Full Scope Implementation of the Alternative Source Term
(661 pages)

Revision 0

August 2013

Full Scope Implementation of Alternative Source Term

Westinghouse

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PROFESSIONAL ENGINEERING STAMPS

This section contains the State of Kansas Professional Engineer certifications for each of the sections pertaining to technical services scope supporting the Wolf Creek Generating Station plant design or design configuration. Each Professional Engineer has designated applicable scope sections for which they provided Practice of Engineering oversight and for which their certification applies.

I, the undersigned, being a registered Professional Engineer, certify that to the best of my knowledge and belief the results herein do not jeopardize the protection of life, health, property, and welfare of the public.

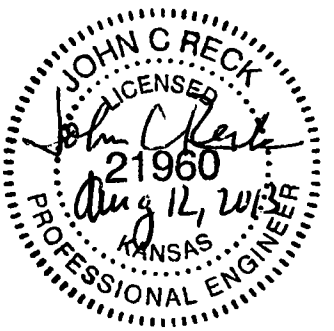
Sections being Certified:

Section 4.1 (excludes the meteorological data that was prepared and provided by WCNOC as input for the atmospheric dispersion calculations)

Section 4.3 (excludes the Current Licensing Basis (CLB) data that was input provided by WCNOC for information only)

Certified By: John C Reck, P.E.

License Number: 21960 State: KS Expiration Date: 2015-04-30

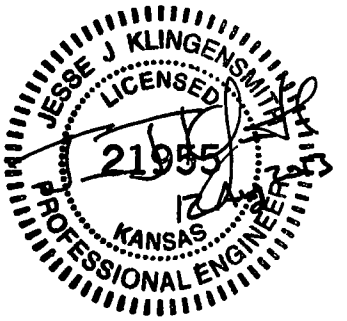


I, the undersigned, being a registered Professional Engineer, certify that to the best of my knowledge and belief the results herein do not jeopardize the protection of life, health, property, and welfare of the public.

Section being Certified: Section 4.2

Certified By: Jesse J Klingensmith, P.E.

License Number: 21955 State: KS Expiration Date: April 30, 2014



I, the undersigned, being a registered Professional Engineer, certify that to the best of my knowledge and belief the results herein do not jeopardize the protection of life, health, property, and welfare of the public.

Sections being Certified: Section 4.4.2, Section 4.4.3

Certified By: Joseph C Adams, P.E.

License Number: 21424 State: KS Expiration Date: 04/30/2014

JOSEPH C ADAMS
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21424
8/12/2013
KANSAS
PROFESSIONAL ENGINEER

1 DESCRIPTION

Pursuant to 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Wolf Creek Nuclear Operating Corporation (WCNOC) hereby requests an amendment to Renewed Facility Operating License No. NPF-42 for the Wolf Creek Generating Station (WCGS). The proposed amendment would include the full scope implementation of the Alternative Source Term (AST) as described in Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Revision 0. In addition to the full scope implementation of the AST, WCNOC is proposing the adoption of Technical Specification Task Force (TSTF)-51-A, Revision 2, "Revise Containment Requirements during Handling Irradiated Fuel and Core Alterations."

In accordance with 10 CFR 50.67, "Accident source term," a licensee may voluntarily revise the accident source term used in design basis radiological consequence analyses. Paragraph 50.67(b) requires that applications under this section contain an evaluation of the consequences of applicable design basis accidents (DBAs) previously analyzed in the plant safety analysis report. RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" provides guidance to licensees on performing evaluations, and re-analyses as required to adopt an AST.

WCNOC proposes the following exceptions to the full scope implementation of the AST methodology: The current Technical Information Document (TID)-14844 accident source term will remain the licensing basis for equipment qualification and NUREG-0737 evaluations other than Control Room Habitability Envelope (CRHE) and Technical Support Center (TSC) doses. This exception is consistent with the guidance provided in RG 1.183, Section 1.3.5, "Equipment Environmental Qualification" and Section 6, "Assumptions for Evaluating the Radiation Doses for Equipment Qualification," which states in part:

> "The NRC staff is assessing the effect of increased cesium releases on EQ doses to determine whether licensee action is warranted. Until such time as this generic issue is resolved, licensees may use either the AST or the TID14844 assumptions for performing the required EQ analyses."

Westinghouse has performed radiological consequence analyses of the applicable pressurized water reactor (PWR) DBAs identified in RG 1.183 for WCGS as well as radiological consequences analyses for the following accidents:

- Letdown line break,
- Loss of AC power, and
- Various tank ruptures.

These analyses are discussed in detail in Section 4.3 of this Enclosure. The analyses were performed using the guidance of RG 1.183, RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants."

A comparison to the guidance contained in the RGs discussed above, is provided in the following Sections:

- A comparison to RG 1.183 is contained in Section 5 of this Enclosure,
- A comparison to RG 1.145 is contained in Section 6 of this Enclosure, and
- A comparison to RG 1.194 is contained in Section 7 of this Enclosure.

In addition, a comparison with the guidance contained in Regulatory Issue Summary 2006-04, "Experience with Implementation of Alternative Source Terms," is provided in Section 8 of this Enclosure.

The proposed full scope implementation of the AST analyses will modify the WCGS licensing bases by adopting the AST methodology to replace the current accident source term with the AST as specified in 10 CFR 50.67. Implementation of the AST establishes the 10 CFR 50.67 total effective dose equivalent (TEDE) dose limits as the new acceptance criterion. The AST is characterized by the composition and magnitude of the radioactive material, the chemical and physical form of the radionuclides, and the timing of the releases of these radionuclides.

Implementation of the AST revises the accident source term used in the design basis radiological analyses. The use of the AST methodology results in changes in the DBA radiological consequences; however, the AST methodology has no direct impact on the probability or initiation of the evaluated design basis accidents. Application of AST methodology and the other changes proposed in this license amendment request (LAR) do not increase the core damage frequency or the large early release frequency. Therefore, this request for a revision to the WCGS licensing basis is not being submitted as a "risk-informed approach" using the guidelines in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Licensing Basis."

This LAR includes the following changes to the WCGS licensing basis:

- The CRHE unfiltered inleakage is revised from 20 scfm to 50 scfm.

- The Control Building unfiltered inleakage is revised from 300 scfm to 400 scfm.

- Revise the USAR Chapter 15 dose analyses in accordance with the guidance in Regulatory Guide 1.183.

- Revise the Technical Specification (TS) to address the update of the accident source term and associated design basis accidents utilizing the guidance provided in Regulatory Guide 1.183 and the associated control room dose limits of General Design Criterion (GDC) 19, and offsite dose limits of 10 CFR 50.67.

- Revise the TS to address the adoption of TSTF-51-A, Revision 2, which allows the elimination of the Technical Specification requirements for certain Engineered Safety Feature (ESF) systems to be OPERABLE, after a sufficient radioactive decay has occurred to ensure that the control room and offsite doses remain below the 10 CFR 50.67 limits.

The proposed TS, TS Bases, and Technical Requirements Manual (TRM) changes, that are associated with the implementation the AST analyses, and TSTF-51-A, are provided in Sections 9, 11, and 12 of this Enclosure, respectively. The clean typed TS pages are provided in Section 10 of this Enclosure. The proposed USAR changes are provided in Section 13 of this Enclosure.

The Regulatory Evaluation for the implementation of the AST (including the No Significant Hazards Consideration) is provided in Attachment 1, Section 4 of this LAR.

The Environmental Consideration for the implementation of the AST is provided in Attachment 1, Section 5 of this LAR.

2 PROPOSED CHANGES

CRHE AND CONTROL BUILDING UNFILTERED INLEAKAGE

The proposed change to the CRHE unfiltered inleakage is from 20 scfm in the current dose consequence analyses, to 50 scfm in the AST analyses.

The proposed change to the Control Building unfiltered inleakage is from 300 scfm in the current dose consequence analyses, to 400 scfm in the AST analyses.

Justification for the Change:

The unfiltered inleakages to the control room and control building that are assumed in each of the events listed below are 50 cfm and 400 cfm, respectively.

These assumptions are discussed in Section 4.3.2.1, and identified in Table 4.3-5 of this Enclosure. The analysis of each of these events demonstrates that the applicable regulatory dose limits in 10 CFR 50.67 are met.

- Main Steamline Break (MSLB) (Section 4.3.3 of Enclosure VI)
- Loss of Non-Emergency AC Power (LOAC) (Section 4.3.4 of Enclosure VI)
- Locked Rotor (Section 4.3.5 of Enclosure VI)
- Rod Ejection (Section 4.3.6 of Enclosure VI)
- Letdown Line Break (Section 4.3.7 of Enclosure VI)
- Steam Generator Tube Rupture (SGTR) (Section 4.3.8 of Enclosure VI)
- Loss-of-Coolant Accident (LOCA) (Section 4.3.9 of Enclosure VI)
- Waste Gas Decay Tank Failure (Section 4.3.10 of Enclosure VI)
- Liquid Waste Tank Failure (Section 4.3.11 of Enclosure VI)
- Fuel Handling Accident (FHA) (Section 4.3.12 of Enclosure VI)

AST ANALYSES

The AST analyses performed in accordance with the guidance in Regulatory Guide 1.183 for the USAR Chapter 15 dose consequence analyses events are described in detail in Section 4.3 of this Enclosure.

AST TS CHANGES

1. DOSE EQUIVALENT I-131 Definition

The current WCGS TS definition of DOSE EQUIVALENT I-131 states:

> "DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using thyroid dose conversion factors from:
>
> 1) Table III of TID-14844, AEC, 1962, Calculation of Distance Factors for Power and Test Reactor Sites," or
>
> 2) Table E-7 of Regulatory Guide 1.109, Revision 1, NRC, 1977, or
>
> 3) ICRP 30, 1979, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity," or
>
> 4) Table 2.1 of EPA Federal Guidance Report No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

The proposed change would revise the WCGS TS DOSE EQUIVALENT I-131 Definition as follows:

> "DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using thyroid dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

The Definition of DOSE EQUIVALENT I-131 would be revised to delete References 1), 2), and 3) for the thyroid dose conversion factors. The number of current Reference 4) was deleted, since it is the only reference left.

Justification for the Change:

NRC Regulatory Issue Summary 2006-04, "Experience with Implementation of Alternative Source Terms," Summary of Issue 10, "Definition of Dose Equivalent ^{131}I" states: "...Although different references are available for dose conversion factors, the TS definition should be based on the same dose conversion factors that are used in the determination of the reactor coolant dose equivalent iodine curie content for the main steamline break and steam generator tube rupture accident analyses."

Therefore the Definition was revised to reflect the reactor coolant system (RCS) dose equivalent iodine curie content for all dose analyses modeling initial RCS activity.

2. DOSE EQUIVALENT XE-133 Definition

The current WCGS TS definition of DOSE EQUIVALENT XE-133 states:

> "DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-87, Kr-88, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using the effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil," or using the dose conversion factors from Table B-1 of Regulatory Guide 1.109, Revision 1, NRC, 1977."

The proposed change would revise the WCGS TS DOSE EQUIVALENT XE-133 Definition as follows:

> "DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-87, Kr-88, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using the effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."

The Definition of DOSE EQUIVALENT XE-133 would be revised to delete the text: "or using the dose conversion factors from Table B-1 of Regulatory Guide 1.109, Revision 1, NRC, 1977."

Justification for the Change:

While not explicitly discussed in NRC Regulatory Issue Summary 2006-04, this change is consistent with the intent of Summary of Issue 10, "Definition of Dose Equivalent ^{131}I" which states: "...Although different references are available for dose conversion factors, the TS definition should be based on the same dose conversion factors that are used in the determination of the reactor coolant dose equivalent iodine curie content for the main steamline break and steam generator tube rupture accident analyses."

Therefore the Definition was revised to reflect the RCS dose equivalent noble gas curie content for all dose analyses modeling initial RCS activity.

3. TS 5.5.12, "Explosive Gas and Storage Tank Radioactivity Monitoring Program."

This WCGS program provides controls for potentially explosive gas mixtures contained in the Waste Gas Holdup System, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The proposed change would affect Section 5.5.12.b which specifies one of the items that must be included in the Program.

The current WCGS TS 5.5.12.b states:

> "b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and"

The proposed change would revise the WCGS TS 5.5.12.b as follows:

> "b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of ≥ 0.1 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and"

The whole body exposure limit of ≥ 0.5 rem to any individual in an unrestricted area, in Specification 5.5.12, "Explosive Gas and Storage Tank Radioactivity Monitoring Program," item b., was revised to ≥ 0.1 rem.

Justification for the Change:

NRC Regulatory Issue Summary 2006-04, "Experience with Implementation of Alternative Source Terms," Summary of Issue 11, "Acceptance Criteria for Off-Gas or Waste Gas System Release," states: "...When the NRC revised 10 CFR Part 20 to incorporate a TEDE dose, the offsite dose to an individual member of the public was changed from 500 mrem whole body to 100 mrem TEDE. Therefore, any licensee who chooses to implement AST for an off-gas or waste gas system release should base its acceptance criteria on 100 mrem TEDE..."

Therefore, the whole body exposure limit of ≥ 0.5 rem to any individual in an unrestricted area, in Specification 5.5.12, "Explosive Gas and Storage Tank Radioactivity Monitoring Program," item b., was revised to reflect the 0.1 rem limit based on the guidance in Regulatory Issue Summary 2006-04.

TSTF-51-A TS CHANGES

The proposed amendment would revise the TS by the adoption of TSTF-51-A, Revision 2, "Revise Containment Requirements during Handling Irradiated Fuel and Core Alterations." TSTF-51-A contains changes that would eliminate the TS requirements for certain Engineered Safety Feature (ESF) systems (e.g., control room emergency ventilation and containment isolation capability) to be OPERABLE after a

sufficient radioactive decay has occurred, to ensure that the control room and offsite doses remain below the applicable regulatory limits. This change also eliminates the OPERABILITY requirements of the affected ESF systems that are required during the Applicability of "CORE ALTERATIONS," for the affected TS. This change would allow the flexibility to move personnel and equipment and perform work which would affect containment OPERABILITY during the handling of irradiated fuel.

TSTF-51-A accomplishes these changes by adding the word "recently" prior to "irradiated fuel assemblies" in the Applicability and Actions of the affected TS. TSTF-51-A also modifies some of the TS Bases to discuss "recently" irradiated fuel assemblies. Consistent with the Bases changes identified in TSTF-51-A, the WCGS Bases discussion of recently irradiated would read as follows:

> "Fuel that has occupied part of a critical reactor core within the previous 76 hours."

The WCGS specific decay time of 76 hours is based on the analysis that demonstrates that the control room and offsite doses resulting from a fuel handling accident (involving fuel removed from a critical core for more than 76 hours) would be within the applicable regulatory limits without the use of these ESF systems to mitigate the accident. As such, the TS (i.e., the Applicability and the Actions) would be modified (i.e., limited) to be applicable when moving "recently" irradiated fuel assemblies, instead of any "irradiated fuel assemblies."

The WCGS specific decay time of 76 hours was used in the fuel handling accident analysis described in Section 4.3.12 of this Enclosure.

Consistent with TSTF-51-A the following WCGS TS changes are proposed:

- 3.3.6, "Containment Purge Isolation Instrumentation,"
  - The Note stating the applicability of Actions Condition B is revised to delete CORE ALTERATIONS and revise the movement of irradiated fuel assemblies to the movement of "recently" irradiated fuel assemblies.
  - The Applicability footnotes on Table 3.3.6-1 are revised to delete the CORE ALTERATIONS footnote and modify the footnote with movement of irradiated fuel assemblies to movement of "recently" irradiated fuel assemblies. The footnotes are re-lettered as necessary due to the deletion of the CORE ALTERATION footnote.
- 3.3.7, "Control Room Emergency Ventilation System (CREVS) Actuation Instrumentation,"
  - Actions Condition E is revised to refer to the movement of "recently" irradiated fuel assemblies.
  - Required Action E.1, "Suspend CORE ALTERATIONS," is deleted.

- Required Action E.2 is revised to suspend movement of "recently" irradiated fuel assemblies and re-numbered to E.1.

- Applicability Footnote (a) on Table 3.3.7-1 is revised to refer to the movement of "recently" irradiated fuel assemblies.

- 3.3.8, "Emergency Exhaust System (EES) Actuation Instrumentation,"

  - Actions Condition D and Required Action D.1 are revised to refer to the movement of "recently" irradiated fuel assemblies.

  - Applicability Footnote (a) on Table 3.3.8-1 is revised to refer to the movement of "recently" irradiated fuel assemblies.

- 3.7.10, "Control Room Emergency Ventilation System (CREVS),"

  - The Applicability is revised to during movement of "recently" irradiated fuel assemblies.

  - Actions Condition D is revised to refer to the movement of "recently" irradiated fuel assemblies.

  - Required Action D.2.1, "Suspend CORE ALTERATIONS," and the following logical connector AND are deleted.

  - Required Action D.2.2 is re-numbered to D.2 and revised to refer to the movement of "recently" irradiated fuel assemblies.

  - Actions Condition E is revised to refer to the movement of "recently" irradiated fuel assemblies.

  - Required Action E.1, "Suspend CORE ALTERATIONS," and the following logical connector AND are deleted.

  - Required Action E.2 is re-numbered to E.1 and revised to refer to the movement of "recently" irradiated fuel assemblies.

- 3.7.11, "Control Room Air Conditioning System (CRACS),"

  - The Applicability is revised to during movement of "recently" irradiated fuel assemblies.

  - Actions Condition C is revised to refer to the movement of "recently" irradiated fuel assemblies.

  - Required Action C.2.1, "Suspend CORE ALTERATIONS," and the following logical connector AND are deleted.

- Required Action C.2.2 is re-numbered to C.2 and revised to refer to the movement of "recently" irradiated fuel assemblies.

- Actions Condition D is revised to refer to the movement of "recently" irradiated fuel assemblies.

- Required Action D.1, "Suspend CORE ALTERATIONS," and the following logical connector AND are deleted.

- Required Action D.2 is re-numbered to D.1 and revised to refer to the movement of "recently" irradiated fuel assemblies.

- 3.7.13, "Emergency Exhaust System (EES),"

  - The Applicability and the Note following the Applicability are revised to during movement of "recently" irradiated fuel assemblies.

  - Actions Conditions D, E and F are revised to refer to the movement of "recently" irradiated fuel assemblies.

  - Required Actions D.2 and F.1 are revised to refer to the movement of "recently" irradiated fuel assemblies.

- 3.9.4, "Containment Penetrations,"

  - The Applicability is revised to delete CORE ALTERATIONS and revise the movement of irradiated fuel assemblies within containment to the movement of "recently" irradiated fuel assemblies within containment.

  - Required Action A.1, "Suspend CORE ALTERATIONS," and the following logical connector AND are deleted.

  - Required Action A.2 is re-numbered to A.1 and revised to refer to the movement of "recently" irradiated fuel assemblies.

TSTF-51-A also includes similar TS changes (as those described above) to the following TS:

- 3.8.2, "AC Sources – Shutdown,"
- 3.8.5, "DC Sources – Shutdown,"
- 3.8.8, "Inverters – Shutdown,"
- 3.8.10, "Distribution Systems – Shutdown," and
- 3.9.7, "Refueling Cavity Water Level."

The TSTF-51-A changes proposed by WCNOC do not include changes to the TS discussed above, for the reasons discussed below.

The TSTF-51-A change to TS 3.9.7, "Refueling Cavity Water Level," deletes the Applicability of CORE ALTERATIONS. The corresponding WCGS TS 3.9.7, "Refueling Pool Water Level," which applies to the refueling cavity water level, does not have the Applicability of CORE ALTERATIONS. Therefore, the TSTF-51-A change does not apply to this WCGS TS.

Regarding the four electrical TS listed above, the primary reason for implementing TSTF-51-A, as stated in the TSTF, is to allow the flexibility to move personnel and equipment and perform work that would affect containment OPERABILITY during the handling of irradiated fuel. The TSTF changes to the electrical TS listed above do not impact the movement of personnel and equipment in and out of containment. In addition, WCNOC chose to conservatively maintain the TS Applicability of electrical system TS requirements whenever irradiated fuel assemblies (independent of whether they are "recently irradiated," or not) are moved. Considering that the deviation from the electrical TS changes in TSTF-51-A results in more conservative WCGS TS, WCNOC finds that this deviation from the TSTF is acceptable, and will continue to ensure the affected electrical system requirements remain applicable, whenever any irradiated fuel is moved.

Justification for the Change:

Following reactor shutdown, the decay of the short-lived fission products greatly reduces the fission product inventory present in irradiated fuel. The proposed changes are based on performing analyses assuming a decay time that takes advantage of the reduced radionuclide inventory available for release in the event of a fuel handling accident. Following a sufficient decay time, the primary success path for mitigating the fuel handling accident no longer includes the functioning of the active ESF systems and verification of containment penetration status as required by the TS addressed by this proposed change. Therefore, the OPERABILITY requirements of the TS are modified to reflect that after the stated decay time, the water level ($\geq$ 23 ft) and decay time are the primary success path for mitigating a fuel handling accident. As such, after the decay time of 76 hours, the ESF systems and containment penetration status requirements addressed in TSTF-51-A are not required to ensure the resulting control room and offsite doses resulting from a fuel handling accident are within the applicable regulatory limits.

The WCGS fuel handling accident analyses were performed assuming a decay time of 76 hours. The results of these analyses demonstrate that the potential offsite and control room doses from a fuel handling accident (inside containment or in the fuel handling building) after a decay time of 76 hours has elapsed are well within the applicable regulatory limits in 10 CFR 50.67 for offsite locations and within the applicable limits in GDC 19 for the control room by maintaining the TS water level limits of $\geq$ 23 ft, as discussed below.

Based on the considerations discussed above regarding the fuel handling accident analyses, decay time, and the following TS water level requirements of:

> TS 3.7.15, "Fuel Storage Pool Water Level," requires that "The fuel storage pool water level shall be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks." TS 3.7.15 is applicable "During movement of irradiated fuel assemblies in the fuel storage pool," and TS 3.9.7, "Refueling Pool Water Level," requires that the "Refueling pool water level shall be maintained ≥ 23 ft above the top of reactor vessel flange." TS 3.9.7 is applicable "During movement of irradiated fuel assemblies within containment."

It can be concluded that the WCGS TS water level requirements of ≥ 23 ft continue to provide adequate assurance that the potential control room and offsite doses from a fuel handling accident (that occurs after 76 hours) is maintained within the applicable regulatory limits.

The WCGS fuel handling accident analyses were performed assuming a decay time of 76 hours. The results of these analyses demonstrate that the potential exclusion area boundary (EAB), low population zone (LPZ) and control room doses from a fuel handling accident (inside containment or in the fuel handling building) after a decay time of 76 hours has elapsed are well within the regulatory limits in 10 CFR 50.67 for the EAB and LPZ and within the applicable limit in 10 CFR 50.67 for the control room.

The fuel handling accident analyses inside containment or inside the fuel handling building assume that the entire fission product inventory of 1 fuel assembly plus 20% of an adjacent assembly is released to the atmosphere within a 2-hour period, after a decay time of 76 hours has elapsed and do not take credit for the following WCGS TS systems:

- 3.3.6, "Containment Purge Isolation Instrumentation,"
- 3.3.7, "Control Room Emergency Ventilation System (CREVS) Actuation Instrumentation,"
- 3.3.8, "Emergency Exhaust System (EES) Actuation Instrumentation,"
- 3.7.10, "Control Room Emergency Ventilation System (CREVS),"
- 3.7.11, "Control Room Air Conditioning System (CRACS),"
- 3.7.13, "Emergency Exhaust System (EES)," and
- 3.9.4, "Containment Penetrations."

Regarding fuel handling of recently irradiated fuel (i.e., "fuel that has occupied part of a critical reactor core within the previous 76 hours") the OPERABILITY requirements for the ESF systems and containment penetration requirements ensure these systems are available and that the containment penetration requirements are met. Therefore, the TS will continue to provide adequate assurance the potential dose from a fuel handling accident involving recently irradiated fuel assemblies will be maintained within the required limits in the same manner as before.

The proposed change also includes the deletion of the OPERABILITY requirements during CORE ALTERATIONS for the ESF systems addressed in TSTF-51-A. The events that could occur during CORE ALTERATIONS, other than a fuel handling accident, would not result in fuel cladding integrity damage. Since the only accident postulated to occur during CORE ALTERATIONS that results in a significant radioactive release is the fuel handling accident, the proposed changes to the TS requirements deleting CORE ALTERATIONS is justified.

TSTF-51 requires licensees adding the term "recently" must make the following commitment which is consistent with NUMARC 93-01, draft Revision 3, Section 11.2.6, "Safety Assessment for Removal of Equipment from Service During Shutdown Conditions," subheading "Containment – Primary (PWR)/ Secondary (BWR)."

"The following guidelines are included in the assessment of systems removed from service during movement irradiated fuel:

- During fuel handling/core alterations, ventilation system and radiation monitor availability (as defined in NUMARC 91-06) should be assessed, with respect to filtration and monitoring of releases from the fuel. Following shutdown, radioactivity in the fuel decays away fairly rapidly. The basis of the Technical Specification OPERABILITY amendment is the reduction in doses due to such decay. The goal of maintaining ventilation system and radiation monitor availability is to reduce doses even further below that provided by the natural decay.

- A single normal or contingency method to promptly close primary or secondary containment penetrations should be developed. Such prompt methods need not completely block the penetration or be capable of resisting pressure."

In WCNOC letter ET 01-0021, "Revision to Technical Specification 3.9.4, "Containment Penetrations"," dated August 7, 2001, WCNOC provided specific details regarding how the guidance in NUMARC 93-01, Section 11.3.6, "Assessment Methods for Shutdown Conditions, sub-section 11.3.6.5, "Containment – Primary (PWR)/Secondary(BWR)," is being met. This is consistent with the requested commitment in TSTF-51. The NRC approved the proposed change in Amendment No. 146 (July 30, 2012). Based on this information, WCNOC considers the guidance in NUMARC 93-01 is being satisfied.

3 BACKGROUND

The current WCGS licensing basis for DBA analysis source terms is U.S. Atomic Energy Commission TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," dated March 23, 1962. This is consistent with 10 CFR Part 100, Section 11 (10 CFR 100.11), "Determination of exclusion area, low population zone, and population center distance," for reactor siting, which contains offsite dose limits in terms of whole body and thyroid dose and further makes reference to TID-14844.

In December 1999, the Nuclear Regulatory Commission (NRC) issued 10 CFR 50.67, "Accident source term," which provides a mechanism for licensed power reactors to replace the traditional accident source term used in their DBA analyses with an AST. Regulatory guidance for the implementation of these ASTs is provided in RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." 10 CFR 50.67 requires a licensee seeking to use an AST to apply for a license amendment and requires that the application contain an evaluation of the consequences of affected DBAs.

Regulatory Guide (RG) 1.183 and Standard Review Plan Section (SRP) 15.0.1 were used by Westinghouse (for WCNOC) in preparing the AST analyses. These documents were prepared by the NRC staff to address the use of ASTs at current operating power reactors. The RG establishes the parameters of an acceptable AST and identifies the significant attributes of an AST acceptable to the NRC staff. In this regard, the RG provides guidance to licensees for operating power reactors on acceptable applications for an AST; the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on risk; and acceptable radiological analysis assumptions. The SRP provides guidance to the staff on the review of AST submittals.

Acceptance criteria consistent with that required by 10 CFR 50.67 were used to replace WCGS current design basis source term acceptance criteria. As part of the implementation of the AST, the TEDE acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11 and 10 CFR Part 50, Appendix A, GDC 19, "Control room," for the DBAs identified in RG 1.183 that could potentially result in control room and offsite doses.

The accident source term is intended to be representative of a major accident involving significant core damage. As a result of significant core damage, fission products are available for release into the containment environment. The proposed AST is an accident source term that is different from the accident source term used in the original design and licensing of WCGS. However, 10 CFR 50.67, as implemented in accordance with RG 1.183, identifies an AST that is acceptable to the NRC staff for use at operating reactors.

The following regulatory requirements and guidance are also considered within this proposed license amendment:

GDC 19, "Control room," of Appendix A to 10 CFR Part 50, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of allowable values.

NUREG-0800, SRP 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms, Revision 0," provides guidance for the safety review of the radiological consequences of DBAs associated with implementing an AST. SRP 15.0.1 supports the guidance outlined in RG 1.183.

NRC Generic Letter 2003-01, "Control Room Habitability," requests addressees to submit information that demonstrates that the control room at each of their respective facilities complies with the current licensing and design bases and applicable regulatory requirements, and that suitable design, maintenance and testing control measures are in place for maintaining this compliance.

RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," provides guidance on determining atmospheric relative concentration (χ/Q) values in support of design basis control room radiological habitability assessments at nuclear power plants. This document describes methods acceptable to the NRC staff for determining χ/Q values that will be used in control room radiological habitability assessments performed in support of applications for licenses and license amendment requests. Many of the regulatory positions presented in this guide represent substantial changes from procedures previously used to determine atmospheric relative concentrations for assessing the potential control room radiological consequences for a range of postulated accidental releases of radioactive material to the atmosphere. These revised procedures are largely based on the NRC sponsored computer code, ARCON96.

RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," provides guidance to determine relative concentrations for assessing the potential offsite radiological consequences for a range of postulated accidental releases of radioactive material to the atmosphere. These procedures include consideration of plume meander, directional dependence of dispersion conditions, and wind frequencies for various locations around actual exclusion area and LPZ boundaries.

4 TECHNICAL ANALYSIS

4.1 ATMOSPHERIC DISPERSION FACTOR – OFFSITE, CRHE, TSC

4.1.1 Atmospheric Dispersion Factors – Offsite

4.1.1.1 Introduction

The section describes a recalculation of atmospheric dispersion factors (χ/Q, sec/m^3) for the EAB and the outer boundary of the LPZ using the most recent meteorological data for WCGS in the event of accidental release of fission products. The existing site boundary χ/Q values are based on meteorological data from the late 1970s. This calculation is known in the Wolf Creek USAR subsection 2.3.4 as "short-term (accident) diffusion estimates." The methodology that is acceptable to the NRC staff for performing the calculation is described by RG 1.145 (Reference 1). Furthermore, the RG 1.145 methodology can be fully implemented through the use of the NRC sponsored PAVAN code (Reference 2). PAVAN is a computer program developed and officially issued in 1983 by Pacific Northwest Nuclear Laboratory under the sponsorship of the NRC. PAVAN is a computer program specifically used for determining atmospheric dispersion factors (χ/Q) at offsite locations for assessment of consequences of design basis accidents according to the methodology described in RG 1.145.

PAVAN implements a straight line Gaussian dispersion model based on the assumption that materials released to the atmosphere will be normally distributed about the plume centerline. A straight-line trajectory is assumed from the point of release to the point of interest. An adjustment factor can be used to include the effect of non-straight trajectories affected by the terrain in the calculation through the user-input options.

4.1.1.2 Input Data and Parameters

Site-specific meteorological data are required as input in the form of joint frequency distributions of wind direction and wind speed for each atmospheric stability class.

4.1.1.3 Meteorological Data

Meteorological data include hourly observations of wind speed and wind direction and a measurement of atmospheric stability. A 15-minute average, for each of the monitoring system's parameters, was used for each hour of data. Values outside of reasonable ranges were removed from the data set and were annotated for future reference and processing. Since redundant instruments are used, if the data from the primary instrument was reporting bad data, the data from the secondary instrument was used instead. Five consecutive years of WCGS site-specific meteorological data from January 1, 2006 to December 31, 2010 were collected and processed into the form of joint frequency distributions. The joint frequency distributions are the average numbers of hours that wind blew in a certain direction for a certain wind speed category and a certain atmospheric stability class. The joint frequency distribution data indicate the existence of winds (in units of hours) in all 16 directions for all wind speed categories (up to 14) and for all 7 atmospheric stability classes. These data are shown in Tables 4.1.1-1 through 4.1.1-7, where the number 7 corresponds to the total number of stability classes. Each of these tables shows the distributions of joint frequency of occurrence of the wind directions and wind speeds for each stability class.

The data recovery for the 5-year period met the 90% recovery criterion. There was only one large gap in recorded data. The missing data occurred when the meteorological data logger failed and had to be replaced. Due to this issue, there is not meteorological data from May 30 2007 to June 7 2007. The total number of hours from the site-specific 5-year data used in the analysis is 40,065 hours and is considered to be sufficiently representative to predict the long-term trend. RG 1.145 did not specifically mention the minimum number of years of data that should be used in the analysis. However according to RG 1.194, which is a guideline for calculating χ/Q for the control room, the NRC considers 5 years (43,824 hours) of hourly data to be representative of long-term trends at most sites.

For the PAVAN input file, the guidance in RG 1.145 was followed. Wind direction was classed into 16 compass directions (22.5° sectors centered on true north). Seven stability class categories were used. The winds speed categories used were: calm, 0.5, 0.75, 1.0, 1.25, 1.5, 2.0, 3.0, 4.0, 5.0, 6.0, 8.0 and 10.0 (meters per second). Calms were defined as hourly average wind speeds below the vane starting speed, since the wind speed threshold of the vane is greater than that of the anemometer.

4.1.1.3.1 Meteorological Monitoring Program

The meteorological tower is located in an open field about 0.5 miles northeast of the plant site. The terrain is flat and undulating, and the tower is located on a flat ridge. There are no variations in topography which exceed 55 feet within a 1 mile radius of the tower.

The meteorological monitoring system was designed to provide a reliable system consistent with the guidance in RG 1.23 (Safety Guide 23) "Onsite Meteorological Programs" as to the scope of monitoring activities and overall quality of the monitored data.

The meteorological tower consists of a 90-meter tower instrumented with the following equipment located at the 10 and 60-meter levels:

- Redundant Wind Speed Instruments
- Redundant Wind Direction Instruments
- Redundant Temperature Instruments

Stability class is determined using the temperature difference measurements between the 10 and 60-meter elevation instruments.

Meteorological monitoring instruments are calibrated on a semi-annual frequency. Calibrations are also performed after major equipment malfunctions, equipment modifications and equipment replacements.

The meteorological data observed by the monitoring equipment is transmitted to a strip chart recorder and the plant computer. An error alarm is generated if the output from a parameter exceeds the operating range for that parameter.

Table 4.1.1-1 Joint Frequency Distribution of Wind Speed and Direction for Stability Class A

Atmospheric Stability: Class A
Period of Record: January 1, 2006 to December 31, 2010
Elevation of Wind Measurement: 13.48 m
This table shows the number of hours of joint occurrence of wind in a certain wind direction and in a certain wind speed category.

Maximum Wind Speed (m/s)	Wind Direction																
	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	Total
0.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0.75	0	0	0	0	0	1	0	0	0	0	0	0	0	0	1	0	2
1.00	1	0	2	1	0	0	0	0	1	1	0	0	0	0	0	0	6
1.25	0	1	4	0	1	0	0	2	0	0	0	0	0	0	0	1	9
1.50	0	1	1	1	1	4	2	3	1	1	4	2	1	2	1	1	26
2.00	8	10	5	3	8	3	6	12	2	3	5	7	2	6	3	2	85
3.00	28	30	42	33	33	20	20	43	23	14	18	17	32	23	19	25	420
4.00	32	41	61	48	51	57	59	126	78	30	36	22	38	28	27	36	770
5.00	40	44	28	19	39	54	53	188	151	76	69	29	35	31	31	44	931
6.00	43	13	14	10	16	37	49	140	232	102	61	15	23	16	40	71	882
8.00	60	14	4	9	18	20	57	154	331	209	28	16	52	37	83	111	1203
10.00	17	8	0	0	2	2	4	53	165	66	2	7	15	25	49	71	486
44.70	8	0	0	0	0	0	2	11	46	12	1	5	14	34	20	9	162
Total	237	162	161	124	169	198	252	732	1030	514	224	120	212	202	274	371	4982

Number of calm hours = 0

Table 4.1.1-2 Joint Frequency Distribution of Wind Speed and Direction for Stability Class B

Atmospheric Stability: Class B
Period of Record: January 1, 2006 to December 31, 2010
Elevation of Wind Measurement: 13.48 m
This table shows the number of hours of joint occurrence of wind in a certain wind direction and in a certain wind speed category.

Maximum Wind Speed (m/s)	Wind Direction																
	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	Total
0.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0.75	0	0	0	0	0	1	0	0	0	0	0	0	0	0	0	0	1
1.00	0	1	0	0	1	0	0	0	1	1	0	1	1	0	0	0	6
1.25	1	0	1	1	0	0	0	0	0	0	0	2	0	0	1	0	6
1.50	1	1	1	2	3	2	3	1	0	0	2	1	0	0	0	4	21
2.00	5	4	6	9	5	4	5	9	5	1	2	11	3	11	6	6	92
3.00	22	28	50	24	25	29	21	49	41	14	14	16	18	8	15	22	396
4.00	24	36	31	29	33	30	29	62	58	29	15	15	13	12	14	26	456
5.00	37	26	6	18	16	20	25	56	65	29	24	13	13	7	19	34	408
6.00	20	12	3	2	11	9	15	22	45	49	13	3	12	8	24	33	281
8.00	24	8	1	4	2	5	19	35	64	64	8	6	11	20	46	55	372
10.00	16	2	0	0	0	1	3	12	35	24	1	2	9	15	35	43	198
44.70	7	0	0	0	1	0	3	6	15	10	0	6	5	19	16	4	92
Total	157	118	99	89	97	101	123	252	329	221	79	76	85	100	176	227	2329

Number of calm hours = 0

Table 4.1.1-3 Joint Frequency Distribution of Wind Speed and Direction for Stability Class C

Atmospheric Stability: Class C
Period of Record: January 1, 2006 to December 31, 2010
Elevation of Wind Measurement: 13.48 m
This table shows the number of hours of joint occurrence of wind in a certain wind direction and in a certain wind speed category.

Maximum Wind Speed (m/s)	Wind Direction																
	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	Total
0.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0.75	0	0	0	0	0	0	1	0	0	0	0	0	0	0	0	0	1
1.00	1	1	0	1	0	0	0	0	1	0	0	1	0	0	1	0	6
1.25	3	3	0	1	1	1	1	5	2	0	0	1	2	1	2	1	24
1.50	1	4	4	1	4	2	6	4	2	1	0	3	3	6	4	1	46
2.00	4	8	12	5	12	11	8	15	9	8	4	8	5	7	3	3	122
3.00	19	40	34	28	39	28	34	75	40	17	16	16	13	18	15	20	452
4.00	40	30	30	47	32	25	20	58	67	32	21	18	24	14	30	27	515
5.00	40	19	7	20	20	18	14	29	61	41	11	9	9	13	18	31	360
6.00	39	6	6	5	16	18	27	33	33	52	10	3	8	13	27	42	338
8.00	36	13	0	5	9	6	14	38	54	68	7	10	10	24	42	56	392
10.00	22	1	0	0	2	1	5	14	28	32	5	4	3	20	38	27	202
44.70	2	0	0	0	1	0	1	4	16	6	1	1	4	9	15	5	65
Total	207	125	93	113	136	110	131	275	313	257	75	74	81	125	195	213	2523

Number of calm hours = 0

Table 4.1.1-4 Joint Frequency Distribution of Wind Speed and Direction for Stability Class D

Atmospheric Stability: Class D
Period of Record: January 1, 2006 to December 31, 2010
Elevation of Wind Measurement: 13.48 m
This table shows the number of hours of joint occurrence of wind in a certain wind direction and in a certain wind speed category.

