

**Duke Energy  
Enclosure 2  
Levy Nuclear Plant Units 1 and 2  
  
Request for Exemption Regarding  
Main Control Room Dose  
  
(9 pages including cover page)**

## 1.0 Summary Description

General Design Criteria 19 requires that operator dose be limited to less than 5 rem following a design basis accident (DBA). The AP1000 main control room (MCR) and operator habitability requirements are met by the non-safety, nuclear island nonradioactive ventilation system (VBS) and the main control room emergency habitability system (VES).

When a source of AC power is available, the VBS provides HVAC service to the MCR. If radioactivity is detected in the MCR supply air duct and the VBS is operable, VBS enters supplemental filtration mode (SFM), during which two supplemental air filtration units automatically start to pressurize the MCR using filtered makeup air. If AC power is unavailable for more than 10 minutes or if a "High-2" MCR radiation signal is received, the MCR is isolated from the VBS and the VES is actuated. Operator habitability requirements are then met by the VES air storage tanks, which are sized to deliver the required air flow to the MCR for 72 hours.

As included in DCD Subsection 6.4.3.2, the major functions of the VES are:

- To provide forced ventilation to deliver an adequate supply of breathable air for the main control room envelope (MCRE) occupants.
- To provide forced ventilation to maintain the MCRE at a 1/8 inch water gauge positive pressure with respect to the surrounding areas.
- Provide passive filtration to filter contaminated air in the MCRE.
- Limit the temperature increase of the MCRE equipment and facilities that must remain functional during an accident, via the heat absorption by passive heat sinks.

The Chapter 15 post-accident dose analyses assume that the VBS is initially in operation, but fails to enter the SFM on a high radioactivity indication in the MCR atmosphere. VES operation is then assumed to be initiated once the High-2 level for control room activity is reached.

Doses are also calculated assuming that the VBS does operate in the SFM as designed, but with no switchover to VES operation. This VBS operating case demonstrates the defense-in-depth that is provided by the system and also shows that, in the event of an accident with realistic assumptions, the VBS is adequate to protect the MCR operators without relying on VES operation. Doses are determined for the following design basis accidents:

1. Large Break LOCA
2. Fuel Handling Accident
3. Steam Generator Tube Rupture
4. Main Steam Line Break
5. Rod Ejection Accident
6. Locked Rotor Accident
7. Small Line Break Outside Containment

For all events, operator doses shall be within the 5 rem acceptance limit of General Design Criteria (GDC) 19. As part of design finalization, however, a number of issues were identified that challenge the ability of the certified design to satisfy GDC-19.

Changes requiring an exemption are a reduction in the allowable secondary coolant iodine activity to meet GDC-19 requirements for the Main Steam Line break accident and revision of the design description of the Nuclear Island Nonradioactive Ventilation System to reflect the correct name of the actuation signal (High-2) for isolating the main control room penetrations.

## **2.0 Description of Licensing Basis Impacts**

### **Tier 1 Changes**

The design description of the Nuclear Island Nonradioactive Ventilation System in Tier 1 Subsection 2.7.1 is revised to reflect the correct name of the actuation signal (High-2) for isolating the main control room penetrations. This signal isolates the HVAC penetration using the VBS and an additional penetration using the Sanitary Drainage System (SDS). DCD Tier 1 information is impacted by changing the name of the VES actuation signal (from High-High to High-2) for isolating the main control room penetrations in Subsection 2.7.1. The Tier 1 actuation name change is required to ensure consistency with Tier 2 design descriptions.

### **Technical Specification Changes**

The Technical Specification (TS) and TS Bases will be updated to lower the allowable specific activity of the secondary coolant. TS 3.7.4 Limiting Condition for Operation (LCO), Surveillance Requirement (SR) 3.7.4.1, Bases Applicable Safety Analyses and LCO for B 3.7.4, and Bases Applicable Safety Analyses for B 3.4.10 are updated. The current Technical Specification (TS) limit for secondary iodine activity is 0.1  $\mu\text{Ci/g}$  dose equivalent (DE) I-131 (LCO 3.7.4). This is a standard value; however, the TS bases refer to the steam line break analysis; in other words, the steam line break analysis defines the level of secondary activity that is acceptable. The maximum secondary activity is also limited by other TS limits. Specifically, the primary coolant specific activity concentration limit of 1  $\mu\text{Ci/g}$  DE I-131 is based on the design basis fuel defect of 0.25% and the TS primary to secondary leakage limit of 300 gallons per day. The TS limit for secondary iodine activity is therefore revised from 0.1  $\mu\text{Ci/g}$  DE I-131 to 0.01  $\mu\text{Ci/g}$  DE I-131.

### 3.0 Technical Evaluation

General Design Criteria 19 requires limiting operator dose to less than 5 rem following design basis accidents (DBAs). Changing the name of the actuation signal (High-2) for isolating the main control room penetrations in the design description of the Nuclear Island Nonradioactive Ventilation System is a conforming change to Tier 2 Chapter 7 actuation setpoint naming convention. The design intent of the AP1000 is to only use VES when absolutely necessary. Instead of immediately actuating VES, isolating the MCR, and emptying the VES air storage tanks, the defense in depth VBS SFM is actuated first. If the non-safety VBS SFM feature is unavailable or insufficient, only then would a High-2 signal be reached and the VES actuated. This change also addresses release scenarios where high concentrations of particulates or iodine may exist with low levels of noble gas. This control logic leverages the defense in depth VBS SFM while at the same time ensuring automatic actuation of VES, if necessary, to maintain compliance with GDC-19.

The current TS 3.7.4 limit for secondary iodine activity is 0.1  $\mu\text{Ci/g}$  dose equivalent (DE) I-131. This is the level of secondary activity that is acceptable as defined by the steam line break analysis. The revised main steam line break analysis uses a secondary iodine activity limit of 0.01  $\mu\text{Ci/g}$  DE I-131. The revised TS limit for secondary iodine activity is acceptable under normal plant operation because the realistic DE I-131 value for secondary iodine activity as presented in DCD Table 11.1-8 is orders of magnitude below the revised TS. The revised DE I-131 value is within the detection capability of existing instrumentation, and is significantly above the secondary coolant iodine activities anticipated for AP1000.

### 4.0 Regulatory Evaluation

#### 4.1 Exemption Justification

- 4.1.1 Pursuant to 10 CFR §52.63(b)(1), an exemption from elements of the design as certified in the 10 CFR Part 52, Appendix D, design certification rule is requested for a plant-specific Tier 1 departure from the AP1000 DCD for Tier 1 information and for a material departure from the generic TS. The Tier 1 departure is contained in Tier 1 Subsection 2.7.1 and involves the revision of VES actuation signal name. The departures also include a change to TS LCO 3.7.4 and TS SR 3.7.4.1 which involves lowering allowable secondary coolant iodine activity. This exemption request is in accordance with the provisions of 10 CFR §50.12, 10 CFR §52.7, and 10 CFR Part 52, Appendix D, as demonstrated below.

Applicable Regulation(s): 10 CFR Part 52, Appendix D, Section III.B

Specific wording from which exemption is requested:

"III. Scope and Contents

- B. An applicant or licensee referencing this appendix, in accordance with Section IV of this appendix, shall incorporate by reference and comply with the requirements of this appendix, including Tier 1, Tier 2 (including the investment protection short-term availability controls in Section 16.3 of the DCD), and the generic TS except as otherwise provided in this appendix. Conceptual design information in the generic DCD and the evaluation of severe accident mitigation design alternatives in appendix 1B of the generic DCD are not part of this appendix."



4.1.2 DEF evaluated this exemption request in accordance with 10 CFR Part 52, Appendix D, Section VIII.A.4, 10 CFR §50.12, 10 CFR §52.7 and 10 CFR §52.63, which state that the NRC may grant exemptions from the requirements of the regulations provided the following six conditions are met: 1) the exemption is authorized by law [§50.12(a)(1)]; 2) the exemption will not present an undue risk to the health and safety of the public [§50.12(a)(1)]; 3) the exemption is consistent with the common defense and security [§50.12(a)(1)]; 4) special circumstances are present [§50.12(a)(2)]; 5) the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption [§52.63(b)(1)]; and 6) the design change will not result in a significant decrease in the level of safety [Part 52, Appendix D, VIII.A.4]. The requested exemption satisfies the criteria for granting specific exemptions, as described below.

**1. This exemption is authorized by law**

The NRC has authority under 10 CFR §§ 50.12, 52.7, and 52.63 to grant exemptions from the requirements of NRC regulations. Specifically, 10 CFR §§50.12 and 52.7 state that the NRC may grant exemptions from the requirements of 10 CFR Part 52 upon a proper showing. No law exists that would preclude the changes covered by this exemption request. Additionally, granting of the proposed exemption does not result in a violation of the Atomic Energy Act of 1954, as amended, or the Commission's regulations.

Accordingly, this requested exemption is "authorized by law," as required by 10 CFR §50.12(a)(1).

**2. This exemption will not present an undue risk to the health and safety of the public**

The proposed exemption from the requirements of 10 CFR 52, Appendix D, Section III.B would allow changes to elements of the plant-specific Tier 1 DCD to depart from the AP1000 certified (Tier 1) design information and a change to a TS LCO and SR to depart from the AP1000 certified (Tier 2) information. The plant-specific Tier 1 DCD will continue to reflect the approved licensing basis for the applicant, and will maintain a consistent level of detail with that which is currently provided elsewhere in Tier 1 of the plant-specific DCD. Because the change ensures the Nuclear Island Nonradioactive Ventilation System will achieve its design functions, the changed design will ensure the protection of the health and safety of the public. Therefore, no adverse safety impact which would present any additional risk to the health and safety is present.

Therefore, the requested exemption from 10 CFR 52, Appendix D, Section III.B would not present an undue risk to the health and safety of the public.

