that an individual at the site boundary can receive for 2 hours during an accident is specified in 10 CFR 50.67 (Ref. 1). The limits on specific activity ensure that the doses are held to the limits specified in Regulatory Guide 1.183 (Ref. 2) during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The allowable levels are intended to limit the 2 hour dose at the site boundary to the dose guideline limits specified in Regulatory Guide 1.183. The limits in the LCO are standardized, based on parametric evaluations of offsite radioactivity dose consequences for typical site locations.

The parametric evaluations showed the potential offsite dose levels for a SGTR accident were less than the dose guideline limits specified in Regulatory Guide 1.183. Each evaluation assumes a broad range of site applicable atmospheric dispersion factors in a parametric evaluation.

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensures that the resulting 2 hour doses at the site boundary will not exceed the dose guideline limits specified in Regulatory Guide 1.183 following a SGTR accident. The SGTR safety analysis (Ref. 3) assumes the specific activity of the reactor coolant at the LCO limit and an existing reactor coolant steam generator (SG) tube leakage rate of 1 gpm. The safety analysis assumes the specific activity of the secondary coolant at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.7, "Secondary Specific Activity."

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The analysis for the SGTR accident establishes the acceptance limits for RCS specific activity. Reference to this analysis is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The analysis is for two cases of reactor coolant specific activity. One case assumes specific activity at 1.0 μCi/gm DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the I-131 release rate to the reactor coolant by a factor of 335 immediately after the accident. The second case assumes the initial reactor coolant iodine activity at 60.0 μCi/gm DOSE EQUIVALENT I-131 due to a pre-accident iodine spike caused by an RCS transient. In both cases, the noble gas activity in the reactor coolant is determined by normalizing the 1% failed fuel inventory from the UFSAR to the amount of failed fuel associated with 1.0 μCi/gm DOSE EQUIVALENT I-131.

The radiologically limiting SGTR analysis also assumes a loss of offsite power at the same time as the SGTR event. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature ΔT signal.

The coincident loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG power operated relief valves and the main steam safety valves. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends.

The safety analysis shows the radiological consequences of an SGTR accident are within the Reference 2 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed the limits shown in Figure 3.4.16-1, in the applicable specification, for more than 48 hours.

The remainder of the above LCO limit permissible iodine levels shown in Figure 3.4.16-1 are acceptable because of the low probability of a SGTR accident occurring during the established 48 hour time limit. The occurrence of an SGTR accident at these permissible levels could increase the site boundary dose levels, but still be within 10 CFR 50.67 dose guideline limits.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices. RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The specific iodine activity is limited to 1.0 µCi/gm DOSE EQUIVALENT I-131, and the gross specific activity in the reactor coolant is limited to the number of µCi/gm equal to 100 divided by E (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on DOSE EQUIVALENT I-131 ensures the 2 hour dose to an individual at the site boundary during the Design Basis Accident (DBA) will be within the limits specified in Regulatory Guide 1.183 (Ref. 2). The SGTR accident analysis (Ref. 3) shows that the 2 hour site boundary dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SGTR, lead to site boundary doses that exceed the Regulatory Guide 1.183 dose guideline limits.
APPLICABILITY	In MODES 1 and 2, and in MODE 3 with RCS average temperature ≥ 500°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and gross specific activity are necessary to contain the potential consequences of an SGTR to within the acceptable site boundary dose values. For operation in MODE 3 with RCS average temperature < 500°F, and in MODES 4 and 5, the release of radioactivity in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves and SG power operated relief valves.
ACTIONS	<u>A.1 and A.2</u> With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the limits of Figure 3.4.16-1 are not exceeded. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is done to continue to provide a trend.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

The DOSE EQUIVALENT I-131 must be restored to within limits within 48 hours. The Completion Time of 48 hours is required, if the limit violation resulted from normal iodine spiking.

B.1

With the gross specific activity in excess of the allowed limit, the unit must be placed in a MODE in which the requirement does not apply.

The change within 6 hours to MODE 3 and RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves and prevents venting the SG to the environment in an SGTR event. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging unit systems.

C.1

If a Required Action and the associated Completion Time of Condition A is not met or if the DOSE EQUIVALENT I-131 is in the unacceptable region of Figure 3.4.16-1, the reactor must be brought to MODE 3 with RCS average temperature < 500°F within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

SR 3.4.16.1 requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant at least once every 7 days. While basically a quantitative measure of radionuclides with half lives longer than 15 minutes, excluding iodines, this measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in gross specific activity.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1 (continued)

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with T_{avg} at least 500°F. The 7 day Frequency considers the unlikelihood of a gross fuel failure during the time.

SR 3.4.16.2

This Surveillance is performed in MODE 1 only to ensure iodine remains within limit during normal operation and following fast power changes when fuel failure is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level, considering gross activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failure; samples at other times would provide inaccurate results.

SR 3.4.16.3

A radiochemical analysis for E determination is required every 184 days (6 months) with the unit operating in MODE 1 equilibrium conditions. The E determination directly relates to the LCO and is required to verify unit operation within the specified gross activity LCO limit. The analysis for E is a measurement of the average energies per disintegration for isotopes with half lives longer than 15 minutes, excluding iodines. The Frequency of 184 days recognizes E does not change rapidly.

This SR has been modified by a Note that indicates sampling is required to be performed within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours. This ensures that the radioactive materials are at equilibrium so the analysis for E is representative and not skewed by a crud burst or other similar abnormal event.

REFERENCES

1. 10 CFR 50.67.

2. Regulatory Guide 1.183, July 2000.

3. UFSAR, Section 15.4.3.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The maximum design internal pressure for the containment is 45.0 psig. The LOCA analyses establish the limits for the containment air partial pressure operating range. The initial conditions used in the containment design basis LOCA analyses were an air partial pressure of 11.7 psia and an air temperature of 120°F. This resulted in a maximum peak containment internal pressure of 44.1 psig, which is less than the maximum design internal pressure for the containment.

The containment was also designed for an external pressure load of 9.2 psid (i.e., a design minimum pressure of 5.5 psia). The inadvertent actuation of the QS System was analyzed to determine the reduction in containment pressure (Ref. 1). The initial conditions used in the analysis were 8.43 psia and 120°F. This resulted in a minimum pressure inside containment of 7.07 psia, which is considerably above the design minimum of 5.5 psia.

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

The radiological consequences analysis demonstrates acceptable results provided the containment pressure decreases to 0.5 psig in 1 hour and does not exceed 0.5 psig for the interval from 1 to 4 hours following the Design Basis Accident (Ref. 3). Beyond 4 hours the containment pressure is assumed to be less than 0.0 psig, terminating leakage from containment.

Containment pressure satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

Maintaining containment pressure within the limits shown in Figure 3.6.4-1 of the LCO ensures that in the event of a DBA the resultant peak containment accident pressure will be maintained below the containment design pressure. These limits also prevent the containment pressure from exceeding

(continued)

BASES

LCO (continued)	the containment design negative pressure differential with respect to the outside atmosphere in the event of inadvertent actuation of the QS System. The LCO limits also ensure the return to subatmospheric conditions within 60 minutes following a DBA.
APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within design basis limits is essential to ensure initial conditions assumed in the accident analyses are maintained, the LCO is applicable in MODES 1, 2, 3, and 4. In MODES 5 and 6, the probability and consequences of these events are reduced due to the Reactor Coolant System pressure and temperature limitations of these MODES. Therefore, maintaining containment pressure within the limits of the LCO is not required in MODE 5 or 6.
ACTIONS	<u>A.1</u> When containment air partial pressure is not within the limits of the LCO, containment pressure must be restored to within these limits within 1 hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour. <u>B.1 and B.2</u> If containment air partial pressure cannot be restored to within limits within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1

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

REFERENCES

1. UFSAR, Section 6.2.
2. 10 CFR 50, Appendix K.
3. UFSAR, Section 15.4.1.7.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Therefore, it is concluded that the calculated transient containment atmosphere temperatures are acceptable for the SLB.

The modeled QS System actuation from the containment analysis is based upon a response time associated with exceeding the containment High-High pressure signal setpoint to achieving full flow through the spray nozzles. A delayed response time initiation provides conservative analyses of peak calculated containment temperature and pressure responses. The QS System total response time of 71.1 seconds comprises the signal delay, diesel generator startup time, and system startup time, including pipe fill time.

For certain aspects of accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 3).

Inadvertent actuation of the QS System is evaluated in the analysis, and the resultant reduction in containment pressure is calculated. The maximum calculated reduction in containment pressure results in containment pressures within the design containment minimum pressure.