Maximum Wind Speed (m/s)	Wind Direction																
	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	Total
0.50	0	0	0	0	1	1	0	0	1	0	0	0	0	0	0	0	3
0.75	2	4	11	2	6	2	2	1	0	1	3	1	1	2	0	1	39
1.00	4	19	18	9	4	9	5	4	1	1	3	1	0	3	1	3	85
1.25	9	18	20	10	14	9	11	9	9	4	5	6	3	2	3	10	142
1.50	8	20	27	18	17	8	13	20	12	6	10	12	4	6	5	12	198
2.00	34	52	67	62	41	39	56	50	31	24	37	34	18	14	17	28	604
3.00	118	164	213	154	134	148	116	127	144	85	109	53	54	41	52	122	1834
4.00	198	124	154	190	157	150	166	264	210	162	93	62	69	78	145	190	2412
5.00	217	119	66	101	102	112	172	276	291	190	46	38	77	107	194	218	2326
6.00	197	85	16	46	77	76	118	250	354	156	45	30	58	103	177	244	2032
8.00	318	61	8	31	65	50	87	279	500	214	14	38	82	162	292	441	2642
10.00	119	21	0	2	12	4	17	113	219	97	7	24	31	70	136	203	1075
44.70	39	0	1	0	2	0	8	39	119	31	4	5	8	21	50	41	368
Total	1263	687	601	625	632	608	771	1432	1891	971	376	304	405	609	1072	1513	13760

Number of calm hours = 1

Table 4.1.1-5 Joint Frequency Distribution of Wind Speed and Direction for Stability Class E

Atmospheric Stability: Class E
Period of Record: January 1, 2006 to December 31, 2010
Elevation of Wind Measurement: 13.48 m
This table shows the number of hours of joint occurrence of wind in a certain wind direction and in a certain wind speed category.

Maximum Wind Speed (m/s)	Wind Direction																
	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	Total
0.50	1	0	3	1	1	0	1	2	0	2	0	0	0	0	0	0	11
0.75	2	6	5	5	3	2	3	1	1	2	0	0	2	2	2	2	38
1.00	12	14	15	10	4	4	3	7	3	6	0	4	1	2	2	6	93
1.25	20	17	20	20	18	12	13	1	7	5	8	3	2	0	5	7	158
1.50	20	31	26	18	28	29	23	9	3	8	15	5	4	3	8	10	240
2.00	62	47	64	70	63	83	62	39	21	36	62	25	10	9	17	46	716
3.00	98	75	98	148	173	184	219	240	163	104	166	56	75	56	114	155	2124
4.00	125	50	46	73	113	169	275	421	270	174	81	53	79	90	151	167	2337
5.00	78	26	11	43	70	104	182	446	324	175	29	40	43	67	100	121	1859
6.00	48	11	4	32	44	60	86	306	246	70	16	15	17	17	46	65	1083
8.00	43	14	3	3	13	22	61	370	330	80	8	6	3	6	20	39	1021
10.00	2	1	1	0	1	4	12	107	142	34	5	3	4	1	2	5	324
44.70	4	0	0	1	0	2	8	26	94	7	1	1	0	0	0	0	144
Total	515	292	296	424	531	675	948	1975	1604	703	391	211	240	253	467	623	10148

Number of calm hours = 1

Table 4.1.1-6 Joint Frequency Distribution of Wind Speed and Direction for Stability Class F

Atmospheric Stability: Class F
Period of Record: January 1, 2006 to December 31, 2010
Elevation of Wind Measurement: 13.48 m
This table shows the number of hours of joint occurrence of wind in a certain wind direction and in a certain wind speed category.

Maximum Wind Speed (m/s)	Wind Direction																
	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	Total
0.50	2	2	1	3	0	3	0	0	1	1	1	0	0	0	0	0	14
0.75	1	7	9	6	1	2	1	3	1	0	4	2	2	2	2	2	45
1.00	9	13	15	4	11	9	5	4	2	3	3	6	3	4	6	3	100
1.25	8	19	23	9	11	12	8	8	8	6	6	5	2	4	5	10	144
1.50	14	29	25	17	30	15	11	17	5	2	9	3	1	6	9	11	204
2.00	44	50	53	58	79	79	122	36	13	13	23	8	4	12	30	37	661
3.00	98	65	49	195	179	221	243	156	59	45	45	18	19	15	84	111	1602
4.00	64	33	16	40	58	64	110	175	85	45	13	11	13	8	53	67	855
5.00	13	3	0	5	7	6	32	83	70	27	4	6	3	2	4	20	285
6.00	3	0	0	0	4	1	8	40	44	15	0	1	0	0	1	4	121
8.00	0	0	0	0	0	0	2	24	30	3	1	1	0	0	0	1	62
10.00	0	0	0	0	0	0	0	3	11	1	2	0	0	0	0	0	17
44.70	0	0	0	0	0	0	0	0	3	1	0	0	0	0	0	0	4
Total	256	221	191	337	380	412	542	549	332	162	111	61	47	53	194	266	4114

Number of calm hours = 5

Table 4.1.1-7 Joint Frequency Distribution of Wind Speed and Direction for Stability Class G

Atmospheric Stability: Class G
Period of Record: January 1, 2006 to December 31, 2010
Elevation of Wind Measurement: 13.48 m
This table shows the number of hours of joint occurrence of wind in a certain wind direction and in a certain wind speed category.

Maximum Wind Speed (m/s)	Wind Direction																
	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	Total
0.50	3	2	2	0	2	0	0	1	1	1	2	0	1	1	0	0	16
0.75	5	3	6	4	2	4	3	2	0	1	2	2	1	2	1	5	43
1.00	4	10	5	5	7	4	5	0	0	4	1	5	3	7	5	7	72
1.25	8	17	16	9	7	8	6	5	4	3	1	2	1	7	7	18	119
1.50	14	25	25	5	11	15	8	4	1	3	5	2	4	3	10	13	148
2.00	33	90	55	20	61	65	46	16	5	6	7	3	2	17	20	41	487
3.00	73	75	60	112	128	162	124	53	35	6	16	3	1	10	50	69	977
4.00	16	20	9	29	22	27	11	39	29	2	8	1	1	2	22	37	275
5.00	6	2	0	1	2	1	3	10	6	2	3	0	0	0	1	5	42
6.00	0	1	0	0	1	1	1	1	5	1	1	0	0	0	0	1	13
8.00	0	0	0	0	0	0	0	4	2	0	0	0	0	0	0	0	6
10.00	0	0	0	0	0	0	0	0	0	1	0	0	0	0	0	0	1
44.70	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Total	162	245	178	185	243	287	207	135	88	30	46	18	14	49	116	196	2199

Number of calm hours = 3

4.1.1.3.2 Input Parameters

The descriptions and justifications of key input parameters to the PAVAN computer code are provided in Table 4.1.1-8.

Table 4.1.1-8 Descriptions and Justifications of Key Input Parameters

Variable Name	Descriptions and justifications
A	This is an input of minimum cross-sectional area (m^2) of the containment structure for use in the building wake effect calculation. In general, the smaller value for this area yields larger, more conservative χ/Q s. For releases from potential sources such as the equipment hatch, unit vent stack, main steam safety valves/atmospheric relief valves (MSSVs/ARVs) vent, or turbine-driven auxiliary feedwater (TDAFW) exhaust vent, the cross-sectional area of the containment building is used. These release sources are located close to the containment building. The containment building wake effect for winds in most directions is applicable to these sources. Effects from other smaller adjacent buildings are ignored. For a release from the refueling water storage tank (RWST), the cross-sectional area of the RWST is used. The RWST is located more than 100 ft from the closest containment building wall. In this case, the wake effects due to the containment building for certain wind sectors and other adjacent buildings are conservatively ignored. Cross-sectional area of the containment building = 2649 m^2 Cross-sectional area of the RWST = 170.94 m^2
D	This input is the height (m) above plant grade of the containment structure used in the building-wake term for the annual-average calculations. The height of the containment building is 63.66 m. The height of the RWST is 14.02 m.
HS	This input is the height above plant grade of the release point. The input value of this height depends upon whether the release is a ground-level release or a stack release. A ground-level release is defined as any release whose plume can be influenced by the airflow wake of the building structures. This is generally all release points that are effectively less than 2½ times the height of adjacent solid structures. For a ground-level release, HS = 10.0 m must be set. This is the release height assumed by the PAVAN code for a ground-level release. All release points (which include the equipment hatch, unit vent stack, MSSVs/ARVs vent, TDAFW exhaust vent, and RWST) considered at WCGS satisfy the ground-level release definition. These release points are right on top of buildings or structures, and will be at the minimum under the influence of the respective building/structure. This is in addition to adjacent buildings. Hence, all release points are classified as a ground-level release.
TOWERH	This is an input for height (m) above ground level at which the wind speed was measured. When TOWERH is larger than HS, the wind speed is adjusted downward by the PAVAN code, so that the wind speed is consistent with the release height. Here, the input height is 13.48 m.

Table 4.1.1-8 (cont.)	Descriptions and Justifications of Key Input Parameters
Variable Name	**Descriptions and justifications**
CALM(J) J = 1,7	This is an input for the number of hours or percent of calm for each stability class: J=1 – stability class A J=2 – stability class B J=3 – stability class C J=4 – stability class D J=5 – stability class E J=6 – stability class F J=7 – stability class G Calm wind speed – Calms are defined in RG 1.145 as hourly average wind speeds below the vane or anemometer starting speed, whichever is higher. At WCGS, 0.75 mph (0.34 m/s) is the calm threshold. This is the value used in the Wolf Creek USAR.
NVEL	Number of wind speed categories into which the joint frequency data are classified (maximum of 14).
UMAX(I) I = 1,NVEL	This is an input for defining the maximum wind speed (meters/second) for each wind speed category. The wind speed data are classified into 14 categories, the maximum number allowed by the PAVAN code, as follows: 0.34 (calm), 0.5, 0.75, 1.0, 1.25, 1.5, 2.0, 3.0, 4.0, 5.0, 6.0, 8.0, 10.0, and 44.7 meters/second. The first wind speed category is calm wind. This way of wind speed classification with higher resolution in the lower wind speed range is recommended by NUREG/CR-2858 (Reference 2) in order to achieve the best results for the χ/Q value at 0.5 percentile. See Table 4.1.1-9.
BDY(K,I) K=1,16 I=1,2	This is an input for downwind distances (meters) at which χ/Q is to be calculated for each wind direction sector (16 in total). K=1 is to the south boundary, K=2 to the SSW, and, K=16 to the SSE. Two sets of distances (I=1 and 2), one for the EAB and another for the LPZ boundary, for each wind direction sector that is used as input. Here, the EAB distance of 1200 meters and the LPZ distance of 4023 meters, as defined in the Wolf Creek USAR, are used for all wind directions. See Table 4.1.1-10.
TAF(K,I) K = 1,16 I = 1,2	This is an input for a site-specific terrain correction factor. See assumption 4 in section 4.1.1.3 for a justification of the values selected.

Table 4.1.1-9 Wind Speed Categories used in the PAVAN Calculations

Category Number	Maximum Wind Speed	
	Meters/Second	Miles/Hour
1 (calm speed)	0.34	0.75
2	0.5	1.12
3	0.75	1.68
4	1.0	2.24
5	1.25	2.80
6	1.5	3.36
7	2.0	4.47
8	3.0	6.71
9	4.0	8.95
10	5.0	11.18
11	6.0	13.42
12	8.0	17.90
13	10.0	22.37
14	44.7	100

Table 4.1.1-10 Downwind Distances to EAB and LPZ at which χ/Q s are Calculated

Downwind Direction	Distance to EAB (meters)	Distance to LPZ (meters)
S	1200	4023
SSW	1200	4023
SW	1200	4023
WSW	1200	4023
W	1200	4023
WNW	1200	4023
NW	1200	4023
NNW	1200	4023
N	1200	4023
NNE	1200	4023
NE	1200	4023
ENE	1200	4023
E	1200	4023
ESE	1200	4023
SE	1200	4023
SSE	1200	4023

4.1.1.4 Assumptions and Acceptance Criteria

Assumptions

1. A release of radioactive materials during an accident from one or more potential release sources is treated as a single ground-level release source (i.e. no multiple releases at the same time) for all cases. The PAVAN code cannot model multiple release sources.

2. The distance from the release point to the EAB is 1200 meters for all directions regardless of the release location. Likewise, the distance from the release point to the LPZ boundary is 4023 meters for all directions. This assumption is based on these two distances that are discussed in the Wolf Creek USAR.

3. A building wake effect is included in the calculation.

   - For a release from potential sources such as the equipment hatch, unit vent stack, MSSVs/ARVs vent, or TDAFW exhaust vent, the cross-sectional area of the containment building is used for the calculation of the building wake term. These release sources are located close to the containment building. Wake effects from other smaller adjacent buildings are ignored.

   - For a release from the RWST, the cross-sectional area of the RWST is used for the building wake term. The RWST is located more than 100 feet from the closest containment building wall. The wake effects due to the containment building for certain wind sectors and other adjacent buildings are conservatively ignored.

4. A straight trajectory of plume movement from the release point to the EAB and LPZ distances is assumed as part of the PAVAN model. The straight trajectory assumption is then supplemented by the use of the terrain correction factors. The terrain correction factors account for the possibility of non-straight trajectory due to temporal and spatial variations in airflow in the EAB and LPZ areas that do not reflect in the meteorological data collected at a single onsite station that is used in this calculation.

   - In the USAR Chapter 2.3.5, on the long-term diffusion estimate, terrain/recirculation correction factors for the 16 wind sectors were calculated based on 1973-1974 data and listed in Table 2.3-61. The maximum value for the EAB in the W sector and the maximum value for the LPZ in the WSW sector were conservatively applied to all wind sectors in the PAVAN calculation.

5. Plume meander in the horizontal direction under low wind conditions is a physical phenomenon that is internally accounted for in the PAVAN calculation for wind speed less than 6 m/s and atmospheric stability conditions being neutral (class D) or stable (class E, F, or G). (This is a code model assumption.)

Acceptance Criteria

The acceptance criteria for determining offsite χ/Q values are that the calculation is performed according to the methodology and guidelines of RG 1.145.

4.1.1.5 Results

The output from the code calculation includes the characteristics of the meteorological data and the relative concentration (χ/Q) of contaminants in the plume as functions of direction for various time periods at the EAB and the boundary of the LPZ. Several definitions of χ/Q values are calculated, and these are described below.

Statistical Characteristics of the 5-Year Meteorological Data

Tables 4.1.1-11 through 4.1.1-17 show the joint frequency distributions that were normalized by PAVAN into a percent of the total wind hours. Tables 4.1.1-11 through 4.1.1-17 are a repeat of Tables 4.1.1-1 through 4.1.1-7, except that the data are presented in percent, rather than the actual number of wind hours. The total percent of wind in each stability class is shown at the lower right corner in each of these tables. The total sum of these numbers adds up to 100 percent. From these tables, meteorological conditions were mostly neutral or stable (i.e., stability class D – neutral (34.35 percent), class E – slightly stable (25.33 percent) and class F – moderately stable (10.28 percent). From these normalized data in Tables 4.1.1-11 through 4.1.1-17, the overall wind direction frequency and the wind speed frequency are calculated as shown in Tables 4.1.1-18 and 4.1.1-19, respectively. The wind direction frequency in Table 4.1.1-18 indicates that there were more winds from the S direction (13.9 percent) and the SSE direction (13.4 percent), than from other directions. The wind speed frequency in Table 4.1.1-19 indicates that the majority of wind speed is from 3 m/s to 8 m/s (i.e., 19.48 percent for 2 to 3 m/s, 19.02 percent for 3 to 4 m/s, 15.50 percent for 4 to 5 m/s, 11.86 percent for 5 to 6 m/s, and 14.22 percent for 6 to 8 m/s).

Maximum Sector χ/Q

For each of the 16 downwind direction sectors, PAVAN calculates χ/Q values for each combination of wind speed category and atmospheric stability class at an EAB distance and an LPZ distance. The χ/Q values calculated for each sector are arranged in order from largest to smallest, and the associated cumulative frequency distribution is derived based on the frequency distribution of wind speed and stability class for that sector. The smallest χ/Q value for the sector has a cumulative frequency of 100 percent for that sector. Then the 0.5 percent χ/Q value for the sector that corresponds to the cumulative frequency of 0.5 percent of the total time of all sectors is determined by logarithmic interpolation. Up to 16 such 0.5 percent χ/Q values are calculated for all 16 sectors. Then the maximum sector χ/Q value is selected from the maximum value of the 16 sectors.

Overall Site χ/Q

PAVAN calculates the overall site χ/Q value by combining χ/Q values from all wind sectors into a cumulative frequency distribution for the entire site. Then the overall site χ/Q value is selected from the value that corresponds to 5.0 percent of the total time.

χ/Q for 0 to 2-Hour Period

The larger of the maximum sector χ/Q value and the overall site value is used to represent the χ/Q for a 0 to 2 hour time period.

Annual Average χ/Q

The annual average χ/Q value is calculated in PAVAN by using a long-term continuous release model (as opposed to a short-term release) described in RG 1.111 (Reference 3). In this model, the plume horizontal dispersion is assumed to be evenly distributed to fill the entire width of the 22.5-degree wind sector.

Intermediate Time Periods χ/Q

The χ/Q values for intermediate time periods including 0 to 8 hours, 8 to 24 hours, 1 to 4 days, and 4 to 30 days are determined by logarithmic interpolation between the 0 to 2 hour period χ/Q value and the annual average χ/Q value.

χ/Q Results

The χ/Q results that account for the building wake effect from the containment building are summarized in Table 4.1.1-20 for the EAB and in Table 4.1.1-21 for the LPZ. These results are applied to most release sources including the unit vent stack, equipment hatch, MSSVs/ARVs vent, and TDAFW exhaust vent. These release sources are located within the close proximity of the containment building and will be impacted by the wake effect of the containment building.

For a release from the RWST, the χ/Q results with the containment building wake effect are not conservative, since the RWST is located more than 100 feet from the closest containment wall. Another set of χ/Q calculations that only accounts for the wake effect of the RWST itself, is summarized in Table 4.1.1-22 for the EAB and in Table 4.1.1-23 for the LPZ.

χ/Q Values for Use in Dose Assessment

In RG 1.145, the χ/Q value used in the dose assessment is the larger of the maximum sector χ/Q value and the overall site value. This is summarized in Table 4.1.1-24.

Two sets of χ/Q values were calculated: (1) for a release from a source located close to containment structure such as the unit vent stack, equipment hatch, MSSVs/ARVs vent, or TDAFW exhaust vent, and (2) for a release from the RWST.

If the release source is not the RWST, the values are given in the upper half of Table 4.1.1-24. If the release source is the RWST, the values are given in the lower half of Table 4.1.1-24. For dose calculations that do not differentiate the release source, the RWST release source will bound all other sources. The difference between the RWST source and other sources is the building wake effect. The RWST source has less of a building wake effect than other potential sources.

Table 4.1.1-25 lists the bounding χ/Q values that are typically used in the dose calculations. (Not all calculated χ/Q values are used in the dose calculations. Unlisted values are typically not used in the dose calculations.) These bounding χ/Q values are associated with the RWST as the release source.

4.1.1.6 Conclusions

The resulting offsite χ/Q values for use in the dose assessment are listed in Table 4.1.1-25.

Table 4.1.1-11 Joint Frequency Distribution (in percent of total hours) for Stability Class A

Atmospheric Stability: Class A
Period of Record: January 1, 2006 to December 31, 2010
Elevation of Wind Measurement: 13.48 m
This table shows the percentage of hours of joint occurrence of wind in a certain wind direction and in a certain wind speed category.

Maximum Wind Speed (m/s)	Wind Direction																
	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	Total
0.34	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
0.50	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
0.75	0.000	0.000	0.000	0.000	0.000	0.002	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.002	0.000	0.005
1.00	0.002	0.000	0.005	0.002	0.000	0.000	0.000	0.000	0.002	0.002	0.000	0.000	0.000	0.000	0.000	0.000	0.015
1.25	0.000	0.002	0.010	0.000	0.002	0.000	0.000	0.005	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.002	0.022
1.50	0.000	0.002	0.002	0.002	0.002	0.010	0.005	0.007	0.002	0.002	0.010	0.005	0.002	0.005	0.002	0.002	0.065
2.00	0.020	0.025	0.012	0.007	0.020	0.007	0.015	0.030	0.005	0.007	0.012	0.017	0.005	0.015	0.007	0.005	0.212
3.00	0.070	0.075	0.105	0.082	0.082	0.050	0.050	0.107	0.057	0.035	0.045	0.042	0.080	0.057	0.047	0.062	1.048
4.00	0.080	0.102	0.152	0.120	0.127	0.142	0.147	0.314	0.195	0.075	0.090	0.055	0.095	0.070	0.067	0.090	1.922
5.00	0.100	0.110	0.070	0.047	0.097	0.135	0.132	0.469	0.377	0.190	0.172	0.072	0.087	0.077	0.077	0.110	2.324
6.00	0.107	0.032	0.035	0.025	0.040	0.092	0.122	0.349	0.579	0.255	0.152	0.037	0.057	0.040	0.100	0.177	2.201
8.00	0.150	0.035	0.010	0.022	0.045	0.050	0.142	0.384	0.826	0.522	0.070	0.040	0.130	0.092	0.207	0.277	3.003
10.00	0.042	0.020	0.000	0.000	0.005	0.005	0.010	0.132	0.412	0.165	0.005	0.017	0.037	0.062	0.122	0.177	1.213
44.70	0.020	0.000	0.000	0.000	0.000	0.000	0.005	0.027	0.115	0.030	0.002	0.012	0.035	0.085	0.050	0.022	0.404
Total	0.59	0.40	0.40	0.31	0.42	0.49	0.63	1.83	2.57	1.28	0.56	0.30	0.53	0.50	0.68	0.93	12.43

Table 4.1.1-12 Joint Frequency Distribution (in percent of total hours) for Stability Class B

Atmospheric Stability: Class B
Period of Record: January 1, 2006 to December 31, 2010
Elevation of Wind Measurement: 13.48 m
This table shows the percentage of hours of joint occurrence of wind in a certain wind direction and in a certain wind speed category.

Maximum Wind Speed (m/s)	Wind Direction																
	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	Total
0.34	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
0.50	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
0.75	0.000	0.000	0.000	0.000	0.000	0.002	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.002
1.00	0.000	0.002	0.000	0.000	0.002	0.000	0.000	0.000	0.002	0.002	0.000	0.002	0.002	0.000	0.000	0.000	0.015
1.25	0.002	0.000	0.002	0.002	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.005	0.000	0.000	0.002	0.000	0.015
1.50	0.002	0.002	0.002	0.005	0.007	0.005	0.007	0.002	0.000	0.000	0.005	0.002	0.000	0.000	0.000	0.010	0.052
2.00	0.012	0.010	0.015	0.022	0.012	0.010	0.012	0.022	0.012	0.002	0.005	0.027	0.007	0.027	0.015	0.015	0.230
3.00	0.055	0.070	0.125	0.060	0.062	0.072	0.052	0.122	0.102	0.035	0.035	0.040	0.045	0.020	0.037	0.055	0.988
4.00	0.060	0.090	0.077	0.072	0.082	0.075	0.072	0.155	0.145	0.072	0.037	0.037	0.032	0.030	0.035	0.065	1.138
5.00	0.092	0.065	0.015	0.045	0.040	0.050	0.062	0.140	0.162	0.072	0.060	0.032	0.032	0.017	0.047	0.085	1.018
6.00	0.050	0.030	0.007	0.005	0.027	0.022	0.037	0.055	0.112	0.122	0.032	0.007	0.030	0.020	0.060	0.082	0.701
8.00	0.060	0.020	0.002	0.010	0.005	0.012	0.047	0.087	0.160	0.160	0.020	0.015	0.027	0.050	0.115	0.137	0.928
10.00	0.040	0.005	0.000	0.000	0.000	0.002	0.007	0.030	0.087	0.060	0.002	0.005	0.022	0.037	0.087	0.107	0.494
44.70	0.017	0.000	0.000	0.000	0.002	0.000	0.007	0.015	0.037	0.025	0.000	0.015	0.012	0.047	0.040	0.010	0.230
Total	0.39	0.29	0.25	0.22	0.24	0.25	0.31	0.63	0.82	0.55	0.20	0.19	0.21	0.25	0.44	0.57	5.81

Table 4.1.1-13 Joint Frequency Distribution (in percent of total hours) for Stability Class C

Atmospheric Stability: Class C
Period of Record: January 1, 2006 to December 31, 2010
Elevation of Wind Measurement: 13.48 m
This table shows the percentage of hours of joint occurrence of wind in a certain wind direction and in a certain wind speed category.

Maximum Wind Speed (m/s)	Wind Direction																
	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	Total
0.34	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
0.50	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
0.75	0.000	0.000	0.000	0.000	0.000	0.000	0.002	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.002
1.00	0.002	0.002	0.000	0.002	0.000	0.000	0.000	0.000	0.002	0.000	0.000	0.002	0.000	0.000	0.002	0.000	0.015
1.25	0.007	0.007	0.000	0.002	0.002	0.002	0.002	0.012	0.005	0.000	0.000	0.002	0.005	0.002	0.005	0.002	0.060
1.50	0.002	0.010	0.010	0.002	0.010	0.005	0.015	0.010	0.005	0.002	0.000	0.007	0.007	0.015	0.010	0.002	0.115
2.00	0.010	0.020	0.030	0.012	0.030	0.027	0.020	0.037	0.022	0.020	0.010	0.020	0.012	0.017	0.007	0.007	0.305
3.00	0.047	0.100	0.085	0.070	0.097	0.070	0.085	0.187	0.100	0.042	0.040	0.040	0.032	0.045	0.037	0.050	1.128
4.00	0.100	0.075	0.075	0.117	0.080	0.062	0.050	0.145	0.167	0.080	0.052	0.045	0.060	0.035	0.075	0.067	1.285
5.00	0.100	0.047	0.017	0.050	0.050	0.045	0.035	0.072	0.152	0.102	0.027	0.022	0.022	0.032	0.045	0.077	0.899
6.00	0.097	0.015	0.015	0.012	0.040	0.045	0.067	0.082	0.082	0.130	0.025	0.007	0.020	0.032	0.067	0.105	0.844
8.00	0.090	0.032	0.000	0.012	0.022	0.015	0.035	0.095	0.135	0.170	0.017	0.025	0.025	0.060	0.105	0.140	0.978
10.00	0.055	0.002	0.000	0.000	0.005	0.002	0.012	0.035	0.070	0.080	0.012	0.010	0.007	0.050	0.095	0.067	0.504
44.70	0.005	0.000	0.000	0.000	0.002	0.000	0.002	0.010	0.040	0.015	0.002	0.002	0.010	0.022	0.037	0.012	0.162
Total	0.52	0.31	0.23	0.28	0.34	0.27	0.33	0.69	0.78	0.64	0.19	0.18	0.20	0.31	0.49	0.53	6.30

Table 4.1.1-14 Joint Frequency Distribution (in percent of total hours) for Stability Class D

Atmospheric Stability: Class D
Period of Record: January 1, 2006 to December 31, 2010
Elevation of Wind Measurement: 13.48 m
This table shows the percentage of hours of joint occurrence of wind in a certain wind direction and in a certain wind speed category.

Maximum Wind Speed (m/s)	Wind Direction																
	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	Total
0.34	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.002
0.50	0.000	0.000	0.000	0.000	0.002	0.002	0.000	0.000	0.002	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.007
0.75	0.005	0.010	0.027	0.005	0.015	0.005	0.005	0.002	0.000	0.002	0.007	0.002	0.002	0.005	0.000	0.002	0.097
1.00	0.010	0.047	0.045	0.022	0.010	0.022	0.012	0.010	0.002	0.002	0.007	0.002	0.000	0.007	0.002	0.007	0.212
1.25	0.022	0.045	0.050	0.025	0.035	0.022	0.027	0.022	0.022	0.010	0.012	0.015	0.007	0.005	0.007	0.025	0.354
1.50	0.020	0.050	0.067	0.045	0.042	0.020	0.032	0.050	0.030	0.015	0.025	0.030	0.010	0.015	0.012	0.030	0.494
2.00	0.085	0.130	0.167	0.155	0.102	0.097	0.140	0.125	0.077	0.060	0.092	0.085	0.045	0.035	0.042	0.070	1.508
3.00	0.295	0.409	0.532	0.384	0.334	0.369	0.290	0.317	0.359	0.212	0.272	0.132	0.135	0.102	0.130	0.305	4.578
4.00	0.494	0.309	0.384	0.474	0.392	0.374	0.414	0.659	0.524	0.404	0.232	0.155	0.172	0.195	0.362	0.474	6.020
5.00	0.542	0.297	0.165	0.252	0.255	0.280	0.429	0.689	0.726	0.474	0.115	0.095	0.192	0.267	0.484	0.544	5.806
6.00	0.492	0.212	0.040	0.115	0.192	0.190	0.295	0.624	0.884	0.389	0.112	0.075	0.145	0.257	0.442	0.609	5.072
8.00	0.794	0.152	0.020	0.077	0.162	0.125	0.217	0.696	1.248	0.534	0.035	0.095	0.205	0.404	0.729	1.101	6.594
10.00	0.297	0.052	0.000	0.005	0.030	0.010	0.042	0.282	0.547	0.242	0.017	0.060	0.077	0.175	0.339	0.507	2.683
44.70	0.097	0.000	0.002	0.000	0.005	0.000	0.020	0.097	0.297	0.077	0.010	0.012	0.020	0.052	0.125	0.102	0.919
Total	3.15	1.72	1.50	1.56	1.58	1.52	1.92	3.57	4.72	2.42	0.94	0.76	1.01	1.52	2.68	3.78	34.35

Table 4.1.1-15 Joint Frequency Distribution (in percent of total hours) for Stability Class E

Atmospheric Stability: Class E
Period of Record: January 1, 2006 to December 31, 2010
Elevation of Wind Measurement: 13.48 m
This table shows the percentage of hours of joint occurrence of wind in a certain wind direction and in a certain wind speed category.

Maximum Wind Speed (m/s)	Wind Direction																
	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	Total
0.34	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.002
0.50	0.002	0.000	0.007	0.002	0.002	0.000	0.002	0.005	0.000	0.005	0.000	0.000	0.000	0.000	0.000	0.000	0.027
0.75	0.005	0.015	0.012	0.012	0.007	0.005	0.007	0.002	0.002	0.005	0.000	0.000	0.005	0.005	0.005	0.005	0.095
1.00	0.030	0.035	0.037	0.025	0.010	0.010	0.007	0.017	0.007	0.015	0.000	0.010	0.002	0.005	0.005	0.015	0.232
1.25	0.050	0.042	0.050	0.050	0.045	0.030	0.032	0.002	0.017	0.012	0.020	0.007	0.005	0.000	0.012	0.017	0.394
1.50	0.050	0.077	0.065	0.045	0.070	0.072	0.057	0.022	0.007	0.020	0.037	0.012	0.010	0.007	0.020	0.025	0.599
2.00	0.155	0.117	0.160	0.175	0.157	0.207	0.155	0.097	0.052	0.090	0.155	0.062	0.025	0.022	0.042	0.115	1.787
3.00	0.245	0.187	0.245	0.369	0.432	0.459	0.547	0.599	0.407	0.260	0.414	0.140	0.187	0.140	0.285	0.387	5.301
4.00	0.312	0.125	0.115	0.182	0.282	0.422	0.686	1.051	0.674	0.434	0.202	0.132	0.197	0.225	0.377	0.417	5.833
5.00	0.195	0.065	0.027	0.107	0.175	0.260	0.454	1.113	0.809	0.437	0.072	0.100	0.107	0.167	0.250	0.302	4.640
6.00	0.120	0.027	0.010	0.080	0.110	0.150	0.215	0.764	0.614	0.175	0.040	0.037	0.042	0.042	0.115	0.162	2.703
8.00	0.107	0.035	0.007	0.007	0.032	0.055	0.152	0.923	0.824	0.200	0.020	0.015	0.007	0.015	0.050	0.097	2.548
10.00	0.005	0.002	0.002	0.000	0.002	0.010	0.030	0.267	0.354	0.085	0.012	0.007	0.010	0.002	0.005	0.012	0.809
44.70	0.010	0.000	0.000	0.002	0.000	0.005	0.020	0.065	0.235	0.017	0.002	0.002	0.000	0.000	0.000	0.000	0.359
Total	1.29	0.73	0.74	1.06	1.33	1.68	2.37	4.93	4.00	1.75	0.98	0.53	0.60	0.63	1.17	1.56	25.33

Table 4.1.1-16 Joint Frequency Distribution (in percent of total hours) for Stability Class F

Atmospheric Stability: Class F
Period of Record: January 1, 2006 to December 31, 2010
Elevation of Wind Measurement: 13.48 m
This table shows the percentage of hours of joint occurrence of wind in a certain wind direction and in a certain wind speed category.

Maximum Wind Speed (m/s)	Wind Direction																
	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	Total
0.34	0.001	0.002	0.002	0.001	0.001	0.001	0.001	0.001	0.000	0.000	0.001	0.001	0.000	0.000	0.001	0.001	0.012
0.50	0.005	0.005	0.002	0.007	0.000	0.007	0.000	0.000	0.002	0.002	0.002	0.000	0.000	0.000	0.000	0.000	0.035
0.75	0.002	0.017	0.022	0.015	0.002	0.005	0.002	0.007	0.002	0.000	0.010	0.005	0.005	0.005	0.005	0.005	0.112
1.00	0.022	0.032	0.037	0.010	0.027	0.022	0.012	0.010	0.005	0.007	0.007	0.015	0.007	0.010	0.015	0.007	0.250
1.25	0.020	0.047	0.057	0.022	0.027	0.030	0.020	0.020	0.020	0.015	0.015	0.012	0.005	0.010	0.012	0.025	0.359
1.50	0.035	0.072	0.062	0.042	0.075	0.037	0.027	0.042	0.012	0.005	0.022	0.007	0.002	0.015	0.022	0.027	0.509
2.00	0.110	0.125	0.132	0.145	0.197	0.197	0.305	0.090	0.032	0.032	0.057	0.020	0.010	0.030	0.075	0.092	1.650
3.00	0.245	0.162	0.122	0.487	0.447	0.552	0.607	0.389	0.147	0.112	0.112	0.045	0.047	0.037	0.210	0.277	3.999
4.00	0.160	0.082	0.040	0.100	0.145	0.160	0.275	0.437	0.212	0.112	0.032	0.027	0.032	0.020	0.132	0.167	2.134
5.00	0.032	0.007	0.000	0.012	0.017	0.015	0.080	0.207	0.175	0.067	0.010	0.015	0.007	0.005	0.010	0.050	0.711
6.00	0.007	0.000	0.000	0.000	0.010	0.002	0.020	0.100	0.110	0.037	0.000	0.002	0.000	0.000	0.002	0.010	0.302
8.00	0.000	0.000	0.000	0.000	0.000	0.000	0.005	0.060	0.075	0.007	0.002	0.002	0.000	0.000	0.000	0.002	0.155
10.00	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.007	0.027	0.002	0.005	0.000	0.000	0.000	0.000	0.000	0.042
44.70	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.007	0.002	0.000	0.000	0.000	0.000	0.000	0.000	0.010
Total	0.64	0.55	0.48	0.84	0.95	1.03	1.35	1.37	0.83	0.40	0.28	0.15	0.12	0.13	0.48	0.66	10.28

Table 4.1.1-17 Joint Frequency Distribution (in percent of total hours) for Stability Class G

Atmospheric Stability: Class G
Period of Record: January 1, 2006 to December 31, 2010
Elevation of Wind Measurement: 13.48 m
This table shows the percentage of hours of joint occurrence of wind in a certain wind direction and in a certain wind speed category.