**3. The exemption is consistent with the common defense and security**

The exemption from the requirements of 10 CFR 52, Appendix D, Section III.B would change elements of the plant-specific Tier 1 DCD by departing from the

AP1000 certified (Tier 1) design information relating to the Nuclear Island Nonradioactive Ventilation System and departing from the generic TS to lower the allowable secondary iodine activity. The exemption does not alter the design, function, or operation of any structures or plant equipment that are necessary to maintain a secure status of the plant. The proposed exemption has no impact on plant security or safeguards procedures.

Therefore, the requested exemption is consistent with the common defense and security.

#### **4. Special circumstances are present**

10 CFR §50.12(a)(2) lists six "special circumstances" for which an exemption may be granted. Pursuant to the regulation, it is necessary for one of these special circumstances to be present in order for the NRC to consider granting an exemption request. The requested exemption meets the special circumstances of 10 CFR §50.12(a)(2)(ii). That subsection defines special circumstances as when "Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule."

The rule under consideration in this request for exemption from Tier 1 Subsection 2.7.1 and the generic TS is 10 CFR 52, Appendix D, Section III.B, which requires that an applicant referencing the AP1000 Design Certification Rule (10 CFR Part 52, Appendix D) shall incorporate by reference and comply with the requirements of Appendix D, including Tier 1 information and generic TS. The Levy Units 1 and 2 COLA references the AP1000 Design Certification Rule and incorporates by reference the requirements of 10 CFR Part 52, Appendix D, including Tier 1 information and generic TS. The underlying purpose of Appendix D, Section III.B is to describe and define the scope and contents of the AP1000 design certification, and to require compliance with the design certification information in Appendix D to maintain the level of safety in the design.

The proposed change to the name of the actuation signal does not impact the design functions for the Nuclear Island Nonradioactive Ventilation System and reducing the TS limit for DE I-131 improves accident consequence margins for DBAs involving secondary coolant release. These changes do not impact the ability of any structures, systems, or components to perform their functions or negatively impact safety. Accordingly, this exemption from the certification information in Tier 1 Subsection 2.7.1, TS LCO 3.7.4, and TS SR 3.7.4.1 will enable the applicant to safely construct and operate the AP1000 facility consistent with the design certified by the NRC in 10 CFR 52, Appendix D.

Therefore, special circumstances are present, because application of the current generic certified design information in Tier 1 and the generic TS as required by 10 CFR Part 52, Appendix D, Section III.B, in the particular circumstances discussed in this request is not necessary to achieve the underlying purpose of the rule.

**5. The special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption**

Based on the nature of the changes to the plant-specific Tier 1 information and generic TS and the understanding that these changes support the design function of the Nuclear Island Nonradioactive Ventilation System and establish limits for the specific activity in the secondary system, it is likely that other AP1000 applicants and licensees will request this exemption. However, if this is not the case, the special circumstances continue to outweigh any decrease in safety from the reduction in standardization because the key design functions of the Nuclear Island Nonradioactive Ventilation System associated with this request will continue to be maintained. This exemption request and the associated marked-up TS LCO and TS SR changes demonstrate that the Nuclear Island Nonradioactive Ventilation System function continues to be maintained following implementation of the change from the generic AP1000 DCD, thereby minimizing the safety impact resulting from any reduction in standardization.

Therefore, the special circumstances associated with the requested exemption outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. In fact, as described in condition 6 below, the exemption will result in no reduction in the level of safety.

**6. The design change will not result in a significant decrease in the level of safety.**

The exemption revises the plant-specific DCD Tier 1 information by changing the name of the actuation signal (from High-High to High-2) for isolating the main control room penetrations in Subsection 2.7.1. This change does not alter the ability of the Nuclear Island Nonradioactive Ventilation System to maintain its design functions. This exemption also revises the generic TS LCO 3.7.4 and TS SR 3.7.4.1 to lower the allowable secondary iodine activity. Because these functions are met, there is no reduction in the level of safety.

Therefore, the design change and change to the TS will not result in a significant decrease in the level of safety.

As demonstrated above, this exemption request satisfies NRC requirements for an exemption to the design certification rule for the AP1000.

**4.2 Significant Hazards Consideration**

**4.2.1 Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

The proposed changes rename the actuation signal (High-2) for isolating the main control room penetrations in the design description of the Nuclear Island Nonradioactive Ventilation System and lower the allowable secondary iodine activity to meet applicable NRC general design criteria requirements. As the proposed changes do not involve any components that could initiate an event by means of component or system failure, the changes do not increase the probability of a previously evaluated accident.

The proposed changes result in reduced consequences for the Main Control Room operator because of lower initial release concentrations for accident scenarios analyzed. The changes do not alter design features available during normal operation or anticipated operational occurrences. The changes do not adversely impact accident source term parameters or affect any release paths used in the safety analyses, which could increase radiological dose consequences. Thus the radiological releases associated with the Chapter 15 accident analyses are not adversely affected.

The proposed changes would not increase the consequences of an accident previously evaluated in the plant-specific DCD. Offsite doses are not adversely affected by the changes proposed. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes do not introduce new failure modes, interactions or dependencies, the malfunction of which could lead to new accident scenarios. One of the changes is administrative and the other lowers an initial source term used in existing accident analyses. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

4.2.3 Does the proposed change involve a significant reduction in a margin of safety?

The proposed changes do not involve a significant reduction in the margin of safety. The proposed changes do not reduce the redundancy or diversity of any safety-related functions. The proposed changes lower the initial source term concentrations used in the accident analyses.

The DCD Chapters 6 and 15 analyses results are not adversely affected. No design basis safety analysis or acceptance criterion is challenged or exceeded by the proposed changes. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

4.3 Applicable Regulatory Requirements/Criteria

10 CFR 52, Appendix D, Section VIII.B.5.a requires that an applicant or licensee who references this appendix may depart from Tier 2 information, without prior NRC approval, unless the proposed departure involves a change to or departure from Tier 1 information, Tier 2\* information, or the Technical Specifications, or requires a license amendment under paragraphs B.5.b or B.5.c of that section. When evaluating the proposed departure, an applicant or licensee shall consider all matters described in the plant-specific DCD. This exemption request involves a departure from Tier 1 Subsection 2.7.1 and the generic TS.

4.4 Precedent

No precedent is cited.

#### **4.5 Conclusions**

Based on the considerations discussed above:

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

The above evaluations demonstrate the requested changes can be accommodated without an increase in the probability or consequences of an accident previously evaluated, without creating the possibility of a new or different kind of accident from any accident previously evaluated, and without a significant reduction in a margin of safety. Having arrived at negative declarations with regard to the criteria of 10 CFR 50.92, this assessment determines the requested change does not involve a Significant Hazards Consideration.

#### **5.0 Risk Assessment**

A risk assessment was determined to be not applicable to address the acceptability of this request.

#### **6.0 References**

- 1) Westinghouse Electric Company, AP1000 Design Control Document, Revision 19, June 2011

**Duke Energy**  
**Enclosure 3**  
**Levy Nuclear Plant Units 1 and 2**  
**Tier 1 and Tier 2 Licensing Basis Documents –**  
**Proposed Changes (Convenience Copy)**

**TIER 1 DCD Changes**



## 2.7 HVAC Systems

### 2.7.1 Nuclear Island Nonradioactive Ventilation System

#### Design Description

The nuclear island nonradioactive ventilation system (VBS) serves the main control room (MCR), control support area (CSA), Class 1E dc equipment rooms, Class 1E instrumentation and control (I&C) rooms, Class 1E electrical penetration rooms, Class 1E battery rooms, remote shutdown room (RSR), reactor coolant pump trip switchgear rooms, adjacent corridors, and passive containment cooling system (PCS) valve room during normal plant operation. The VBS consists of the following independent subsystems: the main control room/control support area HVAC subsystem, the class 1E electrical room HVAC subsystem, and the passive containment cooling system valve room heating and ventilation subsystem. The VBS provides heating, ventilation, and cooling to the areas served when ac power is available. The system provides breathable air to the control room and maintains the main control room and control support area areas at a slightly positive pressure with respect to the adjacent rooms and outside environment during normal operations. The VBS monitors the main control room supply air for radioactive particulate and iodine concentrations and provides filtration of main control room/control support area air during conditions of abnormal (high) airborne radioactivity. In addition, the VBS isolates the HVAC penetrations in the main control room boundary on "~~high-high~~High-2" particulate or iodine radioactivity in the main control room supply air duct or on a loss of ac power for more than 10 minutes. The Sanitary Drainage System (SDS) also isolates a penetration in the main control room boundary on "~~high-high~~High-2" particulate or iodine radioactivity in the main control room supply air duct or on a loss of ac power for more than 10 minutes. Additional penetrations from the SDS and Potable Water System (PWS) into the main control room boundary are maintained leak tight using a loop seal in the piping, and the Waste Water System (WWS) is isolated using a normally closed safety related manual isolation valve. These features support operation of the main control room emergency habitability system (VES), and have been included in Tables 2.7.1-1 and 2.7.1-2.