The radiological consequences analysis demonstrates acceptable results provided the containment pressure decreases to 0.5 psig in 1 hour and does not exceed 0.5 psig for the interval from 1 to 4 hours following the Design Basis Accident (Ref. 4). Beyond 4 hours the containment pressure is assumed to be less than 0.0 psig, terminating leakage from containment.

The QS System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

During a DBA, one train of the QS System is required to provide the heat removal capability assumed in the safety analyses for containment. In addition, one QS System train, with spray pH adjusted by the contents of the chemical addition tank, is required to scavenge iodine fission

(continued)

BASES

LCO (continued)

products from the containment atmosphere and ensure their retention in the containment sump water. To ensure that these requirements are met, two QS System trains must be OPERABLE with power from two safety related, independent power supplies. Therefore, in the event of an accident, at least one train of QS will operate, assuming that the worst case single active failure occurs.

Each QS train includes a spray pump, a dedicated spray header, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST.

APPLICABILITY

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

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

ACTIONS

A.1

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

B.1 and B.2

If the Required Action and associated Completion Time are not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

Verifying the correct alignment of manual, power operated, and automatic valves, excluding check valves, in the QS System provides assurance that the proper flow path exists for QS System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they were verified to be in the correct position prior to being secured. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment and capable of potentially being mispositioned are in the correct position.

SR 3.6.6.2

Verifying that each QS pump's developed head at the flow test point is greater than or equal to the required developed head ensures that QS pump performance is consistent with the safety analysis assumptions. Flow and differential head are normal tests of centrifugal pump performance required by the ASME Code (Ref. 5). Since the QS System pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.6.3 and SR 3.6.6.4

These SRs ensure that each QS automatic valve actuates to its correct position and each QS pump starts upon receipt of an actual or simulated Containment Pressure high-high signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillances when performed at an 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.5

With the quench spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections or an inspection of the nozzles can be performed. This SR ensures that each spray nozzle is unobstructed and that spray coverage of the containment during an accident is not degraded. Due to the passive nature of the design of the nozzle and the non-corrosive design of the system, a test performed following maintenance which could result in nozzle blockage is considered adequate to detect obstruction of the nozzles.

REFERENCES

1. UFSAR, Section 6.2.
2. 10 CFR 50.49.
3. 10 CFR 50, Appendix K.
4. UFSAR, Section 15.4.1.7.
5. ASME Code for Operation and Maintenance of Nuclear Power Plants.

BASES

APPLICABLE SAFETY ANALYSES (continued)

the containment atmosphere temperature exceeds the containment design temperature is short enough that there would be no adverse effect on equipment inside containment. Therefore, it is concluded that the calculated transient containment atmosphere temperatures are acceptable for the SLB and LOCA.

The RS System actuation model from the containment analysis is based upon a response time associated with exceeding the High-High containment pressure signal setpoint to achieving full flow through the RS System spray nozzles. A delay in response time initiation provides conservative analyses of peak calculated containment temperature and pressure. The RS System's total response time is determined by the delay timers and system startup time.

For certain aspects of accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 3).

The radiological consequences analysis demonstrates acceptable results provided the containment pressure decreases to 0.5 psig in 1 hour and does not exceed 0.5 psig for the interval from 1 to 4 hours following the Design Basis Accident (Ref. 4). Beyond 4 hours the containment pressure is assumed to be less than 0.0 psig, terminating leakage from containment.

The RS System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

During a DBA, one train (one inside and one outside RS subsystem in the same train) or two outside RS subsystems of the RS System are required to provide the minimum heat removal capability assumed in the safety analysis. To ensure that this requirement is met, four RS subsystems and the casing cooling tank must be OPERABLE. This will ensure that at least one train will operate assuming the worst case single failure occurs, which is no offsite power and the loss of one emergency diesel generator. Inoperability of the

(continued)

BASES

LCO (continued)

casing cooling tank, the casing cooling pumps, the casing cooling valves, piping, instrumentation, or controls, or of the QS System requires an assessment of the effect on RS subsystem OPERABILITY.

Each RS train consists of one RS subsystem outside containment and one RS subsystem inside containment. Each RS subsystem includes one spray pump, one spray cooler, one 180° coverage spray header, nozzles, valves, piping, instrumentation, and controls to ensure an OPERABLE flow path capable of taking suction from the containment sump.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause an increase in containment pressure and temperature requiring the operation of the RS System.

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

ACTIONS

A.1

With one of the RS subsystems inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. The components in this degraded condition are capable of providing at least 100% of the heat removal needs (i.e., approximately 150% when one RS subsystem is inoperable) after an accident. The 7 day Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the RS and QS systems and the low probability of a DBA occurring during this period.

B.1 and C.1

With two of the required RS subsystems inoperable either in the same train, or both inside RS subsystems, at least one of the inoperable RS subsystems must be restored to OPERABLE status within 72 hours. The components in this degraded condition are capable of providing 100% of the heat removal needs and 360° containment spray coverage after an accident. The 72 hour Completion Time was developed taking into account the redundant heat removal capability afforded by the OPERABLE subsystems, a reasonable amount of time for repairs, and the low probability of a DBA occurring during this period.

BASES

ACTIONS (continued)

D.1

With the casing cooling tank inoperable, the NPSH available to both outside RS subsystem pumps may not be sufficient. The inoperable casing cooling tank must be restored to OPERABLE status within 72 hours. The components in this degraded condition are capable of providing 100% of the heat removal needs after an accident. The casing cooling tank does not affect the OPERABILITY of the inside RS subsystem pumps. The effect on NPSH of the outside RS pumps must be assessed as part of outside RS pump OPERABILITY. The 72 hour Completion Time was chosen based on the same reasons as given in Required Action B.1.

E.1 and E.2

If the inoperable RS subsystem(s) or the casing cooling tank cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 84 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. The extended interval to reach MODE 5 allows additional time and is reasonable considering that the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

F.1

With an inoperable inside RS subsystem in one train, and an inoperable outside RS subsystem in the other train, only 180° containment spray coverage is available. This condition is outside accident analysis. With three or more RS subsystems inoperable, the unit is in a condition outside the accident analysis. With two inoperable outside RS subsystems, less than 100% of required RS flow is available. Therefore, in all three cases, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1

Verifying that the casing cooling tank solution temperature is within the specified tolerances provides assurance that the water injected into the suction of the outside RS pumps will increase the NPSH available as per design. The 24 hour

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1 (continued)

Frequency of this SR was developed considering operating experience related to the parameter variations and instrument drift during the applicable MODES. Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal condition.

SR 3.6.7.2

Verifying the casing cooling tank contained borated water volume provides assurance that sufficient water is available to support the outside RS subsystem pumps during the time they are required to operate. The 7 day Frequency of this SR was developed considering operating experience related to the parameter variations and instrument drift during the applicable MODES. Furthermore, the 7 day Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal condition.

SR 3.6.7.3

Verifying the boron concentration of the solution in the casing cooling tank provides assurance that borated water added from the casing cooling tank to RS subsystems will not dilute the solution being recirculated in the containment sump. A Note states that for Unit 2, until the first entry into MODE 4 following the Unit 2 Fall 2002 refueling outage, the casing cooling tank boron concentration acceptance criteria shall be ≥ 2300 ppm and ≤ 2400 ppm. The 7 day Frequency of this SR was developed considering the known stability of stored borated water and the low probability of any source of diluting pure water.

SR 3.6.7.4

Verifying the correct alignment of manual, power operated, and automatic valves, excluding check valves, in the RS System and casing cooling tank provides assurance that the proper flow path exists for operation of the RS System. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified as being in the correct position prior to being secured. This SR does not require any testing or valve manipulation. Rather,

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.7.4 (continued)

it involves verification, through a system walkdown, that those valves outside containment and capable of potentially being mispositioned are in the correct position.

SR 3.6.7.5

Verifying that each RS and casing cooling pump's developed head at the flow test point is greater than or equal to the required developed head ensures that these pumps' performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by the ASME Code (Ref. 5). Since the RS System pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.7.6

These SRs ensure that each automatic valve actuates and that the RS System and casing cooling pumps start upon receipt of an actual or simulated High-High containment pressure signal. Start delay times are also verified for the RS System pumps. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was considered to be acceptable from a reliability standpoint.

SR 3.6.7.7

This SR ensures that each spray nozzle is unobstructed and that spray coverage of the containment will meet its design bases objective. Either an inspection of the nozzles or an air or smoke test is performed through each spray header. Due

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.7.7 (continued)

to the passive design of the spray header and its normally dry state, a test performed following maintenance which could result in nozzle blockage is considered adequate for detecting obstruction of the nozzles.