Maximum Wind Speed (m/s)	Wind Direction																
	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	Total
0.34	0.001	0.001	0.001	0.001	0.001	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.001	0.000	0.001	0.007
0.50	0.007	0.005	0.005	0.000	0.005	0.000	0.000	0.002	0.002	0.002	0.005	0.000	0.002	0.002	0.000	0.000	0.040
0.75	0.012	0.007	0.015	0.010	0.005	0.010	0.007	0.005	0.000	0.002	0.005	0.005	0.002	0.005	0.002	0.012	0.107
1.00	0.010	0.025	0.012	0.012	0.017	0.010	0.012	0.000	0.000	0.010	0.002	0.012	0.007	0.017	0.012	0.017	0.180
1.25	0.020	0.042	0.040	0.022	0.017	0.020	0.015	0.012	0.010	0.007	0.002	0.005	0.002	0.017	0.017	0.045	0.297
1.50	0.035	0.062	0.062	0.012	0.027	0.037	0.020	0.010	0.002	0.007	0.012	0.005	0.010	0.007	0.025	0.032	0.369
2.00	0.082	0.225	0.137	0.050	0.152	0.162	0.115	0.040	0.012	0.015	0.017	0.007	0.005	0.042	0.050	0.102	1.216
3.00	0.182	0.187	0.150	0.280	0.319	0.404	0.309	0.132	0.087	0.015	0.040	0.007	0.002	0.025	0.125	0.172	2.439
4.00	0.040	0.050	0.022	0.072	0.055	0.067	0.027	0.097	0.072	0.005	0.020	0.002	0.002	0.005	0.055	0.092	0.686
5.00	0.015	0.005	0.000	0.002	0.005	0.002	0.007	0.025	0.015	0.005	0.007	0.000	0.000	0.000	0.002	0.012	0.105
6.00	0.000	0.002	0.000	0.000	0.002	0.002	0.002	0.002	0.012	0.002	0.002	0.000	0.000	0.000	0.000	0.002	0.032
8.00	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.010	0.005	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.015
10.00	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.002	0.000	0.000	0.000	0.000	0.000	0.000	0.002
44.70	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
Total	0.40	0.61	0.45	0.46	0.61	0.72	0.52	0.34	0.22	0.08	0.11	0.05	0.04	0.12	0.29	0.49	5.50

Table 4.1.1-18 Wind Direction Occurrence Frequency

Wind Direction	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
Frequency	7.0	4.6	4.0	4.7	5.5	6.0	7.4	13.4	13.9	7.1	3.3	2.2	2.7	3.5	6.2	8.5
Total Frequency = 100																

Table 4.1.1-19 Wind Speed Occurrence Frequency

Maximum Wind Speed (m/s):	0.34	0.50	0.75	1.00	1.25	1.50	2.00	3.00	4.00	5.00	6.00	8.00	10.00	44.70
Wind Speed Frequency:	0.02	0.11	0.42	0.92	1.50	2.20	6.91	19.48	19.02	15.50	11.86	14.22	5.75	2.08
Total Frequency = 100														

Table 4.1.1-20 Relative Concentration (χ/Q) Results at Exclusion Area Boundary for Release from Sources near Containment

Downwind Distance							
Sector	**(meters)**	**0-2 hours**	**0-8 hours**	**8-24 hours**	**1-4 days**	**4-30 days**	**Annual Average**
S	1200.	1.18E-04	5.69E-05	3.95E-05	1.79E-05	5.73E-06	1.43E-06
SSW	1200.	1.38E-04	6.50E-05	4.47E-05	1.98E-05	6.18E-06	1.48E-06
SW	1200.	1.32E-04	6.25E-05	4.29E-05	1.90E-05	5.90E-06	1.41E-06
WSW	1200.	1.27E-04	6.04E-05	4.16E-05	1.85E-05	5.81E-06	1.41E-06
W	1200.	1.32E-04	6.37E-05	4.42E-05	2.00E-05	6.43E-06	1.60E-06
WNW	1200.	1.36E-04	6.59E-05	4.60E-05	2.10E-05	6.84E-06	1.73E-06
NW	1200.	1.29E-04	6.36E-05	4.48E-05	2.09E-05	6.99E-06	1.83E-06
NNW	1200.	1.10E-04	5.73E-05	4.13E-05	2.03E-05	7.35E-06	2.12E-06
N	1200.	8.77E-05	4.54E-05	3.27E-05	1.60E-05	5.75E-06	1.64E-06
NNE	1200.	6.48E-05	3.23E-05	2.28E-05	1.07E-05	3.62E-06	9.60E-07
NE	1200.	6.76E-05	3.20E-05	2.20E-05	9.80E-06	3.06E-06	7.36E-07
ENE	1200.	4.12E-05	1.95E-05	1.34E-05	5.93E-06	1.85E-06	4.43E-07
E	1200.	4.48E-05	2.07E-05	1.41E-05	6.12E-06	1.84E-06	4.25E-07
ESE	1200.	5.26E-05	2.49E-05	1.72E-05	7.63E-06	2.39E-06	5.76E-07
SE	1200.	9.86E-05	4.64E-05	3.19E-05	1.41E-05	4.35E-06	1.04E-06
SSE	1200.	1.26E-04	6.09E-05	4.23E-05	1.91E-05	6.13E-06	1.52E-06
Maximum sector χ/Q		1.38E-04					
Overall site χ/Q		1.31E-04	6.64E-05	4.72E-05	2.25E-05	7.77E-06	2.12E-06

Table 4.1.1-21 Relative Concentration (χ/Q) Results at Low Population Zone for Release from Sources near Containment

Downwind Distance							
Sector	**(meters)**	**0-2 hours**	**0-8 hours**	**8-24 hours**	**1-4 days**	**4-30 days**	**Annual Average**
S	4023.	3.49E-05	1.53E-05	1.01E-05	4.14E-06	1.15E-06	2.38E-07
SSW	4023.	4.50E-05	1.90E-05	1.24E-05	4.87E-06	1.28E-06	2.48E-07
SW	4023.	4.21E-05	1.79E-05	1.17E-05	4.60E-06	1.21E-06	2.36E-07
WSW	4023.	3.89E-05	1.68E-05	1.10E-05	4.42E-06	1.19E-06	2.40E-07
W	4023.	4.26E-05	1.85E-05	1.22E-05	4.92E-06	1.34E-06	2.72E-07
WNW	4023.	4.33E-05	1.90E-05	1.26E-05	5.15E-06	1.43E-06	2.97E-07
NW	4023.	4.10E-05	1.83E-05	1.23E-05	5.12E-06	1.46E-06	3.16E-07
NNW	4023.	3.06E-05	1.47E-05	1.02E-05	4.58E-06	1.46E-06	3.60E-07
N	4023.	2.20E-05	1.07E-05	7.41E-06	3.37E-06	1.09E-06	2.72E-07
NNE	4023.	1.58E-05	7.37E-06	5.04E-06	2.21E-06	6.76E-07	1.59E-07
NE	4023.	1.67E-05	7.43E-06	4.95E-06	2.06E-06	5.84E-07	1.25E-07
ENE	4023.	9.61E-06	4.29E-06	2.87E-06	1.19E-06	3.40E-07	7.30E-08
E	4023.	9.32E-06	4.15E-06	2.77E-06	1.15E-06	3.26E-07	6.96E-08
ESE	4023.	1.11E-05	5.03E-06	3.39E-06	1.44E-06	4.22E-07	9.38E-08
SE	4023.	2.67E-05	1.16E-05	7.64E-06	3.09E-06	8.44E-07	1.72E-07
SSE	4023.	3.88E-05	1.69E-05	1.11E-05	4.51E-06	1.23E-06	2.53E-07
Maximum sector χ/Q		4.50E-05					
Site Limit		4.19E-05	1.91E-05	1.29E-05	5.49E-06	1.61E-06	3.60E-07

Table 4.1.1-22 Relative Concentration (χ/Q) Results at Exclusion Area Boundary for Release from RWST

Downwind Distance							
Sector	**(meters)**	**0-2 hours**	**0-8 hours**	**8-24 hours**	**1-4 days**	**4-30 days**	**Annual Average**
S	1200.	1.22E-04	5.86E-05	4.05E-05	1.82E-05	5.80E-06	1.43E-06
SSW	1200.	1.40E-04	6.58E-05	4.52E-05	2.00E-05	6.21E-06	1.48E-06
SW	1200.	1.34E-04	6.30E-05	4.32E-05	1.91E-05	5.92E-06	1.41E-06
WSW	1200.	1.28E-04	6.06E-05	4.17E-05	1.86E-05	5.82E-06	1.41E-06
W	1200.	1.34E-04	6.45E-05	4.47E-05	2.02E-05	6.46E-06	1.60E-06
WNW	1200.	1.37E-04	6.67E-05	4.64E-05	2.12E-05	6.87E-06	1.73E-06
NW	1200.	1.30E-04	6.42E-05	4.51E-05	2.10E-05	7.01E-06	1.83E-06
NNW	1200.	1.17E-04	6.02E-05	4.32E-05	2.10E-05	7.49E-06	2.12E-06
N	1200.	9.90E-05	5.03E-05	3.58E-05	1.72E-05	5.97E-06	1.64E-06
NNE	1200.	7.60E-05	3.69E-05	2.57E-05	1.17E-05	3.81E-06	9.60E-07
NE	1200.	6.84E-05	3.23E-05	2.22E-05	9.86E-06	3.07E-06	7.36E-07
ENE	1200.	4.67E-05	2.16E-05	1.47E-05	6.38E-06	1.92E-06	4.43E-07
E	1200.	4.50E-05	2.08E-05	1.42E-05	6.14E-06	1.85E-06	4.25E-07
ESE	1200.	5.28E-05	2.50E-05	1.72E-05	7.66E-06	2.39E-06	5.76E-07
SE	1200.	1.00E-04	4.71E-05	3.23E-05	1.42E-05	4.37E-06	1.04E-06
SSE	1200.	1.28E-04	6.15E-05	4.27E-05	1.93E-05	6.15E-06	1.52E-06
Maximum sector χ/Q		1.40E-04					
Overall site χ/Q		1.35E-04	6.77E-05	4.80E-05	2.28E-05	7.83E-06	2.12E-06

Table 4.1.1-23 Relative Concentration (χ/Q) Results at Low Population Zone for Release from RWST

Downwind Distance							
Sector	**(meters)**	**0-2 hours**	**0-8 hours**	**8-24 hours**	**1-4 days**	**4-30 days**	**Annual Average**
S	4023.	3.50E-05	1.53E-05	1.01E-05	4.14E-06	1.15E-06	2.38E-07
SSW	4023.	4.50E-05	1.90E-05	1.24E-05	4.87E-06	1.28E-06	2.48E-07
SW	4023.	4.21E-05	1.79E-05	1.17E-05	4.60E-06	1.21E-06	2.36E-07
WSW	4023.	3.89E-05	1.68E-05	1.10E-05	4.42E-06	1.19E-06	2.40E-07
W	4023.	4.26E-05	1.85E-05	1.22E-05	4.92E-06	1.34E-06	2.72E-07
WNW	4023.	4.33E-05	1.90E-05	1.26E-05	5.15E-06	1.43E-06	2.97E-07
NW	4023.	4.10E-05	1.83E-05	1.23E-05	5.12E-06	1.46E-06	3.16E-07
NNW	4023.	3.17E-05	1.51E-05	1.04E-05	4.68E-06	1.48E-06	3.60E-07
N	4023.	2.24E-05	1.08E-05	7.50E-06	3.40E-06	1.09E-06	2.72E-07
NNE	4023.	1.58E-05	7.37E-06	5.04E-06	2.21E-06	6.76E-07	1.59E-07
NE	4023.	1.67E-05	7.43E-06	4.95E-06	2.06E-06	5.84E-07	1.25E-07
ENE	4023.	9.61E-06	4.29E-06	2.87E-06	1.19E-06	3.40E-07	7.30E-08
E	4023.	9.32E-06	4.15E-06	2.77E-06	1.15E-06	3.26E-07	6.96E-08
ESE	4023.	1.11E-05	5.03E-06	3.39E-06	1.44E-06	4.22E-07	9.38E-08
SE	4023.	2.67E-05	1.16E-05	7.64E-06	3.09E-06	8.44E-07	1.72E-07
SSE	4023.	3.88E-05	1.69E-05	1.11E-05	4.51E-06	1.23E-06	2.53E-07
Maximum sector χ/Q		4.50E-05					
Overall site χ/Q		4.19E-05	1.91E-05	1.29E-05	5.49E-06	1.61E-06	3.60E-07

Table 4.1.1-24 χ/Q Values for Dose Assessment

Downwind Location		χ/Q values (s/m^3) For release from sources located close to containment structure, such as unit vent stack, equipment hatch, MSSVs/ARVs vent, or TDAFW exhaust vent				
Name	**Distance (meters)**	**0-2 hours**	**0-8 hours**	**8-24 hours**	**1-4 days**	**4-30 days**
EAB	1200	1.38E-4	6.64E-5	4.72E-5	2.25E-5	7.77E-6
LPZ	4023	4.50E-5	1.91E-5	1.29E-5	5.49E-6	1.61E-6
		χ/Q values (s/m^3) For release from RWST				
EAB	1200	1.40E-4	6.77E-5	4.80E-5	2.28E-5	7.83E-6
LPZ	4023	4.50E-5	1.91E-5	1.29E-5	5.49E-6	1.61E-6

Table 4.1.1-25 Bounding χ/Q Values for Dose Assessment

	χ/Q values (s/m³) Bounding values for release from any one of the following sources: unit vent stack, equipment hatch, MSSVs/ARVs vent, TDAFW exhaust vent, or RWST					
Location	**0-2 hours**	**2-8 hours[1]**	**0-8 hours**	**8-24 hours**	**1-4 days**	**4-30 days**
EAB[2]	1.40E-4					
LPZ[3]	4.50E-5	2.39E-5	1.91E-5	1.29E-5	5.49E-6	1.61E-6

Notes:
1. This value was not calculated by PAVAN. It was obtained by logarithmic interpolation between the 2-hour average value $(\chi/Q)_{2hr}$ and the annual average value $(\chi/Q)_{1yr}$.
2. Defined as 1200 m distance.
3. Defined as 4023 m distance.

4.1.1.7 References

1. NRC, Regulatory Guide 1.145, Rev. 1, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," November 1982.

2. Bander, T.J., PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations, NUREG/CR-2858, PNL-4413, November 1982.

3. NRC, Regulatory Guide 1.111, Rev. 1, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," July 1977.

4.1.2 Atmospheric Dispersion Factors – Control Room and Technical Support Center

4.1.2.1 Introduction

Atmospheric dispersion factors (χ/Q, sec/m^3) were determined for the air intake of the control room and the technical support center (TSC) from the following sources for the WCGS: (1) equipment hatch, (2) unit vent stack, (3) MSSVs/ARVs vent, (4) RWST vent, and (5) TD AFW exhaust vent. The calculation was performed with the ARCON96 code (Reference 1).

ARCON96 utilizes an NRC accepted methodology for determining atmospheric dispersion factors χ/Q in the design basis evaluations of control room radiological analyses (Reference 2). ARCON96 is a computer program for calculating atmospheric relative concentrations in plumes in building wakes under a wide range of situations. ARCON96 implements a straight line Gaussian dispersion model with dispersion coefficients that are modified to account for low wind meander and building wake effects. Hourly, normalized concentrations (χ/Q) are calculated from hourly meteorological data. The hourly values are averaged to form χ/Qs for periods ranging from 2 to 720 hours in duration. The calculated values for each period are used to form cumulative frequency distributions and 95th percentile χ/Q values.

All release sources were treated as point source, ground level releases in the calculation. The input parameters to the ARCON96 code were prepared consistent with the guidance on the use of ARCON96, as discussed in RG 1.194 (Reference 2).

4.1.2.2 Input Data and Parameters

Five years (2006 through 2010) of WCGS site-specific meteorological data were used in the calculation. These data include hourly observations of wind speed, wind direction, and a measure of atmospheric stability. A 15-minute average, for each of the monitoring system's parameters, was used for each hour of data. Values outside of reasonable ranges were removed from the data set and were annotated for future reference and processing. Since redundant instruments are used, if the data from the primary instrument was reporting bad data, the data from the secondary instrument was used instead.

The NRC requires a minimum of data for 1 complete year and considers 5 years (43,824 hours) of hourly data to be representative of long-term trends at most sites. The actual number of valid data hours used in

the calculation was 40,065. The total number of invalid data hours was 3759. The data recovery for the 5-year period met the 90% recovery criterion. There was only one large gap in recorded data. The missing data occurred when the meteorological data logger failed and had to be replaced. Due to this issue, there is no meteorological data from May 30 2007 to June 7 2007. The lower measurement height was 13.47 m above grade, and the upper measurement was 63.47 m.

For the ARCON96 input file, the format discussed in RG 1.194 was followed. The wind speed was entered in units of meters per second, the wind direction was entered in units of degrees with 360° corresponding to a wind from the north, and the stability class, as a function of temperature difference, was entered. Bad data was indicated by entering all 9s in the parameter fields.

A number of plant-specific geometric parameters were calculated for individual source-receptor pairs. These parameters are: (1) release height, (2) distance to receptor, and (3) direction to source. The values recommended by RG 1.194 are used for the dispersion model parameters required by the ARCON96 code.

The complete sets of input parameters for individual cases are summarized in Table 4.1.2-1 for the control room and Table 4.1.2-2 for the TSC.

Table 4.1.2-1 Input Parameters for Control Room Vent Intake

Input Parameter	From Equipment Hatch	From Unit Vent Stack	From MSSVs/ ARVs Vent	From RWST Vent	From Turbine-Driven AFW Exhaust Vent
Meteorological data	Determined from data collected: 2006 to 2010				
Height of lower wind speed instrument	13.47 m				
Height of upper wind speed instrument	63.47 m				
Release type	Ground				
Release height	17.37 m	66.25 m	34.29 m	17.39 m	13.87 m
Building area perpendicular to wind direction	2649 m^2			170.94 m^2	2649 m^2
Effluent vertical velocity	0 m/sec				
Vent or stack flow	0 m^3/sec				
Vent or stack radius	0 m				
Direction to source	115°	111°	90°	153°	89°
Wind direction sector width	90°				
Distance to control room air intake	113.17 m	79.01 m	75.66 m	101.71 m	111.95 m

Table 4.1.2-1 Input Parameters for Control Room Vent Intake (cont.)

Input Parameter	From Equipment Hatch	From Unit Vent Stack	From MSSVs/ ARVs Vent	From RWST Vent	From Turbine-Driven AFW Exhaust Vent
Control room air intake height	6.10 m				
Terrain elevation difference	0 m				
Minimum wind speed	0.5 m/sec				
Surface roughness length	0.2 m				
Sector averaging constant	4.3				
Initial values of sigma y and sigma z	0, 0, respectively				

Table 4.1.2-2 Input Parameters for TSC Vent Intake

Input Parameter	From Equipment Hatch	From Unit Vent Stack	From MSSVs/A RVs Vent	From RWST Vent	From Turbine-Driven AFW Exhaust Vent
Meteorological data	Determined from data collected: 2006 to 2010				
Height of lower wind speed instrument	13.47 m				
Height of upper wind speed instrument	63.47 m				
Release type	Ground				
Release height	17.37 m	66.25 m	35.71 m	17.39 m	13.87 m
Building area perpendicular to wind direction	2649 m^2			2649 m^2	2649 m^2
Effluent vertical velocity	0 m/sec				
Vent or stack flow	0 m3/sec				
Vent or stack radius	0 m				
Direction to source	222°	232°	243°	222°	241°
Wind direction sector width	90°				
Distance to TSC air intake	124.28 m	129.29 m	108.26 m	187.33 m	106.67 m
TSC air intake height	2.84 m				
Terrain elevation difference	0 m				
Minimum wind speed	0.5 m/sec				
Surface roughness length	0.2 m				
Sector averaging constant	4.3				
Initial values of sigma y and sigma z	0, 0, respectively				

4.1.2.3 Assumptions and Acceptance Criteria

Assumptions

1. Ground level release is assumed for all cases.

2. The source type for all cases is assumed to be a point source.

3. The shortest source-to-receptor distances are determined from the closest point on the perimeter of the source to the closest point on the perimeter of the receptor for all cases. In the case of the unit vent stack, one corner of the rectangular-shaped unit vent stack is assumed to point in the direction of the control room air intake to yield a conservatively short distance from the unit vent stack to the control room air intake. Similarly, it is also assumed that another corner of the unit vent stack points in the direction of the TSC air intake to yield a conservatively short distance from the unit vent stack to the TSC air intake.

4. Paths that traverse the perimeter of the reactor building cylinder conservatively ignore the buttresses.

5. The MSSV and ARV vents are treated as a single point source. The closest vent is chosen to represent the source for the purposes of calculating the distance to the receiver location. Because the MSSV and ARV vents have different discharge elevations, this results in different release heights being used for the control room and TSC calculations.

6. For the control room calculation, it is assumed that the MSSV/ARV release point exists at a distance west of the reactor building centerline equal to the most westward point that exists at the exit of the MSSV vents. The release point is further assumed to be the same distance north of the reactor building centerline as the receiver location. This yields a conservatively short distance between the MSSV/ARV release point and the control room air intake.

Acceptance Criteria

The acceptance criteria for determining control room χ/Q values are that the calculation is performed consistent with the methodology and guidelines of RG 1.194. Since the χ/Q values are used as input to the downstream analyses, there are no numerical acceptance criteria for χ/Q values.

4.1.2.4 Results and Conclusions

The results for all source-receptor pairs of the 95th percentile χ/Q values averaged over periods of 0 to 2 hours, 2 to 8 hours, 8 to 24 hours, 1 to 4 days, and 4 to 30 days are summarized in Table 4.1.2-3 for the control room and in Table 4.1.2-4 for the TSC.

4.1.2.5 References

1. Ramsdell, J. V. and Simonen, C. A. Atmospheric Relative Concentrations in Building Wakes. ARCON96 Computer Code User's Guide. : Pacific Northwest National Laboratory (PNNL), May 1997. NUREG/CR-6331, PNNL-10521, Rev. 1.

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

Table 4.1.2-3 Calculated χ/Q Values for the Control Room

Equipment Hatch to Control Room Air Intake	
0 to 2 hours	5.31E-04
2 to 8 hours	4.22E-04
8 to 24 hours	1.72E-04
1 to 4 days	1.22E-04
4 to 30 days	9.91E-05
Unit Vent Stack to Control Room Air Intake	
0 to 2 hours	4.41E-04
2 to 8 hours	3.21E-04
8 to 24 hours	1.35E-04
1 to 4 days	8.66E-05
4 to 30 days	7.70E-05
MSSVs/ARVs to Control Room Air Intake	
0 to 2 hours	9.24E-04
2 to 8 hours	6.89E-04
8 to 24 hours	2.94E-04
1 to 4 days	1.88E-04
4 to 30 days	1.56E-04
RWST Vent to Control Room Air Intake	
0 to 2 hours	6.77E-04
2 to 8 hours	5.96E-04
8 to 24 hours	2.42E-04
1 to 4 days	2.04E-04
4 to 30 days	1.59E-04
TDAFW Exhaust to Control Room Air Intake	
0 to 2 hours	5.15E-04
2 to 8 hours	4.08E-04
8 to 24 hours	1.75E-04
1 to 4 days	1.10E-04
4 to 30 days	8.93E-05

Table 4.1.2-4 Calculated χ/Q Values for the TSC

Equipment Hatch to TSC Air Intake	
0 to 2 hours	3.91E-04
2 to 8 hours	2.66E-04
8 to 24 hours	9.62E-05
1 to 4 days	7.05E-05
4 to 30 days	5.52E-05
Unit Vent Stack to TSC Air Intake	
0 to 2 hours	2.80E-04
2 to 8 hours	1.80E-04
8 to 24 hours	6.44E-05
1 to 4 days	4.42E-05
4 to 30 days	3.22E-05
MSSVs/ARVs to TSC Air Intake	
0 to 2 hours	4.14E-04
2 to 8 hours	2.08E-04
8 to 24 hours	8.22E-05
1 to 4 days	5.21E-05
4 to 30 days	4.09E-05
RWST Vent to TSC Air Intake	
0 to 2 hours	1.87E-04
2 to 8 hours	1.25E-04
8 to 24 hours	4.51E-05
1 to 4 days	3.33E-05
4 to 30 days	2.61E-05
TDAFW Exhaust to TSC Air Intake	
0 to 2 hours	4.83E-04
2 to 8 hours	2.58E-04
8 to 24 hours	9.63E-05
1 to 4 days	6.45E-05
4 to 30 days	4.89E-05

4.2 ACCIDENT SOURCE TERMS

The inventory of radionuclides in the reactor core available for release into containment is based on a core thermal power of 3637 MWt (102 percent of rated thermal power). An additional fuel management multiplier is applied to the calculated core activity to account for anticipated variations in fuel cycle design. A list of the radionuclides used in the AST analysis is presented in Table 4.2-1. Using similar assumptions, reactor coolant system specific activity, volume control tank (VCT) liquid and vapor phase specific activity, gas decay tank activity, and liquid waste tank activity were calculated and are presented in Tables 4.2-2 through 4.2-5.

The source term information for use in the radiological consequence analyses was generated using the ORIGEN-S and FIPCO-V computer codes. The ORIGEN-S code is a versatile point depletion and radioactive decay computer code for use in simulating nuclear fuel cycles and calculating the nuclide compositions and characteristics of materials contained therein. The ORIGEN-S code is an industry standard code that incorporates data from modern evaluated nuclear data files.

The FIPCO-V computer code calculates the buildup of fission product activities in plant systems and components, including the reactor coolant system, chemical and volume control system demineralizer resins, VCT liquid and vapor phases, and waste gas decay tank (WGDT). The time-dependent inventory of the core fission products calculated by ORIGEN-S is used as input to the FIPCO-V evaluations. When calculating the radionuclide inventory in the VCT, no purging is assumed through the cycle. The activities for the WGDT are calculated assuming a maximum RCS letdown rate to degas the RCS by repeated purges of the VCT at end of cycle. The fuel management multiplier is applied to all results calculated by FIPCO-V.

The hypothetical liquid waste tank inventory is based on a series of hand calculations and is intended to bound the inventory of several smaller waste tanks (such as the recycle holdup tank, waste holdup tank, and floor drain tank).

Table 4.2-1 Core Inventory used in the AST Analysis

Core Inventory with Fuel Management Multiplier			
Nuclide	**Activity [Ci]**	**Nuclide**	**Activity [Ci]**
Noble Gases		**Barium and Strontium**	
Kr-85	1.10E+06	Sr-89	9.98E+07
Kr-85m	2.69E+07	Sr-90	8.51E+06
Kr-87	5.30E+07	Sr-91	1.25E+08
Kr-88	7.12E+07	Sr-92	1.33E+08
Xe-131m	1.05E+06	Ba-139	1.87E+08
Xe-133	2.01E+08	Ba-140	1.79E+08
Xe-133m	6.06E+06	**Noble Metals**	
Xe-134m	3.00E+06	Tc-99m	1.69E+08
Xe-135	4.06E+07	Mc-99	1.91E+08
Xe-135m	4.39E+07	Ru-103	1.56E+08
Xe-138	1.80E+08	Ru-105	1.08E+08
Halogens		Rh-105	1.00E+08
Br-83	1.24E+07	Ru-106	4.79E+07
Br-84	2.26E+07	**Lanthanides**	
Br-85	2.68E+07	Y-90	8.88E+06
I-129	2.57E+00	Y-91	1.30E+08
I-130	1.98E+06	Y-92	1.35E+08
I-131	1.01E+08	Y-93	1.52E+08
I-132	1.49E+08	Zr-95	1.74E+08
I-133	2.10E+08	Nb-95	1.76E+08
I-134	2.36E+08	Zr-97	1.75E+08
I-135	2.00E+08	La-140	1.85E+08
Alkali Metals		La-141	1.69E+08
Rb-86	1.86E+05	La-142	1.64E+08
Rb-88	7.24E+07	Pr-143	1.55E+08
Cs-134	1.65E+07	Nd-147	6.57E+07
Cs-136	3.95E+06	Am-241	9.94E+03
Cs-137	1.11E+07	Cm-242	2.82E+06
Cs-138	1.96E+08	Cm-244	2.11E+05
Tellurium		**Cerium**	
Te-127	9.09E+06	Ce-141	1.69E+08
Te-127m	1.49E+06	Ce-143	1.59E+08
Te-129	2.66E+07	Ce-144	1.30E+08
Te-129m	5.04E+06	Pu-238	2.10E+05
Te-131m	1.99E+07	Pu-239	2.70E+04
Te-132	1.45E+08	Pu-240	4.20E+04
Sb-127	9.23E+06	Pu-241	1.06E+07
Sb-129	2.85E+07	Np-239	1.94E+09

Table 4.2-2 Reactor Coolant System Specific Activity

Reactor Coolant System Specific Activity			
Nuclide	**S.A. [µCi/g]**	**Nuclide**	**S.A. [µCi/g]**
Br-83	9.86E-02	Sr-89	4.04E-03
Br-84	4.88E-02	SR-90	2.59E-04
Br-85	5.75E-03	Y-90	7.34E-05
I-127 [g]	1.24E-10	Y-91m	3.01E-03
I-129	7.17E-08	Sr-91	5.60E-03
I-130	4.65E-02	Y-91	5.64E-04
I-132	3.39E+00	Sr-92	1.31E-03
I-134	7.30E-01	Y-92	1.13E-03
Kr-83m	4.62E-01	Y-93	3.82E-04
Kr-85m	1.83E+00	Zr-95	7.02E-04
Kr-85	1.00E+01	Nb-95	7.03E-04
Kr-87	1.19E+00	Mo-99	8.94E-01
Kr-88	3.29E+00	Tc-99m	8.22E-01
Kr-89	9.30E-02	Ru-103	7.43E-04
I-131	3.28E+00	Rh-103m	7.44E-04
Xe-131m	3.74E+00	Ru-106	3.25E-04
Xe-133m	5.54E+00	Ag-110m	3.34E-03
I-133	5.04E+00	Te-125m	6.03E-04
Xe-133	3.08E+02	Te-127m	4.49E-03
Xe-135m	6.15E-01	Te-127	1.52E-02
I-135	2.85E+00	Te-129m	1.46E-02
Xe-135	8.12E+00	Te-129	1.63E-02
Xe-137	2.18E-01	Te-131m	4.18E-02
Xe-138	7.59E-01	Te-131	1.63E-02
Rb-86	4.33E-02	Te-132	3.43E-01
Rb-88	4.08E+00	Te-134	3.51E-02
Rb-89	1.89E-01	Ba-140	4.55E-03
Cs-134	4.82E+00	La-140	1.53E-03
Cs-136	4.35E+00	Ce-141	6.94E-04
Cs-137	2.68E+00	Ce-143	5.46E-04
Cs-138	1.16E+00	Pr-143	6.46E-04
		Ce-144	5.37E-04

Table 4.2-3 Volume Control Tank Liquid and Vapor Phase Specific Activity

Volume Control Tank Specific Activity					
Nuclide	**Liquid [µCi/g]**	**Vapor [µCi/cc]**	**Nuclide**	**Liquid [µCi/g]**	**Vapor [µCi/cc]**
Br-83	9.86E-03		Sr-89	4.04E-04	
Br-84	4.88E-03		SR-90	2.59E-05	
Br-85	5.75E-04		Y-90	7.34E-05	
I-127 [g]	1.24E-11		Y-91m	3.01E-03	
I-129	7.17E-09		Sr-91	5.60E-04	
I-130	4.65E-03		Y-91	5.64E-04	
I-131	3.28E-01		Sr-92	1.31E-04	
I-132	3.39E-01		Y-92	1.13E-03	
I-133	5.04E-01		Y-93	3.82E-04	
I-134	7.30E-02		Zr-95	7.02E-04	
I-135	2.85E-01		Nb-95	7.03E-04	
Kr-83m	1.10E-01	2.83E+00	Mo-99	8.94E-01	
Kr-85m	7.92E-01	1.65E+01	Tc-99m	8.22E-01	
Kr-85	1.00E+01	1.35E+01	Ru-103	7.43E-04	
Kr-87	2.13E-01	4.42E+00	Rh-103m	7.44E-04	
Kr-88	1.08E+00	2.24E+01	Ru-106	3.25E-04	
Kr-89	8.33E-04	1.73E-02	Ag-110m	3.34E-03	
Xe-131m	3.69E+00	5.15E+01	Te-125m	6.03E-04	
Xe-133m	5.16E+00	7.52E+01	Te-127m	4.49E-03	
Xe-133	2.98E+02	4.16E+03	Te-127	1.52E-02	
Xe-135m	3.73E-02	8.79E+00	Te-129m	1.46E-02	
Xe-135	5.68E+00	9.77E+01	Te-129	1.63E-02	
Xe-137	3.46E-03	4.85E-02	Te-131m	4.18E-02	
Xe-138	4.26E-02	5.98E-01	Te-131	1.63E-02	
Rb-86	4.33E-02		Te-132	3.43E-01	
Rb-88	4.08E+00		Te-134	3.51E-02	
Rb-89	1.89E-01		Ba-140	4.55E-04	
Cs-134	4.82E+00		La-140	1.53E-03	
Cs-136	4.35E+00		Ce-141	6.94E-04	
Cs-137	2.68E+00		Ce-143	5.46E-04	
Cs-138	1.16E+00		Pr-143	6.46E-04	
			Ce-144	5.37E-04	

Table 4.2-4 Waste Gas Decay Tank Activity	
Waste Gas Decay Tank Inventory	
Nuclide	**Activity [Ci]**
Kr-83m	1.92E+01
Kr-85	5.52E+03
Kr-85m	1.49E+02
Kr-87	3.00E+01
Kr-88	1.79E+02
Kr-89	1.18E-01
Xe-131m	1.07E+03
Xe-133	8.12E+04
Xe-133m	1.27E+03
Xe-135	1.02E+03
Xe-135m	5.97E+01
Xe-137	3.30E-01
Xe-138	4.06E+00
I-131	3.99E-02
I-132	2.30E-02
I-133	4.28E-02
I-134	4.96E-03
I-135	1.94E-02

Table 4.2-5 Liquid Waste Tank Activity	
Hypothetical Liquid Waste Tank Inventory	
Nuclide	**Activity [Ci]**
Br-83	2.28E-04
Br-84	5.55E-06
Br-85	5.32E-09
I-130	2.75E-03
I-131	2.75E+01
I-132	7.10E-03
I-133	8.07E-01
I-134	2.26E-04
I-135	4.88E-02

4.3 DOSE ANALYSES

4.3.1 Introduction

The licensing basis for the radiological consequences analyses for Chapter 15 of the Wolf Creek USAR is currently based on methodologies and assumptions that are derived from TID-14844 (Reference 1) and other early guidance, including a series of RGs and SRP chapters. That guidance was developed to be consistent with the release fractions and timing from the TID-14844 source term and the whole body and thyroid dose limits stated in 10 CFR 100.11.

RG 1.183 (Reference 2) provides guidance on application of ASTs in revising the accident source terms used in design basis radiological consequences analyses, as allowed by 10 CFR 50.67. The AST methodology as established in RG 1.183 is being used to calculate the offsite, control room, and TSC radiological consequences for WCGS to support the accident analysis methods transition. The following accidents are analyzed:

- Main steamline break (MSLB)
- Loss of non-emergency AC power (LOAC)
- Locked rotor
- Rod ejection
- Letdown line break
- Steam generator tube rupture (SGTR)
- Loss-of-coolant accident (LOCA)
- Waste gas decay tank failure
- Liquid waste tank failure
- Fuel handling accident (FHA)

Each accident and the specific input assumptions are described in detail in the following sections.

The analyses are performed using Version 3.03 of the RADTRAD computer code. RADTRAD is described in NUREG/CR-6604 (Reference 3) and its supplements (References 4 and 5). The RADTRAD code uses a combination of tables and/or numerical models of source term reduction phenomena to determine the time-dependent dose at user-specified locations for a given accident scenario. The code system also applies the inventory, decay chain, and dose conversion factor tables needed for the dose calculation. The RADTRAD code can be used to assess occupational radiation exposures, typically in the control room, to estimate offsite doses, and to estimate dose attenuation due to modification of a facility or accident sequence. The calculation models in RADTRAD are consistent with those outlined in RG 1.183.

4.3.2 Common Analysis Inputs and Assumptions

The inputs and assumptions in this section are common to all of the analyses discussed in Section 4.3 of this Enclosure. The accident-specific inputs and assumptions are discussed in Sections 4.3.3 through 4.3.12.

The total effective dose equivalent (TEDE) doses were determined at the EAB for the worst 2-hour interval, unless an exception is noted for the specific accident being analyzed. The TEDE doses at the LPZ are determined for the duration of activity release and doses to the control room personnel are determined for the duration of the event. The interval for determining control room doses may extend beyond the time when the releases are terminated. This accounts for the additional dose to the operators in the control room, which will continue for 30 days, consistent with RG 1.183.

The TEDE dose is equivalent to the committed effective dose equivalent (CEDE) from inhalation plus the deep dose equivalent (DDE) from external exposure. Effective dose equivalent (EDE) is used in lieu of DDE in determining the contribution of external dose to the TEDE, consistent with RG 1.183 guidance. The dose conversion factors (DCFs) used in determining the CEDE dose are from Table 2.1 of Federal Guidance Report (FGR) No. 11 (Reference 6) and the DCFs used in determining the EDE dose are from Table III.1 of FGR No. 12 (Reference 7). The DCFs are listed in Table 4.3-3a. For comparison to the AST parameters, the DCFs for current licensing basis (CLB) analyses are listed in Tables 4.3-3b for waste gas decay tank rupture and fuel handling accident and 4.3-3c for LOCA.

The breathing rates used in the offsite dose calculations are provided in Table 4.3-4a and the atmospheric dispersion factors are presented in Section 4.1.1 of this Enclosure. For comparison to the AST parameters, the breathing rates for CLB analyses are listed in Table 4.3-4b.

No credit is taken for the radioactive decay during release and transport or for cloud depletion by ground deposition during transport to the control room, EAB, or outer boundary of the LPZ.

The reactor coolant activity is based on operation with 1 percent fuel defects. The core and coolant activities are based on a core power of 3637 MWt, which is the nominal core power of 3565 MWt plus a 2 percent calorimetric uncertainty. The core activity at shutdown and coolant activities present in the primary and secondary systems during normal plant operation are listed in Table 4.3-1a. For comparison to the AST parameters, the core activities and coolant activities for CLB analyses are listed in Tables 4.3-1b, 4.3-1c (for SGTR) and 4.3-4c (for FHA).

4.3.2.1 Control Room Model

Parameters modeled in the control room personnel dose calculations are provided in Table 4.3-5. A listing of the CLB parameters is included in Table 4.3-5 for comparison to the AST parameters. Control room modeling with respect to compartments and flow configurations is consistent with that described in Wolf Creek USAR Section 15A.3.

At the start of all the events considered, the control room ventilation system is in normal mode. In this mode, unfiltered air from the environment enters the control building and control room. Receipt of a safety injection (SI) actuation signal or a high radiation signal from the control room air intake monitors will isolate the control room and initiate the emergency mode of operation, including a delay.

After emergency mode is initiated, outside air is brought into the control building through safety grade filters. Makeup air is brought into the control room via both trains of the control room filtration system, which draws in air from the control building. Unfiltered air also leaks into the control building and control room via assumed inleakage rates. In addition, a filtered recirculation flow is modeled for the control

room during emergency mode operation. Air in the control room and control building is discharged at flow rates to match the total inflow to the compartments.

The control room ventilation isolation signal starts both trains of the control room filtration system. However, a failure of one of the filtration fans is assumed at the start of emergency mode and a larger unfiltered inflow to the control room is assumed since only half of the makeup flow to the control room can pass through a filter. After a defined time of 90 minutes, operator action isolates the failed train and reduces the unfiltered inflow to the control, and consequently lowers the filtered inflow to the control building.

The unfiltered inleakages to the control room and control building that are assumed in each of the events discussed in Sections 4.3.3 through 4.3.12 of this Enclosure are 50 cfm and 400 cfm, respectively, as identified in Table 4.3-5. The analysis of these events demonstrates that the applicable regulatory control room dose limit is met.

4.3.2.2 Technical Support Center Model

Parameters modeled in the TSC personnel dose calculations are provided in Table 4.3-16. A listing of the CLB parameters is included in Table 4.3-16 for comparison to the AST parameters.

At the start of all the events considered, the TSC ventilation system is in normal mode. In this mode, unfiltered air from the environment enters the TSC. After emergency mode is manually initiated, outside makeup air is brought into the TSC through safety grade filters. Unfiltered air also leaks into the TSC via an assumed inleakage rate. In addition, a filtered recirculation flow is modeled during emergency mode operation. Air is discharged at a flow rate to match the total inflow to the compartment.

4.3.2.3 References

1. United States Atomic Energy Commission TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," March 1962.

2. NRC, Regulatory Guide 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

3. NUREG/CR-6604, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal and Dose Estimation," December 1997.

4. NUREG/CR-6604, Supplement 1, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal and Dose Estimation," June 1999.

5. NUREG/CR-6604, Supplement 2, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal and Dose Estimation," October 2002.

6. Environmental Protection Agency 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, September 1988.

7. Environmental Protection Agency Federal Guidance Report No. 12, "External Exposure to Radionuclides in Air, Water, and Soil," EPA-402-R-93-081, September 1993.

4.3.3 Main Steamline Break (USAR Chapter 15.1.5.3)

4.3.3.1 Introduction

The complete severance of a main steamline outside containment is assumed to occur. The affected steam generator (SG) will rapidly depressurize and release radionuclides initially contained in the secondary coolant to the outside atmosphere. Primary coolant activity transferred to the affected SG via tube leakage will be released to the outside atmosphere as well. A portion of the activity initially contained in the intact SGs is released to the atmosphere through the atmospheric relief valves (ARVs) and/or the main steam safety valves. A portion of the activity transferred to the intact SGs via tube leakage will also be released to the outside atmosphere. The steamline break outside containment will bound any break inside containment since the outside break provides a means for direct release to the environment.

4.3.3.2 Input Parameters and Assumptions

The analysis of the MSLB radiological consequences uses the analytical methods and assumptions outlined in RG 1.183, Appendix E (Reference 1). A summary of input parameters and assumptions is provided in Table 4.3-6. A listing of the CLB parameters is also included in Table 4.3-6 for comparison to the AST parameters.

4.3.3.2.1 Source Term

The initial RCS and SG activity is provided in Table 4.3-1a. Since no fuel failure results from the accident, pre-accident iodine spike, and accident-initiated iodine spike scenarios are considered, consistent with RG 1.183 (Reference 1). For the pre-accident iodine spike case, it is assumed that a reactor transient has occurred prior to the MSLB and has raised the RCS iodine concentration to the Technical Specification (TS) limit for a transient of 60 μCi/gm DE I-131 (i.e., 60 times the maximum equilibrium RCS concentration). For the accident-initiated iodine spike case, the reactor trip associated with the MSLB creates an iodine spike that is assumed to increase the iodine release rate from the fuel to the RCS to a value 500 times greater than the appearance rate corresponding to the maximum equilibrium RCS concentration of 1.0 μCi/gm DE I-131. The iodine appearance rates are conservatively calculated assuming maximum letdown flow with perfect cleanup. The duration of the accident-initiated iodine spike is assumed to be 8 hours.

The iodine activity concentration of the secondary coolant at the time the MSLB occurs is assumed to be at 0.1 μCi/gm DE I-131, which is 10 percent of the maximum equilibrium RCS concentration. The alkali metal activity in the RCS at the time the MSLB occurs is at a 1 percent fuel defect level and the activity in the secondary coolant is assumed to be in the same ratio as the primary-to-secondary iodine

concentrations. The noble gas activity concentration in the RCS at the time the accident occurs is based on the TS value of 500 μCi/gm DE Xe-133.

Iodine available for release to the atmosphere is assumed to be 97 percent elemental and 3 percent organic.

4.3.3.2.2 Release Models

The SG connected to the broken steamline (i.e., the faulted steam generator) is assumed to boil dry in the initial 2 minutes following the MSLB. The entire liquid inventory of this SG is assumed to be steamed off and all of the iodine and alkali metal initially in this SG is released to the environment. A conservative maximum value for the faulted SG secondary side mass is modeled.

The TS limits the primary-to-secondary leakage up to 1440 gpd (1 gpm) for accident-induced conditions. The total leakage is assumed to enter the faulted SG and is immediately released to the atmosphere with no credit for iodine or alkali metal retention in the SG. Leakage to the intact SGs is not required, although it is assumed to be 450 gpd, so that 150 gpd is modeled to each SG according to the normal operation primary-to-secondary leakage TS limit. Iodine and alkali metal in this leakage mixes with the inventory in the secondary side of the intact SGs, conservatively modeled with a minimum water mass. All primary-to-secondary leakage is modeled with density based on cooled liquid.