**TIER 2 DCD Changes**



Table 1.6-1 (Sheet 15 of 21)

**MATERIAL REFERENCED**

DCD Section Number	Westinghouse Topical Report Number	Title
15.4	WCAP-7979-P-A (P) WCAP-8028-A	TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code, January 1975
	WCAP-7908-A	FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO <sub>2</sub> Fuel Rod, December 1989
	WCAP-7907-P-A (P) WCAP-7907-A	LOFTRAN Code Description, April 1984
	WCAP-15806-P-A (P) WCAP-15807-NP- <del>WCAP-7588</del>	Westinghouse Control Rod Ejection Accident Analysis Methodology Using Multi-Dimensional Kinetics <del>An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods, Revision 1A, January 1975</del>
	WCAP-10965-P-A (P) WCAP-10966-A	ANC: A Westinghouse Advanced Nodal Computer Code, September 1986
	WCAP-11397-P-A (P) WCAP-11397-A	Revised Thermal Design Procedure, April 1989
	WCAP-15644-P (P) WCAP-15644-NP	AP1000 Code Applicability Report, Revision 2, March 2004
	WCAP-11596-P-A (P) WCAP-11597-A	Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized -Water Reactor Cores, June 1988
	WCAP-16045-P-A (P) WCAP-16045-NP-A	Qualification of the Two-Dimensional Transport Code PARAGON, August -2004
	WCAP-10965-P-A, Addendum 1 (P) WCAP-10966-A Addendum -1	ANC – A Westinghouse Advanced Nodal Computer Code; Enhancements to ANC Rod Power Recovery, April 1989
	WCAP-14565-P-A (P) WCAP-15306-NP-A	VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA -Thermal-Hydraulic Safety Analysis, October 1999
	WCAP--15063-P-A, Revision 1 with Errata (P) WCAP-15064-NP-A	Westinghouse Improved Performance Analysis and Design Model (PAD 4.0), July -2000
	WCAP-16045-P-A Addendum -1-A (P) WCAP-16045-NP-A Addendum 1-A	Qualification of the NEXUS Nuclear Data Methodology, August 2007
	WCAP-10965-P-A Addendum 2-A (P)	Qualification of the New Pin Power Recovery Methodology, September 2010
	WCAP-15025-P-A (P) WCAP-15026-NP-A	Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in -17x17 Rod Bundles with Modified LPD Mixing Vane Grids, April- 1999



**1.9.4.2.3 New Generic Issues****Issue 83    Control Room Habitability****Discussion:**

Loss of control room habitability following an accidental release of external toxic or radioactive material or smoke can impair or cause loss of the control room operators' capability to safely control the reactor. Use of the remote shutdown workstation outside the control room following such events is unreliable since this station has no emergency habitability or radiation protection provisions.

**AP1000 Response:**

Habitability of the main control room is provided by the main control room/control support area HVAC subsystem of the nonsafety-related nuclear island nonradioactive ventilation system (VBS). If ac power is unavailable for more than 10 minutes or if "~~high-high~~High-2" particulate or iodine radioactivity is detected in the main control room supply air duct, which would lead to exceeding General Design Criteria 19 operator dose limits, the protection and safety monitoring system automatically isolates the main control room and operator habitability requirements are then met by the main control room emergency habitability system (VES). The safety-related main control room emergency habitability system supplies breathable quality air for the main control room operators while the main control room is isolated.



## Appendix 1A

**Reg. Guide 1.52, Rev. 3, 6/01 – Design, Testing, and Maintenance Criteria for Post Accident  
Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Absorption Units of  
Light-Water-Cooled Nuclear Power Plants**

C.4.9

~~Exception~~Conforms     ~~The credited adsorber efficiencies are 90% for elemental iodine and 30%90% for organic iodine. These efficiencies assume no humidity control.~~

Table 1

~~Exception~~Conforms     ~~The Technical Specification methyl iodide penetration acceptance limit for the AP1000 activated carbon adsorber is 5%, which correlates to 90% removal efficiency of both organic and elemental iodine. The calculated design basis for the AP1000 passive filtration adsorbers assumes a 30% organic iodine removal efficiency and a 90% elemental iodine efficiency. A 1% bypass leakage is accounted for by testing to increased organic iodine removal efficiency.~~



### 3.1.2 Protection by Multiple Fission Product Barriers

#### Criterion 19 – Control Room

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss of coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident.

Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

#### AP1000 Compliance

The AP1000 main control room provides the man-machine interfaces required to operate the plant safely and efficiently under normal conditions and to maintain it in a safe manner under accident conditions, including LOCAs. Simplified passive safety-related system designs are provided that do not rely upon operator action to maintain core cooling for design basis accidents. Operator action outside the main control room to mitigate the consequences of an accident is permitted.

The main control room is shielded by the containment and auxiliary building from direct gamma radiation and inhalation doses resulting from the postulated release of fission products inside containment. Refer to Chapter 15 for additional information on accident conditions. The main control room/control support area HVAC subsystem of the nuclear island nonradioactive ventilation system (VBS) allows access to and occupancy of the main control room under accident conditions as described in subsection 9.4.1. Sufficient shielding and the main control room/control support area HVAC subsystem provide adequate protection so that personnel will not receive radiation exposure in excess of 5 rem whole-body or its equivalent to any part of the body for the duration of the accident.

If ac power is unavailable for more than 10 minutes or if "~~high-high~~High-2" particulate, low pressurizer pressure is detected, or "High-2" iodine radioactivity is detected in the main control room supply air duct, which would lead to exceeding General Design Criteria 19 operator dose limits, the protection and safety monitoring system automatically isolates the main control room and operator habitability requirements are then met by the main control room emergency habitability system (VES). The main control room emergency habitability system also allows access to and occupancy of the main control room under accident conditions. The emergency main control room habitability system is designed to satisfy seismic Category I requirements as described in Section 3.2; the system design is described in Section 6.4.

In the event that the operators are forced to abandon the main control room, a workstation is provided with remote shutdown capability. A main control room evacuation is not assumed to occur simultaneously with design basis events. The remote shutdown workstation is described in Section 7.4.



## 6.4 Habitability Systems

The habitability systems are a set of individual systems that collectively provide the habitability functions for the plant. The systems that make up the habitability systems are the:

- Nuclear island nonradioactive ventilation system (VBS)
- Main control room emergency habitability system (VES)
- Radiation monitoring system (RMS)
- Plant lighting system (ELS)
- Fire Protection System (FPS)

When a source of ac power is available, the nuclear island nonradioactive ventilation system (VBS) provides normal and abnormal HVAC service to the main control room (MCR), control support area (CSA), instrumentation and control rooms, dc equipment rooms, battery rooms, and the nuclear island nonradioactive ventilation system equipment room as described in subsection 9.4.1.

If ac power is unavailable for more than 10 minutes or if “~~high-high~~High-2” particulate or iodine radioactivity is detected in the main control room supply air duct, which would lead to exceeding General Design Criteria 19 operator dose limits, the protection and safety monitoring system automatically isolates the main control room and operator habitability requirements are then met by the main control room emergency habitability system (VES). The main control room emergency habitability system is capable of providing emergency ventilation and pressurization for the main control room. The main control room emergency habitability system also provides emergency passive heat sinks for the main control room, instrumentation and control rooms, and dc equipment rooms.

Radiation monitoring of the main control room environment is provided by the radiation monitoring system. Smoke detection is provided in the VBS system. Emergency lighting is provided by the plant lighting system. Storage capacity is provided in the main control room for personnel support equipment. Manual hose stations outside the MCR and portable fire extinguishers are provided to fight MCR fires.

### 6.4.2.6 Shielding Design

The design basis loss-of-coolant accident (LOCA) dictates the shielding requirements for the main control room). Main control room shielding design bases are discussed in Section 12.3. **In addition to shielding provided by building structural features, consideration is given to shielding provided by the VES filter shielding.** Descriptions of the design basis LOCA source terms, main control room shielding parameters, and evaluation of doses to main control room personnel are presented in Section 15.6.

The main control room and its location in the plant are shown in Figure 12.3-1.



#### 6.4.3.2 Emergency Mode

Operation of the main control room emergency habitability system is automatically initiated by either of the following conditions:

- “~~High-high~~High-2” particulate or iodine radioactivity in the main control room supply air duct
- Loss of ac power for more than 10 minutes

Operation can also be initiated by manual actuation.

The nuclear island nonradioactive ventilation system is isolated from the main control room pressure boundary by automatic closure of the isolation devices located in the nuclear island nonradioactive ventilation system ductwork if radiation levels in the main control room supply air duct exceed the “~~high-high~~High-2” setpoint or if ac power is lost for more than 10 minutes. At the same time, the main control room emergency habitability system begins to deliver air from the emergency air storage tanks to the main control room by automatically opening the isolation valves located in the supply line. The relief damper isolation valves also open allowing the pressure relief dampers to function and discharge the damper flow to purge the vestibule.

#### 6.4.4 System Safety Evaluation

In the event of an accident involving the release of radioactivity to the environment, the nuclear island nonradioactive ventilation system (VBS) is expected to switch from the normal operating mode to the supplemental air filtration mode to protect the main control room personnel. Although the VBS is not a safety-related system, it is expected to be available to provide the necessary protection for realistic events. However, the design basis accident doses reported in Chapter 15 utilize conservative assumptions, and the main control room doses are calculated based on operation of the safety-related emergency habitability system (VES) since this is the system that is relied upon to limit the amount of activity the personnel are exposed to. The analyses assume that the VBS is initially in operation, but fails to enter the supplemental air filtration mode on a High-1 radioactivity indication in the main control room atmosphere. VES operation is then assumed to be initiated once the High-2 level for control room atmosphere activity is reached.

Doses are also calculated assuming that the VBS does operate in the supplemental air filtration mode as designed, but with no switchover to VES operation. This VBS operating case demonstrates the defense-in-depth that is provided by the system and also shows that, in the event of an accident with realistic assumptions, the VBS is adequate to protect the control room operators without depending on VES operation.