REFERENCES

1. UFSAR, Section 6.2.
2. 10 CFR 50.49.
3. 10 CFR 50, Appendix K.
4. UFSAR, Section 15.4.1.7.
5. ASME Code for Operation and Maintenance of Nuclear Power Plants.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

d. Following a steam generator tube rupture, closure of the MSTVs isolates the ruptured steam generator from the intact steam generators to minimize radiological releases.

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

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

LCO

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

This LCO provides assurance that the MSTVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 50.67 (Ref. 4) limits or the NRC staff approved licensing basis.

APPLICABILITY

The MSTVs must be OPERABLE in MODE 1, and in MODES 2 and 3 except when closed and de-activated, when there is significant mass and energy in the RCS and steam generators. When the MSTVs are closed, they are already performing the safety function.

In MODE 4, the steam generator energy is low and the MSTVs are not required to support the safety analyses due to the low probability of a design basis accident.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.2.2

This SR verifies that each MSTV closes on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. The Frequency of MSTV testing is every 18 months. The 18 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 10.3.
2. UFSAR, Section 6.2.
3. UFSAR, Section 15.4.2.
4. 10 CFR 50.67.
5. ASME Code for Operation and Maintenance of Nuclear Power Plants.

B 3.7 PLANT SYSTEMS

B 3.7.7 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicates current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

This limit is lower than the activity value that might be expected from a 1 gpm tube leak (LCO 3.4.13, "RCS Operational LEAKAGE") of primary coolant at the limit of 1.0 μCi/gm (LCO 3.4.16, "RCS Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives, (i.e., < 20 hours).

If the main steam safety valves (MSSVs) open for 2 hours following a trip from full power with the specified activity limit, the resultant 2 hour dose to a person at the exclusion area boundary (EAB) would be less than 0.033 rem TEDE (the consequences of the design basis main steam line break accident).

Operating a unit at the allowable limits could result in a 2 hour EAB exposure at the Regulatory Guide 1.183 (Ref. 1) limits, or the limits established as the NRC staff approved licensing basis.

BASES

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB), as discussed in the UFSAR, Chapter 15 (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 μCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed the limits specified in Regulatory Guide 1.183 (Ref. 1).

With the loss of offsite power, the remaining steam generators are available for core decay heat dissipation by venting steam to the atmosphere through the MSSVs and steam generator power operated relief valves (SG PORVs). The Auxiliary Feedwater System supplies the necessary makeup to the steam generators. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Residual Heat Removal System to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generator is assumed to discharge steam and any entrained activity through the MSSVs and SG PORV during the event. Since no credit is taken in the analysis for activity plateout or retention, the resultant radiological consequences represent a conservative estimate of the potential integrated dose due to the postulated steam line failure.

Secondary specific activity limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

As indicated in the Applicable Safety Analyses, the specific activity of the secondary coolant is required to be ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131 to limit the radiological consequences of a Design Basis Accident (DBA) to the required limit (Ref. 1).

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

In MODES 5 and 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS and contributes to increased post accident doses. If the secondary specific activity cannot be restored to within limits within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

This SR verifies that the secondary specific activity is within the limits of the accident analysis. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.

REFERENCES

1. Regulatory Guide 1.183, July 2000.

2. UFSAR, Chapter 15.

B 3.7 PLANT SYSTEMS

B 3.7.10 Main Control Room/Emergency Switchgear Room (MCR/ESGR) Emergency Ventilation System (EVS)—MODES 1, 2, 3, and 4

BASES

BACKGROUND

The MCR/ESGR Emergency Habitability System (EHS) provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity. The MCR/ESGR EHS consists of the MCR/ESGR bottled air system (LCO 3.7.13) and the MCR/ESGR EVS (LCO 3.7.10 and LCO 3.7.14).

The MCR/ESGR EVS was designed as four redundant trains that can filter and recirculate air inside the MCR/ESGR envelope, or supply filtered air to the MCR/ESGR envelope. The two independent and redundant unit MCR/ESGR EVS trains on the accident unit can actuate automatically in recirculation. Either of these trains can also be aligned to provide filtered outside air for pressurization when appropriate. One train from the other unit can be manually actuated to provide filtered outside air approximately 60 minutes after the event. Each train consists of a heater, demister filter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves, dampers, and instrumentation also form part of the system. One EVS train is capable of performing the safety function of supplying outside filtered air for pressurization. Due to the location of the air intake for 1-HV-F-41, it can not be used to satisfy the requirements of LCO 3.7.10. Two of the three remaining trains (1-HV-F-42, 2-HV-F-41, and 2-HV-F-42) are required for independence and redundancy.

Upon receipt of the actuating signal(s), normal air supply to and exhaust from the MCR/ESGR envelope is isolated, two trains of MCR/ESGR EVS actuate to recirculate air, and airflow from the bottled air banks maintains a positive pressure in the MCR/ESGR envelope. The MCR/ESGR envelope consists of the MCR, ESGRs, computer rooms, logic rooms, instrument rack rooms, air conditioning rooms, battery rooms, the MCR toilet, and the stairwell behind the MCR. Approximately 60 minutes after actuation of the MCR/ESGR bottled air system, a single MCR/ESGR EVS train is manually actuated to provide filtered outside air to the MCR/ESGR envelope through HEPA filters and charcoal adsorbers for

(continued)

BASES

BACKGROUND (continued)

pressurization. The demisters remove any entrained water droplets present, to prevent excessive moisture loading of the HEPA filters and charcoal adsorbers. Continuous operation of each train for at least 10 hours per month, with the heaters on, reduces moisture buildup on the HEPA filters and adsorbers. Both the demister and heater are important to the effectiveness of the HEPA filters and charcoal adsorbers.

Pressurization of the MCR/ESGR envelope prevents infiltration of unfiltered air from the areas adjacent to the envelope.

A single train of the MCR/ESGR EVS will pressurize the MCR/ESGR envelope to ≥ 0.04 inches water gauge. The MCR/ESGR EHS operation in maintaining the MCR/ESGR envelope habitable is discussed in the UFSAR, Section 6.4 (Ref. 1).

Redundant MCR/ESGR EVS supply and recirculation trains provide the required pressurization and filtration should an excessive pressure drop develop across the other filter train. Normally closed isolation dampers are arranged in series pairs so that the failure of one damper to open will not result in an inability of the system to perform the function based on the presence of the redundant train. The MCR/ESGR EHS is designed in accordance with Seismic Category I requirements. The actuation signal will only start the MCR/ESGR EVS trains for the affected unit. Requiring two of the three MCR/ESGR EVS trains provides redundancy, assuring that at least one train is available to be realigned to provide filtered outside air for pressurization.

The MCR/ESGR EHS is designed to maintain the control room environment for 30 days of continuous occupancy after a DBA without exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 3) for alternative source terms.

APPLICABLE SAFETY ANALYSES

The MCR/ESGR EVS components are arranged in redundant, safety related ventilation trains. The location of most components and ducting within the MCR/ESGR envelope ensures an adequate supply of filtered air to all areas requiring access. The MCR/ESGR EHS provides airborne radiological protection for the control room operators, as demonstrated

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

by the control room accident dose analyses for the most limiting design basis accident fission product release presented in the UFSAR, Chapter 15 (Ref. 2). This accident analysis assumes that at least one train is aligned for control room pressurization approximately 60 minutes after actuation of bottled air, but does not take any credit for filtration of recirculated air. Since, the MCR/ESGR EVS train associated with 1-HV-F-41 can not be used to provide outside air for filtered pressurization (due to the location of its air intake with respect to Vent Stack B) it can not be used to satisfy the requirements of LCO 3.7.10.

The worst case single active failure of a component of the MCR/ESGR EVS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

The MCR/ESGR EVS—MODES 1, 2, 3, and 4 satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two independent and redundant MCR/ESGR EVS trains are required to be OPERABLE to ensure that at least one train is available to be manually aligned to pressurize and to provide filtered air to the MCR/ESGR envelope, assuming a single failure disables one of the two required OPERABLE trains. Total system failure could result in exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 3) for alternative source terms, in the event of a large radioactive release.

The MCR/ESGR EVS—MODES 1, 2, 3, and 4 is considered OPERABLE when the individual components necessary to limit operator exposure are OPERABLE in the two required trains of the MCR/ESGR EVS—MODES 1, 2, 3, and 4. 1-HV-F-41 can not be used to satisfy the requirements of LCO 3.7.10.