An iodine partition coefficient of 100 (Ci iodine/gm water) / (Ci iodine/gm steam) is applied to releases from the intact SGs. The release of alkali metals from the secondary side of the intact SGs is limited by applying the plant-specific moisture carryover factor of 0.25 percent to the steam releases. All noble gas activity carried over to the secondary side through SG tube leakage is assumed to be immediately released to the outside atmosphere.

Twelve hours after the accident, the residual heat removal (RHR) system is assumed to be placed into service for heat removal. After 12 hours, there are no further steam releases to the atmosphere from the intact SGs. Within 34 hours after the accident, the RCS has been cooled to below 212°F and there are no further steam releases to atmosphere from the faulted SG.

4.3.3.2.3 Control Room

In the event of an MSLB, the low steamline pressure SI setpoint will be reached almost immediately following the break. The SI signal causes the control room to switch from the normal operation mode to the emergency operation mode. The switchover is conservatively modeled at 90 seconds following event initiation, which includes a 60-second delay from the initiating signal. As discussed in Section 4.3.2.1, operator action is taken 90 minutes after event initiation to isolate the ventilation train with failed filtration.

4.3.3.3 Acceptance Criteria

The EAB and LPZ dose acceptance criterion for an MSLB with an assumed pre-accident iodine spike is 25 rem TEDE per RG 1.183, which is the 10 CFR 50.67 limit. The EAB and LPZ dose acceptance criterion for an MSLB with an assumed accident-initiated iodine spike is 2.5 rem TEDE per RG 1.183,

which is 10 percent of the 10 CFR 50.67 limit. The acceptance criterion for the control room dose is 5 rem TEDE per 10 CFR 50.67. The acceptance criterion for the TSC dose is 5 rem TEDE as allowed by GDC 19, in accordance with Reference 2.

The EAB doses are calculated for the worst 2-hour interval. The LPZ doses are calculated until all releases are terminated, which is the time to cool the RCS to 212°F (34 hours) used in the analysis. The control room and TSC doses are calculated for 30 days.

4.3.3.4 Results and Conclusions

The MSLB accident doses are listed below:

For the pre-accident iodine spike case:

- EAB 0.22 rem TEDE
- LPZ 0.13 rem TEDE
- Control room 0.75 rem TEDE
- TSC 0.31 rem TEDE

For the accident-initiated iodine spike case:

- EAB 0.64 rem TEDE
- LPZ 0.59 rem TEDE
- Control room 1.1 rem TEDE
- TSC 0.48 rem TEDE

The EAB doses reported are for the worst 2-hour interval, determined to be from 0 to 2 hours for the pre-accident iodine spike and from 7.8 to 9.8 hours for the accident-initiated iodine spike.

It is concluded that for an MSLB, the accident doses meet the applicable acceptance criteria.

4.3.3.5 References

1. NRC, Regulatory Guide 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

2. Supplement 1 to NUREG-0737, "Clarification of TMI Action Plan Requirements: Requirements for Emergency Response Capability," January 1983.

4.3.4 Loss of Non-Emergency AC Power (USAR Chapter 15.2.6.3)

4.3.4.1 Introduction

A LOAC is assumed to occur that trips the reactor coolant pumps (RCPs) and decreases forced flow through the reactor core. No fuel cladding damage or fuel melting is assumed to occur as a result of this accident. Due to the pressure differential between the primary and secondary systems and assumed

SG tube leakage, RCS activity passes from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through the ARVs and/or main steam safety valves. In addition, iodine and alkali metal activity is contained in the secondary coolant before the accident and some of this activity is released to the atmosphere as a result of steaming from the SGs following the accident.

4.3.4.2 Input Parameters and Assumptions

The analysis of the LOAC event is not discussed in RG 1.183 (Reference 1). However, the release pathway for this analysis is similar to the locked rotor event and the accident-initiated iodine spike is similar to the MSLB event. Therefore, release pathway models consistent with RG 1.183, Appendix G and accident-initiated iodine spiking models consistent with RG 1.183, Appendix E are applied to this analysis. A summary of input parameters and assumptions is provided in Table 4.3-7. A listing of the CLB parameters is also included in Table 4.3-7 for comparison to the AST parameters.

4.3.4.2.1 Source Term

The initial RCS and SG activity is provided in Table 4.3-1a. Since no fuel failure results from the accident, an accident-initiated iodine spike scenario is considered. The reactor trip associated with the LOAC creates an iodine spike that is assumed to increase the iodine release rate from the fuel to the RCS to a value 500 times greater than the appearance rate corresponding to the maximum equilibrium RCS concentration of 1.0 μCi/gm DE I-131. The iodine appearance rates are conservatively calculated assuming maximum letdown flow with perfect cleanup. The duration of the accident-initiated iodine spike is assumed to be 8 hours.

The iodine activity concentration of the secondary coolant at the time the LOAC occurs is assumed to be at 0.1 μCi/gm DE I-131, which is 10 percent of the maximum equilibrium RCS concentration. The alkali metal activity in the RCS at the time the LOAC occurs is at a 1 percent fuel defect level and the activity in the secondary coolant is assumed to be in the same ratio as the primary-to-secondary iodine concentrations. The noble gas activity concentration in the RCS at the time the accident occurs is based on the TS value of 500 μCi/gm DE Xe-133.

Iodine available for release to the atmosphere is assumed to be 97 percent elemental and 3 percent organic.

4.3.4.2.2 Release Models

The primary-to-secondary accident-induced SG tube leakage to the four SGs is assumed to be at the TS maximum of 1 gpm total. The iodine and alkali metals in the leakage mixes with the inventory in the secondary side of the SGs, conservatively modeled with a minimum water mass. The minimum SG water mass is increased after 2 hours to take credit for operators maintaining level at narrow range just on span. All primary-to-secondary leakage is modeled with a density based on cooled liquid.

An iodine partition coefficient of 100 (Ci iodine/gm water)/(Ci iodine/gm steam) is applied to releases from the SGs. The release of alkali metals from the secondary side is limited by applying the plant-specific moisture carryover factor of 0.25 percent to the steam releases. All noble gas activity

carried over to the secondary side through SG tube leakage is assumed to be immediately released to the outside atmosphere.

Twelve hours after the accident is initiated, the RHR system is assumed to be placed into service for heat removal. After 12 hours, there are no further steam releases to the atmosphere.

4.3.4.2.3 Control Room

The control room intake radiation monitor signal setpoint is not reached in the analysis. As a result, the control room ventilation system remains in normal mode operation.

4.3.4.3 Acceptance Criteria

The EAB and LPZ dose acceptance criterion for a LOAC is 0.1 rem TEDE, consistent with 10 CFR 20. The control room dose acceptance criterion is 5 rem TEDE per 10 CFR 50.67. The acceptance criterion for the TSC dose is 5 rem TEDE as allowed by GDC 19, in accordance with Reference 2.

The EAB and LPZ doses are calculated until all releases are terminated, which is the RHR cut-in time (12 hours) used in the analysis. The control room and TSC doses are calculated for 30 days.

4.3.4.4 Results and Conclusions

The LOAC accident doses are listed below:

- EAB 0.033 rem TEDE
- LPZ 0.0036 rem TEDE
- Control room 0.11 rem TEDE
- TSC 0.0034 rem TEDE

It is concluded that for a LOAC, the accident doses meet the applicable acceptance criteria.

4.3.4.5 References

1. NRC, Regulatory Guide 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

2. Supplement 1 to NUREG-0737, "Clarification of TMI Action Plan Requirements: Requirements for Emergency Response Capability," January 1983.

4.3.5 Locked Rotor (USAR Chapter 15.3.3.3)

4.3.5.1 Introduction

An instantaneous seizure of an RCP is assumed to occur, which rapidly reduces flow through the affected reactor coolant loop. Fuel cladding damage is predicted to occur as a result of this accident. Due to the pressure differential between the primary and secondary systems and assumed SG tube leakage, fission

products pass from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through the ARVs and/or safety valves.

4.3.5.2 Input Parameters and Assumptions

The analysis of the locked rotor radiological consequences uses the analytical methods and assumptions outlined in RG 1.183, Appendix G (Reference 1). A summary of input parameters and assumptions is provided in Table 4.3-8. A listing of the CLB parameters is also included in Table 4.3-8 for comparison to the AST parameters.

4.3.5.2.1 Source Term

The core fission product activity is provided in Table 4.3-1a. The analysis assumes that 5 percent of the fuel rods in the core fail or suffer damage as a result of the locked rotor sufficient that all of their gap activity is released to the RCS. The calculation uses gap fractions of 8 percent for I-131, 10 percent for Kr-85, 12 percent for alkali metals, and 5 percent for all other nuclides. In the calculation of activity releases from the damaged fuel, the maximum radial peaking factor of 1.65 is applied.

The calculation does not model the initial RCS or secondary side system activities as the activities are insignificant compared to the core activity from failed fuel.

Iodine available for release to the atmosphere is assumed to be 97 percent elemental and 3 percent organic.

4.3.5.2.2 Release Models

The primary-to-secondary accident-induced SG tube leakage to the four SGs is assumed to be at the TS maximum of 1 gpm total. The iodine in the leakage mixes with the inventory in the secondary side of the SGs, conservatively modeled with a minimum water mass. The minimum SG water mass is increased after 2 hours to take credit for operators maintaining level at narrow range just on span. All primary-to-secondary leakage is modeled with a density based on cooled liquid.

An iodine partition coefficient of 100 (Ci iodine/gm water) / (Ci iodine/gm steam) is applied to releases from the SGs. The release of alkali metals from the secondary side is limited by applying the plant-specific moisture carryover factor of 0.25 percent to the steam releases. All noble gas activity carried over to the secondary side through SG tube leakage is assumed to be immediately released to the outside atmosphere.

Twelve hours after the accident is initiated, the RHR system is assumed to be placed into service for heat removal. After 12 hours, there are no further steam releases to the atmosphere.

4.3.5.2.3 Control Room

The control room intake radiation monitor signal setpoint is not reached in the analysis. As a result, the control room ventilation system remains in normal mode operation.

4.3.5.3 Acceptance Criteria

The EAB and LPZ dose acceptance criterion for a locked rotor is 2.5 rem TEDE per RG 1.183, which is 10 percent of the 10 CFR 50.67 limit. The control room dose acceptance criterion is 5 rem TEDE per 10 CFR 50.67. The acceptance criterion for the TSC dose is 5 rem TEDE as allowed by GDC 19, in accordance with Reference 2.

The EAB dose is calculated for the worst 2-hour interval. The LPZ dose is calculated until all releases are terminated, which is the RHR cut-in time (12 hours) used in the analysis. The control room and TSC doses are calculated for 30 days.

4.3.5.4 Results and Conclusions

A comparison of the source term, release models, and control room discussions shows that the locked rotor accident is bounded by the primary-to-secondary release pathway for the rod ejection accident described in Section 4.3.6. The inputs and assumptions are the same with the exception of gap fractions and the amount of fuel rods that suffer damage. The rod ejection accident source term bounds the source term for the locked rotor accident due to a higher number of rods that experience fuel damage and higher gap fractions. As a result, the rod ejection accident results are compared against the locked rotor accident acceptance criteria.

The locked rotor accident doses, which are the rod ejection doses from secondary side releases, are listed below:

- EAB 0.38 rem TEDE
- LPZ 0.30 rem TEDE
- Control room 4.7 rem TEDE
- TSC 0.16 rem TEDE

The EAB doses reported are for the worst 2-hour interval, determined to be from 0 to 2 hours.

It is concluded that for a locked RCP rotor, the accident doses meet the applicable acceptance criteria.

4.3.5.5 References

1. NRC, Regulatory Guide 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

2. Supplement 1 to NUREG-0737, "Clarification of TMI Action Plan Requirements: Requirements for Emergency Response Capability," January 1983.

4.3.6 Rod Ejection (USAR Chapter 15.4.8.3)

4.3.6.1 Introduction

It is assumed that a mechanical failure of a control rod mechanism pressure housing has occurred, resulting in the ejection of a rod cluster assembly and drive shaft. As a result of the accident, some fuel cladding damage and a small amount of fuel melt are assumed to occur. Due to the pressure differential between the primary and secondary systems and the assumed SG tube leakage, radioactive reactor coolant passes from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through the ARVs and/or the main steam safety valves (MSSVs). Finally, radioactive reactor coolant is discharged to the containment via the spill from the opening in the reactor vessel head. A portion of this radioactivity is released through containment leakage to the environment.

4.3.6.2 Input Parameters and Assumptions

The analysis of the rod ejection radiological consequences uses the analytical methods and assumptions outlined in RG 1.183, Appendix H (Reference 1). A summary of input parameters and assumptions is provided in Table 4.3-9. A listing of the CLB parameters is also included in Table 4.3-9 for comparison to the AST parameters.

4.3.6.2.1 Source Term

The core fission product activity is provided in Table 4.3-1a. In determining the activity released from the core following the rod ejection accident, it is assumed that 10 percent of the fuel rods in the core fail or suffer damage due to departure from nucleate boiling (DNB) such that all of their gap activity is released. Ten percent of the total core activity of iodine and noble gases and 12 percent of the total core activity of alkali metals is assumed to be in the fuel-cladding gap.

A small fraction of the fuel in the damaged fuel rods is assumed to melt as a result of the rod ejection accident equal to 0.25 percent of the core. This is based on the assumption that 50 percent of the rods in DNB undergo centerline melting, with the melting limited to the inner 10 percent and occurring over 50 percent of the axial length of the affected rods. All noble gas and 50 percent of the iodine and alkali metal activity contained in the melted fuel are released.

In the calculation of activity releases from the damaged/melted fuel, the maximum radial peaking factor of 1.65 is applied.

The calculation does not model the initial RCS or secondary side system activities because the activities are insignificant compared to the core activity from failed fuel.

Iodine available for release to the atmosphere is assumed to be 97 percent elemental and 3 percent organic for the primary-to-secondary release pathway and 95 percent particulate, 4.85 percent elemental, and 0.15 percent organic for the containment leakage pathway.

4.3.6.2.2 Release Models

4.3.6.2.2.1 Containment Leakage

The containment is assumed to leak at the design leak rate of 0.2 percent per day for the first 24 hours of the accident and then to leak at half that rate (0.1 percent per day) for the remainder of the 30-day period following the accident considered in the analysis.

For the containment leakage pathway, no credit is taken for any processes that would remove airborne activity.

4.3.6.2.2.2 Secondary Releases

The primary-to-secondary accident-induced SG tube leakage to the four SGs is assumed to be at the TS maximum of 1 gpm total. The iodine and alkali metals in the leakage mix with the inventory in the secondary side of the SGs, conservatively modeled with a minimum water mass. The minimum SG water mass is increased after 2 hours to take credit for operators maintaining level at narrow range just on span. All primary-to-secondary leakage is modeled with a density based on cooled liquid.

An iodine partition coefficient of 100 (Ci iodine/gm water)/(Ci iodine/gm steam) is applied to releases from the SGs. The release of alkali metals from the secondary side is limited by applying the plant-specific moisture carryover factor of 0.25 percent to the steam releases. All noble gas activity carried over to the secondary side through SG tube leakage is assumed to be immediately released to the outside atmosphere.

Twelve hours after the accident is initiated, the RHR system is assumed to be placed into service for heat removal. After 12 hours, there are no further steam releases to the atmosphere.

4.3.6.2.3 Control Room

In the event of a rod ejection that results in releases to the containment atmosphere, the low pressurizer pressure SI setpoint will be reached within 150 seconds of event initiation due to the loss of RCS mass through the hole in the upper head. The SI signal causes the control room to switch from the normal operation mode to the emergency operation mode. The switch is modeled at 210 seconds following event initiation, which includes a 60-second delay from the initiating signal. As discussed in Section 4.3.2.1, operator action is taken 90 minutes after event initiation to isolate the ventilation train with failed filtration.

For the primary-to-secondary leakage case, where the RCS is assumed intact, there is no SI and the control room intake radiation monitor signal setpoint is not reached in the analysis. As a result, the control room ventilation system remains in normal mode operation.

4.3.6.3 Acceptance Criteria

The EAB and LPZ dose acceptance criterion for a rod ejection is 6.3 rem TEDE per RG 1.183, which is approximately 25 percent of the 10 CFR 50.67 limit. The control room dose acceptance criterion is 5 rem TEDE per 10 CFR 50.67. The acceptance criterion for the TSC dose is 5 rem TEDE as allowed by GDC 19, in accordance with Reference 2. The limits apply to the containment leakage and primary-to-secondary leakage pathways separately.
The EAB doses are calculated for the worst 2-hour interval. The LPZ dose is calculated for 30 days in the containment leakage case and until all releases are terminated, which is the RHR cut-in time (12 hours) for the primary-to-secondary leakage case. The control room and TSC doses are calculated for 30 days.

4.3.6.4 Results and Conclusions

The rod ejection accident doses are listed below:

For the containment leakage case:

- EAB 1.2 rem TEDE
- LPZ 2.0 rem TEDE
- Control room 1.7 rem TEDE
- TSC 2.2 rem TEDE

For the primary-to-secondary leakage case:

- EAB 0.38 rem TEDE
- LPZ 0.30 rem TEDE
- Control room 4.7 rem TEDE
- TSC 0.16 rem TEDE

The EAB doses reported are for the worst 2-hour interval, determined to be from 0 to 2 hours for both the containment leakage case and the primary-to-secondary leakage case.

It is concluded that for a rod ejection, the accident doses meet the applicable acceptance criteria.

4.3.6.5 References

1. NRC, Regulatory Guide 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

2. Supplement 1 to NUREG-0737, "Clarification of TMI Action Plan Requirements: Requirements for Emergency Response Capability," January 1983.

4.3.7 Letdown Line Break (USAR Chapter 15.6.2.1)

4.3.7.1 Introduction

The analysis models the complete severance of the letdown line outside of containment. The occurrence of a complete severance of the letdown line would result in a loss of reactor coolant at the rate of 141 gpm with a time for the operations personnel to isolate the letdown line of 30 minutes, 10 seconds. The letdown line break is used as a representative scenario for all small lines transporting reactor coolant inventory outside of containment as it poses the most severe consequences regarding radioactivity release based upon break size.

4.3.7.2 Input Parameters and Assumptions

The letdown line break event is not discussed in RG 1.183 (Reference 1); however, RG 1.183 models are applied to this analysis in conjunction with event guidance from SRP 15.6.2 (Reference 2). A summary of input parameters and assumptions is provided in Table 4.3-10. A listing of the CLB parameters is also included in Table 4.3-10 for comparison to the AST parameters.

4.3.7.2.1 Source Term

The initial RCS activity is provided in Table 4.3-1a. No fuel failure results from the accident. Consistent with SRP 15.6.2, an accident-initiated spike in the RCS is considered. The iodine spike increases the iodine release rate from the fuel to the RCS to a value 500 times greater than the appearance rate corresponding to the maximum equilibrium RCS concentration of 1.0 μCi/gm of DE I-131. The iodine appearance rates are conservatively calculated assuming maximum letdown flow with perfect cleanup. The duration of the accident-initiated iodine spike has no impact on the results beyond the time at which the release is terminated since the release is terminated well before the assumed 8-hour spike ends.

The noble gas activity concentration in the RCS at the time the accident occurs is based on the TS value of 500 μCi/gm of DE Xe-133. The alkali metal activity concentration in the RCS is at a 1 percent fuel defect level.

Iodine available for release to the atmosphere is assumed to be 97 percent elemental and 3 percent organic.

4.3.7.2.2 Release Model

Reactor coolant is assumed to be released at a rate of 141 gpm until the isolation valve is fully closed. The time required for the operator to identify the accident and close the letdown isolation valve is expected to be within 30 minutes, 10 seconds after accident initiation. It is assumed that 18 percent of the leaking coolant flashes to steam, based on the temperature and pressure conditions of the letdown line flow. Only the iodine and alkali metal in this steam is assumed to become airborne and is available for release to the atmosphere, whereas all noble gases contained in the leaking primary coolant are available for release to the atmosphere.

4.3.7.2.3 Control Room

The control room intake radiation monitor signal setpoint is not reached in the analysis. As a result, the control room ventilation system remains in normal mode operation.

4.3.7.3 Acceptance Criteria

The EAB and LPZ dose acceptance criterion for a letdown line break outside containment is 2.5 rem TEDE, which is a small fraction (i.e., 10 percent) of the 10 CFR 50.67 limit by applying the same basis of acceptance from SRP 15.6.2. The control room dose acceptance criterion is 5 rem TEDE per 10 CFR 50.67. The acceptance criterion for the TSC dose is 5 rem TEDE as allowed by GDC 19, in accordance with Reference 3.

The EAB dose is calculated for the worst 2-hour interval. The LPZ dose is calculated until all releases are terminated, which is the letdown line isolation time (30 minutes, 10 seconds) used in the analysis. The control room and TSC doses are calculated for 30 days.

4.3.7.4 Results and Conclusions

The letdown line break accident doses are listed below:

- EAB 0.23 rem TEDE
- LPZ 0.073 rem TEDE
- Control room 0.83 rem TEDE
- TSC 0.29 rem TEDE

The EAB dose reported is for the worst 2-hour interval, determined to be from 0 to 2 hours.

It is concluded that for a letdown line break, the accident doses meet the applicable acceptance criteria.

4.3.7.5 References

1. NRC, Regulatory Guide 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

2. NUREG-0800, Standard Review Plan 15.6.2, Revision 2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment," July 1981.

3. Supplement 1 to NUREG-0737, "Clarification of TMI Action Plan Requirements: Requirements for Emergency Response Capability," January 1983.

4.3.8 Steam Generator Tube Rupture (USAR Chapter 15.6.3.3)

4.3.8.1 Introduction

For the SGTR, the complete severance of a single SG tube is assumed to occur. Due to the pressure differential between the primary and secondary systems, radioactive reactor coolant is discharged from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through the main condenser, the ARVs, or the MSSVs. In addition, iodine and alkali metal activity is contained in the secondary coolant prior to the accident and some of this activity is released to the atmosphere as a result of steaming from the SGs following the accident.

The analysis of the postulated SGTR assumed a loss of offsite power. The scenario involves the release of steam from the secondary system caused by a turbine trip in conjunction with loss of main steam dump capabilities and the subsequent venting to the atmosphere through the ARVs. The limiting single failure relative to the radiological consequences is chosen in order to maximize the amount of radioactivity released to the atmosphere. It has been demonstrated that no single active failure scenario results in the flooding of the main steamlines. Thus, the limiting single failure for the SGTR is the failure of the ARV to close on the loop with the ruptured SG. After identifying the stuck-open ARV, operations personnel are dispatched to locally close the ARV block valve. It is assumed that the block valve is closed within 30 minutes after the ARV becomes stuck open, thus terminating the release of radioactive steam from the ruptured SG to the atmosphere. Primary-to-secondary break flow will continue following the closure of the block valve until the primary and secondary system pressures are equalized. Equalization of the primary and secondary system pressures terminates the flow of reactor coolant into the ruptured SG in sufficient time to prevent filling it completely.

The time-dependent mass releases used to assess the radiological consequences of this postulated SGTR are calculated from a transient thermal-hydraulic analysis described in Section 2.7.3 in Enclosure I of this LAR. Time-dependent values of the leakage rate into the ruptured SG and the flashing fraction were also used to assess the radiological consequences. Following the closure of the ruptured SG ARV block valve, there is additional radiological dose due to the leakage from the primary system into the intact SGs and the initial concentration of radioactivity contained in the intact SGs as steaming continues to provide plant cooldown.

4.3.8.2 Input Parameters and Assumptions

The analysis of the SGTR radiological consequences uses the analytical methods and assumptions outlined in RG 1.183, Appendix F (Reference 1). A summary of input parameters and assumptions is provided in Table 4.3-11. A listing of the CLB parameters is also included in Table 4.3-11 for comparison to the AST parameters.

4.3.8.2.1 Source Term

The initial RCS and SG activity is provided in Table 4.3-1a. Since no fuel failure results from the accident, pre-accident iodine spike and accident-initiated iodine spike scenarios are considered, consistent with RG 1.183 (Reference 1). For the pre-accident iodine spike case, it is assumed that a reactor transient has occurred prior to the SGTR and has raised the RCS iodine concentration to the TS limit for a transient

of 60 μCi/gm DE I-131 (i.e., 60 times the maximum equilibrium RCS concentration). For the accident-initiated iodine spike case, the reactor trip associated with the SGTR creates an iodine spike that is assumed to increase the iodine release rate from the fuel to the RCS to a value 335 times greater than the appearance rate corresponding to the maximum equilibrium RCS concentration of 1.0 μCi/gm DE I-131. The iodine appearance rates are conservatively calculated assuming maximum letdown flow with perfect cleanup. The duration of the accident-initiated iodine spike is assumed to be 8 hours.

The iodine activity concentration of the secondary coolant at the time the SGTR occurs is assumed to be at 0.1 μCi/gm DE I-131, which is 10 percent of the maximum equilibrium RCS concentration. The alkali metal activity in the RCS at the time the SGTR occurs is at a 1 percent fuel defect level and the activity in the secondary coolant is assumed to be in the same ratio as the primary-to-secondary iodine concentrations. The noble gas activity concentration in the RCS at the time the accident occurs is based on the TS value of 500 μCi/gm DE Xe-133.

Iodine available for release to the atmosphere is assumed to be 97 percent elemental and 3 percent organic.

4.3.8.2.2 Release Model

The mass transfer data provided by the transient thermal-hydraulic analysis described in Section 2.7.3 in Enclosure I of this LAR are increased by 10 percent for use in the analysis of SGTR radiological consequences. The data used for the radiological analysis are shown in Table 4.3-11.

Reactor trip and loss of offsite power occur at approximately 52 seconds. The ARV on the ruptured SG fails open at approximately 1102 seconds and is isolated 30 minutes later, terminating releases from the ruptured SG. The plant cooldown by steaming from the intact SGs is initiated at approximately 3502 seconds and is completed at approximately 4855 seconds. Cooling via the intact SGs continues at a reduced rate until the RHR system is placed into service for heat removal and there is no further steam release to the atmosphere from the secondary system. The RHR system is in service within 12 hours.

Break flow is transferred to the ruptured SG beginning at event initiation and terminating at 7527 seconds. A portion of this break flow is assumed to be released directly to the environment through flashing, which is calculated based on the primary side hot leg and SG secondary side fluid enthalpies. Flashed break flow begins at event initiation and terminates at 3846 seconds. The entire 1 gpm primary-to-secondary accident-induced leakage allowed by the TS is assumed to be leaking into the intact SGs with a density based on cooled liquid, which otherwise is negligible compared to the flow through the ruptured tube. This leakage begins at event initiation and continues throughout the event.

Iodine and alkali metal activity contained in the portion of the break flow that flashed to steam upon entering the ruptured SG is released directly to the atmosphere as long as steam releases from the ruptured SG continue. An iodine partition coefficient in the SGs of 100 (Ci iodine/gm water)/(Ci iodine/gm steam) is applied to releases resulting from steaming of the secondary side fluid. The release of alkali metals from the secondary side is limited by applying the plant-specific moisture carryover factor of 0.25 percent to the steam releases. Prior to reactor trip and concurrent loss of offsite power, a removal factor for both iodine and alkali metal activity of 0.01 is taken from steam released to the condenser.

All noble gas activity carried over to the secondary side through SG tube leakage is assumed to be immediately released to the outside atmosphere as long as steam releases continue.

The WCGS specific transient SGTR analyses for thermal and hydraulic input to dose, described in Section 2.7.3 in Enclosure I of this LAR and margin to overfill, described in Section 2.7.2 in Enclosure I of this LAR confirm that the ruptured SG does not reach water levels that would result in tube uncovery and that margin is maintained in the ruptured SG to prevent water relief via the MSSVs.

4.3.8.2.3 Control Room

The low pressurizer pressure SI setpoint will be reached approximately 325 seconds from event initiation. The SI signal causes the control room to switch from the normal operation mode to the emergency operation mode. The switch is modeled at 385 seconds, which includes a 60-second delay from the initiating signal. As discussed in Section 4.3.2.1, operator action is taken 90 minutes after event initiation to isolate the ventilation train with failed filtration.

4.3.8.3 Acceptance Criteria

The EAB and LPZ dose acceptance criterion for an SGTR with an assumed pre-accident iodine spike is 25 rem TEDE per RG 1.183, which is the 10 CFR 50.67 limit. The EAB and LPZ dose acceptance criterion for an SGTR with an assumed accident-initiated iodine spike is 2.5 rem TEDE per RG 1.183, which is 10 percent of the 10 CFR 50.67 limit. The acceptance criterion for the control room dose is 5 rem TEDE per 10 CFR 50.67. The acceptance criterion for the TSC dose is 5 rem TEDE as allowed by GDC 19, in accordance with Reference 2.

The EAB doses are calculated for the worst 2-hour interval. The LPZ doses are calculated until all releases are terminated, which is the RHR cut-in time (12 hours) used in the analysis. The control room and TSC doses are calculated for 30 days.

4.3.8.4 Results and Conclusions

The SGTR accident doses are listed below.

For the pre-accident iodine spike case:

- EAB 1.1 rem TEDE
- LPZ 0.35 rem TEDE
- Control room 1.1 rem TEDE
- TSC 2.3 rem TEDE

For the accident-initiated iodine spike case:

- EAB 0.86 rem TEDE
- LPZ 0.29 rem TEDE
- Control room 0.45 rem TEDE
- TSC 1.7 rem TEDE

The EAB doses reported are for the worst 2-hour interval, determined to be from 0 to 2 hours for both the pre-accident iodine spike and accident-initiated iodine spike.

It is concluded that for an SGTR, the accident doses meet the applicable acceptance criteria.

4.3.8.5 References

1. NRC, Regulatory Guide 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

2. Supplement 1 to NUREG-0737, "Clarification of TMI Action Plan Requirements: Requirements for Emergency Response Capability," January 1983.

4.3.9 Loss-of-Coolant Accident (USAR Chapter 15.6.5.4)

4.3.9.1 Introduction

An abrupt failure of the main reactor coolant pipe is assumed to occur. Activity from the RCS is released to containment and a portion of this activity is released to the atmosphere via the mini-purge system prior to containment isolation. It is assumed that the emergency core coolant features fail to prevent the core from experiencing significant degradation (i.e., melting). This sequence cannot occur unless there are multiple failures, and thus goes beyond the typical design basis accident that considers a single active failure. Activity from the core is released to the containment and from there is released to the environment by means of containment leakage. In addition, once recirculation of the emergency core cooling system (ECCS) is established, iodine activity in the sump solution may be released to the environment by means of leakage from ESF equipment outside containment in the auxiliary building, and by means of leakage from the ESF to the RWST with subsequent leaking or venting. The total offsite and control room doses are the sum of the doses resulting from the four postulated release paths.

4.3.9.2 Input Parameters and Assumptions

The analysis of the LOCA radiological consequences uses the analytical methods and assumptions outlined in RG 1.183, Appendix A (Reference 1). A summary of input parameters and assumptions is provided in Table 4.3-12. A listing of the CLB parameters is also included in Table 4.3-12 for comparison to the AST parameters.

4.3.9.2.1 Source Term

The iodine activity in the RCS at the time of the accident is assumed to be at the TS limit of 1.0 μCi/gm of DE I-131 for the maximum equilibrium RCS concentration. The noble gas activity concentration in the RCS at the time of the accident is assumed to be at the TS limit of 500 μCi/gm of DE Xe-133 for the maximum equilibrium RCS concentration. The activity for the remaining nuclide groups is at a 1 percent fuel defect level. The RCS activities are listed in Table 4.3-1a. These values are used in the containment purge release pathway.

For modeling the containment leakage, ECCS leakage, and RWST back-leakage release pathways, the analysis assumes fuel melt occurs in the entire core and the release of activity occurs over a 1.8 hour interval. The gap release phase occurs at 30 seconds and ends in the first 0.5 hour and the release from the melted fuel occurs over the subsequent 1.3 hours. This modeling is consistent with RG 1.183. Table 4.3-1a lists the core activities and Table 4.3-12 lists the gap and fuel melt activity release fractions for the various nuclide groups.

Iodine available for release to the atmosphere is assumed to be 95 percent particulate, 4.85 percent elemental, and 0.15 percent organic for the containment leakage pathway and 97 percent elemental and 3 percent organic for the other pathways.

4.3.9.2.2 Release Models

4.3.9.2.2.1 Containment Leakage

For the containment leakage pathway, all activity released from the fuel is assumed to go into the unsprayed portion of containment before being mixed with the sprayed portion of the containment. The time-dependent removal of elemental iodine and particulates from the containment atmosphere is accomplished by containment sprays, particulate sedimentation, radioactive decay, and leakage from containment. The noble gases and organic iodine are subject to removal only by radioactive decay and leakage from containment.

The maximum free volume of the containment modeled in the containment leakage pathway is 2.7E6 ft^3. The portion of this volume covered by spray drops (85 percent) and its unsprayed portion (15 percent) are modeled separately. The mixing rate between the sprayed and unsprayed regions is modeled as 69,400 cfm per fan cooler. Only one fan cooler is assumed to be operating and a conservative delay of 2 minutes is assumed before mixing is credited. Mixing continues for the remainder of the event.

The containment is assumed to leak at the design leak rate of 2 percent per day for the first 24 hours of the accident and then to leak at half that rate (1 percent per day) for the remainder of the 30-day period considered in the analysis.

Containment sprays are actuated at 2 minutes following accident initiation and terminate 5 hours following accident initiation.

SRP 6.5.2 (Reference 2) identifies a methodology for the determination of spray removal of elemental iodine. The removal rate constant is determined by:

$$\lambda_S = 6K_gTF/VD$$

where:

K_g = Gas phase mass transfer coefficient, m/min
T = Time of fall of the spray drops, min
F = Volume flow rate of sprays, m^3/hr
V = Containment sprayed volume, m^3
D = Mass-mean diameter of the spray drops, m

The resulting removal coefficient for elemental iodine is 22.9 hr^{-1}. SRP 6.5.2 (Reference 2) allows for elemental iodine removal credit of up to 20 hr^{-1} during injection spray; however, to avoid sensitivities with spray switchover times from injection to recirculation and to conservatively address iodine loading in the spray fluid during recirculation, the removal is limited to 10 hr^{-1} for either spray mode. The elemental removal is modeled until a decontamination factor (DF) of 200 is reached at 2.335 hours, at which time elemental iodine removal is terminated.

SRP 6.5.2 (Reference 2) identifies a methodology for the determination of spray removal of particulates. The removal rate constant is determined by:

$$\lambda_P = 3hFE / 2VD$$

where:

h = Drop fall height, m
F = Volume flow rate of sprays, m^3/hr
V = Containment sprayed volume, m^3
E/D = Ratio of dimensionless collection efficiency (E) to the average spray drop diameter (D)

The resulting removal coefficient for particulates is 5 hr^{-1} until a DF of 50 is reached at 2.43 hours. After this time the particulate removal coefficient is reduced by a factor of 10 to 0.5 hr^{-1} until sprays are terminated at 5 hours.

Sedimentation is credited in the portion of containment that is not impacted by spray removal and in the sprayed portion when sprays are not on at a rate of 0.1 hr^{-1} until a DF of 1000 is reached at 23.5 hours. After this time sedimentation removal is terminated.

The sump pH is maintained at greater than or equal to 7.0. Therefore, no re-evolution of iodine occurs.

4.3.9.2.2.2 Emergency Core Cooling System Leakage

For the ECCS leakage pathway, all iodine activity released from the fuel is assumed to be in the sump solution immediately. The only removal of activity from the sump is by radioactive decay or leakage to the auxiliary building. The sump volume is 460,000 gallons. When ECCS recirculation is established following the LOCA, leakage is assumed to occur from ESF equipment in the auxiliary building. Recirculation is modeled to initiate at the start of the event and continues throughout the event.

The leakage to the auxiliary building is modeled at a rate of 2 gpm. The leakage value was doubled in accordance with RG 1.183. The analysis assumes that 10 percent of the iodine activity in the leakage becomes airborne and is available for release to the environment. The activity of the airborne leakage is further reduced as it is released through the auxiliary building vent filters with 90 percent efficiency for all forms of iodine.

4.3.9.2.2.3 Refueling Water Storage Tank Back-Leakage

For the RWST back-leakage pathway, a portion of the ECCS recirculation is assumed to leak into the RWST. All iodine activity released from the fuel is assumed to be in the sump solution immediately. The only removal of activity from the sump is by radioactive decay or leakage to the RWST. The sump volume is 460,000 gallons. Recirculation is modeled to initiate at the start of the event and continues throughout the event.

Leakage to the RWST is modeled at a rate of 3.8 gpm. The activity is modeled to be delivered directly to the gas filled portion of the RWST; however, only 10 percent of the activity becomes airborne and is available for release to the environment. The release rate from the RWST to the environment is based on the volume displacement from the incoming leakage. An adjustment is made to account for a reduction in the RWST gas volume available for dilution as the leakage into the RWST increases the water level.

4.3.9.2.2.4 Containment Purge

For the containment purge system release pathway, all of the initial primary coolant activity is instantly released from the RCS and is evenly distributed throughout the containment volume. The minimum free volume of the containment modeled in this pathway is 2.5E6 ft^3. The only removal of activity from containment is by radioactive decay or the purge flow. The maximum flow rate of 4680 cfm is modeled until the purge line is isolated at 10 seconds.

4.3.9.2.3 Control Room

In the event of a LOCA, the low pressurizer pressure SI setpoint will be reached almost immediately following the break. The SI signal causes the control room to switch from the normal operation mode to the emergency operation mode. The switch is conservatively modeled at 120 seconds following event initiation, which includes a 60-second delay from the initiating signal. As discussed in Section 4.3.2.1, operator action is taken 90 minutes after event initiation to isolate the ventilation train with failed filtration.

The calculated dose to control room personnel from external sources was calculated to be 0.132 rem TEDE. These external sources include the activity remaining in containment following the LOCA, the activity cloud outside the control room in the control building, the activity buildup on filters, and the activity in the control building streaming through doors and penetrations. This is added to the dose calculated from the four release paths discussed above.

4.3.9.3 Acceptance Criteria

The EAB and LPZ dose acceptance criterion for a LOCA is 25 rem TEDE per RG 1.183, which is also the 10 CFR 50.67 limit. The acceptance criterion for the control room dose is 5 rem TEDE per 10 CFR 50.67. The acceptance criterion for the TSC dose is 5 rem TEDE as allowed by GDC 19, in accordance with Reference 3.

The EAB dose is calculated for the worst 2-hour interval. The LPZ, control room, and TSC doses are calculated for 30 days.

4.3.9.4 Results and Conclusions

The LOCA doses are listed below:

- EAB 5.1 rem TEDE
- LPZ 4.6 rem TEDE
- Control room 4.7 rem TEDE
- TSC 4.9 rem TEDE

The EAB dose reported is for the worst 2-hour interval, determined to be from 0.4 to 2.4 hours.

It is concluded that for a LOCA, the accident doses meet the applicable acceptance criteria.

4.3.9.5 References

1. NRC, Regulatory Guide 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

2. NUREG-0800, Standard Review Plan 6.5.2, Revision 4, "Containment Spray as a Fission Product Cleanup System," March 2007.

3. Supplement 1 to NUREG-0737, "Clarification of TMI Action Plan Requirements: Requirements for Emergency Response Capability," January 1983.

4.3.10 Waste Gas Decay Tank Failure (USAR Chapter 15.7.1.5)

4.3.10.1 Introduction

For the waste gas decay tank rupture, a failure is assumed that results in the uncontrolled release of the contents of one gas decay tank.

4.3.10.2 Input Parameters and Assumptions

This event is not discussed in RG 1.183 (Reference 1); however, RG 1.183 models are applied to this analysis in conjunction with event guidance from RG 1.24 (Reference 2) and SRP Branch Technical Position (BTP) 11-5 (Reference 3). A summary of input parameters and assumptions is provided in Table 4.3-13. A listing of the CLB parameters is also included in Table 4.3-13 for comparison to the AST parameters.