Doses were determined for the following design basis:

	VES Operating	VBS Operating
Large Break LOCA	4.414.33 rem TEDE	4.734.84 rem TEDE
Fuel Handling Accident	21.5 rem TEDE	1.6 rem TEDE
Steam Generator Tube Rupture		
(Pre-existing iodine spike)	4.33.4 rem TEDE	3.14.0 rem TEDE
(Accident-initiated iodine spike)	1.20 rem TEDE	1.74 rem TEDE
Steam Line Break		
(Pre-existing iodine spike)	3.91.1 rem TEDE	2.10.9 rem TEDE
(Accident-initiated iodine spike)	4.01.3 rem TEDE	4.92.9 rem TEDE



Rod Ejection Accident	<del>1.83.6</del> rem TEDE	<del>2.25</del> rem TEDE
Locked Rotor Accident		
(Accident without feedwater available)	0. <del>74</del> rem TEDE	0. <del>57</del> rem TEDE
(Accident with feedwater available)	0. <del>52</del> rem TEDE	<del>1.50.9</del> rem TEDE
Small Line Break Outside Containment	0. <del>84</del> rem TEDE	0. <del>34</del> rem TEDE

For all events the doses are within the dose acceptance limit of 5.0 rem TEDE. The details of analysis assumptions for modeling the doses to the main control room personnel are delineated in the LOCA dose analysis discussion in subsection 15.6.5.3 for VES operating cases. The analysis assumptions are provided in subsection 9.4.1.2.3.1 for the VBS operating case.

No radioactive materials are stored or transported near the main control room pressure boundary.

As discussed and evaluated in subsection 9.5.1, the use of noncombustible construction and heat and flame resistant materials throughout the plant reduces the likelihood of fire and consequential impact on the main control room atmosphere. Operation of the nuclear island nonradioactive ventilation system in the event of a fire is discussed in subsection 9.4.1.

The exhaust stacks of the onsite standby power diesel generators are located in excess of 150 feet away from the fresh air intakes of the main control room. The onsite standby power system fuel oil storage tanks are located in excess of 300 feet from the main control room fresh air intakes. These separation distances reduce the possibility that combustion fumes or smoke from an oil fire would be drawn into the main control room.

The protection of the operators in the main control room from offsite toxic gas releases is discussed in Section 2.2. The sources of onsite chemicals are described in Table 6.4-1, and their locations are shown on Figure 1.2-2. Analysis of these sources is in accordance with Regulatory Guide 1.78 (Reference 5) and the methodology in NUREG-0570, "Toxic Vapor Concentrations in the Control Room Following a Postulated Accidental Release" (Reference 6), and the analysis shows that these sources do not represent a toxic or flammability hazard to control room personnel.

A supply of protective clothing, respirators, and self-contained breathing apparatus adequate for 11 persons is stored within the main control room pressure boundary.

The main control room emergency habitability system components discussed in subsection 6.4.2.3 are arranged as shown in Figure 6.4-2. The location of components and piping within the main control room pressure boundary provides the required supply of compressed air to the main control room pressure boundary, as shown in Figure 6.4-1.

During emergency operation, the main control room emergency habitability system passive heat sinks are designed to limit the temperature inside the main control room to remain within limits for reliable human performance (References 2 and 3) over 72 hours. The passive heat sinks limit the air temperature inside the instrumentation and control rooms to 120°F and dc equipment rooms to 120°F. The walls and ceilings that act as the passive heat sinks contain sufficient thermal mass to accommodate the heat sources from equipment, personnel, and lighting for 72 hours.

The main control room emergency habitability system nominally provides 65 scfm of ventilation air to the main control room from the compressed air storage tanks. Sixty scfm of supplied ventilation flow is sufficient to induce a filtration flow of at least 600 cfm into the passive air filtration line located inside the main control room envelope. This ventilation flow is also sufficient to pressurize the control room to at



least positive 1/8-inch water gauge differential pressure with respect to the surrounding areas in addition to limiting the carbon dioxide concentration below one-half percent by volume for a maximum occupancy of 11 persons and maintaining air quality within the guidelines of Table 1 and Appendix C, Table C-1, of Reference 1.

Automatic transfer of habitability system functions from the main control room/control support area HVAC subsystem of the nuclear island nonradioactive ventilation system to the main control room emergency habitability system is initiated by either the following conditions:

- “~~High-high~~High-2” particulate or iodine radioactivity in MCR air supply duct
- Loss of ac power for more than 10 minutes

The airborne fission product source term in the reactor containment following the postulated LOCA is assumed to leak from the containment and airborne fission products are assumed to result from spent fuel pool steaming. The concentration of radioactivity, which is assumed to surround the main control room, after the postulated accident, is evaluated as a function of the fission product decay constants, the containment leak rate, and the meteorological conditions assumed. The assessment of the amount of radioactivity within the main control room takes into consideration the radiological decay of fission products and the infiltration/exfiltration rates to and from the main control room pressure boundary.

A single active failure of a component of the main control room emergency habitability system or nuclear island nonradioactive ventilation system does not impair the capability of the systems to accomplish their intended functions. The Class 1E components of the main control room emergency habitability system are connected to independent Class 1E power supplies. Both the main control room emergency habitability system and the portions of the nuclear island nonradioactive ventilation system which isolates the main control room are designed to remain functional during an SSE or design-basis tornado.

In accordance with SECY-77-439 (Reference 13), a single passive failure of a component in the passive filtration line in the main control room emergency habitability system does not impair the capability of the system to accomplish its intended function. There is no source that could create line blockage in the VES line from the air bottles to the eductor. Thus potential blockage in the filtration line does not preclude breathable air from the emergency air storage tanks from being delivered to the main control room envelope for 72 hours during VES operation. The following cases are evaluated since they involve releases that extend beyond 24 hours after the initiation of the event:

Large Break LOCA	4.54 rem TEDE
Steam Line Break	
(Pre-existing iodine spike)	4.01.2 rem TEDE
(Accident-initiated iodine spike)	4.52.0 rem TEDE



Table 6.4-2

**MAIN CONTROL ROOM HABITABILITY INDICATIONS AND ALARMS**

VES emergency air storage tank pressure (indication and low and low-low alarms)

VES MCR pressure boundary differential pressure (indication and high and low alarms)

VES air delivery line flowrate (indication and high and low alarms)

VES passive filtration flow rate (indication and high and low alarms)

VBS main control room supply air radiation level (~~high-high~~High-1 and High-2 alarms)

VBS outside air intake smoke level (high alarm)

VBS isolation valve position

VBS MCR pressure boundary differential pressure



### 7.3.1.2.17 Control Room Isolation and Air Supply Initiation

Signals to initiate isolation of the main control room, to initiate the air supply, and to open the control room pressure relief isolation valves are generated from either of the following conditions:

1. High-2 control room air supply radioactivity level
2. Loss of ac power sources (low Class 1E battery charger input voltage)
3. Manual initiation

Condition 1 is the occurrence one of two main control room air supply radioactivity monitors detecting a the iodine or particulate radioactivity level above the High-2 setpoint.

### 9.2.6.1.1 Safety Design Basis

The sanitary drainage system isolates the SDS vent penetration in the main control room boundary on ~~high-high~~High-2 particulate or iodine concentrations in the main control room air supply or on extended loss of ac power to support operation of the main control room emergency habitability system as described in Section 6.4. The SDS vent line that penetrates the main control room envelope is safety related and designed as seismic Category I to provide isolation of the main control room envelope from the surrounding areas and outside environment in the event of a design basis accident. An additional penetration from the SDS into the main control room envelope is maintained leak tight using a loop seal in the safety-related seismic Category I piping.

### 9.4.1.1.1 Safety Design Basis

The nuclear island nonradioactive ventilation system provides the following nuclear safety-related design basis functions:

- Monitors the main control room supply air for radioactive particulate and iodine concentrations
- Isolates the HVAC penetrations in the main control room boundary on ~~high-high~~High-2 particulate or iodine concentrations in the main control room supply air or on extended loss of ac power to support operation of the main control room emergency habitability system as described in Section 6.4

### 9.4.1.1.2 Power Generation Design Basis Main Control Room/Control Support Area (CSA) Areas

The nuclear island nonradioactive ventilation system provides the following specific functions:

- Controls the main control room and control support area relative humidity between 25 to 60 percent
- Maintains the main control room and CSA areas at a slightly positive pressure with respect to the adjacent rooms and outside environment during normal operations to prevent infiltration of unmonitored air into the main control room and CSA areas
- Isolates the main control room and/or CSA area from the normal outdoor air intake and provides filtered outdoor air to pressurize the main control room and CSA areas to a positive pressure of at least 1/8 inch wg when a ~~high-High-1 gaseous~~radioactivity



concentration (~~gaseous, particulate, or iodine~~) is detected in the main control room supply duct

- Isolates the main control room and/or CSA area from the normal outdoor air intake and provides 100 percent recirculation air to the main control room and CSA areas when a high concentration of smoke is detected in the outside air intake
- Provides smoke removal capability for the main control room and control support area
- Maintains the main control room emergency habitability system passive cooling heat sink below its initial design ambient air temperature limit of 75°F
- Maintains the main control room/control support area carbon dioxide levels below 0.5 percent concentration and the air quality within the guidelines of Table 1 and Appendix C, Table C-1 of Reference 32.

#### 9.4.1.2.1.1 Main Control Room/Control Support Area HVAC Subsystem

The main control room/control support area HVAC subsystem serves the main control room and control support area with two 100 percent capacity supply air handling units, return/exhaust air fans, supplemental air filtration units, associated dampers, instrumentation and controls, and common ductwork. The supply air handling units and return/exhaust air fans are connected to common ductwork which distributes air to the main control room and CSA areas. The main control room envelope consists of the main control room, shift manager's office, operation work area, toilet, and operations break room area. The CSA area consists of the main control support area operations area, conference rooms, NRC room, computer rooms, shift turnover room, kitchen/rest area, and restrooms. The main control room and control support area toilets have separate exhaust fans.