An MCR/ESGR EVS train is OPERABLE when the associated:

a. Fan is OPERABLE;

b. Demister filters, HEPA filters and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions; and

c. Heater, ductwork, valves, and dampers are OPERABLE, and air flow can be maintained.

BASES

LCO (continued)	The MCR/ESGR EVS is shared by Unit 1 and Unit 2. In addition, the MCR/ESGR boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors. The LCO is modified by a Note allowing the MCR/ESGR boundary to be opened intermittently under administrative controls. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for MCR/ESGR isolation is indicated.
APPLICABILITY	In MODES 1, 2, 3, and 4, MCR/ESGR EVS must be OPERABLE to control operator exposure during and following a DBA.
ACTIONS	<u>A.1</u> When one required MCR/ESGR EVS train is inoperable, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining required OPERABLE MCR/ESGR EVS train is adequate to perform the MCR/ESGR envelope protection function. However, the overall reliability is reduced because a single failure in the required OPERABLE EVS trains could result in loss of MCR/ESGR EVS function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining trains to provide the required capability. <u>B.1</u> If the MCR/ESGR boundary is inoperable in MODE 1, 2, 3, or 4 (e.g., excessive control room inleakage or excessive Emergency Core Cooling System leakage), the MCR/ESGR EVS cannot perform its intended function. Actions must be taken to restore an OPERABLE MCR/ESGR boundary within 24 hours. During the period that the MCR/ESGR boundary is inoperable, appropriate compensatory measures (consistent with the intent of GDC 19) should be utilized to protect control room operators from potential hazards such as radioactive contamination. Preplanned measures should be available to address these concerns for intentional and unintentional

(continued)

BASES

ACTIONS

B.1 (continued)

entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan, and possibly repair, and test most problems with the MCR/ESGR boundary.

C.1 and C.2

In MODE 1, 2, 3, or 4, if the inoperable required MCR/ESGR EVS train or the inoperable MCR/ESGR boundary cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE that minimizes accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

D.1

When two required MCR/ESGR EVS trains are inoperable in MODE 1, 2, 3, or 4 for reasons other than an inoperable MCR/ESGR boundary (i.e., Condition B), the MCR/ESGR EVS may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on the MCR/ESGR EVS are not too severe, testing each required train once every month provides an adequate check of this system. Monthly heater operations dry out any moisture accumulated in the charcoal and HEPA filters from humidity in the ambient air. Each required train must be operated for ≥ 10 continuous hours with the heaters energized. The 31 day Frequency is based on the reliability of the equipment and the one train redundancy availability.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.10.2

This SR verifies that the required MCR/ESGR EVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing the performance of the demister filter, HEPA filter, charcoal adsorber efficiency, minimum and maximum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.10.3

Not Used

SR 3.7.10.4

This SR verifies, by pressurizing the MCR/ESGR envelope, the integrity of the MCR/ESGR envelope, and the assumed inleakage rates of the potentially contaminated air. The MCR/ESGR envelope positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper functioning of the MCR/ESGR EVS. During the emergency mode of operation, the MCR/ESGR EVS is designed to pressurize the MCR/ESGR envelope ≥ 0.04 inches water gauge positive pressure with respect to adjacent areas in order to prevent unfiltered inleakage. The MCR/ESGR EVS is designed to maintain this positive pressure with one train at a makeup flow rate of ≥ 900 cfm and ≤ 1100 cfm. The Frequency of 18 months on a STAGGERED TEST BASIS is consistent with the guidance provided in NUREG-0800 (Ref. 4).

REFERENCES

1. UFSAR, Section 6.4.
2. UFSAR, Chapter 15.
3. 10 CFR 50, Appendix A.
4. NUREG-0800, Rev. 2, July 1981.

BASES

APPLICABLE SAFETY ANALYSES (continued)	The MCR/ESGR ACS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	Two independent and redundant subsystems of the MCR/ESGR ACS, providing cooling to the unit ESGR and associated portion of the MCR, are required to be OPERABLE to ensure that at least one is available, assuming a single failure disabling the other subsystem. Total system failure could result in the equipment operating temperature exceeding limits in the event of an accident. The MCR/ESGR ACS is considered to be OPERABLE when the individual components necessary to cool the MCR/ESGR envelope air are OPERABLE in both required subsystems. Each subsystem consists of two air handling units (one for the MCR and one for the ESGR), one chiller, valves, piping, instrumentation and controls. The two subsystems provide air temperature cooling to the portion of the MCR/ESGR envelope associated with the unit. In addition, an OPERABLE MCR/ESGR ACS must be capable of maintaining air circulation. An MCR/ESGR ACS subsystem does not have to be in operation to be considered OPERABLE. The MCR/ESGR ACS is considered OPERABLE when it is capable of being started by manual actions within 10 minutes. The time of 10 minutes is based on the time required to start the system manually following required testing.
APPLICABILITY	In MODES 1, 2, 3, and 4, and during movement of recently irradiated fuel assemblies, the MCR/ESGR ACS must be OPERABLE to ensure that the MCR/ESGR envelope temperature will not exceed equipment operational requirements following isolation of the MCR/ESGR envelope. The MCR/ESGR ACS is only required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 300 hours), due to radioactive decay.
ACTIONS	<u>A.1</u> With one or more required MCR/ESGR ACS subsystem inoperable, and at least 100% of the MCR/ESGR ACS cooling equivalent to a single OPERABLE MCR/ESGR ACS subsystem available, action must be taken to restore OPERABLE status within 30 days. In

(continued)

BASES

ACTIONS

A.1 (continued)

this Condition, the remaining OPERABLE MCR/ESGR ACS subsystem is adequate to maintain the MCR/ESGR envelope temperature within limits. However, the overall reliability is reduced because a single failure in the OPERABLE MCR/ESGR ACS subsystem could result in loss of MCR/ESGR ACS function. The 30 day Completion Time is based on the low probability of an event requiring MCR/ESGR envelope isolation, the consideration that the remaining subsystem can provide the required protection, and that alternate safety or nonsafety related cooling means are available.

The LCO requires the OPERABILITY of a number of independent components. Due to the redundancy of subsystems and the diversity of components, the inoperability of one active component in a subsystem does not render the MCR/ESGR ACS incapable of performing its function. Neither does the inoperability of two different components, each in a different subsystem, necessarily result in a loss of function for the MCR/ESGR ACS (e.g., an inoperable chiller in one subsystem, and an inoperable air handler in the other). This allows increased flexibility in unit operations under circumstances when components in opposite subsystems are inoperable.

B.1 and B.2

In MODE 1, 2, 3, or 4, if the inoperable MCR/ESGR ACS subsystem cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE that minimizes the risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1 and C.2

During movement of recently irradiated fuel, if the required inoperable MCR/ESGR ACS subsystems cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE MCR/ESGR ACS subsystem must be placed in operation immediately. This action ensures that the remaining subsystem is OPERABLE and that active failures will be readily detected.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the MCR/ESGR envelope. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

D.1

During movement of recently irradiated fuel assemblies, with less than 100% of the MCR/ESGR ACS cooling equivalent to a single OPERABLE MCR/ESGR ACS subsystem available, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the MCR/ESGR envelope. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

E.1

With less than 100% of the MCR/ESGR ACS cooling equivalent to a single OPERABLE MCR/ESGR ACS subsystem available in MODE 1, 2, 3, or 4, the MCR/ESGR ACS may not be capable of performing its intended function. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1

This SR verifies that the heat removal capability of any one of the three chillers for the unit is sufficient to remove the heat load assumed in the safety analyses in the MCR/ESGR envelope. This SR consists of a combination of testing and calculations. The 18 month on a STAGGERED TEST BASIS Frequency is appropriate since significant degradation of the MCR/ESGR ACS is slow and is not expected over this time period.

REFERENCES

1. UFSAR, Section 9.4.

BASES

BACKGROUND (continued)

Building General area exhaust, fuel building exhaust, decontamination building exhaust, and containment purge exhaust.

One Safeguards Area exhaust fan is normally operating and dampers are aligned to bypass the HEPA filters and charcoal adsorbers. During emergency operations, the ECCS PREACS dampers are realigned to begin filtration. Upon receipt of the actuating Engineered Safety Feature Actuation System signal(s), normal air discharges from the Safeguards Area room are diverted through the filter banks. Two Auxiliary Building Central Exhaust fans are normally operating. Air discharges from the Auxiliary Building Central Exhaust area are manually diverted through the filter banks. Required Safeguards Area and Auxiliary Building Central Exhaust area fans are manually actuated if they are not already operating. The prefilters remove any large particles in the air to prevent excessive loading of the HEPA filters and charcoal adsorbers.