4.3.10.2.1 Source Term

The iodine and noble gas activity associated with the gas decay tank is presented in Table 4.3-2a. For comparison to the AST parameters, the tank activities for CLB analyses are listed in Table 4.3-2b.The iodine is assumed to be 100 percent elemental; however, the chemical species of iodine has no impact on the calculation since no removal processes are modeled and the control room filters have the same efficiencies for all forms of iodine.

4.3.10.2.2 Release Model

As a result of the failure, all of the activity in the tank is assumed to be released to the atmosphere over a period of 2 hours. A linear release model is used.

4.3.10.2.3 Control Room

The control room is not credited to isolate following a tank failure; therefore, the control room ventilation remains in normal operation mode.

4.3.10.3 Acceptance Criteria

The EAB and LPZ dose acceptance criterion for a waste gas decay tank failure is 0.1 rem TEDE, consistent with SRP BTP 11-5, which is the 10 CFR 20 limit. The control room dose acceptance criterion is 5 rem TEDE per 10 CFR 50.67. The acceptance criterion for the TSC dose is 5 rem TEDE as allowed by GDC 19, in accordance with Reference 4.

The EAB dose is calculated for the worst 2-hour interval. The LPZ dose is calculated until all releases are terminated, which is at 2 hours. The control room and TSC doses are calculated for 30 days.

4.3.10.4 Results and Conclusions

The waste gas decay tank failure accident doses are listed below:

- EAB 0.090 rem TEDE
- LPZ 0.029 rem TEDE
- Control room 0.019 rem TEDE
- TSC 0.0037 rem TEDE

The EAB dose reported is for the worst 2-hour interval, determined to be from 0 to 2 hours.

It is concluded that for a waste gas decay tank failure, the accident doses meet the applicable acceptance criteria.

4.3.10.5 References

1. NRC, Regulatory Guide 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

2. NRC, Regulatory Guide 1.24, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure," March 1972.

3. NUREG-0800, Standard Review Plan Branch Technical Position 11-5, Revision 3, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," March 2007.

4. Supplement 1 to NUREG-0737, "Clarification of TMI Action Plan Requirements: Requirements for Emergency Response Capability," January 1983.

4.3.11 Liquid Waste Tank Failure (USAR Chapter 15.7.2.5)

4.3.11.1 Introduction

The accident is defined as the uncontrolled atmospheric release from a recycle holdup tank or a hypothetical tank containing a maximized quantity of iodine. Both tanks are analyzed to determine the bounding liquid waste tank rupture from a dose perspective.

4.3.11.2 Input Parameters and Assumptions

This event is not discussed in RG 1.183 (Reference 1); however, RG 1.183 models are applied to this analysis in conjunction with the current analysis of record presented in USAR Chapter 15.7.2.5. A summary of input parameters and assumptions is provided in Table 4.3-14. A listing of the CLB parameters is also included in Table 4.3-14 for comparison to the AST parameters.

4.3.11.2.1 Source Term

The source terms for the recycle holdup tank and hypothetical tank are presented in Table 4.3-2a. The iodine is assumed to be 100 percent elemental; however, the chemical species of iodine has no impact on the calculation since no removal processes are modeled and the control room filters have the same efficiencies for all forms of iodine.

4.3.11.2.2 Release Model

As a result of the failure, all of the activity, in either the recycle holdup tank or hypothetical tank, is assumed to be released to the atmosphere and become airborne over a period of 2 hours. A linear release model is used. This model is used for both tank failures.

4.3.11.2.3 Control Room

The control room is not credited to isolate following a tank failure; therefore, the control room ventilation remains in normal operation mode.

4.3.11.3 Acceptance Criteria

The EAB and LPZ dose acceptance criterion for a liquid waste tank failure is 0.1 rem TEDE, which is the 10 CFR 20 limit by applying the same basis of acceptance from Reference 2. The control room dose acceptance criterion is 5 rem TEDE per 10 CFR 50.67. The acceptance criterion for the TSC dose is 5 rem TEDE as allowed by GDC 19, in accordance with Reference 3.

The EAB doses are calculated for the worst 2-hour interval. The LPZ doses are calculated until all releases are terminated, which is at 2 hours. The control room and TSC doses are calculated for 30 days.

4.3.11.4 Results and Conclusions

The accident doses from both liquid waste tank failures are listed below.

Recycle holdup tank:

- EAB 0.028 rem TEDE
- LPZ 0.0088 rem TEDE
- Control room 0.055 rem TEDE
- TSC 0.0061 rem TEDE

Hypothetical tank maximizing iodine:

- EAB 0.050 rem TEDE
- LPZ 0.016 rem TEDE
- Control room 0.24 rem TEDE
- TSC 0.025 rem TEDE

The EAB doses reported are for the worst 2-hour interval, determined to be from 0 to 2 hours for both tank rupture scenarios.

It is concluded that for a liquid waste tank failure, the accident doses meet the applicable acceptance criteria.

4.3.11.5 References

1. NRC, Regulatory Guide 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

2. NRC Regulatory Issue Summary 2006-04, "Experience with Implementation of Alternative Source Terms," March 2006.

3. Supplement 1 to NUREG-0737, "Clarification of TMI Action Plan Requirements: Requirements for Emergency Response Capability," January 1983.

4.3.12 Fuel Handling Accident (USAR Chapter 15.7.4.5)

4.3.12.1 Introduction

A fuel assembly is assumed to be dropped and damaged during refueling, along with some of the fuel rods from a neighboring assembly. Analysis of the accident is performed with assumptions selected so that the results are bounding for the accident occurring either inside containment or in the fuel building. The bounding activity pathway modeled releases of damaged fuel activity through the pool water to the building air space and then to the environment without crediting containment isolation or filtration by the fuel pool ventilation system.

4.3.12.2 Input Parameters and Assumptions

The analysis of the FHA radiological consequences uses the analytical methods and assumptions outlined in RG 1.183, Appendix B (Reference 1). A summary of input parameters and assumptions is provided in Table 4.3-15. A listing of the CLB parameters is also included in Table 4.3-15 for comparison to the AST parameters.

4.3.12.2.1 Source Term

The core fission product activity for an average assembly at a specified decay time is provided in Table 4.3-15. It is assumed that all fuel rods in the equivalent of 1.2 fuel assemblies are damaged to the extent that all their gap activity is released. The assembly inventory is based on the assumption that the subject fuel assembly has been operated at 1.65 times the core average power.

The decay time used in determining the inventory of the damaged rods is 76 hours. Thus, the analysis supports the TS limit of 76 hours decay time prior to fuel movement.

The calculation uses the defined gap fractions in NUREG/CR-5009 (Reference 2) of 12 percent for I-131, 30 percent for Kr-85, and 10 percent for all other iodines and noble gases as these gap fractions correspond to high burnup fuel, which supports the conservative assumption that 100 percent of the rods do not meet the burnup and kW/ft limits set forth in Footnote 11 of RG 1.183 (Reference 1).

RG 1.183 allows credit for an overall pool DF for iodine of 200 for a pool depth of 23 feet. Although not explicitly discussed, the specified overall DF also applies to rod internal pressures up to 1500 psig. Both of these criteria are met for this analysis. The overall DF is based on DFs of 285 for elemental iodine and 1 for organic iodine and iodine chemical fractions of 99.85 percent for elemental iodine and 0.15 percent for organic iodine. Thus, the normalized split between elemental and organic iodine leaving the pool is 70 percent for elemental iodine and 30 percent for organic iodine.

4.3.12.2.2 Release Model

All activity released from the fuel pool is assumed to be released to the atmosphere in 2 hours using a linear release model. No credit is taken for spent fuel pool ventilation system operation for the FHA in the fuel building. No credit is taken for isolation of containment for the FHA in containment. For these conditions, the assumptions and parameters for a FHA inside containment are identical to those for an FHA in the fuel building, and therefore, the radiological consequences are the same regardless of the accident location.

4.3.12.2.3 Control Room

The analysis showed that an acceptable control room dose could be obtained without crediting the control room switching to emergency mode operation, so the reported dose reflects this model.

4.3.12.3 Acceptance Criteria

The EAB and LPZ dose acceptance criterion for a FHA is 6.3 rem TEDE per RG 1.183, which is approximately 25 percent of the 10 CFR 50.67 limit. The control room dose acceptance criterion is 5 rem TEDE per 10 CFR 50.67. The acceptance criterion for the TSC dose is 5 rem TEDE as allowed by GDC 19, in accordance with Reference 3.

The EAB dose is calculated for the worst 2-hour interval. The LPZ dose is calculated until all releases are terminated, which is at 2 hours. The control room and TSC doses are calculated for 30 days.

4.3.12.4 Results and Conclusions

The FHA doses are listed below:

- EAB 1.1 rem TEDE
- LPZ 0.35 rem TEDE
- Control room 4.0 rem TEDE
- TSC 0.96 rem TEDE

The EAB dose reported is for the worst 2-hour interval, determined to be from 0 to 2 hours.

It is concluded that for a FHA, the accident doses meet the applicable acceptance criteria.

4.3.12.5 References

1. NRC, Regulatory Guide 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

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

3. Supplement 1 to NUREG-0737, "Clarification of TMI Action Plan Requirements: Requirements for Emergency Response Capability," January 1983.

Table 4.3-1a Core Activities and RCS and SG Coolant Activity Concentrations – AST

Isotope	Core Activity (Ci)	RCS Activity Concentration (µCi/gm)	SG Activity Concentration (µCi/gm)
Kr-85m	2.69E+07	1.37E+00	0.0
Kr-85	1.10E+06	7.51E+00	0.0
Kr-87	5.30E+07	8.94E-01	0.0
Kr-88	7.12E+07	2.47E+00	0.0
Xe-131m	1.05E+06	2.81E+00	0.0
Xe-133m	6.06E+06	4.16E+00	0.0
Xe-133	2.01E+08	2.31E+02	0.0
Xe-135m	4.39E+07	4.62E-01	0.0
Xe-135	4.06E+07	6.10E+00	0.0
Xe-138	1.80E+08	5.70E-01	0.0
I-130	1.98E+06	1.10E-02	1.10E-03
I-131	1.01E+08	7.77E-01	7.77E-02
I-132	1.49E+08	8.03E-01	8.03E-02
I-133	2.10E+08	1.19E+00	1.19E-01
I-134	2.36E+08	1.73E-01	1.73E-02
I-135	2.00E+08	6.75E-01	6.75E-02
Cs-134	1.65E+07	4.82E+00	4.82E-01
Cs-136	3.95E+06	4.35E+00	4.35E-01
Cs-137	1.11E+07	2.68E+00	2.68E-01
Cs-138	1.96E+08	1.16E+00	1.16E-01
Rb-86	1.86E+05	4.33E-02	4.33E-03
Te-127m	1.49E+06	0.0	0.0
Te-127	9.09E+06	0.0	0.0
Te-129m	5.04E+06	0.0	0.0
Te-129	2.66E+07	0.0	0.0
Te-131m	1.99E+07	0.0	0.0
Te-132	1.45E+08	0.0	0.0
Sb-127	9.23E+06	0.0	0.0
Sb-129	2.85E+07	0.0	0.0
Sr-89	9.98E+07	0.0	0.0
Sr-90	8.51E+06	0.0	0.0
Sr-91	1.25E+08	0.0	0.0
Sr-92	1.33E+08	0.0	0.0

Table 4.3-1a (cont.) Core Activities and RCS and SG Coolant Activity Concentrations – AST

Isotope	Core Activity (Ci)	RCS Activity Concentration (µCi/gm)	SG Activity Concentration (µCi/gm)
Ba-139	1.87E+08	0.0	0.0
Ba-140	1.79E+08	0.0	0.0
Ru-103	1.56E+08	0.0	0.0
Ru-105	1.08E+08	0.0	0.0
Ru-106	4.79E+07	0.0	0.0
Rh-105	1.00E+08	0.0	0.0
Mo-99	1.91E+08	0.0	0.0
Tc-99m	1.69E+08	0.0	0.0
Ce-141	1.69E+08	0.0	0.0
Ce-143	1.59E+08	0.0	0.0
Ce-144	1.30E+08	0.0	0.0
Pu-238	2.10E+05	0.0	0.0
Pu-239	2.70E+04	0.0	0.0
Pu-240	4.20E+04	0.0	0.0
Pu-241	1.06E+07	0.0	0.0
Np-239	1.94E+09	0.0	0.0
Y-90	8.88E+06	0.0	0.0
Y-91	1.30E+08	0.0	0.0
Y-92	1.35E+08	0.0	0.0
Y-93	1.52E+08	0.0	0.0
Nb-95	1.76E+08	0.0	0.0
Zr-95	1.74E+08	0.0	0.0
Zr-97	1.75E+08	0.0	0.0
La-140	1.85E+08	0.0	0.0
La-142	1.64E+08	0.0	0.0
Nd-147	6.57E+07	0.0	0.0
Pr-143	1.55E+08	0.0	0.0
Am-241	9.94E+03	0.0	0.0
Cm-242	2.82E+06	0.0	0.0
Cm-244	2.11E+05	0.0	0.0

Table 4.3-1b Core Activities and RCS and SG Coolant Activity Concentrations – CLB

Isotope	Core Activity (Ci)	RCS Activity Concentration (μCi/gm)	SG Activity Concentration (μCi/gm)
I-131	9.460E+07	7.310E-01	7.310E-02
I-132	1.370E+08	8.190E-01	8.190E-02
I-133	1.950E+08	1.287E+00	1.287E-01
I-134	2.150E+08	1.960E-01	1.96E-02
I-135	1.830E+08	7.510E-01	7.51E-02
Xe-131m	1.010E+06	3.410E+00	0.0
Xe-133m	6.060E+06	5.370E+00	0.0
Xe-133	1.950E+08	2.900E+02	0.0
Xe-135m	3.770E+07	6.040E-01	0.0
Xe-135	4.700E+07	9.820E+00	0.0
Xe-137	1.710E+08	2.240E-01	0.0
Xe-138	1.640E+08	8.150E-01	0.0
Kr-83m	1.240E+07	5.540E-01	0.0
Kr-85m	2.670E+07	2.260E+00	0.0
Kr-85	1.020E+06	9.410E+00	0.0
Kr-87	5.160E+07	1.470E+00	0.0
Kr-88	7.280E+07	4.260E+00	0.0
Kr-89	8.940E+07	1.21E-01	0.0

Table 4.3-1c Core Activities and RCS and SG Coolant Activity Concentrations for SGTR – CLB

Isotope	RCS Activity Concentration (µCi/gm)	SG Activity Concentration (µCi/gm)
I-131	0.7518	0.07518
I-132	0.8431	0.08431
I-133	1.3237	0.13237
I-134	0.2019	0.02019
I-135	0.7733	0.07733
Xe-131m	3.41	1.07E-04
Xe-133m	5.37	1.73E-04
Xe-133	290.0	9.12E-03
Xe-135m	0.604	8.62E-05
Xe-135	9.82	3.20E-04
Xe-137	0.224	6.90E-06
Xe-138	0.815	2.55E-05
Kr-83m	0.554	1.92E-05
Kr-85m	2.26	7.10E-05
Kr-85	9.41	2.96E-04
Kr-87	1.47	4.62E-05
Kr-88	4.26	1.34E-04
Kr-89	0.121	3.71E-06

Table 4.3-2a Tank Activities – AST

Isotope	Waste Gas Decay Tank (Ci)	Recycle Holdup Tank (Ci)	Hypothetical Tank to Maximize Iodine (Ci)
Kr-85m	1.49E+02	4.00E+00	0.0
Kr-85	5.52E+03	1.69E+03	0.0
Kr-87	3.00E+01	7.34E-01	0.0
Kr-88	1.79E+02	4.49E+00	0.0
Xe-131m	1.07E+03	3.67E+02	0.0
Xe-133m	1.27E+03	1.44E+02	0.0
Xe-133	8.12E+04	1.79E+04	0.0
Xe-135m	5.97E+01	1.09E-01	0.0
Xe-135	1.02E+03	3.64E+01	0.0
Xe-138	4.06E+00	8.74E-02	0.0
I-130	0.0	6.36E-03	2.75E-03
I-131	3.99E-02	5.83E+00	2.75E+01
I-132	2.30E-02	8.52E-02	7.10E-03
I-133	4.28E-02	1.15E+00	8.07E-01
I-134	4.96E-03	7.04E-03	2.26E-04
I-135	1.94E-02	2.07E-01	4.88E-02

Table 4.3-2b Tank Activities – CLB

Isotope	Waste Gas Decay Tank (Ci)	Recycle Holdup Tank (Ci)	Hypothetical Tank to Maximize Iodine (Ci)
I-131	4.02E-02	4.96E+00	1.92E+01
I-132	0.0	7.90E-02	0.0
I-133	3.50E-02	1.13E+00	7.46E-03
I-134	0.0	7.25E-03	0.0
I-135	1.39E-02	2.09E-01	7.91E-09
Kr-83m	2.26E+01	5.02E-01	0
Kr-85m	1.85E+02	4.93E+00	0
Kr-85	4.75E+03	1.59E+03	0
Kr-87	3.64E+01	9.06E-01	0
Kr-88	2.25E+02	5.80E+00	0
Kr-89	1.76E-01	3.10E-03	0
Xe-131m	9.05E+02	3.35E+02	0
Xe-133m	1.14E+03	1.40E+02	0
Xe-133	7.09E+04	1.68E+04	0
Xe-135m	5.99E+01	1.06E-01	0
Xe-135	1.15E+03	4.40E+01	0
Xe-137	3.42E-01	6.97E-03	0
Xe-138	4.40E+00	9.38E-02	0

Table 4.3-3a Dose Conversion Factors – AST

Isotope	CEDE (Sievert/Becquerel [Sv/Bq])	EDE (Sv-m^3/Bq-sec)
Kr-85m	N/A	7.48E-15
Kr-85	N/A	1.19E-16
Kr-87	N/A	4.12E-14
Kr-88	N/A	1.02E-13
Xe-131m	N/A	3.89E-16
Xe-133m	N/A	1.37E-15
Xe-133	N/A	1.56E-15
Xe-135m	N/A	2.04E-14
Xe-135	N/A	1.19E-14
Xe-138	N/A	5.77E-14
I-130	7.14E-10	1.04E-13
I-131	8.89E-09	1.82E-14
I-132	1.03E-10	1.12E-13
I-133	1.58E-09	2.94E-14
I-134	3.55E-11	1.30E-13
I-135	3.32E-10	7.98E-14
Cs-134	1.25E-08	7.57E-14
Cs-136	1.98E-09	1.06E-13
Cs-137	8.63E-09	2.88E-14
Cs-138	2.74E-11	1.21E-13
Rb-86	1.79E-09	4.81E-15
Te-127m	5.81E-09	1.47E-16
Te-127	8.60E-11	2.42E-16
Te-129m	6.47E-09	1.55E-15
Te-129	2.42E-11	2.75E-15
Te-131m	1.73E-09	7.01E-14
Te-132	2.55E-09	1.03E-14
Sb-127	1.63E-09	3.33E-14
Sb-129	1.74E-10	7.14E-14
Sr-89	1.12E-08	7.73E-17
Sr-90	3.51E-07	7.53E-18
Sr-91	4.49E-10	3.45E-14
Sr-92	2.18E-10	6.79E-14

Table 4.3-3a Dose Conversion Factors – AST (cont.)

Isotope	CEDE (Sievert/Becquerel [Sv/Bq])	EDE (Sv-m^3/Bq-sec)
Ba-139	4.64E-11	2.17E-15
Ba-140	1.01E-09	8.58E-15
Ru-103	2.42E-09	2.25E-14
Ru-105	1.23E-10	3.81E-14
Ru-106	1.29E-07	0.0
Rh-105	2.58E-10	3.72E-15
Mo-99	1.07E-09	7.28E-15
Tc-99m	8.80E-12	5.89E-15
Ce-141	2.42E-09	3.43E-15
Ce-143	9.16E-10	1.29E-14
Ce-144	1.01E-07	8.53E-16
Pu-238	1.06E-04	4.88E-18
Pu-239	1.16E-04	4.24E-18
Pu-240	1.16E-04	4.75E-18
Pu-241	2.23E-06	7.25E-20
Np-239	6.78E-10	7.69E-15
Y-90	2.28E-09	1.90E-16
Y-91	1.32E-08	2.60E-16
Y-92	2.11E-10	1.30E-14
Y-93	5.82E-10	4.80E-15
Nb-95	1.57E-09	3.74E-14
Zr-95	6.39E-09	3.60E-14
Zr-97	1.17E-09	9.02E-15
La-140	1.31E-09	1.17E-13
La-142	6.84E-11	1.44E-13
Nd-147	1.85E-09	6.19E-15
Pr-143	2.19E-09	2.10E-17
Am-241	1.20E-04	8.18E-16
Cm-242	4.67E-06	5.69E-18
Cm-244	6.70E-05	4.91E-18

Table 4.3-3b Dose Conversion Factors – CLB

Isotope	Thyroid (rem/ci)	Whole Body (rem-m^3/ci-sec)	Beta Skin (rem-m^3/ci-sec)[1]
I-131	1.49E+06	8.72E-02	3.17E-02
I-132	1.43E+04	5.13E-01	1.32E-01
I-133	2.69E+05	1.55E-01	7.35E-02
I-134	3.73E+03	5.32E-01	9.23E-02
I-135	5.60E+04	4.21E-01	1.29E-01
Kr-83m	N/A	2.40E-06	0.00E+00
Kr-85m	N/A	7.10E-03	4.63E-02
Kr-85	N/A	5.11E-04	4.25E-02
Kr-87	N/A	1.88E-01	3.09E-01
Kr-88	N/A	4.67E-01	7.52E-02
Kr-89	N/A	5.27E-01	3.20E-01
Xe-131m	N/A	2.91E-03	1.51E-02
Xe-133m	N/A	7.97E-03	3.15E-02
Xe-133	N/A	9.33E-03	9.70E-03
Xe-135m	N/A	9.91E-02	2.25E-02
Xe-135	N/A	5.75E-02	5.90E-02
Xe-137	N/A	4.51E-02	3.87E-01
Xe-138	N/A	2.80E-01	1.31E-01

Note:
1. Only applicable to Waste Gas Decay Tank Failure and FHA for control room habitability

Table 4.3-3c Dose Conversion Factors (LOCA & SGTR) – CLB

Isotope	Thyroid (rem/ci)	Whole Body (rem-m^3/ci-sec)	Beta Skin (rem-m^3/ci-sec)[1]
I-131	1.08E+06	6.73E-02	3.17E-02
I-132	6.44E+03	4.14E-01	1.32E-01
I-133	1.80E+05	1.09E-01	7.35E-02
I-134	1.07E+03	4.81E-01	9.23E-02
I-135	3.13E+04	2.95E-01	1.29E-01
Kr-83m	0.00E+00	5.55E-06	0.00E+00
Kr-85m	0.00E+00	2.77E-02	4.63E-02
Kr-85	0.00E+00	4.40E-04	4.25E-02
Kr-87	0.00E+00	1.52E-01	3.09E-01
Kr-88	0.00E+00	3.77E-01	7.52E-02
Kr-89	0.00E+00	3.23E-01	3.20E-01
Xe-131m	0.00E+00	1.44E-03	1.51E-02
Xe-133m	0.00E+00	5.07E-03	3.15E-02
Xe-133	0.00E+00	5.77E-03	9.70E-03
Xe-135m	0.00E+00	7.55E-02	2.25E-02
Xe-135	0.00E+00	4.40E-02	5.90E-02
Xe-137	0.00E+00	3.03E-02	3.87E-01
Xe-138	0.00E+00	2.13E-01	1.31E-01

Note:
1. Only applicable to LOCA

Table 4.3-4a Offsite Breathing Rates – AST

Time	Offsite Breathing Rate (m^3/sec)
0 – 8 hours	3.5E-04
8 – 24 hours	1.8E-04
> 24 hours	2.3E-04

Table 4.3-4b Offsite Breathing Rates – CLB

Time	Offsite Breathing Rate (m^3/sec)
0 – 2 hours	3.47E-04
2 – 8 hours	3.47E-04
8 – 24 hours	1.75E-04
24 – 96 hours	2.32E-04
96 – 720 hours	2.32E-04

Table 4.3-4c Core Activity at end of cycle – CLB

Isotope	Activity (Ci)
I-131	9.460E+07
I-132	1.370E+08
I-133	1.950E+08
I-134	2.150E+08
I-135	1.830E+08
Kr-83m	1.240E+07
Kr-85m	2.670E+07
Kr-85	1.020E+06
Kr-87	5.160E+07
Kr-88	7.280E+07
Kr-89	8.940E+07
Xe-131m	1.010E+06
Xe-133m	6.060E+06
Xe-133	1.950E+08
Xe-135m	3.770E+07
Xe-135	4.700E+07
Xe-137	1.710E+08
Xe-138	1.640E+08

Table 4.3-5 Control Room and Control Building Parameters

	AST	CLB
Control room volume (ft^3)	100,000	100,000
Control building volume (ft^3)	239,000	239,000
Normal ventilation flow rates (cfm)		
Unfiltered makeup flow rate from environment to control building	13,050	13,050
Unfiltered makeup flow rate from environment to control room	1950	1950
Unfiltered inleakage to control room	50	10
Emergency mode flow rates prior to operator action (cfm)		
Filtered makeup flow rate from environment to control building	1350	1350
Filtered makeup flow rate from control building to control room	550	550
Unfiltered makeup flow rate from environment to control building	400	300
Unfiltered makeup flow rate from control building to control room	550	550
Unfiltered inleakage to control room	50	20
Filtered control room recirculation flow	1250	1250
Emergency mode flow rates following operator action (cfm)		
Filtered makeup flow rate from environment to control building	675	675
Filtered makeup flow rate from control building to control room	550	550
Unfiltered makeup flow rate from environment to control building	400	300
Unfiltered makeup flow rate from control building to control room	0	0
Unfiltered inleakage to control room	50	20
Filtered control room recirculation flow	1250	1250
Operator action time to terminate failed train of filtered makeup flow from start of event (minutes)	90	90
Filter efficiencies (%)		
Elemental iodine	95	95
Organic iodine	95	95
Particulates	95	95
Isolation setpoint for R-23 detector (μCi/cc Xe-133)	1.35E-03	N/A
Delay to switch to emergency mode operation following receipt of isolation signal (seconds)	60	N/A
Control room breathing rate for duration of the event (m^3/sec)	3.5E-04	3.47E-04
Control room occupancy factors		
0 – 24 hours	1.0	1.0
1 – 4 days	0.6	0.6
4 – 30 days	0.4	0.4

Table 4.3-6 Assumptions Used for Main Steamline Break Analysis

	AST	CLB
RCS activity	See Table 4.3-1a	See Table 4.3-1b
Initial secondary system activity	See Table 4.3-1a	See Table 4.3-1b
Pre-accident iodine spike factor	60	60
Accident-initiated iodine spike appearance rate calculations		
Letdown flow, maximum (gpm)	132	75
Letdown flow decontamination (%)	100	N/A
RCS leakage (gpm)	11	1
Spike factor	500	500
Duration of accident-initiated iodine spike (hr)	8	8
RCS mass, maximum (lbm)	8.42E+05	4.94E+05
Equilibrium appearance rates (Ci/min)		
I-130	9.87E-03	N/A
I-131	4.39E-01	N/A
I-132	1.98E+00	N/A
I-133	8.93E-01	N/A
I-134	9.65E-01	N/A
I-135	8.17E-01	N/A
Iodine chemical form of releases (%)		
Elemental	97	N/A
Organic	3	N/A
Particulate	0	N/A
Approximate timing of events		
Safety injection (SI) signal (sec)	30	N/A
Control room isolation (including delay) (sec)	90	N/A
Faulted SG releases all initial activity (min)	2	N/A
RHR cooling takes over (releases from intact SGs terminated) (hr)	12	N/A
RCS cooled below 212°F (releases from faulted SG terminated) (hr)	34	N/A
Mass transfer data		
Initial faulted SG release (in first 2 minutes) (lbm)	165,000	164,500
Total primary-to-secondary leakage		
Leakage through faulted SG to atmosphere (gpm)	1	1
Leakage into intact SGs (gpd, total)	450	N/A
Steam Released from Intact SGs to Atmosphere		
0 to 2 hours (lbm)	419,340	404,452
2 to 12 hours (lbm)	1,310,269	945,973

Table 4.3-6 Assumptions Used for Main Steamline Break Analysis (cont.)

	AST	CLB
RCS Mass, Minimum (lbm)	3.99E+05	4.94E+05 CLB does not use maximum and minimum
Faulted SG Mass, Maximum (lbm)	1.65E+05	164,500
Intact SGs Mass, Minimum (lbm, Total)	2.47E+05	286,500
SG iodine water/steam partition coefficient	100	100
Moisture carryover (%)	0.25	0.25
Control room atmospheric dispersion factors (sec/m^3)		
Intact SGs		
0 – 2 hours	1.04E-03	N/A
2 – 8 hours	7.46E-04	N/A
8 – 24 hours	3.03E-04	N/A
24 – 96 hours	1.90E-04	N/A
96 – 720 hours	1.39E-04	N/A
Faulted SG		
0 – 2 hours	6.12E-04	N/A
2 – 8 hours	4.38E-04	N/A
8 – 24 hours	1.79E-04	N/A
24 – 96 hours	1.14E-04	N/A
96 – 720 hours	8.94E-05	N/A
TSC atmospheric dispersion factors (sec/m^3)		
Intact SGs		
0 – 2 hours	4.83E-04	N/A
2 – 8 hours	2.58E-04	N/A
8 – 24 hours	9.63E-05	N/A
24 – 96 hours	6.45E-05	N/A
96 – 720 hours	4.89E-05	N/A
Faulted SGs		
0 – 2 hours	2.80E-04	N/A
2 – 8 hours	1.80E-04	N/A
8 – 24 hours	6.44E-05	N/A
24 – 96 hours	4.42E-05	N/A
96 – 720 hours	3.22E-05	N/A

Table 4.3-7 Assumptions Used for Loss of Non-Emergency AC Power Analysis

	AST	CLB
RCS activity	See Table 4.3-1a	See Table 4.3-1b
Initial secondary system activity	See Table 4.3-1a	See Table 4.3-1b
Accident-initiated iodine spike appearance rate calculations		
Letdown flow, maximum (gpm)	132	75
Letdown flow decontamination (%)	100	N/A
RCS leakage (gpm)	11	1
Spike factor	500	N/A
Duration of accident-initiated iodine spike (hr)	8	N/A
RCS mass, maximum (lbm)	8.42E+05	4.94E+05 CLB does not use max and min
Equilibrium appearance rates (Ci/min)		
I-130	9.87E-03	N/A
I-131	4.39E-01	N/A
I-132	1.98E+00	N/A
I-133	8.93E-01	N/A
I-134	9.65E-01	N/A
I-135	8.17E-01	N/A
Iodine chemical form of releases (%)		
Elemental	97	N/A
Organic	3	N/A
Particulate	0	N/A
Time RHR cooling matched decay heat (SG releases terminated) (hr)	12	8
Mass transfer data		
Total primary-to-secondary leakage (gpm)	1	1
Steam released from SGs to atmosphere		
0 to 2 hours (lbm)	419,846	549,000
2 to 12 hours (lbm)	1,352,918	1,030,000
RCS mass, minimum (lbm)	3.99E+05	4.94E+05 CLB does not use max and min

Table 4.3-7 Assumptions Used for Loss of Non-Emergency AC Power Analysis (cont.)

Plant total SG mass, minimum (lbm)	**AST**	**CLB**
Until 2 hours (lbm)	3.30E+05	382,000
After 2 hours (lbm)	4.85E+05	382,000
SG iodine water/steam partition coefficient	100	100
Moisture carryover (%)	0.25	0.25
Control room isolation	None	N/A
Control room atmospheric dispersion factors (sec/m^3)		
0 – 2 hours	1.04E-03	N/A
2 – 8 hours	7.46E-04	N/A
8 – 24 hours	3.03E-04	N/A
24 – 96 hours	1.90E-04	N/A
96 – 720 hours	1.39E-04	N/A
TSC atmospheric dispersion factors (sec/m^3)		
0 – 2 hours	4.83E-04	N/A
2 – 8 hours	2.58E-04	N/A
8 – 24 hours	9.63E-05	N/A
24 – 96 hours	6.45E-05	N/A
96 – 720 hours	4.89E-05	N/A

Table 4.3-8 Assumptions Used for Locked Rotor Analysis

	AST	CLB
Core activity	See Table 4.3-1a	See Table 4.3-1b
Failed fuel (% of Core)	5	5
Melted fuel (% of Core)	0	0
Peaking factor	1.65	1.65
Gap fractions		
I-131	0.08	0.12
Kr-85	0.10	0.30
Other iodines and noble gases	0.05	0.10
Iodine chemical form of releases (%)		
Elemental	97	N/A
Organic	3	N/A
Particulate	0	N/A
Time RHR cooling matched decay heat (SG releases terminated) (hr)	12	8
Mass transfer data		
Total primary-to-secondary leakage (gpm)	1	1
Steam released from SGs to atmosphere		
0 to 2 hours (lbm)	419,846	5.49E+05
2 to 12 hours (lbm)	1,352,918	1.03E+06
RCS mass, minimum (lbm)	3.99E+05	4.94E+05
Plant total SG mass, minimum (lbm)		
Until 2 hours (lbm)	3.30E+05	3.82E+05
After 2 hours (lbm)	4.85E+05	3.82E+05
SG iodine water/steam partition coefficient	100	100
Moisture carryover (%)	0.25	0.25
Control room isolation	None	N/A
Control room atmospheric dispersion factors (sec/m^3)		
0 – 2 hours	1.04E-03	N/A
2 – 8 hours	7.46E-04	N/A
8 – 24 hours	3.03E-04	N/A
24 – 96 hours	1.90E-04	N/A
96 – 720 hours	1.39E-04	N/A
TSC atmospheric dispersion factors (sec/m^3)		
0 – 2 hours	4.83E-04	N/A
2 – 8 hours	2.58E-04	N/A
8 – 24 hours	9.63E-05	N/A
24 – 96 hours	6.45E-05	N/A
96 – 720 hours	4.89E-05	N/A

Table 4.3-9 Assumptions Used for Rod Ejection Analysis

	AST	CLB
Core activity	See Table 4.3-1a	See Table 4.3-1b
Failed fuel (% of core)	10	10
Melted fuel (% of core)	0.25	0.25
Peaking factor	1.65	1.65
Gap fractions		
Iodines and noble gases	0.10	0.10
Alkali metals	0.12	N/A
Containment Leakage		
Activity released to containment from failed fuel (%)		
Iodines and noble gases	10	10
Alkali metals	12	N/A
Activity released to containment from melted fuel (%)		
Iodines and alkali metals	50	50
Noble gas	100	100
Iodine chemical form of releases (%)		
Elemental	4.85	91
Particulate	95	5
Organic	0.15	4
Containment leak rates (weight %/day)		
0 – 24 hours	0.2	0.2
1 – 30 days	0.1	0.1
Containment volume (ft^3)	2.5E+06	2.5E+06
Removal of airborne activity in containment (other than leakage or decay)	None	None
SI signal (sec)	150	N/A
Time of control room isolation (including delays) (sec)	210	N/A
Control room atmospheric dispersion factors (sec/m^3)		
0 – 2 hours	5.44E-04	N/A
2 – 8 hours	4.35E-04	N/A
8 – 24 hours	1.62E-04	N/A
24 – 96 hours	1.22E-04	N/A
96 – 720 hours	8.70E-05	N/A

Table 4.3-9 Assumptions Used for Rod Ejection Analysis (cont.)

	AST	CLB
TSC atmospheric dispersion factors (sec/m^3)		
0 – 2 hours	3.91E-04	N/A
2 – 8 hours	2.66E-04	N/A
8 – 24 hours	9.62E-05	N/A
24 – 96 hours	7.05E-05	N/A
96 – 720 hours	5.52E-05	N/A
Primary-to-Secondary Leakage		
Activity released to RCS from failed fuel (%)		
Iodines and noble gases	10	10
Alkali metals	12	N/A
Activity released to RCS from melted fuel (%)		
Iodines and alkali metals	50	50
Noble gas	100	100
Iodine chemical form of releases (%)		
Elemental	97	N/A
Organic	3	N/A
Particulate	0	N/A
Time RHR cooling matched decay heat (SG releases terminated) (hr)	12	8
Mass transfer data		
Total primary-to-secondary leakage (gpm)	1	1
Steam released from SGs to atmosphere		
0 to 2 hours (lbm)	419,846	48,600 (140 sec)
2 to 12 hours (lbm)	1,352,918	N/A
RCS mass, minimum (lbm)	3.99E+05	4.94E+05
Plant total SG mass, minimum (lbm)		
Until 2 hours (lbm)	3.30E+05	4.16E+05
After 2 hours (lbm)	4.85E+05	4.16E+05
SG iodine water/steam partition coefficient	100	100
Moisture carryover (%)	0.25	0.25
Control room isolation	None	N/A

Table 4.3-9 Assumptions Used for Rod Ejection Analysis (cont.)

	AST	CLB
Control room atmospheric dispersion factors (sec/m^3)		
0 – 2 hours	1.04E-03	N/A
2 – 8 hours	7.46E-04	N/A
8 – 24 hours	3.03E-04	N/A
24 – 96 hours	1.90E-04	N/A
96 – 720 hours	1.39E-04	N/A
TSC atmospheric dispersion factors (sec/m^3)		
0 – 2 hours	4.83E-04	N/A
2 – 8 hours	2.58E-04	N/A
8 – 24 hours	9.63E-05	N/A
24 – 96 hours	6.45E-05	N/A
96 – 720 hours	4.89E-05	N/A

Table 4.3-10 Assumptions Used for Letdown Line Break Analysis

	AST	CLB
RCS activity	See Table 4.3-1a	See Table 4.3-1b
Accident-initiated iodine spike appearance rate calculations		
Letdown flow, maximum (gpm)	132	195
Letdown flow decontamination (%)	100	N/A
RCS leakage (gpm)	11	N/A
Spike factor	500	N/A
Duration of accident-initiated iodine spike (hr)	8	N/A
Reactor coolant mass, maximum (lbm)	8.42E+05	4.94E+05
Equilibrium appearance rates (Ci/min)		
I-130	9.87E-03	N/A
I-131	4.39E-01	N/A
I-132	1.98E+00	N/A
I-133	8.93E-01	N/A
I-134	9.65E-01	N/A
I-135	8.17E-01	N/A
Iodine chemical form of releases (%)		
Elemental	97	N/A
Organic	3	N/A
Particulate	0	N/A
Reactor coolant mass, minimum (lbm)	3.99E+05	N/A
Flow rate out of broken line (gpm)	141	141
Iodine and alkali metal airborne fraction	0.18	N/A
Maximum RCS letdown pressure (psig)	600	2200
Maximum RCS letdown temperature (°F)	380	286
Time to isolate break flow (terminating releases) (min)	30.167	30.167
Control room isolation	None	N/A
Control room atmospheric dispersion factor (sec/m^3)		N/A
0 – 2 hours	6.12E-04	N/A
TSC atmospheric dispersion factor (sec/m^3)		
0 – 2 hours	2.80E-04	N/A

Table 4.3-11 Assumptions Used for SGTR Dose Analysis

	AST	CLB
RCS activity	See Table 4.3-1a	See Table 4.3-1c
Initial secondary system activity	See Table 4.3-1a	See Table 4.3-1c
Pre-accident iodine spike factor	60	60
Accident-initiated iodine spike appearance rate calculations		
Letdown flow, maximum (gpm)	132	120
Letdown flow decontamination (%)	100	N/A
RCS leakage (gpm)	11	1
Spike factor	335	335
Reactor coolant mass, maximum (lbm)	8.42E+05	5.05E+5
Duration of accident-initiated iodine spike (hr)	8	8
Equilibrium appearance rates (Ci/min)		
I-130	9.87E-03	N/A
I-131	4.39E-01	3.54E-1
I-132	1.98E+00	1.36E+00
I-133	8.93E-01	7.72E+00
I-134	9.65E-01	7.02E-01
I-135	8.17E-01	6.63E-01
Iodine chemical form of releases (%)		
Elemental	97	N/A
Organic	3	N/A
Particulate	0	N/A
Approximate timing of events (sec)	See Table 2.7.3-2 in Enclosure I of this LAR (For dose input, exclude the 100 seconds of steady-state operation.)	N/A
Time of control room isolation (including delay) (sec)	385	N/A

Table 4.3-11 Assumptions Used for SGTR Dose Analysis (cont.)