Outside supply air is provided to the plant areas served by the main control room/control support area HVAC subsystem through an outside air intake duct that is protected by an intake enclosure located on the roof of the auxiliary building at elevation 153'-0". The outside air intake duct is located more than 50 feet below and more than 100 feet laterally away from the plant vent discharge. The supply, return, and toilet exhaust are the only HVAC penetrations in the main control room envelope and include redundant safety-related seismic Category I isolation valves that are physically located within the main control room envelope. Redundant safety-related radiation monitor sample line connections are located upstream of the VBS supply air isolation valves. These monitors initiate operation of the nonsafety-related supplemental air filtration units on ~~high-High-1~~ ~~gaseous~~-radioactivity concentrations (~~gaseous, particulate, or iodine~~) and isolate the main control room from the nuclear island nonradioactive ventilation system on ~~high-highHigh-2~~ particulate or iodine radioactivity concentrations. See Section 11.5 for a description of the main control room supply air radiation monitors.

#### 9.4.1.2.3.1 Main Control Room/Control Support Area HVAC Subsystem Abnormal Plant Operation

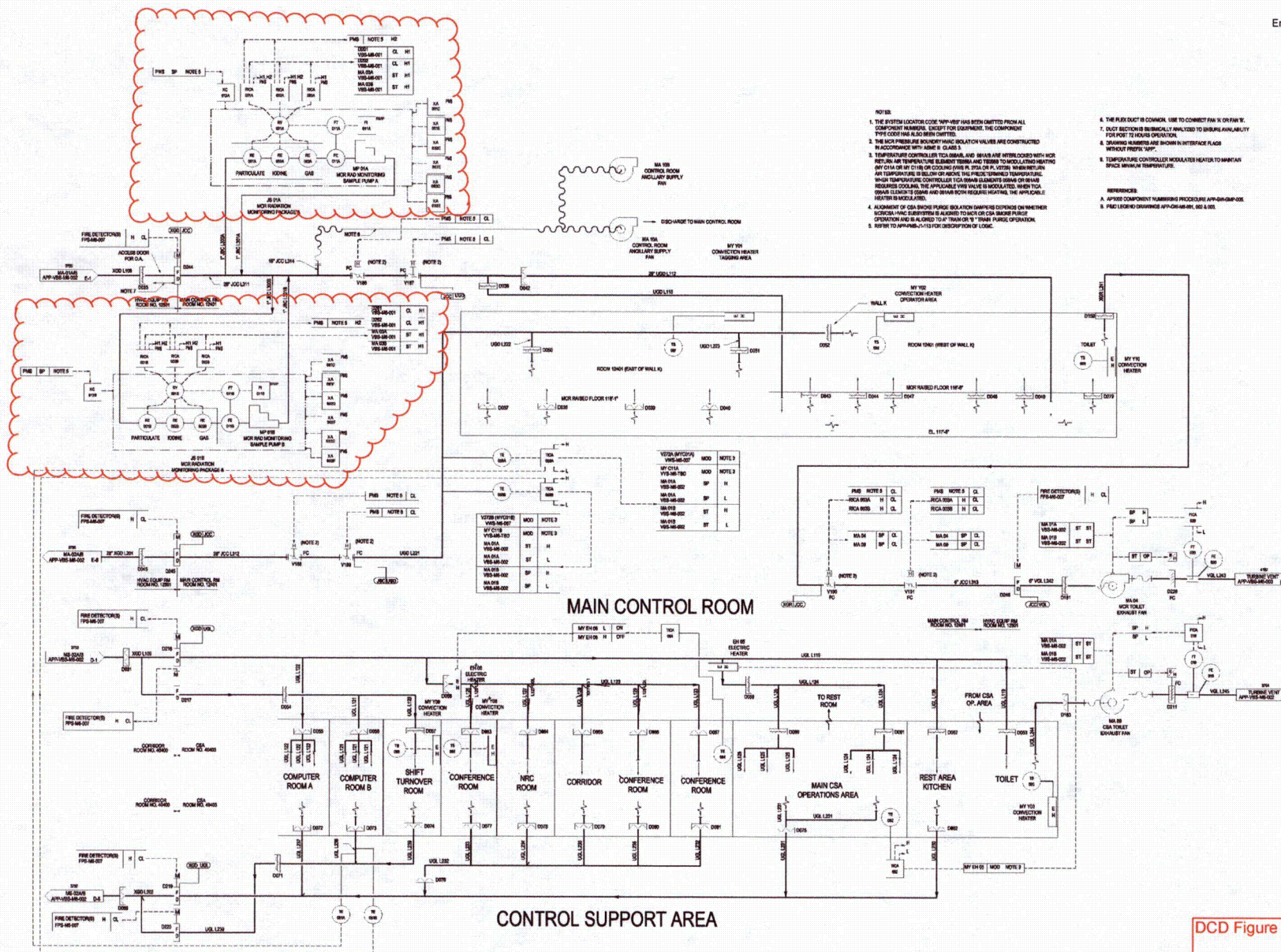
Control actions are taken at two levels of radioactivity as detected in the main control room supply air duct. The first is "~~highHigh-1~~" radioactivity based upon ~~gaseous~~-radioactivity instrumentation (~~gaseous, particulate, or iodine~~). The second is "~~high-highHigh-2~~" radioactivity based upon either particulate or iodine radioactivity instruments.



If "~~high~~High-1" ~~gaseous~~ radioactivity is detected in the main control room supply air duct and the main control room/control support area HVAC subsystem is operable, both supplemental air filtration units automatically start to pressurize the main control room and CSA areas to at least 1/8 inch wg with respect to the surrounding areas and the outside environment using filtered makeup air. The normal outside air makeup duct and the main control room and control support area toilet exhaust duct isolation dampers close. The smoke/purge exhaust isolation dampers close, if open. The main control room/control support area supply air handling unit continues to provide cooling with recirculation air to maintain the main control room passive heat sink below its initial ambient air design temperature and maintains the main control room and CSA areas within their design temperatures. The supplemental air filtration subsystem pressurizes the combined volume of the main control room and control support area concurrently with filtered outside air. A portion of the recirculation air from the main control room and control support area is also filtered for cleanup of airborne radioactivity. The main control room/control support area HVAC equipment and ductwork that form an extension of the main control room/control support area pressure boundary limit the overall infiltration (negative operating pressure) and exfiltration (positive operating pressure) rates to those values shown in Table 9.4.1-1. Based on these values, the system is designed to maintain personnel doses within allowable General Design Criteria (GDC) 19 limits during design basis accidents in both the main control room and the control support area.

If ac power is unavailable for more than 10 minutes or if "~~high~~ ~~high~~High-2h" particulate or iodine radioactivity is detected in the main control room supply air duct, which would lead to exceeding GDC 19 operator dose limits, the protection and safety monitoring system automatically isolates the main control room from the normal main control room/control support area HVAC subsystem by closing the supply, return, and toilet exhaust isolation valves. Main control room habitability is maintained by the main control room emergency habitability system, which is discussed in Section 6.4.





DCD Figure 9.4.1-1 Sheet 5



Table 11.1-4

**PARAMETERS USED TO CALCULATE SECONDARY COOLANT ACTIVITY**

Total secondary side water mass (lb/steam generator)	<del>1.761</del> 1.68 x 10 <sup>5</sup>
Steam generator steam fraction	0. <del>955</del> 058
Total steam flow rate (lb/hr)	1.5 x 10 <sup>7</sup>
Moisture carryover (percent)	0.1
Total makeup water feed rate (lb/hr)	<del>732</del> 700
Total blowdown rate (gpm)	186
Total primary-to-secondary leak rate (gpd)	<del>500</del> 300
Iodine partition factor (mass basis)	100