The ECCS PREACS is discussed in the UFSAR, Section 9.4 (Ref. 1) and it may be used for normal, as well as post accident, atmospheric cleanup functions. The primary purpose of the heaters is to maintain the relative humidity at an acceptable level during normal operations, generally consistent with iodine removal efficiencies per Regulatory Guide 1.52 (Ref. 3). The heaters are not required for post-accident conditions.

APPLICABLE SAFETY ANALYSES

The design basis of the ECCS PREACS is established by the large break LOCA. The system evaluation assumes ECCS leakage outside containment, such as safety injection pump leakage, during the recirculation mode. In such a case, the system limits radioactive release to within the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 4) for alternative source terms. The analysis of the effects and consequences of a large break LOCA is presented in Reference 2. The ECCS PREACS also may actuate following a small break LOCA, in those cases where the ECCS goes into the recirculation mode of long term cooling, to clean up releases of smaller leaks, such as from valve stem packing. The analyses assume the filtration by the ECCS PREACS does not begin for 60 minutes following an accident.

The ECCS PREACS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

Two redundant trains of the ECCS PREACS are required to be OPERABLE to ensure that at least one is available. Total system failure could result in exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 4) for alternative source terms.

ECCS PREACS is considered OPERABLE when the individual components necessary to maintain the ECCS pump room filtration are OPERABLE in both trains.

An ECCS PREACS train is considered OPERABLE when its associated:

a. Safeguards Area exhaust fan is OPERABLE;

b. One Auxiliary Building HEPA filter and charcoal adsorber assembly (shared with the other unit) is OPERABLE;

c. One Auxiliary Building Central exhaust system fan (shared with other unit) is OPERABLE;

d. Controls for the Auxiliary Building Central exhaust system filter and bypass dampers (shared with the other unit) are OPERABLE;

e. HEPA filter and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions; and

f. Ductwork, valves, and dampers are OPERABLE.

The Auxiliary Building Central Exhaust subsystem may be removed from service (e.g., tag out fans, open ductwork, etc.), in order to perform required testing and maintenance. The Auxiliary Building Central Exhaust subsystem is OPERABLE in this condition if it can be restored to service and perform its function within 60 minutes following an accident.

In addition, the required Safeguards Area and charging pump cubicle boundaries for charging pumps not isolated from the Reactor Coolant System must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors, except for those openings which are left open by design, including charging pump ladder wells.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.13 Main Control Room/Emergency Switchgear Room (MCR/ESGR) Bottled Air System

BASES

BACKGROUND

The MCR/ESGR Emergency Habitability System (EHS) provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity. The MCR/ESGR EHS consists of the MCR/ESGR bottled air system (LCO 3.7.13) and the MCR/ESGR Emergency Ventilation System (EVS) (LCO 3.7.10 and LCO 3.7.14).

The MCR/ESGR bottled air system consists of four trains of bottled air lined up to provide air to the MCR/ESGR envelope when the system actuates. The air is provided via four trains which feed a common header, supplying air to the Unit 1 and Unit 2 ESGRs. The header is also capable of being aligned to supply air directly to the MCR. Each train is provided air by one of the bottled air banks. Unit 1 and Unit 2 each provide two trains of bottled air. Two bottled air trains are capable of providing dry air of breathing quality to maintain a positive interior pressure in the MCR/ESGR envelope for Unit 1 and Unit 2 for a period of one hour following a Design Basis Accident (DBA).

In MODES 1, 2, 3, or 4, upon receipt of the actuating signal(s), normal air supply to and exhaust from the MCR/ESGR envelope is isolated, two trains of MCR/ESGR EVS actuate to recirculate air, and airflow from the bottled air banks maintains a positive pressure in the MCR/ESGR envelope. In case of a fuel handling accident (FHA) in the fuel building, automatic actuation of bottled air is possible. A FHA in containment can not cause an automatic actuation of bottled air, but manual actuation can be initiated. After 300 hours of decay, actuation of bottled air is not required. The MCR/ESGR envelope consists of the MCR, ESGRs, computer rooms, logic rooms, instrument rack rooms, air conditioning rooms, battery rooms, the MCR toilet, and the stairwell behind the MCR. Approximately 60 minutes after actuation of the MCR/ESGR bottled air system, a single MCR/ESGR EVS train is manually actuated to provide filtered outside air to the MCR/ESGR envelope through high efficiency particulate air (HEPA) filters and charcoal adsorbers for pressurization.

(continued)

BASES

BACKGROUND (continued)	Pressurization of the MCR/ESGR envelope prevents infiltration of unfiltered air from the areas adjacent to the envelope. Two trains of the MCR/ESGR bottled air system will pressurize the MCR/ESGR envelope to ≥ 0.05 inches water gauge. The MCR/ESGR EHS operation in maintaining the MCR/ESGR envelope habitable is discussed in the UFSAR, Section 6.4 (Ref. 1). The MCR/ESGR EHS is designed in accordance with Seismic Category I requirements. The MCR/ESGR EHS is designed to maintain the MCR/ESGR envelope environment for 30 days of continuous occupancy after a DBA without exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 2) for alternative source terms.
APPLICABLE SAFETY ANALYSES	The MCR/ESGR bottled air system is arranged in redundant, safety related trains providing pressurized air from the required bottled air banks to maintain a habitable environment in the MCR/ESGR envelope. The MCR/ESGR EHS provides airborne radiological protection for the control room operators, as demonstrated by the control room accident dose analyses for the most limiting design basis accident fission product release presented in the UFSAR, Chapter 15 (Ref. 4). The worst case single active failure of a component of the MCR/ESGR bottled air system, assuming a loss of offsite power, does not impair the ability of the system to perform its design function. The MCR/ESGR bottled air system satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	Three independent and redundant MCR/ESGR bottled air system trains are required to be OPERABLE to ensure that at least two are available assuming a single failure disables one train. Total system failure could result in exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 2) for alternative source terms, in the event of a large radioactive release.

(continued)

BASES

LCO (continued)

The MCR/ESGR bottled air system is considered OPERABLE when the individual components necessary to limit operator exposure are OPERABLE in the three required trains of the MCR/ESGR bottled air system.

A MCR/ESGR bottled air system train is OPERABLE when:

a. One OPERABLE bottled air bank of 69 bottles is in service;

b. A flow path, including associated valves and piping, is OPERABLE; and

c. The common exhaust header is OPERABLE.

The MCR/ESGR bottled air system trains are shared by Unit 1 and Unit 2.

In addition, the MCR/ESGR boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.

The LCO is modified by a Note allowing the MCR/ESGR boundary to be opened intermittently under administrative controls. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for MCR/ESGR isolation is indicated.

APPLICABILITY

In MODES 1, 2, 3, and 4, and during movement of recently irradiated fuel assemblies, MCR/ESGR bottled air system must be OPERABLE to control operator exposure during and following a DBA.

During movement of recently irradiated fuel assemblies, the MCR/ESGR bottled air system must be OPERABLE to respond to the release from a fuel handling accident involving handling recently irradiated fuel. The MCR/ESGR bottled air system is only required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 300 hours), due to radioactive decay.

BASES

ACTIONS

A.1

When one required MCR/ESGR bottled air system train is inoperable, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining required OPERABLE MCR/ESGR bottled air system trains are adequate to perform the MCR/ESGR envelope protection function. However, the overall reliability is reduced because a single failure in one of the remaining required OPERABLE trains could result in loss of MCR/ESGR bottled air system function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining trains to provide the required capability.

B.1

If the MCR/ESGR boundary is inoperable in MODE 1, 2, 3, or 4 (e.g., excessive control room inleakage or excessive Emergency Core Cooling System leakage), the MCR/ESGR bottled air system cannot perform its intended function. Actions must be taken to restore an OPERABLE MCR/ESGR boundary within 24 hours. During the period that the MCR/ESGR boundary is inoperable, appropriate compensatory measures (consistent with the intent of GDC 19) should be utilized to protect control room operators from potential hazards such as radioactive contamination. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan, and possibly repair, and test most problems with the MCR/ESGR boundary.

C.1

When two or more required trains of the MCR/ESGR bottled air system are inoperable in MODE 1, 2, 3, or 4 for reasons other than an inoperable MCR/ESGR boundary (i.e., Condition B), action must be taken to restore at least two of the required MCR/ESGR bottled air system trains to OPERABLE status within 24 hours. During the period that two or more required trains of the MCR/ESGR bottled air system are inoperable, appropriate compensatory measures (consistent with the intent of GDC 19) should be utilized to protect control room operators from potential hazards such as radioactive contamination. Preplanned measures should be available to

(continued)

BASES

ACTIONS

C.1 (continued)

address these concerns for intentional and unintentional entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan, restore, and possibly repair, and test most problems with the MCR/ESGR bottled air system, such as repressurizing the system after an inadvertent actuation.