	AST	CLB
Transient mass transfer data		
Non-flashed break flow (lbm)		
0 – 52 seconds	2227.5	N/A
52 – 1102 seconds	43,129.9	N/A
1102 – 2902 seconds	88,387.2	N/A
2902 – 3502 seconds	32,991.2	N/A
3502 – 3846 seconds	18,224.8	N/A
3846 – 5155 seconds	61,523.0	N/A
5155 – 7527 seconds	41,166.4	N/A
Flashed break flow (lbm)		
0 – 52 seconds	438.9	N/A
52 – 1102 seconds	2,901.8	N/A
1102 – 2902 seconds	13,432.1	N/A
2902 – 3502 seconds	2,635.6	N/A
3502 – 3846 seconds	606.1	N/A
Steam released from ruptured SG (lbm)		
0 – 52 seconds	188,100	N/A
52 – 1102 seconds	27,469.2	N/A
1102 – 2902 seconds	149,850.8	N/A
2902 – 7527 seconds	0	N/A
7527 – 43,200 seconds	2530	N/A
Steam released from intact SGs (lbm)		
0 – 52 seconds	562,650	N/A
52 – 1102 seconds	69,877.5	N/A
1102 – 3502 seconds	0	N/A
3502 – 3846 seconds	94,307.4	N/A
3846 – 5155 seconds	130,799.9	N/A
5155 – 7527 seconds	98,156.3	N/A
7527 – 43,200 seconds	1,645,930	N/A
Reactor coolant mass, minimum (lbm)	3.99E+05	5.05E+05 CLB does not use max and min
Ruptured SG mass, minimum (lbm)	7.00E+04	N/A

Table 4.3-11 Assumptions Used for SGTR Dose Analysis (cont.)

	AST	CLB
Intact SGs mass, minimum (lbm, total)	1.95E+05	N/A
Condenser iodine and alkali metal removal factor	100	N/A
SG iodine water/steam partition coefficient	100	N/A
Moisture carryover (%)	0.25	0.25
Control room atmospheric dispersion factors (sec/m^3)		
0 – 2 hours	1.04E-03	N/A
2 – 8 hours	7.46E-04	N/A
8 – 24 hours	3.03E-04	N/A
24 – 96 hours	1.90E-04	N/A
96 – 720 hours	1.39E-04	N/A
TSC atmospheric dispersion factors (sec/m^3)		
0 – 2 hours	4.83E-04	N/A
2 – 8 hours	2.58E-04	N/A
8 – 24 hours	9.63E-05	N/A
24 – 96 hours	6.45E-05	N/A
96 – 720 hours	4.89E-05	N/A

Table 4.3-12 Assumptions Used for LOCA Analysis

	AST		CLB	
Core activity (containment leakage, ECCS leakage, and RWST back-leakage)	See Table 4.3-1a		See Table 4.3-1b	
RCS activity (containment purge)	See Table 4.3-1a		See Table 4.3-1b	
Fuel release fractions and timing			100% of the core activity is released immediately following event initiation	
Nuclide Group	Gap Release Phase Fraction	Early In-Vessel Phase Fraction	Gap Release Phase Fraction	Early In-Vessel Phase Fraction
Noble gases	0.05	0.95	N/A	N/A
Iodines	0.05	0.35	N/A	N/A
Alkali metals	0.05	0.25	N/A	N/A
Tellurium metals	0.00	0.05	N/A	N/A
Barium and strontium	0.00	0.02	N/A	N/A
Noble metals	0.00	0.0025	N/A	N/A
Cerium	0.00	0.0005	N/A	N/A
Lanthanides	0.00	0.0002	N/A	N/A
Duration of phases				
Phase	Onset	Duration	Onset	Duration
Gap release	30 sec	0.49167 hr	N/A	N/A
Early in-vessel	0.5 hr	1.3 hr	N/A	N/A
SI signal (sec)	0		N/A	
Time of control room isolation (including delays) (sec)	120		0	
Containment Leakage				
Iodine chemical form of releases (%)				
Elemental	4.85		91	
Organic	0.15		4	
Particulate	95		5	
Containment volume, maximum (ft^3)	2.7E+06		2.5E+06	
% Sprayed	85		85	
% Unsprayed	15		15	

Table 4.3-12 Assumptions Used for LOCA Analysis (cont.)

	AST	CLB
Mixing between sprayed and unsprayed containment volumes (cfm)	6.94E+04	8.50E+04
Start of fan cooler mixing (min)	2	N/A
Containment leak rates (weight %/day)		
0 – 24 hours	0.2	0.2
1 – 30 days	0.1	0.1
Spray timing		
Initiation (min)	2	0
Termination (hr)	5	720
Spray removal coefficients		
Organic iodine spray removal coefficient (hr^{-1})	0.0	0.0
Elemental iodine spray removal coefficient calculations		
Spray removal coefficient (hr^{-1}) , DF < 200	10	10
Gas phase mass transfer coefficient (m/min)	3	N/A
Time of fall of the spray drops (min)	0.146	N/A
Volume flow rate of sprays (m^3/hr)	658.66	711.13
Containment sprayed volume (m^3)	6.5E+04	6.0E+04
Mass-mean diameter of the spray drops (m)	0.00116	N/A
Particulate spray removal coefficient calculations		
Spray removal coefficient (hr^{-1}), DF < 50	5	0.45 (Used DFs up to 100)
Spray removal coefficient (hr^{-1}), DF > 50	0.5	N/A
Drop fall height (m)	35.966	36.017
Volume flow rate of sprays (m^3/hr)	658.66	711.13
Containment sprayed volume (m^3)	6.5E+04	6.0E+04
Ratio of dimensionless collection efficiency to average spray drop diameter		
Prior to DF of 50 (m^{-1})	10	N/A
After DF of 50 (m^{-1})	1	N/A
Particulate sedimentation removal coefficient (hr^{-1}), DF < 1000	0.1	N/A

Table 4.3-12 Assumptions Used for LOCA Analysis (cont.)

	AST	CLB
pH of sump	≥ 7.0	≥ 8.5
Control room atmospheric dispersion factors (sec/m^3)		
0 – 2 hours	5.44E-04	5.30E-04
2 – 8 hours	4.35E-04	5.30E-04
8 – 24 hours	1.62E-04	3.6E-04
24 – 96 hours	1.22E-04	6.60E-05
96 – 720 hours	8.70E-05	0
TSC atmospheric dispersion factors (sec/m^3)		
0 – 2 hours	3.91E-04	2.2E-04
2 – 8 hours	2.66E-04	2.2E-04
8 – 24 hours	9.62E-05	1.17E-04
24 – 96 hours	7.05E-05	2.04E-05
96 – 720 hours	5.52E-05	0
ECCS Leakage		
Iodine chemical form of releases (%)		
Elemental	97	N/A
Organic	3	N/A
Particulate	0	N/A
Sump volume (gal)	4.60E+05	4.60E+05
Time to initiate ECCS recirculation (min)	0	28.2
ECCS leakage to auxiliary building (gpm)	2	2
Iodine airborne fraction	0.10	0.10
Auxiliary building exhaust filter efficiency (all forms of iodine) (%)	90	90
Control room atmospheric dispersion factors (sec/m^3)		
0 – 2 hours	6.12E-04	1.10E-04
2 – 8 hours	4.38E-04	1.10E-04
8 – 24 hours	1.79E-04	6.80E-05
24 – 96 hours	1.14E-04	1.70E-05
96 – 720 hours	8.94E-05	0

Table 4.3-12 Assumptions Used for LOCA Analysis (cont.)

	AST	CLB
TSC atmospheric dispersion factors (sec/m^3)		
0 – 2 hours	2.80E-04	2.2E-04
2 – 8 hours	1.80E-04	2.2E-04
8 – 24 hours	6.44E-05	1.17E-04
24 – 96 hours	4.42E-05	2.04E-05
96 – 720 hours	3.22E-05	0
RWST Back-Leakage		
RWST initial gas volume, minimum (gal)	3.54E+05	N/A
Time to initiate ECCS recirculation (min)	0	28.2
ECCS leakage to RWST (gpm)	3.8	3.8
Iodine airborne fraction	0.10	0.10
Release from RWST gas space (gpm)	3.8	3.8
Iodine chemical form of releases (%)		
Elemental	97	91
Organic	3	4
Particulate	0	5
Control room atmospheric dispersion factors (sec/m^3)		
0 – 2 hours	6.80E-04	1.10E-04
2 – 8 hours	6.19E-04	1.10E-04
8 – 24 hours	2.27E-04	6.80E-05
24 – 96 hours	1.96E-04	1.70E-05
96 – 720 hours	1.53E-04	0
TSC atmospheric dispersion factors (sec/m^3)		
0 – 2 hours	1.87E-04	2.2E-04
2 – 8 hours	1.25E-04	2.2E-04
8 – 24 hours	4.51E-05	1.17E-04
24 – 96 hours	3.33E-05	2.04E-05
96 – 720 hours	2.61E-05	0

Table 4.3-12 Assumptions Used for LOCA Analysis (cont.)

	AST	CLB
Containment Purge		
RCS activity released (%)	100	100
Iodine chemical form of releases (%)		
Elemental	97	91
Organic	3	4
Particulate	0	5
RCS mass, maximum (lbm)	8.42E+05	4.94E+05 CLB does not use maximum and minimum
Containment volume, minimum (ft^3)	2.5E+06	2.5E+06
Maximum purge flow rate, unfiltered (cfm)	4,680	4,680
Duration of purge release (sec)	10	8
Control room atmospheric dispersion factors (sec/m^3)		
0 – 2 hours	6.12E-04	1.10E-04
2 – 8 hours	4.38E-04	1.10E-04
8 – 24 hours	1.79E-04	6.80E-05
24 – 96 hours	1.14E-04	6.60E-05
96 – 720 hours	8.94E-05	0
TSC atmospheric dispersion factors (sec/m^3)		
0 – 2 hours	2.80E-04	2.2E-04
2 – 8 hours	1.80E-04	2.2E-04
8 – 24 hours	6.44E-05	1.17E-04
24 – 96 hours	4.42E-05	2.04E-05
96 – 720 hours	3.22E-05	0

Table 4.3-13 Assumptions Used for Waste Gas Decay Tank Failure Analysis		
	AST	**CLB**
Activity in ruptured tank	See Table 4.3-2a	See Table 4.3-2b
Iodine chemical form of releases (%)		
Elemental	100	N/A
Duration of release (hr)	2	2
Control room isolation	None	N/A
Control room atmospheric dispersion factor (sec/m^3)		
0 – 2 hours	6.80E-04	5.30E-04
TSC atmospheric dispersion factor (sec/m^3)		
0 – 2 hours	1.87E-04	N/A

Table 4.3-14 Assumptions Used for Liquid Waste Tank Failure Analysis		
	AST	**CLB**
Activity in ruptured tank		
Recycle holdup tank	See Table 4.3-2a	See Table 4.3-2b
Hypothetical tank maximizing iodine	See Table 4.3-2a	N/A
Iodine chemical form of releases (%)		
Elemental	100	N/A
Duration of release (hr)	2	2
Control room isolation	None	N/A
Control room atmospheric dispersion factor (sec/m^3)		
0 – 2 hours	6.80E-04	N/A
TSC atmospheric dispersion factor (sec/m^3)		
0 – 2 hours	1.87E-04	N/A

Table 4.3-15 Assumptions Used for Fuel Handling Accident Analysis

	AST	CLB
Core activity for one assembly at minimum time prior to fuel movement (Ci)		
Kr-85m	1.10E+00	See Table 4.3-4c
Kr-85	5.69E+03	See Table 4.3-4c
Xe-131m	5.38E+03	See Table 4.3-4c
Xe-133m	1.76E+04	See Table 4.3-4c
Xe-133	8.19E+05	See Table 4.3-4c
Xe-135m	5.58E+01	See Table 4.3-4c
Xe-135	8.14E+03	See Table 4.3-4c
I-130	1.45E+02	See Table 4.3-4c
I-131	4.12E+05	See Table 4.3-4c
I-132	3.95E+05	See Table 4.3-4c
I-133	8.90E+04	See Table 4.3-4c
I-135	3.42E+02	See Table 4.3-4c
Number of fuel assemblies damaged	1.2	1.2
Peaking factor	1.65	1.65
Control room isolation	None	N/A
Decay time prior to fuel movement, minimum (hr)	76	76
Gap fractions		
I-131	0.12	0.12
Kr-85	0.30	0.30
Other iodines and noble gases	0.10	0.10
Iodine chemical form in gap (%)		
Elemental	99.85	N/A
Organic	0.15	N/A
Fuel pool water depth, minimum (ft)	23	23
Fuel rod internal pressure, maximum (psig)	1500	1200
Overall pool iodine DF	200	100
Iodine airborne fractions (%)		
Elemental	70	N/A
Organic	30	N/A

Table 4.3-15 Assumptions Used for Fuel Handling Accident Analysis (cont.)

	AST	CLB
Duration of release (hr)	2	2
Removal of airborne activity in containment/fuel building (other than decay)	None	None
Control room atmospheric dispersion factor (sec/m^3)		
0 – 2 hours	6.12E-04	5.30E-04
TSC atmospheric dispersion factor (sec/m^3)		
0 – 2 hours	3.91E-04	N/A

Table 4.3-16 Technical Support Center Parameters

	AST	CLB
TSC volume (ft^3)	44,000	52,800
Normal ventilation flow rates (cfm)		
Unfiltered makeup flow rate	550	550
Unfiltered inleakage	20	20
Emergency mode flow rates (cfm)		
Filtered makeup flow rate	550	550
Unfiltered makeup flow rate	0	N/A
Unfiltered inleakage	20	20
Filtered recirculation flow	450	550
Filter efficiencies (%)		
Elemental iodine	95	90
Organic iodine	95	90
Particulates	95	90
Delay to switch to emergency mode operation after event initiation (minutes)	60	0
TSC breathing rate for duration of the event (m^3/sec)	3.5E-04	3.5E-04
TSC occupancy factors		
0 – 24 hours	1.0	1.0
1 – 4 days	0.6	0.6
4 – 30 days	0.4	0.4

4.4 CONTAINMENT SUMP PH

4.4.1 Sampling System

This section describes the ability of WCNOC to sample the containment sumps in post-accident configurations.

Following the transition to cold leg recirculation, the WCGS Procedures (e.g., CHS AC-001, CHS SJ-143A) provide instructions for sampling the containment sumps. Information from the sample analysis may be used by the operating staff, in concert with the Technical Support Center personnel, to evaluate the long term plant status including sump pH. An initial sample could be taken per plant procedures consistent with plant equipment availability and existing environmental conditions. The time needed for the initial sample would be event-specific, however it is reasonable to assume that the sampling activity could be completed within 24 hours of the accident.

As approved per Amendment No. 137 to the Operating License, WCNOC eliminated the Post Accident Sampling System (PASS). Consistent with this amendment WCNOC maintains contingency plans for obtaining and analyzing highly radioactive samples from the containment sump.

Samples would be taken from two points on the RHR Loops A and B within the Auxiliary Building following the transition to cold leg recirculation. With suction flow coming from the containment sump(s) and a fully mixed sump solution, these sampling points provide a good indication of conditions within the sump solution.

4.4.2 Containment Sump pH Evaluation

4.4.2.1 Analysis Bases Summary

The following summary of the containment sump pH analysis describes the performance of the systems and components that are employed to control the pH of the containment sump water inventory under post-accident conditions.

Background

Under LOCA conditions, buffering agents must be added to the containment sump fluid that is recirculated by the ECCS to increase the coolant pH to greater than 7.0. Buffering agent addition is mainly required to reduce release of iodine fission products from the coolant to the containment atmosphere as iodine gas. Thus, pH control is primarily an offsite dose control measure. Increasing the coolant pH also reduces the corrosion rates of most materials in the containment sump, most notably stainless steel structural members and components. Sodium hydroxide (NaOH) is used as the buffering agent at the WCGS. The NaOH is introduced into the containment sump fluid via the Spray Additive Tank (SAT) and associated spray eductor in the containment spray system (CSS).

Methodology

In order to calculate the minimum sump pH, the maximum amount of boric acid from the various sources of borated water that enter the containment sump post-LOCA combined with a minimum amount of caustic from various sources (mainly the SAT) will yield a minimum pH value. The concentrations of these substances are used to compute the value of the sump pH as a function of time using verified titration curve data for aqueous solutions of boric acid and sodium hydroxide.

The CLB for the WCGS sump pH analysis forms the basis of existing USAR Figure 6.5-5. Two scenarios are analyzed: both Containment Spray trains operating with one NaOH eductor in service, and both Containment Spray trains operating with both NaOH eductors in service.

No computer codes were used in this analysis. To determine the pH in the CLB analysis, NaOH and H_3BO_3 molarities were first calculated and then used with Oak Ridge National Laboratory (ORNL) titration curve data to determine the pH of the spray and sump solutions during the injection and recirculation phases of ECCS operation.

Westinghouse has subsequently performed an independent analysis to update the conclusions of the CLB analysis. The Westinghouse analysis used pH data for boric acid/sodium hydroxide solutions from verified titration curve data for aqueous solutions of boric acid and sodium hydroxide.

Assumptions and Inputs

Among the various assumptions that were made in the sump pH analysis is that of perfect mixing of the water inventory in the sump. This is a valid assumption because of the long term operation of the recirculation spray system and the uniform dispersion of spray over the containment cross section.

The minimum long-term sump pH in the CLB calculation uses the following conservative bases:

1. Maximum Refueling Water RWST and SI accumulator Technical Specification boron concentration of 2500 ppm.

2. A conservatively high RCS boron concentration of 1900 ppm in the CLB, and 1980 ppm in the Westinghouse analysis. Given that the RCS is only ~15% of the sump fluid, this difference in boron concentration is a small effect.

3. The SAT is assumed to contain the Technical Specification minimum 28% NaOH solution.

While the SAT Technical Specification minimum contained volume is 4340 gal., the CLB assumed a conservative minimum delivered volume of only 2960 gal. The Westinghouse analysis used an updated delivered volume of 3060 gallons.

Results and Conclusions

Existing Figure 6.5-5 in the Wolf Creek USAR shows that the post-LOCA sump pH remains well above 7 throughout the event. This conclusion was based on the current licensing basis discussed in USAR Section 6.5.2.3, Table 3.11(B)-5, and original project calculations.

The Westinghouse analysis determined a long-term sump pH of 8.8, while the CLB analysis determined a minimum value of 8.6. The slightly higher pH in the Westinghouse analysis is attributed to two factors: 1) a higher minimum volume of NaOH and 2) a different boron/NaOH/pH correlation. In either case, the results show that the sump pH remains well above the Nuclear Regulatory Commission (NRC)-required value of 7.0 after the minimum amount of NaOH is injected into the sump.

In the CLB analysis, the limiting system alignment of both Containment Spray trains operating with one NaOH educator in service results in a sump pH of 7 in approximately 15 minutes, with the final long-term pH of 8.6 in approximately 80 minutes. The less limiting case of both Containment Spray trains operating with both NaOH supplies in service results in a sump pH of 7 in approximately 11 minutes, with the final long-term pH of 8.6 in approximately 45 minutes. For the Westinghouse analysis, the limiting system alignment of both Containment Spray trains operating with one NaOH educator in service results in a sump pH of 7 in approximately 11 minutes, with the final long-term pH of 8.8 in approximately 80 minutes. The less limiting case of both Containment Spray trains operating with both NaOH supplies in service results in a sump pH of 7 in approximately 9 minutes, with the final long-term pH of 8.6 in approximately 40 minutes.

The calculations did not include consideration of acid generation (nitric acid produced by the irradiation of water and air or hydrochloric acid produced by the radiolysis of chlorine bearing materials) as they were considered secondary effects. Recently, the staff summarized the effects of acid generation for a Westinghouse PWR in the Byron Safety Evaluation (ML062340420). The conclusion reached therein was that the effect of these phenomena on sump pH would be to decrease the pH value less than 0.1 pH unit, a negligible effect. It is expected that a similar result would be the case for WCGS from inspection of Table 4.4-1, which contains a comparison of the parameters listed in the Byron Safety Evaluation with the WCGS parameters used in the CLB analysis for the conservatively calculated minimum pH case.

Table 4.4-1 Comparison of Plant Design Parameters Used in Minimum Sump pH Analyses

	Design Parameters			
	Byron		WCGS CLB	
Component	Vol. or Mass	Concentration	Vol. or Mass	Concentration
RWST	457,904 gal	2500 ppm boron	379,800 gallons	2500 ppm boron
RCS	620,800 lb_m	2300 ppm boron	504,520 lb_m	1900 ppm boron
SI Accumulators	28,868 gal	2400 ppm boron	26,376 gal	2500 ppm boron
SAT	2500 gal	30 wt% NaOH	2960 gal	28 wt% NaOH

Furthermore, back leakage of sump fluid through the RWST is not considered in the post-LOCA analysis based on the plant emergency procedure for transfer to cold leg recirculation that requires closure of the 24" RWST outlet valve within 16 hours of SI initiation to limit releases from the RWST to the atmosphere. This results in three valve isolation and a minimum of two valve isolation in the long term with a single failure. Refer to USAR Table 6.3-5. Therefore, there is no need to address the potential issue of reduced pH in the RWST, which could lead to a potential radioactive iodine release from the RWST to the environment.

4.4.3 Containment Sump pH Calculation

This section summarizes the calculation to determine the minimum sump pH versus time after a large LOCA.

Inputs

Sources of boron include the RCS, the RWST, and Safety Injection Accumulators. Volumes of these sources and their boron concentrations are maximized to minimize the calculated sump pH. Sodium hydroxide (NaOH), from the Spray Additive Tank, is injected at a minimum volume and concentration to minimize the sump pH results.

Methodology

The RCS and accumulator volumes are assumed to enter the sump at time = 0. The RWST volume then enters the sump at maximum ECCS and CSS flow rates. WCNOC provided inputs for the RWST volume, and ECCS and CSS flows. The NaOH flow enters the sump at a constant rate of 40 gpm per train. An average boron and NaOH concentration is then calculated at various time intervals, and a Westinghouse proprietary empirical correlation of boron, NaOH, and pH is used to determine sump pH.

Acceptance Criteria

Per the Wolf Creek USAR and Technical Specification Bases, the minimum long-term sump pH is 8.5.

Results

Figure 4-1 depicts the results, and is also listed as a proposed USAR markup in Section 13 to replace the existing USAR Figure 6.5-5.

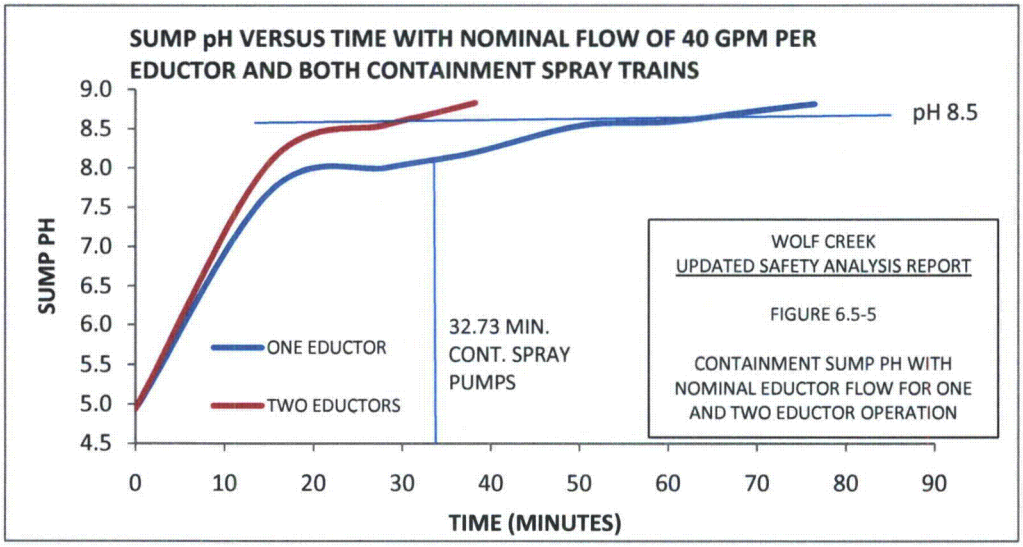


Figure 4-1 Containment Sump pH versus Time

Figure 4-1 shows that the acceptance criteria of 8.5 (minimum) is met for one or two eductors in service. The final sump pH of 8.8 is maintained after approximately 40 minutes after the LOCA occurs (two eductors) or approximately 80 minutes (one eductor). This result slightly exceeds the current WCGS analysis of record pH value, which is 8.6 achieved at times of approximately 40 and 80 minutes, respectively.

Licensing Documentation Changes

Only USAR Figure 6.5-5 needs to be changed, and the USAR markup is provided in Section 13. No other USAR or Technical Specification changes are required.

5 REGULATORY GUIDE 1.183 CONFORMANCE TABLE

Regulatory Guide 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants" – Conformance Tables

NOTE: In Tables A – G, the text shown in the "RG Position" columns is taken from Regulatory Guide 1.183. Therefore, references to footnotes, tables, and numbered references may be found in the regulatory guide.

Table A Conformance with Regulatory Guide 1.183 Main Sections

Table B Conformance with Regulatory Guide 1.183 Appendix A (Loss-of-Coolant-Accident)

Table C Conformance with Regulatory Guide 1.183 Appendix B (Fuel Handling Accident)

Table D Conformance with Regulatory Guide 1.183 Appendix E (PWR Main Steam Line Break)

Table E Conformance with Regulatory Guide 1.183 Appendix F (PWR Steam Generator Tube Rupture Accident)

Table F Conformance with Regulatory Guide 1.183 Appendix G (PWR Locked Rotor Accident)

Table G Conformance with Regulatory Guide 1.183 Appendix H (PWR Rod Ejection Accident)

REGULATORY GUIDE 1.183 COMPARISON

Table A	Conformance with Regulatory Guide 1.183 Main Sections		
RG Section	**RG Position**	**Analysis**	**Comments**
3.1	The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty.[8] The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values.[9] The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 (Ref. 17) or ORIGEN-ARP (Ref. 18). Core inventory factors (Ci/MWt) provided in TID14844 and used in some analysis computer codes were derived for low burnup, low enrichment fuel and should not be used with higher burnup and higher enrichment fuels.	Conforms	The inventory of fission products in the reactor core and available for release to the containment was based on the maximum full power operation with a core thermal power of 3637 MWt (102% of 3565 MWt nominal power). Core design parameters (enrichment, burnup, and MTU loading) are based on the cycle 19 core design. Margin is added to the EOC core inventory, calculated with ORIGEN-S, to account for potential core design differences in future cycles. The magnitude of this margin is based on sensitivity studies that consider variations in enrichment and burnup.
3.1	For the DBA LOCA, all fuel assemblies in the core are assumed to be affected and the core average inventory should be used. For DBA events that do not involve the entire core, the fission product inventory of each of the damaged fuel rods is determined by dividing the total core inventory by the number of fuel rods in the core. To account for differences in power level across the core, radial peaking factors from the facility's core operating limits report (COLR) or technical specifications should be applied in determining the inventory of the damaged rods.	Conforms	For the DBA LOCA, all fuel assemblies were assumed to be affected and the core average inventory was used. A peaking factor of 1.65 was used for DBA events that do not involve the entire core (fuel handling accident, rod ejection, locked rotor), with fission product inventories for damages fuel rods determined by multiplying the total core inventory by the fraction of damaged rods.
3.1	No adjustment to the fission product inventory should be made for events postulated to occur during power operations at less than full rated power or those postulated to occur at the beginning of core life. For events postulated to occur while the facility is shutdown, e.g., a fuel handling accident, radioactive decay from the time of shutdown may be modeled.	Conforms	No adjustments for less than full power were made in any analysis. For the fuel handling accident, 76-hours of radioactive decay after shutdown was modeled.

Table A (cont.) Conformance with Regulatory Guide 1.183 Main Sections

RG Section	RG Position	Analysis	Comments
3.2	The core inventory release fractions[10], by radionuclide groups, for the gap release and early in-vessel damage phases for DBA LOCAs are listed in Table 1 for BWRs and Table 2 for PWRs. These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1. **Table 2** **PWR Core Inventory Fraction Released Into Containment** **Group** / **Gap Release Phase** / **Early In-Vessel Phase** / **Total** Noble Gases 0.05 0.95 1.0 Halogens 0.05 0.35 0.4 Alkali Metals 0.05 0.25 0.3 Tellurium Metals 0.00 0.05 0.05 Ba, Sr 0.00 0.02 0.02 Noble Metals 0.00 0.0025 0.0025 Cerium Group 0.00 0.0005 0.0005 Lanthanides 0.00 0.0002 0.0002	Conforms	For the LOCA event, the core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage phases in Table 2 were utilized.
3.2	For non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in Table 3. The release fractions from Table 3 are used in conjunction with the fission product inventory calculated with the maximum core radial peaking factor. **Table 3[11]** **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	Conforms	For non-LOCA events, the fraction of the core inventory assumed to be in the gap by radionuclide group in Table 3 were utilized in conjunction with the maximum core radial peaking factor of 1.65. The control rod ejection accident was evaluated per Footnote 11 of RG 1.183 (the gap fractions are assumed to be 10% for iodines and noble gases). To account for possible damage to an assembly with high burnup and rod power and to address Footnote 11, the fuel handling accident used conservatively high gap fractions of 12% for I-131, 30% for Kr-85, and 10% for all other iodines and noble gases. These gap fractions were obtained from NUREG/CR-5009.

Table A (cont.) **Conformance with Regulatory Guide 1.183 Main Sections**

RG Section	RG Position	Analysis	Comments
3.3	Table 4 tabulates the onset and duration of each sequential release phase for DBA LOCAs at PWRs and BWRs. The specified onset is the time following the initiation of the accident (i.e., time = 0). The early in-vessel phase immediately follows the gap release phase. The activity released from the core during each release phase should be modeled as increasing in a linear fashion over the duration of the phase.[12] For non-LOCA DBAs in which fuel damage is projected, the release from the fuel gap and the fuel pellet should be assumed to occur instantaneously with the onset of the projected damage. **Table 4** **LOCA Release Phases** **PWRs** / **BWRs** **Phase** / **Onset** / **Duration** / **Onset** / **Duration** Gap Release / 30 sec / 0.5 hr / 2 min / 0.5 hr Early In-Vessel / 0.5 hr / 1.3 hr / 0.5 hr / 1.5 hr	Conforms	The Table 4 PWR onset and durations for the DBA LOCA releases were utilized in the analysis. Note that the gap release was modeled beginning at 30 seconds and ending in the first half hour in order to model the early in-vessel release beginning at 0.5 hr.
3.3	For facilities licensed with leak-before-break methodology, the onset of the gap release phase may be assumed to be 10 minutes. A licensee may propose an alternative time for the onset of the gap release phase, based on facility-specific calculations using suitable analysis codes or on an accepted topical report shown to be applicable to the specific facility. In the absence of approved alternatives, the gap release phase onsets in Table 4 should be used.	Not Applicable	No additional delays in gap release were assumed for the DBA analyses.

Table A (cont.)	Conformance with Regulatory Guide 1.183 Main Sections		
RG Section	**RG Position**	**Analysis**	**Comments**
3.4	Table 5 lists the elements in each radionuclide group that should be considered in design basis analyses. **Table 5** **Radionuclide Groups** **Group** — **Elements** Noble Gases — Xe, Kr Halogens — I, Br Alkali Metals — Cs, Rb Tellurium Group — Te, Sb, Se, Ba, Sr Noble Metals — Ru, Rh, Pd, Mo, Tc, Co Lanthanides — La, Zr, Nd, Eu, Nb, Pm, Pr Sm, Y, Cm, Am Cerium — Ce, Pu, Np	Conforms	The Table 5 elements in each radionuclide group were utilized in DBA analyses. Note that since RADTRAD is limited to modeling 63 nuclides, certain nuclides which were deemed to be insignificant from a dose perspective were not included.
3.5	Of the radioiodine released from the reactor coolant system (RCS) to the containment in a postulated accident, 95 percent of the iodine released should be assumed to be cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. This includes releases from the gap and the fuel pellets. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form. The same chemical form is assumed in releases from fuel pins in FHAs and from releases from the fuel pins through the RCS in DBAs other than FHAs or LOCAs. However, the transport of these iodine species following release from the fuel may affect these assumed fractions. The accident-specific appendices to this regulatory guide provide additional details.	Conforms	For releases from the reactor coolant system (RCS) to the containment, 95% of the iodine released was assumed to be cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. Fission products were assumed to be in particulate form with the exception of elemental and organic iodine and noble gases,

Table A (cont.)	Conformance with Regulatory Guide 1.183 Main Sections		
RG Section	**RG Position**	**Analysis**	**Comments**
3.6	The amount of fuel damage caused by non-LOCA design basis events should be analyzed to determine, for the case resulting in the highest radioactivity release, the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and the fraction of fuel elements for which the fuel clad is breached. Although the NRC staff has traditionally relied upon the departure from nucleate boiling ratio (DNBR) as a fuel damage criterion, licensees may propose other methods to the NRC staff, such as those based upon enthalpy deposition, for estimating fuel damage for the purpose of establishing radioactivity releases.	Conforms	The amount of fuel damage caused by non-LOCA design basis events was analyzed. The conservatively calculated values were reflected in the rod ejection and locked rotor DBA analyses.
4.1.1	The dose calculations should determine the TEDE. TEDE is the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure. The calculation of these two components of the TEDE should consider all radionuclides, including progeny from the decay of parent radionuclides, that are significant with regard to dose consequences and the released radioactivity.[13]	Conforms	The dose calculations determine the TEDE and consider all radionuclides that are significant with regard to dose consequences. Progeny was not included in the dose calculations consistent with previously approved submittals, including: • Point Beach Units 1 & 2 – April 2011 (ADAMS Accession Number ML110240054) • Arkansas Nuclear One, Unit 2 – April 2011 (ADAMS Accession Number ML110980197)
4.1.2	The exposure-to-CEDE factors for inhalation of radioactive material should be derived from the data provided in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers" (Ref. 19). Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (Ref. 20), provides tables of conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the CEDE.	Conforms	CEDE Conversion factors for isotopes were taken from Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

Table A (cont.)	Conformance with Regulatory Guide 1.183 Main Sections		
RG Section	**RG Position**	**Analysis**	**Comments**
4.1.3	For the first 8 hours, the breathing rate of persons offsite should be assumed to be 3.5 x 10^{-4} cubic meters per second. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.8 x 10-4 cubic meters per second. After that and until the end of the accident, the rate should be assumed to be 2.3 x 10^{-4} cubic meters per second.	Conforms	The breathing rates provided were utilized to calculate the offsite dose consequences. For determining a limiting 2-hour EAB dose, a constant breathing rate of 3.5 x 10^{-4} cubic meters per second was used.
4.1.4	The DDE should be calculated assuming submergence in semi-infinite cloud assumptions with appropriate credit for attenuation by body tissue. 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 may be used in lieu of DDE in determining the contribution of external dose to the TEDE. Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil" (Ref. 21), provides external EDE conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the EDE.	Conforms	EDE Conversion factors for isotopes were taken from Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil."
4.1.5	The TEDE should be determined for the most limiting person at the EAB. The maximum EAB TEDE for any two-hour period following the start of the radioactivity release should be determined and used in determining compliance with the dose criteria in 10 CFR 50.67.[14] The maximum two-hour TEDE should be determined by calculating the postulated dose for a series of small time increments and performing a "sliding" sum over the increments for successive two-hour periods. The maximum TEDE obtained is submitted. The time increments should appropriately reflect the progression of the accident to capture the peak dose interval between the start of the event and the end of radioactivity release (see also Table 6).	Conforms	The TEDE was determined for the most limiting person at the EAB. The maximum two-hour TEDE was determined by calculating the postulated dose for a series of small time increments and performing a "sliding" sum over the increments for successive two-hour periods. This was performed by the RADTRAD computer code with constant inputs for atmospheric dispersion factors and breathing rates.
4.1.6	TEDE should be determined for the most limiting receptor at the outer boundary of the low population zone (LPZ) and should be used in determining compliance with the dose criteria in 10 CFR 50.67.	Conforms	The TEDE was determined for the most limiting receptor at the outer boundary of the low population zone (LPZ).

Table A (cont.)	Conformance with Regulatory Guide 1.183 Main Sections		
RG Section	**RG Position**	**Analysis**	**Comments**
4.1.7	No correction should be made for depletion of the effluent plume by deposition on the ground.	Conforms	No correction was made for the depletion of the effluent plume by deposition on the ground.
4.2.1	The TEDE analysis should consider all sources of radiation that will cause exposure to control room personnel. The applicable sources will vary from facility to facility, but typically will include: • Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility, • Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope, • Radiation shine from the external radioactive plume released from the facility, • Radiation shine from radioactive material in the reactor containment, • Radiation shine from radioactive material in systems and components inside or external to the control room envelope, e.g., radioactive material buildup in recirculation filters.	Conforms	The TEDE analysis considered all significant sources of radiation that would cause exposure to Control Room personnel. For WCGS, the limiting Control Room dose included: • Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility, • Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from the Control Building, • Radiation shine from the external radioactive plume released from the facility, • Radiation shine from radioactive material in the reactor containment, • Radiation shine from radioactive material in Control Room recirculation filters and radioactive material in the Control Building.
4.2.2	The radioactive material releases and radiation levels used in the control room dose analysis should be determined using the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values, unless these assumptions would result in nonconservative results for the control room.	Conforms	The radioactive material releases and radiation levels used in the Control Room dose analyses were determined using the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values.

Table A (cont.)	Conformance with Regulatory Guide 1.183 Main Sections		
RG Section	**RG Position**	**Analysis**	**Comments**
4.2.3	The models used to transport radioactive material into and through the control room,[15] and the shielding models used to determine radiation dose rates from external sources, should be structured to provide suitably conservative estimates of the exposure to control room personnel.	Conforms	The models used to transport radioactive material into and through the Control Room, and the shielding models used to determine radiation dose rates from external sources, were developed to provide suitably conservative estimates of the exposure to Control Room personnel.
4.2.4	Credit for engineered safety features that mitigate airborne radioactive material within the control room may be assumed. Such features may include control room isolation or pressurization, or intake or recirculation filtration. Refer to Section 6.5.1, "ESF Atmospheric Cleanup System," of the SRP (Ref. 3) and Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Postaccident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" (Ref. 25), for guidance. The control room design is often optimized for the DBA LOCA and the protection afforded for other accident sequences may not be as advantageous. In most designs, control room isolation is actuated by engineered safeguards feature (ESF) signals or radiation monitors (RMs). In some cases, the ESF signal is effective only for selected accidents, placing reliance on the RMs for the remaining accidents. Several aspects of RMs can delay the control room isolation, including the delay for activity to build up to concentrations equivalent to the alarm setpoint and the effects of different radionuclide accident isotopic mixes on monitor response.	Conforms	Credit for engineered safety features that mitigate airborne radioactive material within the Control Room and Control Building were assumed as appropriate. Note that no credit for Control Room isolation was modeled for events that rely solely on radiation monitors.
4.2.5	Credit should generally not be taken for the use of personal protective equipment or prophylactic drugs. Deviations may be considered on a case-by-case basis.	Conforms	Credit was not taken for the use of personnel protective equipment or prophylactic drugs.