Table 11.1-5

## DESIGN BASIS STEAM GENERATOR SECONDARY SIDE LIQUID ACTIVITY

Nuclide	Activity ( $\mu\text{Ci/g}$ )	Nuclide	Activity ( $\mu\text{Ci/g}$ )
Br-83	$1.4 \times 10^{-5}$ <del><math>2.3 \times 10^{-5}</math></del>	Y-93	$8.2 \times 10^{-8}$ <del><math>1.5 \times 10^{-7}</math></del>
Br-84	$2.4 \times 10^{-6}$ <del><math>4.0 \times 10^{-6}</math></del>	Zr-95	$1.5 \times 10^{-7}$ <del><math>2.7 \times 10^{-7}</math></del>
Br-85	$3.1 \times 10^{-8}$ <del><math>4.9 \times 10^{-8}</math></del>	Nb-95	$1.5 \times 10^{-7}$ <del><math>2.7 \times 10^{-7}</math></del>
I-129	$1.3 \times 10^{-11}$ <del><math>2.4 \times 10^{-11}</math></del>	Mo-99	$1.9 \times 10^{-4}$ <del><math>3.4 \times 10^{-4}</math></del>
I-130	$7.9 \times 10^{-6}$ <del><math>1.4 \times 10^{-5}</math></del>	Tc-99m	$1.7 \times 10^{-4}$ <del><math>3.2 \times 10^{-4}</math></del>
I-131	$6.3 \times 10^{-4}$ <del><math>1.1 \times 10^{-3}</math></del>	Ru-103	$1.2 \times 10^{-7}$ <del><math>2.3 \times 10^{-7}</math></del>
I-132	$4.2 \times 10^{-4}$ <del><math>7.3 \times 10^{-4}</math></del>	Ru-106	$4.1 \times 10^{-8}$
I-133	$1.0 \times 10^{-3}$ <del><math>1.8 \times 10^{-3}</math></del>	Rh-103m	$1.2 \times 10^{-7}$ <del><math>2.3 \times 10^{-7}</math></del>
I-134	$4.9 \times 10^{-5}$ <del><math>8.1 \times 10^{-5}</math></del>	Rh-106	$4.1 \times 10^{-8}$ <del><math>2.0 \times 10^{-10}</math></del>
I-135	$5.0 \times 10^{-4}$ <del><math>8.7 \times 10^{-4}</math></del>	Ag-110m	$3.0 \times 10^{-6}$ <del><math>6.7 \times 10^{-7}</math></del>
Rb-86	$1.4 \times 10^{-5}$	Te-125m	$1.5 \times 10^{-7}$
Rb-88	$1.4 \times 10^{-4}$ <del><math>2.3 \times 10^{-4}</math></del>	Te-127m	$7.0 \times 10^{-7}$ <del><math>1.3 \times 10^{-6}</math></del>
Rb-89	$5.6 \times 10^{-6}$ <del><math>8.9 \times 10^{-6}</math></del>	Te-127	$2.2 \times 10^{-6}$ <del><math>3.2 \times 10^{-7}</math></del>
Cs-134	$1.1 \times 10^{-3}$ <del><math>2.1 \times 10^{-3}</math></del>	Te-129m	$2.4 \times 10^{-6}$ <del><math>4.4 \times 10^{-6}</math></del>
Cs-136	$1.7 \times 10^{-3}$ <del><math>3.0 \times 10^{-3}</math></del>	Te-129	$2.1 \times 10^{-6}$ <del><math>3.8 \times 10^{-6}</math></del>
Cs-137	$8.2 \times 10^{-4}$ <del><math>1.5 \times 10^{-3}</math></del>	Te-131m	$5.6 \times 10^{-6}$ <del><math>1.0 \times 10^{-5}</math></del>
Cs-138	$5.9 \times 10^{-5}$ <del><math>9.5 \times 10^{-5}</math></del>	Te-131	$1.6 \times 10^{-6}$ <del><math>2.8 \times 10^{-6}</math></del>
H-3	$3.8 \times 10^{-1}$ <del><math>1.0</math></del>	Te-132	$7.0 \times 10^{-5}$ <del><math>1.3 \times 10^{-4}</math></del>
Cr-51	$1.3 \times 10^{-6}$ <del><math>2.2 \times 10^{-6}</math></del>	Te-134	$2.0 \times 10^{-6}$ <del><math>3.2 \times 10^{-6}</math></del>
Mn-54	$6.6 \times 10^{-7}$ <del><math>1.1 \times 10^{-6}</math></del>	Ba-137m	$7.7 \times 10^{-4}$ <del><math>1.4 \times 10^{-3}</math></del>
Mn-56	$7.8 \times 10^{-5}$ <del><math>1.3 \times 10^{-4}</math></del>	Ba-140	$9.4 \times 10^{-7}$ <del><math>1.7 \times 10^{-6}</math></del>
Fe-55	$5.0 \times 10^{-7}$ <del><math>8.4 \times 10^{-7}</math></del>	La-140	$3.3 \times 10^{-7}$ <del><math>6.0 \times 10^{-7}</math></del>
Fe-59	$1.3 \times 10^{-7}$ <del><math>2.2 \times 10^{-7}</math></del>	Ce-141	$1.4 \times 10^{-7}$ <del><math>2.6 \times 10^{-7}</math></del>
Co-58	$1.9 \times 10^{-6}$ <del><math>3.2 \times 10^{-6}</math></del>	Ce-143	$1.2 \times 10^{-7}$ <del><math>2.2 \times 10^{-7}</math></del>
Co-60	$2.2 \times 10^{-7}$ <del><math>3.7 \times 10^{-7}</math></del>	Ce-144	$1.1 \times 10^{-7}$ <del><math>1.9 \times 10^{-7}</math></del>
Sr-89	$1.8 \times 10^{-6}$ <del><math>3.3 \times 10^{-6}</math></del>	Pr-143	$1.4 \times 10^{-7}$ <del><math>2.5 \times 10^{-7}</math></del>
Sr-90	$8.0 \times 10^{-8}$ <del><math>1.5 \times 10^{-7}</math></del>	Pr-144	$1.1 \times 10^{-7}$ <del><math>1.9 \times 10^{-7}</math></del>
Sr-91	$1.9 \times 10^{-6}$ <del><math>3.3 \times 10^{-6}</math></del>		
Sr-92	$2.4 \times 10^{-7}$ <del><math>4.0 \times 10^{-7}</math></del>		
Y-90	$1.4 \times 10^{-8}$ <del><math>2.7 \times 10^{-8}</math></del>		
Y-91m	$1.0 \times 10^{-6}$ <del><math>1.8 \times 10^{-6}</math></del>		
Y-91	$1.3 \times 10^{-7}$ <del><math>2.3 \times 10^{-7}</math></del>		
Y-92	$2.8 \times 10^{-7}$ <del><math>4.9 \times 10^{-7}</math></del>		



Table 11.1-6

**DESIGN BASIS STEAM GENERATOR SECONDARY SIDE STEAM ACTIVITY**

Nuclide	Activity ( $\mu\text{Ci/g}$ )
Kr-83m	$1.1 \times 10^{-6}$ <del><math>1.8 \times 10^{-6}</math></del>
Kr-85m	$4.3 \times 10^{-6}$ <del><math>7.2 \times 10^{-6}</math></del>
Kr-85	$1.5 \times 10^{-5}$ <del><math>2.5 \times 10^{-5}</math></del>
Kr-87	$2.4 \times 10^{-6}$ <del><math>4.1 \times 10^{-6}</math></del>
Kr-88	$7.7 \times 10^{-6}$ <del><math>1.3 \times 10^{-5}</math></del>
Kr-89	$1.8 \times 10^{-7}$ <del><math>3.0 \times 10^{-7}</math></del>
Xe-131m	$6.9 \times 10^{-6}$ <del><math>1.2 \times 10^{-5}</math></del>
Xe-133m	$8.7 \times 10^{-6}$ <del><math>1.4 \times 10^{-5}</math></del>
Xe-133	$6.4 \times 10^{-4}$ <del><math>1.1 \times 10^{-3}</math></del>
Xe-135m	$5.5 \times 10^{-6}$ <del><math>1.0 \times 10^{-5}</math></del>
Xe-135	$1.9 \times 10^{-5}$ <del><math>3.1 \times 10^{-5}</math></del>
Xe-137	$3.4 \times 10^{-7}$ <del><math>5.7 \times 10^{-7}</math></del>
Xe-138	$1.3 \times 10^{-6}$ <del><math>2.1 \times 10^{-6}</math></del>
I-129	$1.5 \times 10^{-13}$ <del><math>2.7 \times 10^{-13}</math></del>
I-130	$8.7 \times 10^{-8}$ <del><math>1.5 \times 10^{-7}</math></del>
I-131	$6.9 \times 10^{-6}$ <del><math>1.3 \times 10^{-5}</math></del>
I-132	$4.7 \times 10^{-6}$ <del><math>8.0 \times 10^{-6}</math></del>
I-133	$1.1 \times 10^{-5}$ <del><math>2.0 \times 10^{-5}</math></del>
I-134	$5.4 \times 10^{-7}$ <del><math>8.9 \times 10^{-7}</math></del>
I-135	$5.5 \times 10^{-6}$ <del><math>9.5 \times 10^{-6}</math></del>
H-3	$3.8 \times 10^{-1}$ <del>4.0</del>



#### 11.5.1.1 Safety Design Basis

While the radiation monitoring system is primarily a surveillance system, certain detector channels perform safety-related functions. The components used in these channels meet the qualification requirements for safety-related equipment as described in subsection 7.1.4.

Channel and equipment redundancy is provided for safety-related monitors to maintain the safety-related function in case of a single failure.

The design objectives of the radiation monitoring system during postulated accidents are:

- Initiate containment air filtration isolation in the event of abnormally high radiation inside the containment (High-1)
- Initiate normal residual heat removal system suction line containment isolation in the event of abnormally high radiation inside the containment (High-2)
- Initiate main control room supplemental filtration in the event of abnormally high particulate, iodine, or gaseous radioactivity in the main control room supply air (High-1)
- Initiate main control room ventilation isolation and actuate the main control room emergency habitability system in the event of abnormally high particulate or iodine radioactivity in the main control room supply air (High-2)
- Provide long-term post-accident monitoring (using both safety-related and nonsafety-related monitors)

The scope of the radiation monitoring system for post-accident monitoring is set forth in General Design Criterion 64 and in the provisions of Regulatory Guide 1.97.

#### 11.5.2.3.1 Fluid Process Monitors

##### Main Control Room Supply Air Duct Radiation Monitors

The main control room supply air duct radiation monitors (particulate detectors VBS-JE-RE001A and VBS-JE-RE001B, iodine detectors VBS-JE-RE002A and VBS-JE-RE002B, and noble gas detectors VBS-JE-RE003A and VBS-JE-RE003B) are offline monitors that continuously measure the concentration of radioactive materials in the air that is supplied to the main control room by the nuclear island nonradioactive ventilation system air handling units. The control support area ventilation is also part of this air supply system. The air supply is partially outside air. Refer to subsection 9.4.1 for system details. The main control room supply air duct radiation monitors receive safety-related power. When predetermined setpoints are exceeded, the monitors provide signals to initiate the supplemental air filtration system on ~~high-a~~ High-1 gaseous, particulate, or iodine concentration, and to isolate the main control room air intake and exhaust ducts and activate the main control room emergency habitability system on ~~high~~ High-2 particulate or iodine concentrations. Alarms are also provided in the main control room for these high concentrations.



### 12.2.1.3 Sources for the Core Melt Accident

#### 12.2.1.3.1 Containment

If there is core degradation, core cooling would be provided by the passive core cooling system which is totally inside the containment such that no high activity sump solution would be recirculated outside the containment. The shielding provided for the containment addresses this post-LOCA source term. The source strengths as a function of time are provided in Table 12.2-20 and the integrated source strengths are provided in Table 12.2-21.

#### 12.2.1.3.2 Main Control Room HVAC Filters

During operation of the nuclear island nonradioactive ventilation system (VBS) supplemental filtration or the main control room emergency habitability system (VES), filters in the control room HVAC work to remove particulate and iodine from the air. As radioactivity accumulates within the filters, this becomes a potential source of dose. These source strengths as a function of time are provided in Table 12.2-28 and the integrated source strengths are provided in Table 12.2-29.