D.1 and D.2

In MODE 1, 2, 3, or 4, if the inoperable required MCR/ESGR bottled air system trains or the inoperable MCR/ESGR boundary cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE that minimizes accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

E.1 and E.2

During movement of recently irradiated fuel assemblies, if the required inoperable MCR/ESGR bottled air system train cannot be restored to OPERABLE status within the required Completion Time or two or more required MCR/ESGR bottled air system trains are inoperable, action must be taken to immediately suspend activities that could result in a release of radioactivity that might require isolation of the MCR/ESGR envelope. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

This SR verifies that each required MCR/ESGR bottled air bank is at the proper pressure. This ensures that when combined with the required number of OPERABLE air bottles, the minimum required air flow will be maintained to ensure the required MCR/ESGR envelope pressurization for

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1 (continued)

approximately 60 minutes when the MCR/ESGR bottled air system is actuated. The 31 day Frequency is based on engineering judgement.

SR 3.7.13.2

This SR verifies that the proper number of MCR/ESGR air bottles are in service, with one bank of 69 air bottles in each required train. This SR requires verification that each bottled air bank manual valve not locked, sealed, or otherwise secured and required to be open during accident conditions is open. This SR helps to ensure that the bottled air banks required to be OPERABLE to pressurize the MCR/ESGR boundary are in service. The 31 day Frequency is based on engineering judgment and was chosen to provide added assurance of the correct positions. This SR does not apply to valves that are locked, sealed, or otherwise secured in the open position, since these were verified to be in the correct position prior to locking, sealing, or securing.

SR 3.7.13.3

This SR verifies that each required MCR/ESGR bottled air system train actuates by verifying the flow path is opened and that the normal air supply to and exhaust from the MCR/ESGR envelope is isolated on an actual or simulated actuation signal. The Frequency of 18 months is consistent with performing this test on a refueling interval basis.

SR 3.7.13.4

This SR verifies, by pressurizing the MCR/ESGR envelope, the integrity of the MCR/ESGR envelope, and the assumed inleakage rates of the potentially contaminated air. The MCR/ESGR envelope positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper functioning of the MCR/ESGR bottled air system. During the emergency mode of operation, the MCR/ESGR bottled air system is designed to pressurize the MCR/ESGR envelope to ≥ 0.05 inches water gauge positive pressure with respect to adjacent areas in order to prevent unfiltered inleakage. The MCR/ESGR bottled air system is designed to maintain this positive pressure with two trains for at least 60 minutes. Testing two trains at a time at the

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.7.13.4 (continued) Frequency of 18 months on a STAGGERED TEST BASIS is consistent with the guidance provided in NUREG-0800 (Ref. 3).
REFERENCES	1. UFSAR, Section 6.4. 2. 10 CFR 50, Appendix A. 3. NUREG-0800, Rev. 2, July 1981. 4. UFSAR, Chapter 15.

B 3.7 PLANT SYSTEMS

B 3.7.14 Main Control Room/Emergency Switchgear Room (MCR/ESGR) Emergency Ventilation System (EVS)–During Movement of Recently Irradiated Fuel Assemblies

BASES

BACKGROUND

The MCR/ESGR Emergency Habitability System (EHS) provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity. The MCR/ESGR EHS consists of the MCR/ESGR bottled air system (LCO 3.7.13) and the MCR/ESGR EVS (LCO 3.7.10 and LCO 3.7.14).

The MCR/ESGR EVS was designed as four independent, redundant trains that can filter and recirculate air inside the MCR/ESGR envelope, or supply filtered air to the MCR/ESGR envelope. Each train consists of a heater, demister filter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves and dampers, and instrumentation also form part of the system. One EVS train is capable of performing the safety function of supplying filtered air for pressurization. Due to the location of the air intake for 1-HV-F-41, it can not be used to satisfy the requirements of LCO 3.7.14. Two of the three remaining trains (1-HV-F-42, 2-HV-F-41, and 2-HV-F-42) are required for independence and redundancy.

In case of a Design Basis Accident (DBA) during movement of recently irradiated fuel assemblies, an automatic (signal from the fuel building radiation monitors) or manual actuation of airflow from the bottled air banks is required. Actuation of airflow from the bottled air banks also automatically isolates the MCR/ESGR envelope to maintain positive pressure in the envelope and automatically starts all available EVS trains in recirculation mode.

The MCR/ESGR envelope consists of the MCR, ESGRs, computer rooms, logic rooms, instrument rack rooms, air conditioning rooms, battery rooms, the MCR toilet, and the stairwell behind the MCR. Approximately 60 minutes after actuation of the MCR/ESGR bottled air system, a single MCR/ESGR EVS train is manually actuated or aligned to provide filtered outside air to the MCR/ESGR envelope through HEPA filters and charcoal adsorbers for pressurization. Due to the location

(continued)

BASES

BACKGROUND (continued)

of the air intake for 1-HV-F-41, it should not be used in pressurization mode during a design basis fuel handling accident. There is no restriction on the use of 1-HV-F-41 in the recirculation mode. The demisters remove any entrained water droplets present in the air, to prevent excessive moisture loading of the HEPA filters and charcoal adsorbers. Continuous operation of each train for at least 10 hours per month, with the heaters on, reduces moisture buildup on the HEPA filters and adsorbers. Both the demister and heater are important to the effectiveness of the HEPA filters and charcoal adsorbers.

Pressurization of the MCR/ESGR envelope prevents infiltration of unfiltered air from the areas adjacent to the envelope.

A single train of the MCR/ESGR EVS will pressurize the MCR/ESGR envelope to ≥ 0.04 inches water gauge. The MCR/ESGR EHS operation in maintaining the MCR/ESGR envelope habitable is discussed in the UFSAR, Section 6.4 (Ref. 1).

Redundant MCR/ESGR EVS supply trains provide the required pressurization and filtration should an excessive pressure drop develop across the other filter train. Normally closed isolation dampers are arranged in series pairs so that the failure of one damper to open will not result in an inability of the system to perform the function based on the presence of the redundant train. The MCR/ESGR EHS is designed in accordance with Seismic Category I requirements.

The MCR/ESGR EHS is designed to maintain the control room environment for 30 days of continuous occupancy after a DBA without exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 2) for alternative source terms.

APPLICABLE SAFETY ANALYSES

The MCR/ESGR EVS components are arranged in redundant, safety related ventilation trains. The location of most components and ducting within the MCR/ESGR envelope ensures an adequate supply of filtered air to all areas requiring access. The MCR/ESGR EHS provides airborne radiological protection for the control room operators, as demonstrated by the control room accident dose analyses for the most limiting design basis accident fission product release presented in the UFSAR, Chapter 15 (Ref. 4).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The worst case single active failure of a component of the MCR/ESGR EVS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function. The MCR/ESGR EVS–During Movement of Recently Irradiated Fuel Assemblies satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two independent and redundant MCR/ESGR EVS trains are required to be OPERABLE to ensure that at least one is available assuming a single failure disables the other train. Total system failure could result in exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 2), for alternative source terms in the event of a large radioactive release.

The MCR/ESGR EVS–During Movement of Recently Irradiated Fuel Assemblies is considered OPERABLE when the individual components necessary to limit operator exposure are OPERABLE in the two required trains of the MCR/ESGR EVS–During Movement of Recently Irradiated Fuel Assemblies.

An MCR/ESGR EVS train is OPERABLE when the associated:

a. Fan is OPERABLE;

b. Demister filters, HEPA filters and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions; and

c. Heater, ductwork, valves, and dampers are OPERABLE, and air flow can be maintained.

The MCR/ESGR EVS is shared by Unit 1 and Unit 2.

In addition, the MCR/ESGR boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.

The LCO is modified by a Note allowing the MCR/ESGR boundary to be opened intermittently under administrative controls. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in

(continued)

BASES

LCO (continued)

continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for MCR/ESGR isolation is indicated.

APPLICABILITY

During movement of recently irradiated fuel assemblies, MCR/ESGR EVS–During Movement of Recently Irradiated Fuel Assemblies must be OPERABLE to control operator exposure during and following a DBA.