Table A (cont.)	Conformance with Regulatory Guide 1.183 Main Sections		
RG Section	**RG Position**	**Analysis**	**Comments**
4.2.6	The dose receptor for these analyses is the hypothetical maximum exposed individual who is present in the control room for 100% of the time during the first 24 hours after the event, 60% of the time between 1 and 4 days, and 40% of the time from 4 days to 30 days.[16] For the duration of the event, the breathing rate of this individual should be assumed to be 3.5×10^{-4} cubic meters per second.	Conforms	The occupancy factors and breathing rate were utilized to determine the doses to the hypothetical maximum exposed individual who is present in the Control Room. Control Room χ/Q values were determined utilizing the ARCON96 computer code which does not incorporate occupancy factors. Occupancy factors were included in the RADTRAD computer code for the dose evaluations.
4.2.7	Control room doses should be calculated using dose conversion factors identified in Regulatory Position 4.1 above for use in offsite dose analyses. The DDE from photons may be corrected for the difference between finite cloud geometry in the control room and the semi-infinite cloud assumption used in calculating the dose conversion factors. The following expression may be used to correct the semi-infinite cloud dose, DDE_{∞}, to a finite cloud dose, DDE_{finite}, where the control room is modeled as a hemisphere that has a volume, V, in cubic feet, equivalent to that of the control room (Ref. 22). $DDE_{finite} = \frac{DDE_{\infty} V^{0.338}}{1173}$	Conforms	The DDE from photons was corrected for the difference between finite cloud geometry in the Control Room and the semi-infinite cloud assumption used in calculating the dose conversion factors by the given equation. This correction was performed by the RADTRAD computer code.
4.3	The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737 (Ref. 2). Design envelope source terms provided in NUREG-0737 should be updated for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of TEDE. Integrated radiation exposure of plant equipment should be determined using the guidance of Appendix I of this guide.	Conforms	Exception – The current TID-14844 accident source term will remain the licensing basis for equipment qualification and NUREG-0737 evaluations other than Control Room and Technical Support Center doses.

Table A (cont.)	Conformance with Regulatory Guide 1.183 Main Sections		
RG Section	**RG Position**	**Analysis**	**Comments**
4.4	The radiological criteria for the EAB, the outer boundary of the LPZ, and for the control room are in 10 CFR 50.67. These criteria are stated for evaluating reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation, e.g., a large-break LOCA. The control room criterion applies to all accidents. For events with a higher probability of occurrence, postulated EAB and LPZ doses should not exceed the criteria tabulated in Table 6. The acceptance criteria for the various NUREG-0737 (Ref. 2) items generally reference General Design Criteria (GDC 19) from Appendix A to 10 CFR Part 50 or specify criteria derived from GDC-19. These criteria are generally specified in terms of whole body dose, or its equivalent to any body organ. For facilities applying for, or having received, approval for the use of AST, the applicable criteria should be updated for consistency with the TEDE criterion in 10 CFR 50.67(b)(2)(iii). **Table 6[17]** **Accident Dose Criteria** **Accident or Case** / **EAB and LPZ Dose Criteria** / **Analysis Release Duration** LOCA / 25 rem TEDE / 30 days for containment, ECCS, and MSIV (BWR) leakage BWR Main Steam Line Break / / Instantaneous puff Fuel Damage or Pre-incident Spike / 25 rem TEDE / / Equilibrium Iodine Activity / 2.5 rem TEDE / / BWR Rod Drop Accident / 6.3 rem TEDE / 24 hours PWR Steam Generator Tube Rupture / / Affected SG: time to isolate. Unaffected SG(s): until cold shutdown is established Fuel Damage or Pre-incident Spike / 25 rem TEDE / / Coincident Iodine Spike / 2.5 rem TEDE / / PWR Main Steam Line Break / / Until cold shutdown is established Fuel Damage or Pre-incident Spike / 25 rem TEDE / / Coincident Iodine Spike / 2.5 rem TEDE / / PWR Locked Rotor Accident / 2.5 rem TEDE / Until cold shutdown is established PWR Rod Ejection Accident / 6.3 rem TEDE / 30 days for containment pathway; until cold shutdown is established for secondary pathway Fuel Handling Accident / 6.3 rem TEDE / 2 hours The column labeled "Analysis Release Duration' is a summary of the assumed radioactivity release durations identified in the individual appendices to this guide. Refer to these appendices for complete descriptions of the release pathways and durations.	Conforms	The DBAs were updated for consistency with the TEDE criterion in Table 6 for offsite doses and in 10 CFR 50.67(b)(2)(iii) for the Control Room and Technical Support Center doses.

Table A (cont.)	Conformance with Regulatory Guide 1.183 Main Sections		
RG Section	**RG Position**	**Analysis**	**Comments**
5.1.1	The evaluations required by 10 CFR 50.67 are re-analyses of the design basis safety analyses and evaluations required by 10 CFR 50.34; they are considered to be a significant input to the evaluations required by 10 CFR 50.92 or 10 CFR 50.59. These analyses should be prepared, reviewed, and maintained in accordance with quality assurance programs that comply with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.	Conforms	The DBA analyses were prepared, reviewed, and maintained per 10 CFR 50 Appendix B and the guidance consistent with RG 1.183.
5.1.2	Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The single active component failure that results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences.	Conforms	Credit was taken for Engineered Safeguard Features with failure assumptions to maximize the calculated doses. Assumptions regarding the occurrence and timing of a loss of offsite power were also selected with the objective of maximizing the postulated radiological consequences.
5.1.3	The numeric values that are chosen as inputs to the analyses required by 10 CFR 50.67 should be selected with the objective of determining a conservative postulated dose. In some instances, a particular parameter may be conservative in one portion of an analysis but be nonconservative in another portion of the same analysis.	Conforms	The numeric values that were chosen as inputs to the analyses required by 10 CFR 50.67 were selected with the objective of determining a conservative postulated dose. For a range of values, the value that resulted in a conservative postulated dose was used.
5.1.4	Licensees should ensure that analysis assumptions and methods are compatible with the ASTs and the TEDE criteria.	Conforms	Licensee has ensured that analysis assumptions and methods are compatible with the AST and the TEDE criteria.

Table A (cont.)	Conformance with Regulatory Guide 1.183 Main Sections		
RG Section	**RG Position**	**Analysis**	**Comments**
5.3	Atmospheric dispersion values (χ/Q) for the EAB, the LPZ, and the control room that were approved by the staff during initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified by this guide. Methodologies that have been used for determining χ/Q values are documented in Regulatory Guides 1.3 and 1.4, Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and the paper, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19" (Refs. 6, 7, 22, and 28). References 22 and 28 should be used if the FSAR χ/Q values are to be revised or if values are to be determined for new release points or receptor distances. Fumigation should be considered where applicable for the EAB and LPZ. For the EAB, the assumed fumigation period should be timed to be included in the worst 2-hour exposure period. The NRC computer code PAVAN (Ref. 29) implements Regulatory Guide 1.145 (Ref. 28) and its use is acceptable to the NRC staff. The methodology of the NRC computer code ARCON96[19] (Ref. 26) is generally acceptable to the NRC staff for use in determining control room χ/Q values. Meteorological data collected in accordance with the site-specific meteorological measurements program described in the facility FSAR should be used in generating accident χ/Q values. Additional guidance is provided in Regulatory Guide 1.23, "Onsite Meteorological Programs" (Ref. 30). All changes in χ/Q analysis methodology should be reviewed by the NRC staff.	Conform	The re-calculation of atmospheric dispersion factors was performed for the EAB and LPZ using the NRC computer code PAVAN according to the guidance of RG 1.145 and for the control room and TSC intakes with new release points using the NRC computer code ARCON96 according to the guidance of RG 1.194. The meteorological data used in the calculation were collected in accordance with WCGS site-specific measurements program and RG 1.23.

Table B	Conformance with Regulatory Guide 1.183 Appendix A (Loss-of-Coolant-Accident)		
RG Section	**RG Position**	**Analysis**	**Comments**
1	Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.	Conforms	Assumptions regarding core inventory and the release of radionuclides from the fuel were obtained from Regulatory Position 3 of RG 1.183.
2	If the sump or suppression pool pH is controlled at values of 7 or greater, the chemical form of radioiodine released to the containment should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine species, including those from iodine re-evolution, for sump or suppression pool pH values less than 7 will be evaluated on a case-by-case basis. Evaluations of pH should consider the effect of acids and bases created during the LOCA event, e.g., radiolysis products. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form.	Conforms	The sump pH is controlled at values of 7 or greater.
3.1	The radioactivity released from the fuel should be assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment in PWRs or the drywell in BWRs as it is released. This distribution should be adjusted if there are internal compartments that have limited ventilation exchange. The suppression pool free air volume may be included provided there is a mechanism to ensure mixing between the drywell to the wetwell. The release into the containment or drywell should be assumed to terminate at the end of the early in-vessel phase.	Conforms	The radioactivity released from the fuel was assumed to mix instantaneously and homogeneously throughout the region of containment not impacted by sprays as it was released. Recirculation fans provide a mechanism for mixing between the sprayed and unsprayed portions of containment.
3.2	Reduction in airborne radioactivity in the containment by natural deposition within the containment may be credited. Acceptable models for removal of iodine and aerosols are described in Chapter 6.5.2, "Containment Spray as a Fission Product Cleanup System," of the Standard Review Plan (SRP), NUREG-0800 (Ref. A-1) and in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments" (Ref. A-2). The latter model is incorporated into the analysis code RADTRAD (Ref. A-3). The prior practice of deterministically assuming that a 50% plateout of iodine is released from the fuel is no longer acceptable to the NRC staff as it is inconsistent with the characteristics of the revised source terms.	Conforms	No natural deposition was assumed for elemental or organic iodine. A sedimentation removal coefficient of 0.1 hr^{-1} for particulates was credited. This is consistent with previously approved submittals, including: • Point Beach Units 1 & 2 – April 2011 (ADAMS Accession Number ML110240054)

Table B (cont.)	Conformance with Regulatory Guide 1.183 Appendix A (Loss-of-Coolant-Accident)		
RG Section	**RG Position**	**Analysis**	**Comments**
3.3	Reduction in airborne radioactivity in the containment by containment spray systems that have been designed and are maintained in accordance with Chapter 6.5.2 of the SRP (Ref. A-1) may be credited. Acceptable models for the removal of iodine and aerosols are described in Chapter 6.5.2 of the SRP and NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays"[1] (Ref. A-4). This simplified model is incorporated into the analysis code RADTRAD (Refs. A-1 to A-3). The evaluation of the containment sprays should address areas within the primary containment that are not covered by the spray drops. The mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building, provided that adequate flow exists between these regions, is assumed to be two turnovers of the unsprayed regions per hour, unless other rates are justified. The containment building atmosphere may be considered a single, well-mixed volume if the spray covers at least 90% of the volume and if adequate mixing of unsprayed compartments can be shown. The SRP sets forth a maximum decontamination factor (DF) for elemental iodine based on the maximum iodine activity in the primary containment atmosphere when the sprays actuate, divided by the activity of iodine remaining at some time after decontamination. The SRP also states that the particulate iodine removal rate should be reduced by a factor of 10 when a DF of 50 is reached. The reduction in the removal rate is not required if the removal rate is based on the calculated time-dependent airborne aerosol mass. There is no specified maximum DF for aerosol removal by sprays. The maximum activity to be used in determining the DF is defined as the iodine activity in the columns labeled "Total" in Tables 1 and 2 of this guide multiplied by 0.05 for elemental iodine and by 0.95 for particulate iodine (i.e., aerosol treated as particulate in SRP methodology).	Conforms	A spray system is available in containment. The spray removal model from Chapter 6.5.2 of NUREG-0800 was used in the analysis. The containment sprays cover 85% of containment. The mixing within containment is promoted by fan coolers which begin operation 2 minutes into the accident and are assumed on for the duration of the event. The elemental iodine removal coefficient was limited to 10 hr^{-1} and removal was terminated when a DF of 200 was reached. A particulate removal coefficient of 5 hr^{-1} was modeled until a DF of 50 was reached, at which time the coefficient was reduced to 0.5 hr^{-1} until sprays were terminated at 5 hours.

Table B (cont.)	Conformance with Regulatory Guide 1.183 Appendix A (Loss-of-Coolant-Accident)		
RG Section	**RG Position**	**Analysis**	**Comments**
3.4	Reduction in airborne radioactivity in the containment by in-containment recirculation filter systems may be credited if these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. A-5 and A-6). The filter media loading caused by the increased aerosol release associated with the revised source term should be addressed.	Not Applicable	In-containment recirculation filters were not credited.
3.5	Reduction in airborne radioactivity in the containment by suppression pool scrubbing in BWRs should generally not be credited. However, the staff may consider such reduction on an individual case basis. The evaluation should consider the relative timing of the blowdown and the fission product release from the fuel, the force driving the release through the pool, and the potential for any bypass of the suppression pool (Ref. 7). Analyses should consider iodine re-evolution if the suppression pool liquid pH is not maintained greater than 7.	Not Applicable	Suppression pool scrubbing was not applicable to the WCGS design.
3.6	Reduction in airborne radioactivity in the containment by retention in ice condensers, or other engineering safety features not addressed above, should be evaluated on an individual case basis. See Section 6.5.4 of the SRP (Ref. A-1).	Not Applicable	No credit was taken for reduction of airborne radioactivity in the containment by retention in ice condensers or other engineering safety features not addressed above.

Table B (cont.)	Conformance with Regulatory Guide 1.183 Appendix A (Loss-of-Coolant-Accident)		
RG Section	**RG Position**	**Analysis**	**Comments**
3.7	The primary containment (i.e., drywell for Mark I and II containment designs) should be assumed to leak at the peak pressure technical specification leak rate for the first 24 hours. For PWRs, the leak rate may be reduced after the first 24 hours to 50% of the technical specification leak rate. For BWRs, leakage may be reduced after the first 24 hours, if supported by plant configuration and analyses, to a value not less than 50% of the technical specification leak rate. Leakage from subatmospheric containments is assumed to terminate when the containment is brought to and maintained at a subatmospheric condition as defined by technical specifications. For BWRs with Mark III containments, the leakage from the drywell into the primary containment should be based on the steaming rate of the heated reactor core, with no credit for core debris relocation. This leakage should be assumed during the two-hour period between the initial blowdown and termination of the fuel radioactivity release (gap and early in-vessel release phases). After two hours, the radioactivity is assumed to be uniformly distributed throughout the drywell and the primary containment.	Conforms	The primary containment peak pressure leak rate was defined as 0.2% by weight of containment air. This leak rate was reduced to 0.1% after the first 24 hours of the accident.
3.8	If the primary containment is routinely purged during power operations, releases via the purge system prior to containment isolation should be analyzed and the resulting doses summed with the postulated doses from other release paths. The purge release evaluation should assume that 100% of the radionuclide inventory in the reactor coolant system liquid is released to the containment at the initiation of the LOCA. This inventory should be based on the technical specification reactor coolant system equilibrium activity. Iodine spikes need not be considered. If the purge system is not isolated before the onset of the gap release phase, the release fractions associated with the gap release and early in-vessel phases should be considered as applicable.	Conforms	Releases via the containment mini-purge system were calculated assuming 100% of the RCS activity (based on the Technical Specifications) was released to containment. The mini-purge release is isolated before the onset of the gap release phase.

Table B (cont.)	Conformance with Regulatory Guide 1.183 Appendix A (Loss-of-Coolant-Accident)		
RG Section	**RG Position**	**Analysis**	**Comments**
4.1	Leakage from the primary containment should be considered to be collected, processed by engineered safety feature (ESF) filters, if any, and released to the environment via the secondary containment exhaust system during periods in which the secondary containment has a negative pressure as defined in technical specifications. Credit for an elevated release should be assumed only if the point of physical release is more than two and one-half times the height of any adjacent structure.	Not Applicable	A dual containment is not applicable to the WCGS design.
4.2	Leakage from the primary containment is assumed to be released directly to the environment as a ground-level release during any period in which the secondary containment does not have a negative pressure as defined in technical specifications.	Not Applicable	A dual containment is not applicable to the WCGS design.
4.3	The effect of high wind speeds on the ability of the secondary containment to maintain a negative pressure should be evaluated on an individual case basis. The wind speed to be assumed is the 1-hour average value that is exceeded only 5% of the total number of hours in the data set. Ambient temperatures used in these assessments should be the 1-hour average value that is exceeded only 5% or 95% of the total numbers of hours in the data set, whichever is conservative for the intended use (e.g., if high temperatures are limiting, use those exceeded only 5%).	Not Applicable	A dual containment is not applicable to the WCGS design.
4.4	Credit for dilution in the secondary containment may be allowed when adequate means to cause mixing can be demonstrated. Otherwise, the leakage from the primary containment should be assumed to be transported directly to exhaust systems without mixing. Credit for mixing, if found to be appropriate, should generally be limited to 50%. This evaluation should consider the magnitude of the containment leakage in relation to contiguous building volume or exhaust rate, the location of exhaust plenums relative to projected release locations, the recirculation ventilation systems, and internal walls and floors that impede stream flow between the release and the exhaust.	Not Applicable	A dual containment is not applicable to the WCGS design.

Table B (cont.)	Conformance with Regulatory Guide 1.183 Appendix A (Loss-of-Coolant-Accident)		
RG Section	**RG Position**	**Analysis**	**Comments**
4.5	Primary containment leakage that bypasses the secondary containment should be evaluated at the bypass leak rate incorporated in the technical specifications. If the bypass leakage is through water, e.g., via a filled piping run that is maintained full, credit for retention of iodine and aerosols may be considered on a case-by-case basis. Similarly, deposition of aerosol radioactivity in gas-filled lines may be considered on a case-by-case basis.	Not Applicable	A dual containment is not applicable to the WCGS design.
4.6	Reduction in the amount of radioactive material released from the secondary containment because of ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).	Not Applicable	A dual containment is not applicable to the WCGS design.
5.1	With the exception of noble gases, all the fission products released from the fuel to the containment (as defined in Tables 1 and 2 of this guide) should be assumed to instantaneously and homogeneously mix in the primary containment sump water (in PWRs) or suppression pool (in BWRs) at the time of release from the core. In lieu of this deterministic approach, suitably conservative mechanistic models for the transport of airborne activity in containment to the sump water may be used. Note that many of the parameters that make spray and deposition models conservative with regard to containment airborne leakage are nonconservative with regard to the buildup of sump activity.	Conforms	With the exception of noble gases, all the fission products released from the fuel to the containment (as defined in Table 1 of RG 1.183) were assumed to instantaneously and homogenously mix in the primary containment sump.
5.2	The leakage should be taken as two times the sum of the simultaneous leakage from all components in the ESF recirculation systems above which the technical specifications, or licensee commitments to item III.D.1.1 of NUREG-0737 (Ref. A-8), would require declaring such systems inoperable. The leakage should be assumed to start at the earliest time the recirculation flow occurs in these systems and end at the latest time the releases from these systems are terminated. Consideration should also be given to design leakage through valves isolating ESF recirculation systems from tanks vented to atmosphere, e.g., emergency core cooling system (ECCS) pump miniflow return to the refueling water storage tank.	Conforms	The ESF leakage to the Plant Auxiliary Building was doubled in the analysis and was assumed to conservatively begin at the start of the event. The dose from ESF leakage to the refueling water storage tank was also included in the analysis.

Table B (cont.)	Conformance with Regulatory Guide 1.183 Appendix A (Loss-of-Coolant-Accident)		
RG Section	**RG Position**	**Analysis**	**Comments**
5.3	With the exception of iodine, all radioactive materials in the recirculating liquid should be assumed to be retained in the liquid phase.	Conforms	Iodines were the only nuclides modeled to be released from the liquid.
5.4	If the temperature of the leakage exceeds 212°F, the fraction of total iodine in the liquid that becomes airborne should be assumed equal to the fraction of the leakage that flashes to vapor. This flash fraction, FF, should be determined using a constant enthalpy, h, process, based on the maximum time-dependent temperature of the sump water circulating outside the containment: $FF = \frac{h_{f_1} - h_{f_2}}{h_{fg}}$ Where: hf1 is the enthalpy of liquid at system design temperature and pressure; hf2 is the enthalpy of liquid at saturation conditions (14.7 psia, 212°F); and hfg is the heat of vaporization at 212°F.	Conforms	The calculated flashing fraction based on the maximum sump temperature was less than 0.10.
5.5	If the temperature of the leakage is less than 212°F or the calculated flash fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid, unless a smaller amount can be justified based on the actual sump pH history and area ventilation rates.	Conforms	An airborne fraction of 10% was assumed for the iodine in the liquid.
5.6	The radioiodine that is postulated to be available for release to the environment is assumed to be 97% elemental and 3% organic. Reduction in release activity by dilution or holdup within buildings, or by ESF ventilation filtration systems, may be credited where applicable. Filter systems used in these applications should be evaluated against the guidance of Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).	Conforms	The radioiodine that was postulated to be available for release to the environment was assumed to be 97% elemental and 3% organic. No reductions due to dilution or holdup were assumed. Credit was taken for the ESF ventilation filtration system in the Plant Auxiliary Building (PAB) exhaust.

Table B (cont.)	Conformance with Regulatory Guide 1.183 Appendix A (Loss-of-Coolant-Accident)		
RG Section	**RG Position**	**Analysis**	**Comments**
7.0	The radiological consequences from post-LOCA primary containment purging as a combustible gas or pressure control measure should be analyzed. If the installed containment purging capabilities are maintained for purposes of severe accident management and are not credited in any design basis analysis, radiological consequences need not be evaluated. If the primary containment purging is required within 30 days of the LOCA, the results of this analysis should be combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. Reduction in the amount of radioactive material released via ESF filter systems may be taken into account provided that these systems meet the guidance in Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).	Not Applicable	The containment is not purged post-LOCA.

Table C	Conformance with Regulatory Guide 1.183 Appendix B (Fuel Handling Accident)		
RG Section	**RG Position**	**Analysis**	**Comments**
1	Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.	Conforms	Assumptions regarding core inventory and the release of radionuclides from the fuel were obtained from Regulatory Position 3 of RG 1.183.
1.1	The number of fuel rods damaged during the accident should be based on a conservative analysis that considers the most limiting case. This analysis should consider parameters such as the weight of the dropped heavy load or the weight of a dropped fuel assembly (plus any attached handling grapples), the height of the drop, and the compression, torsion, and shear stresses on the irradiated fuel rods. Damage to adjacent fuel assemblies, if applicable (e.g., events over the reactor vessel), should be considered.	Conforms	For the postulated fuel handling accident, one entire assembly and 20% of an adjacent assembly were assumed to be damaged as a result of this event.
1.2	The fission product release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. All the gap activity in the damaged rods is assumed to be instantaneously released. Radionuclides that should be considered include xenons, kryptons, halogens, cesiums, and rubidiums.	Conforms	The fission product release from the breached fuel was based on Regulatory Position 3.2 of RG 1.183 and the estimate of the number of fuel rods breached. All the gap activity in the damaged rods was assumed to be instantaneously released. Radionuclides that were considered include xenons, kryptons, halogens, cesiums, and rubidiums.
1.3	The chemical form of radioiodine released from the fuel to the spent fuel pool should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. The CsI released from the fuel is assumed to completely dissociate in the pool water. Because of the low pH of the pool water, the iodine re-evolves as elemental iodine. This is assumed to occur instantaneously. The NRC staff will consider, on a case-by-case basis, justifiable mechanistic treatment of the iodine release from the pool.	Conforms	All particulate iodine released to the spent fuel pool was assumed to dissociate and instantaneously re-evolve as elemental iodine.

Table C (cont.)	Conformance with Regulatory Guide 1.183 Appendix B (Fuel Handling Accident)		
RG Section	**RG Position**	**Analysis**	**Comments**
2	If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 500 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85%) and organic iodine (0.15%) species results in the iodine above the water being composed of 57% elemental and 43% organic species. If the depth of water is not 23 feet, the decontamination factor will have to be determined on a case-by-case method (Ref. B-1).	Conforms	The minimum water depth over the reactor core when handling fuel and over the spent fuel in the fuel handling building is 23 feet, so the overall DF of 200 was applied.
3	The retention of noble gases in the water in the fuel pool or reactor cavity is negligible (i.e., decontamination factor of 1). Particulate radionuclides are assumed to be retained by the water in the fuel pool or reactor cavity (i.e., infinite decontamination factor).	Conforms	The analysis modeled a noble gas DF of 1 and an infinite DF for particulates.
4.1	The radioactive material that escapes from the fuel pool to the fuel building is assumed to be released to the environment over a 2-hour time period.	Conforms	The release of radioactive material was modeled as a linear release over a 2-hour period.
4.2	A reduction in the amount of radioactive material released from the fuel pool by engineered safety feature (ESF) filter systems may be taken into account provided these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. B-2, B-3). Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system1 should be determined and accounted for in the radioactivity release analyses.	Not Applicable	No engineered safety features were credited for the releases from the fuel building.

Table C (cont.)	Conformance with Regulatory Guide 1.183 Appendix B (Fuel Handling Accident)		
RG Section	**RG Position**	**Analysis**	**Comments**
4.3	The radioactivity release from the fuel pool should be assumed to be drawn into the ESF filtration system without mixing or dilution in the fuel building. If mixing can be demonstrated, credit for mixing and dilution may be considered on a case-by-case basis. This evaluation should consider the magnitude of the building volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the pool, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the pool and the exhaust plenums.	Conforms	No credit was taken for mixing or dilution in the fuel building.
5.1	If the containment is isolated[2] during fuel handling operations, no radiological consequences need to be analyzed.	Not Applicable	No credit was taken for containment isolation in the analysis.
5.2	If the containment is open during fuel handling operations, but designed to automatically isolate in the event of a fuel handling accident, the release duration should be based on delays in radiation detection and completion of containment isolation. If it can be shown that containment isolation occurs before radioactivity is released to the environment,[1] no radiological consequences need to be analyzed.	Not Applicable	No credit was taken for containment isolation in the analysis.
5.3	If the containment is open during fuel handling operations (e.g., personnel air lock or equipment hatch is open),[3] the radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period.	Conforms	The containment was assumed to be open and the release of radioactive material was modeled as a linear release over a 2-hour period.
5.4	A reduction in the amount of radioactive material released from the containment by ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. B-2 and B-3). Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system should be determined and accounted for in the radioactivity release analyses.[1]	Not Applicable	No engineered safety features were credited for the releases from containment.

Table C (cont.)	Conformance with Regulatory Guide 1.183 Appendix B (Fuel Handling Accident)		
RG Section	**RG Position**	**Analysis**	**Comments**
5.5	Credit for dilution or mixing of the activity released from the reactor cavity by natural or forced convection inside the containment may be considered on a case-by-case basis. Such credit is generally limited to 50% of the containment free volume. This evaluation should consider the magnitude of the containment volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the reactor cavity, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the reactor cavity and the exhaust plenums.	Not Applicable	No credit was taken for mixing or dilution in containment.

Table D	Conformance with Regulatory Guide 1.183 Appendix E (PWR Main Steam Line Break)		
RG Section	**RG Position**	**Analysis**	**Comments**
1	Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this regulatory guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. The fuel damage estimate should assume that the highest worth control rod is stuck at its fully withdrawn position.	Not Applicable	No fuel damage was postulated to occur during the MSLB.
2	If no or minimal[2] fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by the technical specifications. Two cases of iodine spiking should be assumed.	Conforms	Since no fuel damage occurs, two cases of iodine spiking (pre-accident and accident-initiated) were modeled.
2.1	A reactor transient has occurred prior to the postulated main steam line break (MSLB) and has raised the primary coolant iodine concentration to the maximum value (typically 60 μCi/gm DE I-131) permitted by the technical specifications (i.e., a preaccident iodine spike case).	Conforms	The pre-accident iodine spike was modeled with a primary coolant iodine concentration of 60 μCi/gm DE I-131, consistent with the Technical Specification limit.
2.2	The primary system transient associated with the MSLB causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (typically 1.0 μCi/gm DE I-131) specified in technical specifications (i.e., concurrent iodine spike case). A concurrent iodine spike need not be considered if fuel damage is postulated. The assumed iodine spike duration should be 8 hours. Shorter spike durations may be considered on a case-by-case basis if it can be shown that the activity released by the 8-hour spike exceeds that available for release from the fuel gap of all fuel pins.	Conforms	The accident-initiated iodine spike was modeled with a spike factor of 500 and spike duration of 8 hours. The initial activity was based on 1.0 μCi/gm DE I-131, consistent with the Technical Specification limit.
3	The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant.	Conforms	The activity was modeled to be released instantaneously and homogenously throughout the primary coolant.

Table D (cont.)	Conformance with Regulatory Guide 1.183 Appendix E (PWR Main Steam Line Break)		
RG Section	**RG Position**	**Analysis**	**Comments**
4	The chemical form of radioiodine released from the fuel should be 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 should be assumed to be 97% elemental and 3% organic. These fractions apply to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking.	Conforms	Iodine chemical fractions for releases to the environment (97% elemental and 3% organic) were modeled in the analysis.
5.1	For facilities that have not implemented alternative repair criteria (see Ref. E-1, DG-1074), the primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate limiting condition for operation specified in the technical specifications. For facilities with traditional generator specifications (both per generator and total of all generators), the leakage should be apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized.	Conforms	The accident-induced Technical Specification Bases leakage of 1 gpm was assumed to be entirely to the faulted steam generator and the normal operation Technical Specification primary-to-secondary leakage of 150 gpd/SG was assumed to be evenly apportioned between the intact steam generators.
5.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of the parameter being converted. The ARC leak rate correlations are generally based on the collection of cooled liquid. Surveillance tests and facility instrumentation used to show compliance with leak rate technical specifications are typically based on cooled liquid. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).	Conforms	A density of 1.0 gm/cc (62.4 lbm/ft^3) was used.
5.3	The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212°F). The release of radioactivity from unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms	The primary-to-secondary leakage was terminated at 34 hours when the reactor coolant system was cooled to 212°F. The release of radioactivity from the unaffected steam generators was terminated at 12 hours when the residual heat removal system was in service and removing all decay heat.
5.4	All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.	Conforms	No reduction or mitigation of noble gas activity in releases from the primary system was modeled.

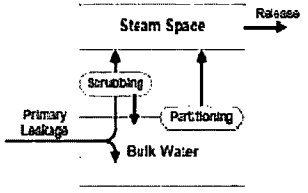
Table D (cont.)	Conformance with Regulatory Guide 1.183 Appendix E (PWR Main Steam Line Break)		
RG Section	**RG Position**	**Analysis**	**Comments**
5.5	The transport model described in this section should be utilized for iodine and particulate releases from the steam generators. This model is shown in Figure E-1 and summarized below:  	Conforms	The transport model was utilized in the analysis.
5.5.1	A portion of the primary-to-secondary leakage will flash to vapor, based on the thermodynamic conditions in the reactor and secondary coolant. • During periods of steam generator dryout, all of the primary-to-secondary leakage is assumed to flash to vapor and be released to the environment with no mitigation. • With regard to the unaffected steam generators used for plant cooldown, the primary-to-secondary leakage can be assumed to mix with the secondary water without flashing during periods of total tube submergence.	Conforms	The faulted steam generator was assumed to blowdown to dryout conditions and any primary-to-secondary leakage was modeled as a release to the environment with no mitigation. Leakage to the intact steam generators was modeled to mix with the secondary water without flashing since the steam generator tubes are submerged.
5.5.2	The leakage that immediately flashes to vapor will rise through the bulk water of the steam generator and enter the steam space. Credit may be taken for scrubbing in the generator, using the models in NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident" (Ref. E-2), during periods of total submergence of the tubes.	Not Applicable	See Position 5.5.1 above.
5.5.3	The leakage that does not immediately flash is assumed to mix with the bulk water.	Not Applicable	See Position 5.5.1 above.

Table D (cont.)	Conformance with Regulatory Guide 1.183 Appendix E (PWR Main Steam Line Break)		
RG Section	**RG Position**	**Analysis**	**Comments**
5.5.4	The radioactivity in the bulk water is assumed to become vapor at a rate that is the function of the steaming rate and the partition coefficient. A partition coefficient for iodine of 100 may be assumed. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators.	Conforms	A partition coefficient of 100 was modeled for iodine and the particulate radionuclides release was limited by the moisture carryover of 0.25%.
5.6	Operating experience and analyses have shown that for some steam generator designs, tube uncovery may occur for a short period following any reactor trip (Ref. E-3). The potential impact of tube uncovery on the transport model parameters (e.g., flash fraction, scrubbing credit) needs to be considered. The impact of emergency operating procedure restoration strategies on steam generator water levels should be evaluated.	Not Applicable	The issue of tube uncovery was addressed by the Westinghouse Owners Group (WOG) in WCAP-13247, "Report on the Methodology for the Resolution of the Steam Generator Tube Uncovery Issue," March 1992. The WOG program concluded that the effect of tube uncovery would be essentially negligible and the issue could be closed without any further investigation or generic restrictions. This position was accepted by the NRC in a letter dated March 10, 1993, from Robert C. Jones, Chief of the Reactor System Branch, to Lawrence A. Walsh, Chairman of the WOG. The letter states "… the Westinghouse analyses demonstrate that the effects of partial steam generator tube uncovery on the iodine release for SGTR and non-SGTR events is negligible. Therefore, we agree with your position on this matter and consider this issue to be resolved." Consistent with this position, the MSLB dose analysis did not model tube uncovery in the intact steam generators.

Table E	Conformance with Regulatory Guide 1.183 Appendix F (PWR Steam Generator Tube Rupture Accident)		
RG Section	**RG Position**	**Analysis**	**Comments**
1	Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are in Regulatory Position 3 of this guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.	Not Applicable	No fuel damage was postulated to occur during the SGTR.
2	If no or minimal[2] fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by technical specification. Two cases of iodine spiking should be assumed.	Conforms	Since no fuel damage occurs, two cases of iodine spiking (pre-accident and accident-initiated) were modeled.
2.1	A reactor transient has occurred prior to the postulated steam generator tube rupture (SGTR) and has raised the primary coolant iodine concentration to the maximum value (typically 60 μCi/gm DE I-131) permitted by the technical specifications (i.e., a preaccident iodine spike case).	Conforms	The pre-accident iodine spike was modeled with a primary coolant iodine concentration of 60 μCi/gm DE I-131, consistent with the Technical Specification limit.
2.2	The primary system transient associated with the SGTR causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 335 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (typically 1.0 μCi/gm DE I-131) specified in technical specifications (i.e., concurrent iodine spike case). A concurrent iodine spike need not be considered if fuel damage is postulated. The assumed iodine spike duration should be 8 hours. Shorter spike durations may be considered on a case-by-case basis if it can be shown that the activity released by the 8-hour spike exceeds that available for release from the fuel gap of all fuel pins.	Conforms	The accident-initiated iodine spike was modeled with a spike factor of 335 and spike duration of 8 hours. The initial activity was based on 1.0 μCi/gm DE I-131, consistent with the Technical Specification limit.
3	The activity released from the fuel, if any, should be assumed to be released instantaneously and homogeneously through the primary coolant.	Conforms	The activity was modeled to be released instantaneously and homogenously throughout the primary coolant.

Table E (cont.)	Conformance with Regulatory Guide 1.183 Appendix F (PWR Steam Generator Tube Rupture Accident)		
RG Section	**RG Position**	**Analysis**	**Comments**
4	Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic.	Conforms	Iodine chemical fractions for releases to the environment (97% elemental and 3% organic) were modeled in the analysis.
5.1	The primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate limiting condition for operation specified in the technical specifications. The leakage should be apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized.	Conforms	The accident-induced Technical Specification Bases leakage of 1 gpm was assumed to be evenly apportioned between the intact steam generators. This leakage modeling was used since the leakage is negligible compared to the flow through the ruptured tube.
5.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests are typically based on cool liquid. Facility instrumentation used to determine leakage is typically located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).	Conforms	A density of 1.0 gm/cc (62.4 lbm/ft^3) was used.
5.3	The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212° F). The release of radioactivity from the unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms	The primary-to-secondary break flow was terminated when the reactor coolant system pressure equalized with the ruptured steam generator secondary side pressure. The release of radioactivity from the ruptured and intact steam generators was terminated at 12 hours when the residual heat removal system was in service and removing all decay heat.
5.4	The release of fission products from the secondary system should be evaluated with the assumption of a coincident loss of offsite power.	Conforms	A loss of offsite power was assumed coincident with reactor trip.
5.5	All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.	Conforms	No reduction or mitigation of noble gas activity in releases from the primary system was modeled.

Table E (cont.)	Conformance with Regulatory Guide 1.183 Appendix F (PWR Steam Generator Tube Rupture Accident)		
RG Section	**RG Position**	**Analysis**	**Comments**
5.6	The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.	Conforms	The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E for iodine and particulates was considered as appropriate in the SGTR. In addition, flashing of break flow in the ruptured steam generator with a time dependent flashing fraction was considered and all activity in the flashed break flow was released to the environment with no mitigation, dilution, or credit for scrubbing.

Table F	Conformance with Regulatory Guide 1.183 Appendix G (PWR Locked Rotor Accident)		
RG Section	**RG Position**	**Analysis**	**Comments**
1	Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are in Regulatory Position 3 of this regulatory guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.	Conforms	Assumptions regarding core inventory and the release of radionuclides from the fuel were obtained from Regulatory Position 3 of RG 1.183. The analysis modeled the fraction of the core inventory assumed to be in the gap by radionuclide group consistent with Table 3 in Regulatory Position 3.2. The assumed number of fuel rods breached was 5% of the core and the radial peaking factor of 1.65 was applied.
2	If no fuel damage is postulated for the limiting event, a radiological analysis is not required as the consequences of this event are bounded by the consequences projected for the main steam line break outside containment.	Not Applicable	Fuel damage was postulated; therefore, an analysis was performed.
3	The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant.	Conforms	The activity was modeled to be released instantaneously and homogenously throughout the primary coolant.
4	The chemical form of radioiodine released from the fuel should be 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 should be assumed to be 97% elemental and 3% organic. These fractions apply to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking.	Conforms	Iodine chemical fractions for releases to the environment (97% elemental and 3% organic) were modeled in the analysis.
5.1	The primary-to-secondary leak rate in the steam generators should be assumed to be the leak-rate-limiting condition for operation specified in the technical specifications. The leakage should be apportioned between the steam generators in such a manner that the calculated dose is maximized.	Conforms	The accident-induced Technical Specification Bases leakage of 1 gpm was assumed to be evenly apportioned to the steam generators.

Table F (cont.)	Conformance with Regulatory Guide 1.183 Appendix G (PWR Locked Rotor Accident)		
RG Section	**RG Position**	**Analysis**	**Comments**
5.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests are typically based on cool liquid. Facility instrumentation used to determine leakage is typically located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).	Conforms	A density of 1.0 gm/cc (62.4 lbm/ft^3) was used.
5.3	The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212° F). The release of radioactivity should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms	The primary-to-secondary leakage was terminated when the release of radioactivity from the steam generators was terminated. This occurred at 12 hours when the residual heat removal system was in service and removing all decay heat.
5.4	The release of fission products from the secondary system should be evaluated with the assumption of a coincident loss of offsite power.	Conforms	A loss of offsite power was assumed.
5.5	All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.	Conforms	No reduction or mitigation of noble gas activity in releases from the primary system was modeled.
5.6	The transport model described in assumptions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.	Conforms	The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E for iodine and particulates was considered as appropriate in the locked rotor event.