### 12.3.2.2.7 Control Room Shielding Design

The design basis loss-of-coolant accident dictates the shielding requirements for the control room. The rod ejection accident dictates the shielding requirements for the main control room emergency habitability (VES) filter in the operator break room. Consideration is given to shielding provided by the shield building structure. Shielding combined with other engineered safety features is provided to permit access and occupancy of the control room following a postulated loss-of-coolant accident, so that radiation doses are limited to five rem whole body from contributing modes of exposure for the duration of the accident, in accordance with General Design Criterion 19.

Shielding of the VES filtration unit is accomplished by safety-related metal shielding. This shielding is composed of either tungsten that is 0.25 inches thick or stainless steel shown to provide an equivalent amount of shielding. The length and width of the shielding are designed to match the length and width of the filtration unit being shielded.



**TABLE 12.2-28 (SHEET 1 OF 2)**  
**CORE MELT ACCIDENT SOURCE STRENGTHS FROM MCR HVAC FILTERS AS A**  
**FUNCTION OF TIME**

**VES Filter <sup>(1)</sup> Source Strengths after a Loss of Coolant Accident**

Energy Group (Mev/gamma)	Source Strength (Mev/sec)			
	2 hours	8 hours	24 hours	30 days
0.01-0.02	1.19E+06	3.11E+06	1.81E+06	1.97E+05
0.02-0.03	1.47E+06	5.26E+06	3.89E+06	2.65E+05
0.03-0.06	2.87E+06	8.30E+06	5.46E+06	6.74E+05
0.06-0.1	3.03E+06	8.13E+06	5.22E+06	5.41E+05
0.1-0.2	5.76E+06	1.41E+07	8.76E+06	9.02E+05
0.2-0.4	6.14E+07	2.61E+08	2.46E+08	1.87E+07
0.4-0.6	1.86E+08	6.02E+08	3.60E+08	1.83E+07
0.6-0.7	1.47E+08	2.33E+08	1.47E+08	1.03E+08
0.7-0.8	1.09E+08	1.80E+08	1.05E+08	7.30E+07
0.8-1.0	1.85E+08	1.67E+08	6.99E+07	7.13E+06
1.0-1.5	3.36E+08	6.99E+08	1.85E+08	1.22E+07
1.5-2.0	1.21E+08	2.55E+08	4.97E+07	2.69E+04
2.0-3.0	3.13E+07	3.87E+07	7.28E+06	9.07E+03
3.0-4.0	3.68E+05	5.98E+03	5.56E+02	1.41E+02
4.0-5.0	1.42E+04	3.16E+01	8.55E-04	7.80E-04
5.0-6.0	3.31E-05	3.12E-04	3.35E-04	3.21E-04
6.0-7.0	1.32E-05	1.24E-04	1.33E-04	1.28E-04
7.0-8.0	5.11E-06	4.82E-05	5.17E-05	4.96E-05
8.0-10.0	2.68E-06	2.53E-05	2.71E-05	2.60E-05
10.0-14.0	1.69E-07	1.60E-06	1.71E-06	1.64E-06
Total	1.19E+09	2.47E+09	1.19E+09	2.35E+08

Notes: 1) Based upon a particulate filter density of 0.212 g/cc and charcoal filter density of 0.440 g/cc.



TABLE 12.2-28 (SHEET 2 OF 2)				
CORE MELT ACCIDENT SOURCE STRENGTHS FROM MCR HVAC FILTERS AS A FUNCTION OF TIME				
VBS Filter <sup>(2)</sup> Source Strengths after a Loss of Coolant Accident				
Energy Group (Mev/gamma)	Source Strength (Mev/sec)			
	2 hours	8 hours	24 hours	30 days
0.01-0.02	6.86E+08	1.00E+09	5.75E+08	6.21E+07
0.02-0.03	9.55E+08	1.76E+09	1.27E+09	8.46E+07
0.03-0.06	1.71E+09	2.71E+09	1.75E+09	2.10E+08
0.06-0.1	1.72E+09	2.60E+09	1.63E+09	1.70E+08
0.1-0.2	3.49E+09	4.61E+09	2.81E+09	2.91E+08
0.2-0.4	3.54E+10	8.45E+10	7.59E+10	5.76E+09
0.4-0.6	1.03E+11	1.91E+11	1.10E+11	5.61E+09
0.6-0.7	7.99E+10	7.20E+10	4.39E+10	3.04E+10
0.7-0.8	5.97E+10	5.62E+10	3.17E+10	2.16E+10
0.8-1.0	1.03E+11	5.23E+10	2.13E+10	2.11E+09
1.0-1.5	1.86E+11	2.20E+11	5.64E+10	3.62E+09
1.5-2.0	6.71E+10	8.03E+10	1.53E+10	8.78E+06
2.0-3.0	1.66E+10	1.22E+10	2.24E+09	3.09E+06
3.0-4.0	1.82E+08	1.93E+06	1.89E+05	4.81E+04
4.0-5.0	6.86E+06	7.65E+03	2.91E-01	2.65E-01
5.0-6.0	3.74E-02	1.12E-01	1.14E-01	1.09E-01
6.0-7.0	1.49E-02	4.47E-02	4.54E-02	4.35E-02
7.0-8.0	5.78E-03	1.74E-02	1.76E-02	1.69E-02
8.0-10.0	3.03E-03	9.11E-03	9.24E-03	8.86E-03
10.0-14.0	1.92E-04	5.75E-04	5.84E-04	5.60E-04
Total	6.59E+11	7.82E+11	3.65E+11	7.00E+10
Notes: 2) Based upon a particulate filter density of 0.230 g/cc and charcoal filter density of 0.632 g/cc.				



**TABLE 12.2-29**  
**CORE MELT ACCIDENT INTEGRATED SOURCE STRENGTHS FROM MCR HVAC**  
**FILTERS**

Energy Group (Mev/gamma)	30-Day Integrated Source Strength (Mev)	
	VES <sup>(1)</sup>	VBS <sup>(2)</sup>
0.01-0.02	1.75E+08	5.65E+10
0.02-0.03	3.81E+08	1.26E+11
0.03-0.06	5.89E+08	1.90E+11
0.06-0.1	5.77E+08	1.84E+11
0.1-0.2	9.03E+08	2.95E+11
0.2-0.4	3.34E+10	1.05E+13
0.4-0.6	2.36E+10	7.44E+12
0.6-0.7	3.81E+10	1.15E+13
0.7-0.8	2.63E+10	7.92E+12
0.8-1.0	7.57E+09	2.39E+12
1.0-1.5	1.77E+10	5.67E+12
1.5-2.0	4.03E+09	1.34E+12
2.0-3.0	6.47E+08	2.18E+11
3.0-4.0	1.20E+06	4.46E+08
4.0-5.0	4.17E+04	1.52E+07
5.0-6.0	1.03E-01	3.51E+01
6.0-7.0	4.08E-02	1.40E+01
7.0-8.0	1.59E-02	5.42E+00
8.0-10.0	8.32E-03	2.84E+00
10.0-14.0	5.25E-04	1.80E-01
Total	1.54E+11	4.79E+13

Notes: 1) Based upon a particulate filter density of 0.212 g/cc and charcoal filter density of 0.440 g/cc.

2) Based upon a particulate filter density of 0.230 g/cc and charcoal filter density of 0.632 g/cc.



From DCD Figure 12.3-1 Sheet 6

**NOTES:**

1. DURING SPENT RESIN WASTE DISPOSAL CONTAINER TRANSFER OR LOADING, THIS AREA CAN BE AS HIGH AS ZONE IX. THE CONTACT DOSE RATE OF THE SPENT RESIN CONTAINER IS >1000 REM/HR.
2. DURING CASK HANDLING OPERATIONS, AREAS OUTSIDE THE RAIL BAY DOORS CAN BE GREATER THAN THEN II LEVEL.
3. UNDERWATER SPENT FUEL ASSEMBLIES ARE AT MINIMUM 2'-6" FROM INSIDE SURFACE OF WEST WALL.
4. UNDERWATER SPENT FUEL ASSEMBLY IS AT MINIMUM 5'-0" FROM NORTH, SOUTH, EAST WALL.
5. ZONE IV WHEN FRESHLY DISCHARGED ASSEMBLIES ARE NOT IN THE OUTER ROW OF SPENT FUEL STORAGE RACK AND ZONE V WHEN FRESHLY DISCHARGED ASSEMBLIES ARE IN THE OUTER ROW OF SPENT FUEL STORAGE RACK
6. DURING UNDERWATER SPENT FUEL TRANSFER OPERATIONS, THIS AREA CAN BE AS HIGH AS ZONE IX.
7. DURING UNDERWATER REACTOR INTERNALS TRANSFER/STORAGE, THIS AREA CAN BE AS HIGH AS ZONE IX.
8. AREAS OUTSIDE SPENT FUEL POOL WALL CAN BE ZONE V WITH FRESHLY DISCHARGED ASSEMBLIES IN OUTER ROW OF SPENT FUEL STORAGE RACK.
9. BLOWDOWN PIPING MAY REACH ZONE III LEVELS WITH CONCURRENT FUEL CLADDING DEFECTS OF 0.25% AND STEAM GENERATOR TUBE LEAKAGE OF 500 gpd.
10. PASSIVE RHR PIPING MAY REACH ZON **300** DURING HEAT EXCHANGER TESTING.
11. THE PURPOSE OF THIS DRAWING IS FOR IDENTIFICATION OF RADIATION ZONES ONLY. BACKGROUND INFORMATION MAY CHANGE AND LEGIBILITY OF THE BACKGROUND IS NOT REQUIRED.