During movement of recently irradiated fuel assemblies, the MCR/ESGR EVS must be OPERABLE to respond to the release from a fuel handling accident involving handling recently irradiated fuel. The MCR/ESGR EVS is only required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 300 hours), due to radioactive decay.

Regarding the MCR/ESGR EVS, it should be noted that they are required to be OPERABLE by other LCOs in other MODES.

ACTIONS

A.1

When one required MCR/ESGR EVS train is inoperable, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining required OPERABLE MCR/ESGR EVS train is adequate to perform the MCR/ESGR envelope protection function. However, the overall reliability is reduced because a single failure in the required OPERABLE MCR/ESGR EVS train could result in loss of MCR/ESGR EVS function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining trains to provide the required capability.

B.1

During movement of recently irradiated fuel assemblies, if the required inoperable MCR/ESGR EVS train cannot be restored to OPERABLE status within the required Completion Time or two required MCR/ESGR EVS trains are inoperable, action must be taken to immediately suspend activities that could result in a release of radioactivity that might require isolation of the MCR/ESGR envelope. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on the MCR/ESGR EVS are not too severe, testing each required train once every month provides an adequate check of this system. Monthly heater operations dry out any moisture accumulated in the charcoal and HEPA filters from humidity in the ambient air. Each required train must be operated for ≥ 10 continuous hours with the heaters energized. The 31 day Frequency is based on the reliability of the equipment and the one train redundancy availability.

SR 3.7.14.2

This SR verifies that the required MCR/ESGR EVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing the performance of the demister filter, HEPA filter, charcoal adsorber efficiency, minimum and maximum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.14.3

This SR verifies, by pressurizing the MCR/ESGR envelope, the integrity of the MCR/ESGR envelope, and the assumed inleakage rates of the potentially contaminated air. The MCR/ESGR envelope positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper functioning of the MCR/ESGR EVS. During the emergency mode of operation, the MCR/ESGR EVS is designed to pressurize the MCR/ESGR envelope ≥ 0.04 inches water gauge positive pressure with respect to adjacent areas in order to prevent unfiltered inleakage. The MCR/ESGR EVS is designed to maintain this positive pressure with one train at a makeup flow rate of ≥ 900 cfm and ≤ 1100 cfm. The Frequency of 18 months on a STAGGERED TEST BASIS is consistent with the guidance provided in NUREG-0800 (Ref. 3).

REFERENCES

1. UFSAR, Section 6.4.

2. 10 CFR 50, Appendix A.

BASES

REFERENCES (continued)	3. NUREG-0800, Rev. 2, July 1981. 4. UFSAR, Chapter 15.

B 3.7 PLANT SYSTEMS

B 3.7.15 Fuel Building Ventilation System (FBVS)

BASES

BACKGROUND

The FBVS discharges airborne radioactive particulates from the area of the fuel pool following a fuel handling accident. The FBVS, in conjunction with other normally operating systems, also provides environmental control of temperature and humidity in the fuel pool area.

The FBVS consists of ductwork, valves and dampers, instrumentation, and two fans.

The FBVS, which may also be operated during normal plant operations, discharges air from the fuel building.

The FBVS is discussed in the UFSAR, Sections 9.4.5 and 15.4.5 (Refs. 1 and 2, respectively) because it may be used for normal, as well as post accident functions.

APPLICABLE SAFETY ANALYSES

The FBVS design basis is established by the consequences of the limiting Design Basis Accident (DBA), which is a fuel handling accident involving handling recently irradiated fuel. The analysis of the fuel handling accident, given in Reference 2, assumes that all fuel rods in an assembly are damaged. The DBA analysis of the fuel handling accident assumes that the FBVS is functional with at least one fan operating. The amount of fission products available for release from the fuel building is determined for a fuel handling accident. Due to radioactive decay, FBVS is only required to be OPERABLE during fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours). These assumptions and the analysis follow the guidance provided in Regulatory Guide 1.183 (Ref. 3).

The fuel handling accident analysis for the fuel building assumes all of the radioactive material available for release is discharged from the fuel building by the FBVS.

The FBVS satisfies Criterion 3 of the 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The FBVS is required to be OPERABLE and in operation. Total system failure could result in the atmospheric release from the fuel building exceeding the 10 CFR 50, Appendix A, GDC-19 (Ref. 4) limits for alternative source terms, in the event of a fuel handling accident involving handling recently irradiated fuel.

The FBVS is considered OPERABLE when the individual components are OPERABLE. The FBVS is considered OPERABLE when at least one fan is OPERABLE and in operation, the associated FBVS ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained. In addition, an OPERABLE FBVS must maintain a pressure in the fuel building pressure envelope £ -0.125 inches water gauge with respect to atmospheric pressure.

The LCO is modified by a Note allowing the fuel building boundary to be opened intermittently under administrative controls. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for fuel building isolation is indicated.

APPLICABILITY

During movement of recently irradiated fuel in the fuel handling area, the FBVS is required to be OPERABLE to alleviate the consequences of a fuel handling accident.

ACTIONS

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

BASES

ACTIONS (continued)	<u>A.1</u> When the FBVS is inoperable or not in operation during movement of recently irradiated fuel assemblies in the fuel building, 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 recently irradiated fuel assemblies in the fuel building. This does not preclude the movement of fuel to a safe position.
SURVEILLANCE REQUIREMENTS	<u>SR 3.7.15.1</u> This SR verifies the integrity of the fuel building pressure envelope. The ability of the fuel building to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the FBVS. The FBVS is designed to maintain a slight negative pressure in the fuel building, to prevent unfiltered LEAKAGE. The FBVS is designed to maintain a ≤ -0.125 inches water gauge with respect to atmospheric pressure. The Frequency of 18 months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 5).
REFERENCES	1. UFSAR, Section 9.4.5. 2. UFSAR, Section 15.4.5. 3. Regulatory Guide 1.183, July 2000. 4. 10 CFR 50, Appendix A, GDC-19. 5. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.

B 3.7 PLANT SYSTEMS

B 3.7.16 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 UFSAR, Section 9.1.2 (Ref. 1). A description of the Spent Fuel Pool Cooling and Cleanup System is given in the UFSAR, Section 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in the UFSAR, 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. 4). The resultant 2 hour dose per person at the exclusion area boundary is within the Regulatory Guide 1.183 limits.

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

The fuel storage pool water level satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

BASES

SURVEILLANCE REQUIREMENTS	<u>SR 3.7.16.1</u> (continued) 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. UFSAR, Section 9.1.2. 2. UFSAR, Section 9.1.3. 3. UFSAR, Section 15.4.5. 4. Regulatory Guide 1.183, July 2000.

BASES

LCO (continued)	It is acceptable for trains to be cross tied during shutdown conditions, allowing a single offsite power circuit to supply all required trains.
APPLICABILITY	The AC sources required to be OPERABLE in MODES 5 and 6 and during movement of recently irradiated fuel assemblies provide assurance that: a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core; b. Systems needed to mitigate a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 300 hours.) are available; c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition. The AC power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.1.
ACTIONS	<u>A.1</u> An offsite circuit would be considered inoperable if it were not available to the necessary portions of the electrical power distribution subsystem(s). One train with offsite power available may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS and recently irradiated fuel movement. By the allowance of the option to declare required features inoperable, with no offsite power available, appropriate restrictions will be implemented in accordance with the affected required features LCO's ACTIONS.

BASES

APPLICABLE SAFETY ANALYSES (continued)	The DC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	The DC electrical power subsystem(s), each subsystem consisting of two batteries, one battery charger per battery, and the corresponding control equipment and interconnecting cabling within the train, are required to be OPERABLE to support required trains of the distribution systems required OPERABLE by LCO 3.8.10, "Distribution Systems-Shutdown." The EDG DC system, consisting of a battery, battery charger, and the corresponding control equipment and interconnection cabling for the EDG, are required to be OPERABLE to support the EDG required by LCO 3.8.2, "AC Sources-Shutdown." This ensures the availability of sufficient DC electrical power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents involving handling recently irradiated fuel).

APPLICABILITY

The DC electrical power sources and EDG DC system required to be OPERABLE in MODES 5 and 6, and during movement of recently irradiated fuel assemblies, provide assurance that:

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

b. Required features needed to mitigate a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 300 hours.) are available;

c. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

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

BASES

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

The train with DC power available may be capable of supporting sufficient systems to allow continuation of CORE ALTERATIONS and recently irradiated fuel movement. By allowing the option to declare required features inoperable with the associated DC power source(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCO ACTIONS. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of recently irradiated fuel assemblies, and operations involving positive reactivity additions) that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

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

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required DC electrical power subsystems should be completed as quickly as possible in order to minimize the time during which the unit safety systems may be without sufficient power.

BASES

APPLICABLE SAFETY ANALYSES (continued)	The inverters were previously identified as part of the distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	The required inverter(s) ensure the availability of electrical power for the instrumentation for systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. The battery powered inverters provide uninterruptible supply of AC electrical power to the AC vital buses even if the 4.16 kV safety buses are de-energized. OPERABILITY of the inverters requires that the AC vital bus be powered by the inverter. This ensures the availability of sufficient inverter power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents involving handling recently irradiated fuel). Supported system(s) that do not provide automatic function(s) may be connected to a vital bus that is powered by a constant voltage transformer (example: Low Temperature Overpressure Protection, when not in automatic).
APPLICABILITY	The inverters required to be OPERABLE in MODES 5 and 6 and during movement of recently irradiated fuel assemblies provide assurance that: a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core; b. Systems needed to mitigate a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical core within the previous 300 hours.) are available; c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition. Inverter requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.7.

BASES

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

The required OPERABLE Inverters are capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, recently irradiated fuel movement, and operations with a potential for positive reactivity additions. By the allowance of the option to declare required features inoperable with the associated inverter(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCOs' Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of recently irradiated fuel assemblies, and operations involving positive reactivity additions) that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

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

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required inverters should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power or powered from a constant voltage source transformer.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and AC vital buses energized from the inverter. The verification of proper voltage output ensures that the required power is readily available for the instrumentation connected to the AC vital buses. The 7 day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions.

REFERENCES

1. UFSAR, Chapter 6.
2. UFSAR, Chapter 15.

BASES

APPLICABLE SAFETY ANALYSES (continued)	The AC and DC electrical power distribution systems satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	Various combinations of subsystems, equipment, and components are required OPERABLE by other LCOs, depending on the specific unit condition. Implicit in those requirements is the required OPERABILITY of necessary support required features. This LCO explicitly requires energization of the portions of the electrical distribution system necessary to support OPERABILITY of required systems, equipment, and components—all specifically addressed in each LCO and implicitly required via the definition of OPERABILITY. Maintaining these portions of the distribution system energized ensures the availability of sufficient power to operate the unit in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents involving handling recently irradiated fuel).
APPLICABILITY	The AC and DC electrical power distribution subsystems required to be OPERABLE in MODES 5 and 6, and during movement of recently irradiated fuel assemblies, provide assurance that: a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core; b. Systems needed to mitigate a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical core within the previous 300 hours.) are available; c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition and refueling condition. The AC, DC, and AC vital bus electrical power distribution subsystems requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.9.

BASES

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, A.2.4, and A.2.5

Although redundant required features may require redundant trains of electrical power distribution subsystems to be OPERABLE, one OPERABLE distribution subsystem train may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS and recently irradiated fuel movement. By allowing the option to declare required features associated with an inoperable distribution subsystem inoperable, appropriate restrictions are implemented in accordance with the affected distribution subsystem LCO's Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of recently irradiated fuel assemblies, and operations involving positive reactivity additions) that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

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

Notwithstanding performance of the above conservative Required Actions, a required residual heat removal (RHR) subsystem may be inoperable. In this case, Required Actions A.2.1 through A.2.4 do not adequately address the concerns relating to coolant circulation and heat removal. Pursuant to LCO 3.0.6, the RHR ACTIONS would not be entered.

(continued)

BASES

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, A.2.4, and A.2.5 (continued)

Therefore, Required Action A.2.5 is provided to direct declaring RHR inoperable, which results in taking the appropriate RHR actions.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required distribution subsystems should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power.

SURVEILLANCE REQUIREMENTS

SR 3.8.10.1

This Surveillance verifies that the required AC, DC, and AC vital bus electrical power distribution subsystems are functioning properly, with all the buses energized. The verification of proper voltage availability on the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. Verification of proper voltage availability for 480 volt buses and load centers may be performed by indirect methods. The 7 day Frequency takes into account the capability of the electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

REFERENCES

1. UFSAR, Chapter 6.

2. UFSAR, Chapter 15.

B 3.9 REFUELING OPERATIONS

B 3.9.4 Containment Penetrations

BASES

BACKGROUND

During movement of recently 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 within the requirements of Regulatory Guide 1.183 (Ref. 2). 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 recently 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." One of the containment air locks is an integral part of the containment equipment hatch. During refueling the air lock

(continued)

BASES

BACKGROUND (continued)

that is part of the containment equipment hatch is typically replaced by a temporary hatch plate. While the temporary hatch plate is installed, there is only one air lock by which to enter containment. The LCO only applies to containment air locks that are installed. 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 recently irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain closed.

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted from escaping to the environment. The closure restrictions are sufficient to restrict fission product radioactivity release from the containment due to a fuel handling accident involving handling of recently irradiated fuel.

The Containment Purge and Exhaust System includes a 36 inch purge penetration and a 36 inch exhaust penetration. During MODES 1, 2, 3, and 4, the two valves in each of the purge and exhaust flow paths are secured in the closed position. The Containment Purge and Exhaust System is not subject to a Specification in MODE 5.

In MODE 6, large air exchanges are necessary to conduct refueling operations. The 36 inch purge system is used for this purpose.

The 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 recently irradiated fuel movements.

BASES

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident involving handling recently irradiated fuel. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 1). Fuel handling accidents, analyzed in Reference 2, involve dropping a single irradiated fuel assembly and handling tool. The requirements of LCO 3.9.7, "Refueling Cavity Water Level," in conjunction with a minimum decay time of 100 hours prior to movement of irradiated fuel (i.e., fuel that has not been recently irradiated) without containment closure capability ensures that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are within the guideline values specified in Regulatory Guide 1.183 (Ref. 2).

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

LCO

This LCO limits the consequences of a fuel handling accident involving handling recently irradiated fuel in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge and exhaust penetrations. For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that these penetrations are isolable by a containment purge and exhaust isolation valve.

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 recently irradiated fuel assemblies within containment, and 2) specified individuals are designated and readily available to isolate the flow path in the event of a fuel handling accident.

APPLICABILITY

The containment penetration requirements are applicable during movement of recently irradiated fuel assemblies within containment because this is when there is a potential for the limiting fuel handling accident. In MODES 1, 2, 3,

(continued)

BASES

APPLICABILITY (continued)

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 design basis fuel handling accident does not exist. Additionally, due to radioactive decay, containment closure capability is only required during a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours). A fuel handling accident involving fuel with a minimum decay time of 100 hours prior to movement will result in doses that are within the guideline values specified in Regulatory Guide 1.183 (Ref. 2) even without containment closure capability. 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 Purge and Exhaust Isolation System not capable of manual 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 recently 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 manually closed.

The Surveillance is performed every 7 days during movement of recently 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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1 (continued)

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 involving handling recently irradiated fuel 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 Regulatory Guide 1.183 (Ref. 2).

SR 3.9.4.2

This Surveillance demonstrates that each containment purge and exhaust valve actuates to its isolation position on manual initiation. The 18 month Frequency maintains consistency with other similar valve testing requirements. This Surveillance performed during MODE 6 will ensure that the valves are capable of being closed after a postulated fuel handling accident involving handling recently irradiated fuel to limit a release of fission product radioactivity from the containment. The SR is modified by a Note stating that this Surveillance is not required to be met for valves in isolated penetrations. The LCO provides the option to close penetrations in lieu of requiring manual initiation capability.

REFERENCES

1. UFSAR, Section 15.4.7.

2. Regulatory Guide 1.183, July 2000.

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. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to the limits of Regulatory Guide 1.183.
APPLICABLE SAFETY ANALYSES	During movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment, as postulated by Regulatory Guide 1.183 (Ref. 1). A minimum water level of 23 ft allows an effective iodine decontamination factor of 200 (Appendix B Assumption 2 of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99.5% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 8% of the fuel rod I-131 inventory and 5% of all other iodine isotopes, which are included as other halogens (Ref. 1). The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained within allowable limits (Ref. 1). Refueling cavity water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits.

BASES

APPLICABILITY	LCO 3.9.7 is applicable when moving irradiated fuel assemblies within containment. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel assemblies are not present in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.16, "Fuel Storage Pool Water Level."
ACTIONS	A.1 With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving movement of irradiated fuel assemblies within the containment shall be suspended immediately to ensure that a fuel handling accident cannot occur. The suspension of fuel movement shall not preclude completion of movement of a component to a safe position.
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. Regulatory Guide 1.183, July 2000. 2. UFSAR, Section 15.4.7.