Table 14.3-7 (Sheet 2 of 3)		
RADIOLOGICAL ANALYSIS		
Reference	Design Feature	Value
Section 8.3.1.1.6	Electrical penetrations through the containment can withstand the maximum short-circuit currents available either continuously without exceeding their thermal limit, or at least longer than the field cables of the circuits so that the fault or overload currents are interrupted by the protective devices prior to a potential failure of a penetration.	
Section 9.4.1.1.1	The VBS isolates the HVAC ductwork that penetrates the main control room boundary on <del>high</del> High-2 particulate or iodine concentrations in the main control room supply air or on extended loss of ac power to support operation of the main control room emergency habitability system.	
Section 12.3.2.2.1	During reactor operation, the shield building protects personnel occupying adjacent plant structures and yard areas from radiation originating in the reactor vessel and primary loop components. The concrete shield building wall and the reactor vessel and steam generator compartment shield walls reduce radiation levels outside the shield building to less than 0.25 mrem/hr from sources inside containment. The shield building completely surrounds the reactor components.	
Section 12.3.2.2.2	The reactor vessel is shielded by the concrete primary shield and by the concrete secondary shield which also surrounds other primary loop components. The secondary shield is a structural module filled with concrete surrounding the reactor coolant system equipment, including piping, pumps and steam generators. Extensive shielding is provided for areas surrounding the refueling cavity and the fuel transfer canal to limit the radiation levels.	
Section 12.3.2.2.3	Shielding is provided for the liquid radwaste, gaseous radwaste and spent resin handling systems consistent with the maximum postulated activity. Corridors are generally shielded to allow Zone II access, and operator areas for valve modules are generally Zone II or III for access. Shielding is provided to attenuate radiation from normal residual heat removal equipment during shutdown cooling operations to levels consistent with radiation zoning requirements of adjacent areas.	



**15.0.11.1 FACTRAN Computer Code**

FACTRAN (Reference 5) calculates the transient temperature distribution in a cross section of a metal-clad  $\text{UO}_2$  fuel rod and the transient heat flux at the surface of the cladding using as input the nuclear power and the time-dependent coolant parameters (pressure, flow, temperature, and density). The code uses a fuel model which simultaneously exhibits the following features:

- A sufficiently large number of radial space increments to handle fast transients ~~such as rod ejection accidents~~

**15.0.11.6 ANC Computer Code**

The ANC -computer code is used to solve the two-group neutron diffusion equation in three spatial dimensions. -ANC can also solve the three-dimensional kinetics equations for six delayed -neutron groups.



Table 15.0-2 (Sheet 4 of 5)

**SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED**

Section	Faults	Computer Codes Used	Reactivity Coefficients Assumed			Initial Thermal Power Output Assumed (MWt)
			Moderator Density ( $\Delta k/\text{gm}/\text{cm}^3$ )	Moderator Temperature (pcm/ $^{\circ}\text{F}$ )	Doppler	
15.4	Spectrum of RCCA ejection accidents	<del>TWINKLE,</del> <del>FACTRAN</del> ANC, VIPRE	Refer to subsection 15.4.8	Refer to subsection 15.4.8	<del>Coefficient consistent with a Doppler defect of -0.90% <math>\Delta K</math> at BOC<sup>(b)</sup> and -0.87% <math>\Delta K</math> at EOC<sup>(b)</sup></del> Refer to subsection 15.4.8	<del>0 and 3483.3 (a)</del> Refer to subsection 15.4.8



#### 15.1.5.4.1 Source Term

The only significant radionuclide releases due to the main steam line break are the iodines and alkali metals that become airborne and are released to the environment as a result of the accident. Noble gases are also released to the environment. Their impact is secondary because any noble gases entering the secondary side during normal operation are rapidly released to the environment.

The analysis considers two different reactor coolant iodine source terms, both of which consider the iodine spiking phenomenon. In one case, the initial iodine concentrations are assumed to be those associated with equilibrium operating limits for primary coolant iodine activity. The iodine spike is assumed to be initiated by the accident with the spike causing an increasing level of iodine in the reactor coolant.

The second case assumes that the iodine spike occurs prior to the accident and that the maximum resulting reactor coolant iodine concentration exists at the time the accident occurs.

The reactor coolant noble gas ~~and~~ concentrations are assumed to be those associated with equilibrium operating limits for primary coolant noble gas activity. The reactor coolant alkali metal concentrations are ~~assumed to be based on~~ those associated with the design basis fuel defect level.

The secondary coolant is assumed to have an iodine source term of ~~0-10.01~~  $\mu\text{Ci/g}$  dose equivalent I-131. This is ~~+0-1~~ percent of the maximum primary coolant activity at equilibrium operating conditions. The secondary coolant alkali metal concentration is also assumed to be ~~+0-1~~ percent of the primary concentration.



#### 15.1.5.4.6 Doses

Using the assumptions from Table 15.1.5-1, the calculated total effective dose equivalent (TEDE) doses for the case with accident-initiated iodine spike are determined to be less than 0.6 rem at the site boundary for the limiting 2-hour interval (~~0-4.8~~ to ~~2-6.8~~ hours) and 1.1 rem at the low population zone outer boundary. These doses are small fractions of the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. A "small fraction" is defined, consistent with the Standard Review Plan, as being 10 percent or less. The TEDE doses for the case with pre-existing iodine spike are determined to be less than 0.5 rem at the site boundary for the limiting 2-hour interval (0 to 2 hours) and 0.4 rem at the low population zone outer boundary. These doses are within the dose guidelines of 10 CFR Part 50.34.

At the time the main steam line break occurs, the potential exists for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. The 30-day contribution to the dose at ~~the~~ **site boundary** and the low population zone boundary is less than 0.01 rem TEDE. When this is added to the dose calculated for the main steam line break, the resulting total dose remains less than the values reported above.



Table 15.1.5-1

**PARAMETERS USED IN EVALUATING THE RADIOLOGICAL  
CONSEQUENCES OF A MAIN STEAM LINE BREAK**

Reactor coolant iodine activity	
– Accident-initiated spike	Initial activity equal to the equilibrium operating limit for reactor coolant activity of 1.0 $\mu\text{Ci/g}$ dose equivalent I-131 with an assumed iodine spike that increases the rate of iodine release from fuel into the coolant by a factor of 500 (see Appendix 15A). Duration of spike is <del>3.6</del> 5 hours.
– Preaccident spike	An assumed iodine spike that has resulted in an increase in the reactor coolant activity to 60 $\mu\text{Ci/g}$ of dose equivalent I-131 (see Appendix 15A)
Reactor coolant noble gas activity	Equal to the operating limit for reactor coolant activity of 280 $\mu\text{Ci/g}$ dose equivalent Xe-133
Reactor coolant alkali metal activity	Design basis activity (see Table 11.1-2)
Secondary coolant initial iodine and alkali metal activity	<del>40</del> 1% of reactor coolant concentrations at maximum equilibrium conditions
Duration of accident (hr)	72
Atmospheric dispersion ( $\chi/Q$ ) factors	See Table 15A-5 in Appendix 15A
Steam generator in faulted loop	
– Initial water mass (lb)	<del>3.0</del> 3.32 E+05
– Primary to secondary leak rate (lb/hr)	<del>52.14</del> 52.25 <sup>(a)</sup>
– Iodine partition coefficient	1.0
– Steam released (lb)	
0 - 2 hr	<del>3.0</del> 3.321E+05
2 - 72 hr	<del>3.6</del> 3.66 E+03
Steam generator in intact loop	
– Primary to secondary leak rate (lb/hr)	<del>52.14</del> 52.25 <sup>(a)</sup>
– Iodine partition coefficient	1.0
– Steam released (lb)	
0 - 2 hr	<del>3.0</del> 3.321E+05
2 - 72 hr	<del>3.6</del> 3.66 E+03
Nuclide data	See Table 15A-4

**Note:**

a. Equivalent to 150 gpd cooled liquid at 62.4 lb/ft<sup>3</sup>.



**15.3.3.3.1 Source Term**

The initial secondary coolant activity is assumed to be ~~10~~ percent of the maximum equilibrium primary coolant activity for iodines and alkali metals.

Table 15.3-3 (Sheet 1 of 2)	
<b>PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LOCKED ROTOR ACCIDENT</b>	
Initial reactor coolant iodine activity	An assumed iodine spike that has resulted in an increase in the reactor coolant activity to 60 $\mu\text{Ci/gm}$ of dose equivalent I-131 (see Appendix 15A) <sup>(a)</sup>
Reactor coolant noble gas activity	Equal to the operating limit for reactor coolant activity of 280 $\mu\text{Ci/gm}$ dose equivalent Xe-133
Reactor coolant alkali metal activity	Design basis activity (see Table 11.1-2)
Secondary coolant initial iodine and alkali metal activity	<del>10</del> % of design basis reactor coolant concentrations at maximum equilibrium conditions
Fraction of fuel rods assumed to fail	0.10
Core activity	See Table 15A-3
Radial peaking factor (for determination of activity in failed fuel rods)	<del>1.65</del> 75
Fission product gap fractions	
I-131	0.08
Kr-85	0.10
Other iodines and noble gases	0.05
Alkali metals	0.12
Reactor coolant mass (lb)	3.7 E+05
Secondary coolant mass (lb)	<del>6.06-04</del> E+05
Condenser	Not available
Atmospheric dispersion factors	See Table 15A-5
Primary to secondary leak rate (lb/hr)	104. <del>53</del> <sup>(b)</sup>
Partition coefficient in steam generators	
iodine	0.01
alkali metals	<del>0.004</del> 0035



Table 15.3-3 (Sheet 1 of 2)

**PARAMETERS USED IN EVALUATING THE RADIOLOGICAL  
CONSEQUENCES OF A LOCKED ROTOR ACCIDENT**

Accident scenario in which startup feedwater is not available	
Duration of accident (hr)	1.5 hr
Steam released (lb)	
0-1.5 hours(c)	6.48 E+05
Leak flashing fraction(d)	
0-60 minutes	0.04
> 60 minutes	0