Table G	Conformance with Regulatory Guide 1.183 Appendix H (PWR Rod Ejection Accident)		
RG Section	**RG Position**	**Analysis**	**Comments**
1	Assumptions acceptable to the NRC staff regarding core inventory are in Regulatory Position 3 of this guide. For the rod ejection accident, the release from the breached fuel is based on the estimate of the number of fuel rods breached and the assumption that 10% of the core inventory of the noble gases and iodines is in the fuel gap. The release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and the assumption that 100% of the noble gases and 25% of the iodines contained in that fraction are available for release from containment. For the secondary system release pathway, 100% of the noble gases and 50% of the iodines in that fraction are released to the reactor coolant.	Conforms	Assumptions regarding core inventory and the release of radionuclides from the fuel were obtained from Regulatory Position 3 of RG 1.183. The analysis modeled gap release fractions of 10% for iodines and noble gases and 12% for alkali metals. The assumed number of fuel rods breached was 10% of the core and the radial peaking factor of 1.65 was applied. The analysis also modeled fuel melting release fractions of 100% of noble gases and 50% of iodines and alkali metals in the melted fuel rods. Note that the assumption of 50% iodine and alkali metal release from melted fuel was conservatively modeled for each pathway.
2	If no fuel damage is postulated for the limiting event, a radiological analysis is not required as the consequences of this event are bounded by the consequences projected for the loss-of-coolant accident (LOCA), main steam line break, and steam generator tube rupture.	Not Applicable	Fuel damage was postulated; therefore, an analysis was performed.
3	Two release cases are to be considered. In the first, 100% of the activity released from the fuel should be assumed to be released instantaneously and homogeneously through the containment atmosphere. In the second, 100% of the activity released from the fuel should be assumed to be completely dissolved in the primary coolant and available for release to the secondary system.	Conforms	Both cases were considered separately in the analysis.
4	The chemical form of radioiodine released to the containment atmosphere should be assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. If containment sprays do not actuate or are terminated prior to accumulating sump water, or if the containment sump pH is not controlled at values of 7 or greater, the iodine species should be evaluated on an individual case basis. Evaluations of pH should consider the effect of acids created during the rod ejection accident event, e.g., pyrolysis and radiolysis products. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form.	Conforms	The iodine chemical fractions for releases to containment were modeled as 95% CsI, 4.85% elemental, and 0.15% organic. All fission products, with the exception of elemental and organic iodine and noble gases, were assumed to be in particulate form. No removal processes were modeled in containment besides leakage and decay, so sump pH has no impact.

Table G (cont.)	Conformance with Regulatory Guide 1.183 Appendix H (PWR Rod Ejection Accident)		
RG Section	**RG Position**	**Analysis**	**Comments**
5	Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic.	Conforms	Iodine chemical fractions for releases to the environment (97% elemental and 3% organic) were modeled in the analysis.
6	Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material in and from the containment are as follows.	Conforms	See below.
6.1	A reduction in the amount of radioactive material available for leakage from the containment that is due to natural deposition, containment sprays, recirculating filter systems, dual containments, or other engineered safety features may be taken into account. Refer to Appendix A to this guide for guidance on acceptable methods and assumptions for evaluating these mechanisms.	Not Applicable	No removal processes were modeled in containment besides leakage and decay.
6.2	The containment should be assumed to leak at the leak rate incorporated in the technical specifications at peak accident pressure for the first 24 hours, and at 50% of this leak rate for the remaining duration of the accident. Peak accident pressure is the maximum pressure defined in the technical specifications for containment leak testing. Leakage from subatmospheric containments is assumed to be terminated when the containment is brought to a subatmospheric condition as defined in technical specifications.	Conforms	The primary containment peak pressure leak rate was defined as 0.2% by weight of containment air. This leak rate was reduced to 0.1% after the first 24 hours of the accident.
7.1	A leak rate equivalent to the primary-to-secondary leak rate limiting condition for operation specified in the technical specifications should be assumed to exist until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms	The accident-induced Technical Specification Bases leakage of 1 gpm was assumed to be evenly apportioned to the steam generators. The primary-to-secondary leakage was terminated when the release of radioactivity from the steam generators was terminated. This occurred at 12 hours when the residual heat removal system was in service and removing all decay heat.

Table G (cont.)	Conformance with Regulatory Guide 1.183 Appendix H (PWR Rod Ejection Accident)		
RG Section	**RG Position**	**Analysis**	**Comments**
7.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests typically are based on cooled liquid. The facility's instrumentation used to determine leakage typically is located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).	Conforms	A density of 1.0 gm/cc (62.4 lbm/ft^3) was used.
7.3	All noble gas radionuclides released to the secondary system are assumed to be released to the environment without reduction or mitigation.	Conforms	No reduction or mitigation of noble gas activity in releases from the primary system was modeled.
7.4	The transport model described in assumptions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.	Conforms	The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E for iodine and particulates was considered as appropriate in the rod ejection event.

6 REGULATORY GUIDE 1.145 CONFORMANCE TABLE

Regulatory Guide 1.145, Revision 1, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants" – Conformance Table

NOTE: The text shown in the "RG Position" columns is taken from Regulatory Guide 1.145. Therefore, references to footnotes, tables, and numbered references may be found in the regulatory guide.

REGULATORY GUIDE 1.145 COMPARISON

Conformance with Regulatory Guide 1.145			
RG Section	**RG Position**	**Analysis**	**Comments**
1	Equations and parameters presented in this section should be used unless unusual sitting, meteorological, or terrain conditions dictate the use of other models or considerations. Site-specific atmospheric diffusion tests covering a full range of conditions may be used as a basis for modifying the equations and parameters.	Conform	No equations and parameters were modified. The analysis conforms by the use of PAVAN.
1.1	The meteorological data needed for χ/Q calculations include windspeed, wind direction, and a measure of atmospheric stability. These data should represent hourly averages as defined in Regulatory Guide 1.23.	Conform	The meteorological data were collected in accordance with Regulatory Guide 1.23.
1.1	Wind direction should be classed into 16 compass directions (22.5-degree sectors centered on true north, north-northeast, etc.).	Conform	The analysis conforms by the use of PAVAN.
1.1	Atmospheric stability should be determined by vertical ΔT between the release height and the 10-meter level. Acceptable stability classes are given in Regulatory Guide 1.23. If other well-documented parameters are used to determine plume dispersion (with appropriate justification), the models described in this guide may require modification. A well-documented parameter is one that is substantiated by diffusion data collected in terrain conditions similar to those at the nuclear power plant site being considered.	Conform	Atmospheric stability was determined by vertical ΔT between the 60-meter level and the 10-meter level. The range of possible release height 17.37 m to 66.25 m is approximately in the same range as the measurement level.
1.1	Calms should be defined as hourly average windspeeds below the vane or anemometer starting speed, whichever is higher (to reflect limitations in instrumentation). If the instrumentation program conforms to the regulatory position in Regulatory Guide 1.23, calms should be assigned a windspeed equal to the vane or anemometer starting speed, whichever is higher. Otherwise, consideration of a conservative evaluation of calms, taking into account the limitations of the windspeed measurement system, will be necessary. Wind directions during calm conditions should be assigned in proportion to the directional distribution of noncalm winds with speeds less than 1.5 meters per second.[3]	Conform	Calms were defined as stated by the Regulatory Guide. The instrumentation program conforms to Regulatory Guide 1.23.

Conformance with Regulatory Guide 1.145 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
1.2	For each wind direction sector, χ/Q values for each significant release point should be calculated at an appropriate exclusion area boundary distance and outer low population zone (LPZ) boundary distance. The following procedure should be used to determine these distances. The procedure takes into consideration the possibility of curved airflow trajectories, plume segmentation (particularly in light wind, stable conditions), and the potential for windspeed and direction frequency shifts from year to year.	Conform	The EAB distance and the LPZ distance have been defined in the USAR as a uniform distance in all directions. All release points are assumed to be at the same point. No differentiation was made among different release points in terms of EAB and LPZ distances.
1.2	For each of the 16 sectors, the distance for exclusion area boundary or outer LPZ boundary χ/Q calculation should be the minimum distance from the stack or, in the case of releases through vents or building penetrations, the nearest point on the building to the exclusion area boundary or outer LPZ boundary within a 45-degree sector centered on the compass direction of interest.	Conform	The EAB distance and the LPZ distance have been defined in the USAR as a uniform radius of a circle in all directions. No differentiation was made among different release points in terms of EAB and LPZ distances. The radius is the minimum straight line distance.
1.2	For stack releases, the maximum ground-level concentration in a sector may occur beyond the exclusion area boundary distance or outer LPZ boundary distance. Therefore, for stack releases, χ/Q calculations should be made in each sector at each boundary distance and at various distances beyond the exclusion area boundary distance to determine the maximum relative concentration for consideration in subsequent calculations.	Not applicable	Stack releases are not considered in the calculation.
1.3	Relative concentrations that can be assumed to apply at the exclusion area boundary for 2 hours immediately following an accident should be determined.[4] Calculations based on meteorological data averaged over a 1-hour period should be assumed to apply for the entire 2-hour period. This assumption is reasonably conservative considering the small variation of χ/Q values with averaging time (Ref. 8). If releases associated with a postulated event are estimated to occur in a period of less than 20 minutes, the applicability of the models should be evaluated on a case-by-case basis.	Conform	The analysis conforms by the use of PAVAN.

Conformance with Regulatory Guide 1.145 (cont.)

RG Section	RG Position	Analysis	Comments
1.3.1	This class of release modes includes all release points or areas that are effectively lower than two and one-half times the height of adjacent solid structures (Ref. 9). Within this class, two sets of meteorological conditions are treated differently , as follows: a. During neutral (D) or stable (E, F, or G) atmospheric stability conditions when the windspeed at the 10-meter level is less than 6 meters per second, horizontal plume meander can be considered. χ/Q values may be determined through selective use of the following set of equations for ground-level relative concentrations at the plume centerline: $\chi/Q = \frac{1}{\bar{U}_{10}(\pi\sigma_y\sigma_z + A/2)}$ (1) $\chi/Q = \frac{1}{\bar{U}_{10}(2\pi\sigma_y\sigma_z)}$ (2) $\chi/Q = \frac{1}{\bar{U}_{10}\pi\Sigma_y\sigma_z}$ (3) χ/Q values should be calculated using Equations 1, 2, and 3. The values from Equations 1 and 2 should be compared and the higher value selected. This value should be compared with the value from Equation 3, and the lower value of these two should be selected as the appropriate χ/Q value. Examples and a detailed explanation of the rationale for determining the controlling conditions are given in Appendix A to this guide.	Conform	The analysis conforms by the use of PAVAN.
1.3.1	b. During all other meteorological conditions, plume meander should not be considered. The appropriate χ/Q value for these conditions is the higher value calculated from Equation 1 or 2.	Conform	The analysis conforms by the use of PAVAN.

Conformance with Regulatory Guide 1.145 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
1.3.2	This class of release modes includes all release points at levels that are two and one-half times the height of adjacent solid structures or higher (Ref. 9). Nonfumigation conditions are treated separately.	Not applicable.	Stack releases are not considered in the calculation.
1.3.2	a. For nonfumigation conditions, the equation for ground-level relative concentration at the plume centerline for stack releases is: $$\chi/Q = \frac{1}{\pi \bar{U}_h \sigma_y \sigma_z} \exp\left[\frac{-h_e^2}{2\sigma_z^2}\right] \quad (4)$$ For those cases in which the applicant can demonstrate that the vertical velocity of effluent plumes from the plant (because of either buoyancy or mechanical jet effects] will be maintained during the course of the accident, the additional velocity may be considered in the determination of the effective stack height (he) using the same procedures described in regulatory position 2.a of Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors."	Not applicable.	Stack releases are not considered in the calculation.

Conformance with Regulatory Guide 1.145 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
1.3.2	b. For fumigation conditions; a "fumigation χ/Q" should be calculated for each sector as follows. The equation for ground-level relative concentration at the plume centerline for stack releases during fumigation conditions is: $$\chi/Q = \frac{1}{(2\pi)^{1/2} U_{h_e} \sigma_y h_e}, h_e > 0 \quad (5)$$ Equation 5 cannot be applied indiscriminately because the χ/Q values calculated, using this equation, become unrealistically large as h_e becomes small (on the order of 10 meters). The χ/Q values calculated using Equation 5 must therefore be limited by certain physical restrictions. The highest ground-level χ/Q values from elevated releases are expected to occur during stable conditions with low windspeeds when the effluent plume impacts on a terrain obstruction (i.e., $h_e = 0$). However, elevated plumes diffuse upward through the stable layer aloft as well as downward through the fumigation layer. Thus ground-level relative concentrations for elevated releases under fumigation conditions cannot be higher than those produced by nonfumigation, stable atmospheric conditions with $h_e = 0$. For the fumigation case that assumes F stability and a windspeed of 2 meters per second, Equation 4 should be used instead of Equation 5 at distances greater than the distance at which the χ/Q values determined using Equation 4 with $h_e = 0$, and Equation 5 are equal.	Not applicable.	Stack releases are not considered in the calculation.
1.4	Two-hour χ/Q values should also be calculated at outer LPZ boundary distances. The procedures described above for exclusion area boundary distances (see regulatory position 1.3) should be used.	Conform	The analysis conforms by the use of PAVAN.

Conformance with Regulatory Guide 1.145 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
1.4	An annual average (8760-hour) χ/Q should be calculated for each sector at the outer LPZ boundary distance for that sector, using the method described in regulatory position 1.c of Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors." (For stack releases, he should be determined as described in regulatory position 1.3.2 above.)	Conform	The analysis conforms by the use of PAVAN.
1.4	These calculated 2-hour and annual average values are used in regulatory position 2.2 to determine sector χ/Q values at outer LPZ boundary distances for various intermediate time periods periods.[6]	Conform	The analysis conforms by the use of PAVAN.
2.1.1	Using the χ/Q values calculated for each hour of data according to regulatory position 1.3, a cumulative probability distribution of χ/Q values should be constructed for each of the 16 sectors. Each distribution should be described in terms of probabilities of given χ/Q values being exceeded in that sector during the total time. A plot of χ/Q versus probability of being exceeded should be made for each sector, and a curve should be drawn to form an upper bound of the data points. For each of the 16 curves, the χ/Q value that is exceeded 0.5 percent[7] of the total number of hours in the data set should be selected (Ref. 10). These are the sector χ/Q values. The highest of the 16 sector values is defined as the maximum sector χ/Q value.	Conform	The analysis conforms by the use of PAVAN.
2.1.2	Regulatory position 1.3.2 describes procedures for calculating a fumigation χ/Q for each sector. These sector fumigation values, and the general (nonfumigation) sector values obtained in regulatory position 2.1.1, are used to determine appropriate sector χ/Qs. Conservative assumptions for fumigation conditions, which differ for inland and coastal sites, are described below. Modifications may be appropriate for specific sites.	Not applicable.	Stack releases are not considered in the calculation.

Conformance with Regulatory Guide 1.145 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
2.1.2	a. Inland Sites: For stack releases at sites located 3.2 kilometers or more from large bodies of water (e.g., oceans or Great Lakes), a fumigation condition should be assumed to exist at the time of the accident and continue for 1/2 hour (Ref. 11). For each sector, if the sector fumigation χ/Q exceeds the sector nonfumigation χ/Q, use the fumigation value for the 0 to 1/2-hour time period and the nonfumigation value for the 1/2-hour to 2-hour time period. Otherwise, use the nonfumigation sector value for the entire 0 to 2-hour time period. The 16 (sets of) values thus determined will be used in dose assessments requiring time-integrated concentration considerations.	Not applicable.	Stack releases are not considered in the calculation.
2.1.2	b. Coastal Sites: For stack releases at sites located less than 3.2 kilometers from large bodies of water, a fumigation condition should be assumed to exist at the exclusion area boundary at the time of the accident and continue for the entire 2-hour period. For each sector, the larger of the sector fumigation χ/Q and the sector nonfumigation χ/Q should be used for the 2-hour period. Of the 16 sector values, the highest is the maximum sector χ/Q value.	Not applicable.	Stack releases are not considered in the calculation.
2.1.2	c. Modifications: These conservative assumptions do not consider frequency and duration of fumigation conditions as a function of airflow direction. If information can be presented to substantiate the likely directional occurrence and duration of fumigation conditions at a site, the assumptions of fumigation in all appropriate directions and of duration of 1/2 hour and 2 hours for the exclusion area boundary may be modified. Then fumigation need only be considered for airflow directions in which fumigation has been determined to occur and of a duration determined from the study of site conditions.[8]	Not applicable.	Stack releases are not considered in the calculation.

Conformance with Regulatory Guide 1.145 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
2.2.1	Sector χ/Q values for the outer LPZ boundary should be determined for various time periods throughout the course of the postulated accident.[9] The time periods should represent appropriate meteorological regimes, e.g. 8 and 16 hours and 3 and 26 days as presented in Section 2.3.4 of Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants--LWR Edition," or other time periods appropriate to release durations.	Conform	The analysis conforms by the use of PAVAN.
2.2.1	For a given sector, the average χ/Q values for the various time periods may be approximated by a logarithmic interpolation between the 2-hour[10] sector χ/Q and the annual average (8760-hour) χ/Q for the same sector. The 2-hour sector χ/Q for the outer LPZ boundary is determined using the general method given for the exclusion area boundary in regulatory position 2.1. The annual average χ/Q for a given sector is determined as described in regulatory position 1.4.	Conform	The analysis conforms by the use of PAVAN.
2.2.1	The logarithmic interpolation procedure produces results that are consistent with studies of variations of average concentrations with time periods up to 100 hours (Ref. 8). Alternative methods should also be consistent with these studies and should produce results that provide a monotonic decrease in average χ/Q with time.	Conform	The analysis conforms by the use of PAVAN.
2.2.1	For each time period, the highest of the 16 sector χ/Q values should be identified. In most cases, these highest values will occur in the same sector for all time periods. These are then the maximum sector χ/Q values. However, if the highest sector χ/Qs do not all occur in the same sector, the 16 (sets of) values will be used in dose assessments requiring time-integrated concentration considerations. The set of χ/Q values resulting in the highest time-integrated dose within a sector should be considered the maximum sector χ/Q values.	Conform	The analysis conforms by the use of PAVAN.
2.2.2	Determination of sector χ/Q values for fumigation conditions at the outer LPZ boundary involves the following assumptions concerning the duration of fumigation for inland and coastal sites:	Not applicable.	Stack releases are not considered in the calculation.

Conformance with Regulatory Guide 1.145 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
2.2.2	a. Inland sites: For stack releases at sites located 3.2 kilometers or more from large bodies of water, a fumigation condition should be assumed to exist at the outer LPZ boundary at the time of the accident and continue for 1/2 hour. Sector χ/Q values for fumigation should be determined as for the exclusion area boundary in regulatory position 2.1.2.	Not applicable.	Stack releases are not considered in the calculation.
2.2.2	b. Coastal Sites: For stack releases at sites located less than 3200 meters from large bodies of water, a fumigation condition should be assumed to exist at the outer LPZ boundary following the arrival of the plume and continue for a 4-hour period. Sector χ/Q values for fumigation should be determined as for the exclusion area boundary in regulatory position 2.1.2.	Not applicable.	Stack releases are not considered in the calculation.
2.2.2	c.The modifications discussed in regulatory position 2.1.2 may also be considered for the outer LPZ boundary.	Not applicable.	Stack releases are not considered in the calculation.
3	The χ/Q values that are exceeded no more than 5 percent of the total number of hours in the data set around the exclusion area boundary and around the outer LPZ boundary should be determined as follows (Ref. 10):	Conform	The analysis conforms by the use of PAVAN.
3	Using the χ/Q values calculated according to regulatory position 1, an overall cumulative probability distribution for all directions combined should be constructed. A plot of χ/Q versus probability of being exceeded should be made, and an upper bound curve should be drawn. The 2-hour χ/Q value that is exceeded 5 percent of the time should be selected from this curve as representing the dispersion condition indicative of the type of release being considered. In addition, for the outer LPZ boundary the maximum of the 16 annual average χ/Q values should be used along with the 5 percent 2-hour χ/Q value to determine χ/Q values for the appropriate time periods by logarithmic interpolation.	Conform	The analysis conforms by the use of PAVAN.

Conformance with Regulatory Guide 1.145 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
4	The χ/Q value for exclusion area boundary or outer LPZ boundary evaluations should be the maximum sector χ/Q (regulatory position 2) or the 5 percent overall site χ/Q (regulatory position 3), whichever is higher. All direction-dependent sector values should be presented for consideration of the appropriateness of the exclusion area and outer LPZ boundaries. Where the basic meteorological data necessary for the analyses described herein substantially deviate from the regulatory position stated in Regulatory Guide 1.23, consideration should be given to the resulting uncertainties in dispersion estimates.	Conform	The χ/Q values for dose evaluations are taken as the larger of the maximum sector χ/Q or the 5 percent overall site χ/Q.

7 REGULATORY GUIDE 1.194 CONFORMANCE TABLE

Regulatory Guide 1.194, Revision 0, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants" – Conformance Table

NOTE: The text shown in the "RG Position" columns is taken from Regulatory Guide 1.194. Therefore, references to footnotes, tables, and numbered references may be found in the regulatory guide.

REGULATORY GUIDE 1.194 COMPARISON

Conformance with Regulatory Guide 1.194			
RG Section	**RG Position**	**Analysis**	**Comments**
1	*95th-percentile χ/Q value*: The χ/Q value that is not exceeded by more than 5.0 percent of the χ/Q values generated with the meteorological observations in the data set. Unless otherwise stated, all χ/Q values referred to in this guide are 95th-percentile values.	Conform	The analysis is performed using the ARCON96 code which provides results that conform to this definition.
1	*Control Room*: The plant area, defined in the facility licensing basis, in which actions can be taken to operate the plant safely under normal conditions and to maintain the reactor in a safe condition during accident situations. It encompasses the instrumentation and controls necessary for a safe shutdown of the plant and typically includes the critical document reference file, the computer room (if used as an integral part of the emergency response plan), shift supervisor's office, the operator wash room and kitchen, and other critical areas to which frequent personnel access or continuous occupancy may be necessary in the event of an accident.	Conform	The analysis is performed to determine the dose received by personnel within the control room.
1	*Control Room Envelope (CRE)*: The plant area, defined in the facility licensing basis, that in the event of an emergency can be isolated from the plant areas and the environment external to the CRE. This area is served by an emergency ventilation system, with the intent of maintaining the habitability of the control room. This area encompasses the control room, and may encompass other non-critical areas to which frequent personnel access or continuous occupancy is not necessary in the event of an accident.	Conform	The control room envelope includes the control room and all areas in or adjacent to the control room containing plant information and equipment that may be needed during an emergency, including pantry, sanitary facilities, and Class 1E air-conditioning equipment rooms.

Conformance with Regulatory Guide 1.194 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
1	*Control Room Intake*: The location at which the released radioactive material enters the CRE. Includes intentional ventilation system outside air intakes and other locations of significant infiltration into the CRE.	Conform	Control Room pressurization fans move outside air into the control building. Once inside the control building, the control room filtration fans are used to move the air into the control room. The HVAC intake for the control room pressurization fans is used for the χ/Q receptor location. The analysis considers the Control Room air intakes (vents 302 and 303) which supply the Control Room ventilation system.
1	*Freestanding Stack*: A stack located outside the zone of influence of structures in the vicinity of the stack. (See Regulatory Position C.3.2.2)	Not Applicable	All releases in the analysis are ground level releases. No stack releases are analyzed.
1	*Infiltration (or Inleakage)*: The transport of released radioactive materials into the CRE via interstices in the structures, systems, and components that comprise the CRE. Such a transport is driven by pressure differentials between the CRE and areas external to the CRE.	Conform	This is a definition of inleakage. The analysis of inleakage conforms to this definition.
1	*Slant Path*: The shortest line-of-sight distance from the release point to the control room intake, based on the differences in elevation and the horizontal intervening distance; calculated and used as the source-to-receptor distance by ARCON96.	Conform	The analysis is performed using source to receptor distances that are calculated along paths that conform to this definition.

Conformance with Regulatory Guide 1.194 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
2	The May 9, 1997, version of the ARCON96 code as described in Revision 1 of NUREG/CR-6331 (Ref. 1) is an acceptable methodology for assessing control room χ/Q values for use in design basis accident (DBA) radiological analyses, subject to the positions in this guide, unless unusual sitting, building arrangement, release characterization, source-receptor configuration, meteorological regimes, or terrain conditions indicate otherwise. These latter situations need to be addressed on a case-by-case basis.	Conform	The analysis follows the recommendations of RG 1.194. In addition, the configuration of the WCGS site (terrain, building arrangement, source-receptor configuration, and meteorological regimes) are conventional and do not require additional special consideration.
2	Although the ARCON96 code, when used as described in this guide, can provide an improved basis for determining site-specific χ/Q values, holders of operating licenses may continue to use χ/Q values determined with methodologies previously approved by the NRC staff and documented in the facility's final safety analysis report (FSAR) to the extent that these values are appropriate for the application in which they are being used.[3] Licensees may also continue to use the licensing basis methodology for determining χ/Q values for newly identified source-receptor combinations or re-generating the approved χ/Q values using more recently collected meteorological data sets. The ARCON96 code and the other models addressed in this guide may be used voluntarily, subject to the guidance herein, as a replacement for the existing licensing basis methodology for determining χ/Q values for design basis control room radiological habitability. Since the existing licensing basis methodology remains valid, a licensee may use the ARCON96 code and the other models addressed in this guide on a selective basis, that is, it is not necessary that existing χ/Q values be updated. The NRC staff does expect that the methodologies will be applied consistently in any particular accident assessment.	Conform	The analysis uses the ARCON96 code to replace the existing licensing basis methodology in order to determine χ/Q values for recently collected meteorological data.

Conformance with Regulatory Guide 1.194 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
2	For each of the source-to-receptor combinations, the 95th percentile χ/Q should be determined. Values for parameters used as input to the χ/Q assessment should be selected consistent with achieving this confidence level. Selection of conservative, bounding source-to-receptor combinations and less detailed site parameters for the χ/Q evaluation may be sufficient to establish compliance with regulatory guidelines.	Conform	The analysis calculates 95th percentile χ/Q values for each source-to-receptor combination.
2	Control room χ/Q values are generally determined for each of the following averaging periods: 0-8 hours (or 0-2 hours and 2-8 hours), 8-24 hours, 24-96 hours, 96-720 hours. The period of the most adverse release of radioactive materials to the environment should be assumed to occur coincident with the period of most unfavorable atmospheric dispersion. If the 0-2 hour χ/Q is calculated, this value should be used coincident with the limiting portion of the release to the environment. The 2-8 hour χ/Q value is used for the remaining 6 hours of the first 8-hour time period. Part of this 6-hour interval may occur before or after the limiting 2-hour period. The 8-24, 24-96, and 96-720 hour χ/Q values should similarly be used for the remainder of the release duration. For facilities using the traditional TID-14844 (Ref. 11) source term, the 2-hour period will generally coincide with the start of the event. For facilities with design basis analyses that include an alternative source term, the 2-hour period is often the onset of the in-vessel release phase. In any case, the start of this period should be determined as a part of the analyses for each facility.	For information	The analysis determines χ/Q values for the following averaging periods: 0-2 hours, 2-8 hours, 8-24 hours, 24-96 hours, and 96-720 hours.
3	This section addresses the use of the ARCON96 code for calculating χ/Q values for design basis control room radiological habitability assessments. The ARCON96 code should be obtained and maintained under an appropriate software quality assurance program that complies with the applicable criteria of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50 and applicable industry consensus standards to which the licensee has committed.	Conform	The ARCON96 code used to perform the analysis has been obtained and maintained under the Fauske and Associates quality assurance program. This quality assurance program complies with the criteria stated in Appendix B to 10 CFR Part 50.

Conformance with Regulatory Guide 1.194 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
3.1	The meteorological data needed for χ/Q calculations include wind speed, wind direction, and a measure of atmospheric stability. These data should be obtained from an onsite meteorological measurement program based on the guidance of Safety Guide 23, "Onsite Meteorological Programs" (Ref. 12), that includes quality assurance provisions consistent with Appendix B to 10 CFR Part 50. The meteorological data set used in these assessments should represent hourly averages as defined in Safety Guide 23. Data should be representative of the overall site conditions and be free from local effects such as building and cooling tower wakes, brush and vegetation, or terrain. Collected data should be reviewed to identify instrumentation problems and missing or anomalous observations (see Ref. 13). The size of the data set used in the χ/Q assessments should be sufficiently large such that it is representative of long-term meteorological trends at the site. The NRC staff considers 5 years of hourly observations to be representative of long-term trends at most sites. With sufficient justification of its representativeness, however, the minimum meteorological data set is one complete year (including all four seasons) of hourly observations.	Conform	The meteorological data used in the analysis represent hourly averages as defined in Safety Guide 23. The data has been reviewed and any anomalous observations have been removed. Data for a five year period from 2006 through 2010 are used in the analysis.
3.1	Wind direction should be expressed as the direction from which the wind is blowing (i.e., the upwind direction from the center of the site) referenced from true north.	Conform	The meteorological data conform to this wind direction convention.
3.1	Atmospheric stability should be determined by the vertical temperature difference (ΔT) measured over the difference in height appropriate for the projected release height (including plume rise as applicable). A table of ΔT values in units of degrees Centigrade per 100 meters (°C/100m) versus stability class is given in Safety Guide 23 (Ref. 12). If other well-documented methodologies are used to estimate atmospheric stability (with appropriate justification), the models described in this guide may require modification. A well-documented methodology is one that is substantiated by diffusion data for conditions similar to those at the nuclear power plant site involved.	Conform	Atmospheric stability was determined by vertical ΔT between the 60-meter level and the 10-meter level. The range of possible release height 17.37 m to 66.25 m is approximately in the same range as the measurement level.

Conformance with Regulatory Guide 1.194 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
3.1	Appendix A provides information on the structure and content of the meteorological data set and input parameters used by the ARCON96 code.	Conform	The ARCON96 meteorological data input files have been created in conformance with the format specified Appendix A to RG 1.194.
3.2	A 95th-percentile χ/Q value should be determined for each identified source-receptor combination. However, it may be possible to identify bounding combinations in order to reduce the needed calculational effort. In determining the bounding combinations it will be necessary to consider the distance, direction, release mode, and height of the various release points to the environment in relation to the various control room intakes. Additional parameters, such as those used in establishing plume rise, may need to be considered in determining the bounding combination.	Conform	The analysis calculates 95th percentile χ/Q values for each source-to-receptor combination. All sources are treated as point sources. The source-to-receptor distance has been calculated using points on the source and receptor that yield the shortest distance. The analysis methodology yields conservative χ/Q values.
3.2	For cases involving two or more release pathways associated with a single release source, a calculated composite value of χ/Q may be considered on a case-by-case basis if the licensee can demonstrate an acceptable modeling approach and justify the conservatism of any assumed weighting factors.	Not Applicable	The analysis treats all sources as point sources. The source-to-receptor distance has been calculated using points on the source and receptor that yield the shortest distance. The analysis methodology yields conservative χ/Q values.

Conformance with Regulatory Guide 1.194 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
3.2	Changes in associated parameters that could occur as a result of differences between normal operation and accident conditions, differences between accidents, differences that occur over the duration of the accident, single failure considerations, and considerations of loss of offsite power, consistent with accident sequences and descriptions, must all be considered in the characterization of the release points.	Conform	Five locations have been characterized as potential release points. During accident and normal operations, air located in the auxiliary building and fuel building is exhausted through the unit vent. During emergency modes of operation, the air is moved through a filter prior to being exhausted; however, the location of the exhaust does not change. The remaining release points are not impacted by a loss of offsite power.
3.2	The ARCON96 code provides options that allow an analyst to model ground-level, elevated stack, and vent-point source releases. In addition, the analyst can model diffuse area sources as a sub mode of the ground-level release type. These modes and limitations on their use are discussed in the positions that follow.	Not Applicable	This statement is a description of the ARCON96 code and does not require conformance.
3.2.1	The ground-level release mode is appropriate for the majority of control room χ/Q assessments. If the release type is ground level, ARCON96 ignores all user inputs related to release velocity and radius. Release height is used to establish the plume slant path.	Conform	The analysis considers all releases to be ground-level releases.

Conformance with Regulatory Guide 1.194 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
3.2.2	The stack release mode is appropriate for releases from a freestanding, vertical, uncapped stack that is outside the directionally dependent zone of influence of adjacent structures. Such a stack should be more than 2-1/2 times the height of the adjacent structures or be located • more than 5L downwind of the trailing edge of upwind buildings, and • more than 2L upwind of the leading edge of downwind buildings, and • more than 0.5L crosswind of the closest edge of crosswind buildings where L is the lesser of the height or width of the building creating the downwind, upwind, or crosswind wake. Since L will be dependent on wind direction for most building clusters, it will generally be necessary to assess the zone of influence for all directions within the 90° wind direction sector centered on the line of sight between the stack and the control room intake. If multiple intakes are involved such that upwind, downwind, and crosswind orientations are confounded, 5L could be used for each orientation. See Figure 1. Plume rise from buoyancy or mechanical jet effects are not calculated by ARCON96. The analyst may determine plume rise and add the amount of rise to the physical height of the stack to obtain an effective plume height as described in Regulatory Position 6 of this guide.4 Although ARCON96 does not determine plume rise, the input values of stack flow, radius, and vertical velocity are used by ARCON96 to assess downwash and to estimate a limiting χ/Q value.	Not Applicable	The analysis considers all releases to be ground-level releases.

Conformance with Regulatory Guide 1.194 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
3.2.2	If the control room intake is located close to the base of a tall stack, the elevated release model in ARCON96 generates negligibly low χ/Q values. Although perhaps numerically correct, these model results may not be sufficiently conservative for a design basis assessment since the model does not adequately address meteorological conditions that could result in higher χ/Q values. Although the staff has previously suggested that licensees model fumigation as a mechanism to address this situation, the fumigation model did not appear to adequately estimate the effluent concentrations at the bases of industrial stacks. Concentrations greater than those predicted by ARCON96 could result from diurnal wind direction changes, meander, or stagnation. Therefore, the following procedure should be used to assess whether a particular stack-intake configuration is subject to this concern and to determine the appropriate χ/Q values.	Not Applicable	The analysis considers all releases to be ground-level releases.
3.2.2	In addition to running ARCON96 to determine the elevated stack χ/Q values for the control room assessment, the analyst should calculate the maximum elevated stack χ/Q value (non-fumigation) using the methodology of Regulatory Guide 1.145 (Ref. 9) to determine the maximum χ/Q value at ground level for the 0-2 hour interval and for the 24-96 and 96-720 hour intervals. The NRC-sponsored code, PAVAN (Ref. 14), is acceptable to the staff for this assessment. For this assessment, the input parameters should be adjusted such that the effective release height is measured from the elevation of the control room outside air intake rather than plant grade. The same release point characterization and meteorological data sets used in ARCON96 should be used to determine the χ/Q values for several distances in each wind direction sector with the objective of identifying the maximum χ/Q value. Figure A.4 of Reference 15 may be useful in this regard. The maximum χ/Q value obtained for the 0-2 hour interval should be compared to the corresponding χ/Q value generated by ARCON96 and the higher value used in habitability assessments. The χ/Q values generated by ARCON96 for the 2-8 and the 8-24 hour intervals may be used without adjustment.	Not Applicable	The analysis considers all releases to be ground-level releases.

Conformance with Regulatory Guide 1.194 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
3.2.2	For the 24-96 hour and 96-720 hour intervals, the following expressions may be used to determine the effective χ/Q. This deterministic approach assumes that the stack plume reverses direction for 1 hour of each day for the duration of the event. The plume is assumed to fold over itself such that the ground level concentration is at its maximum value at the control room intake. $\left(\frac{\chi}{Q}\right)_{24\to 96\,hr} = \frac{1\cdot\left(\frac{\chi}{Q}\right)^{PAVAN}_{24\to 96\,hr} + 23\cdot\left(\frac{\chi}{Q}\right)^{ARCON96}_{24\to 96\,hr}}{24}$ (1) $\left(\frac{\chi}{Q}\right)_{96\to 720\,hr} = \frac{1\cdot\left(\frac{\chi}{Q}\right)^{PAVAN}_{96\to 720\,hr} + 23\cdot\left(\frac{\chi}{Q}\right)^{ARCON96}_{96\to 720\,hr}}{24}$ (2)	Not Applicable	The analysis considers all releases to be ground-level releases.
3.2.3	The ARCON96 calculation of vent releases includes an algorithm to model mixed-mode releases as described in Regulatory Guide 1.111 (Ref. 10), which addresses χ/Q values used in the assessment of routine effluent releases. The development of this algorithm was based in part on limited field experiments. Given the limited experiment set, the results obtained with this algorithm may not be sufficiently conservative for accident evaluations. For this reason, the vent release mode should not be used in design basis assessments. This position is consistent with the guidance of Regulatory Guide 1.145 (Ref. 9) for offsite χ/Q values. These releases should be treated as a ground level release (Section 3.2.1) or as an elevated release (Section 3.2.2).	Not Applicable	The analysis considers all releases to be ground-level releases.

Conformance with Regulatory Guide 1.194 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
3.2.4	The diffusion models in ARCON96 are based on point-source formulations. However, some release sources may be better characterized as area sources. Examples of possible area sources are postulated releases from the surface of a reactor or a secondary containment building. Typical assessments for loss-of-coolant accidents (LOCAs) have conservatively assumed that the containment structure could leak anywhere on the exposed surface. As such, these assessments typically used the shortest distance between the building surface and the control room intake and have treated the building as a point source. This approach may be unnecessarily conservative. A more reasonable approach, while still maintaining adequate conservatism, would be to model the building surface as a vertical planar area source. This approach is not intended to address dispersion resulting from building-induced turbulence. Treatment of a release as a diffuse source will be acceptable for design basis calculations if the guidance herein is followed. The staff may consider deviations from this guidance on a case-by-case basis.	Not Applicable	The analysis considers all releases to be point-source releases.
3.2.4.1	Diffuse source modeling should be used only for those situations in which the activity being released is homogeneously distributed throughout the building and when the assumed release rate from the building surface would be reasonably constant over the surface of the building. For example, steam releases within a turbine building with roof ventilators or louvered walls would generally not be suitable for modeling as a diffuse source. (See Regulatory Positions 3.2.4.7 and 3.2.4.8.)	Not Applicable	The analysis considers all releases to be point-source releases.

Conformance with Regulatory Guide 1.194 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
3.2.4.2	Since leakage is more likely to occur at a penetration, analysts must consider the potential impact of building penetrations exposed to the environment5 within this modeled area. If the penetration release would be more limiting, the diffuse area source model should not be used. Releases from personnel air locks and equipment hatches exposed to the environment, or containment purge releases prior to containment isolation, may need to be treated differently. It may be necessary to consider several cases to ensure that the χ/Q value for the most limiting location is identified.	Conform	The analysis considers several possible release points including the reactor building equipment hatch. All releases are considered to be point-source releases.
3.2.4.3	The total release rate (e.g., $Ci \cdot s^{-1}$) from the building atmosphere is to be used in conjunction with the diffuse area source χ/Q in assessments. This release rate is assumed to be equally distributed over the entire diffuse source area from which the radioactivity release can enter the environment. For freestanding containments, this would be the entire periphery above grade or above a building that surrounds the lower elevations of the containment. When a licensee can justify assuming collection of a portion of the release from the containment within the surrounding building, the total release from the containment may be apportioned between the exposed and enclosed building surfaces. Similarly, if the building atmosphere release is modeled through more than one simultaneous pathway (e.g., drywell leakage and main steam safety valve leakage in a BWR), only that portion of the total release released through the building surface should be used with the diffuse area χ/Q. The release rate should not be averaged or otherwise apportioned over the surface area of the building. For example, reducing the release rate by 50 percent because only 50 percent of the surface faces the control room intake would be inappropriate.	Not Applicable	The analysis considers all releases to be point-source releases.

Conformance with Regulatory Guide 1.194 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
3.2.4.4	ARCON96 uses two initial diffusion coefficients entered by the user to represent the area source. There are insufficient field measurements to mechanistically model these initial diffusion coefficients. The following deterministic equations should be used in the absence of site-specific empirical data.[6] $\sigma_{Y_0} = \frac{\text{Width}_{\text{area source}}}{6}$ (3) $\sigma_{Z_0} = \frac{\text{Height}_{\text{area source}}}{6}$ (4)	Not Applicable	The analysis considers all releases to be point-source releases.
3.2.4.5	The height and width of the area source (e.g., the building surface) are taken as the maximum vertical and horizontal dimensions of the above-grade building cross-sectional area perpendicular to the line of sight from the building center to the control room intake (see Figure 2). These dimensions are projected onto a vertical plane perpendicular to the line of sight and located at the closest point on the building surface to the control room intake. The release height is set at the vertical center of the projected plane. The source-to-receptor distance (slant path) is measured from this point to the control room intake.	Not Applicable	The analysis considers all releases to be point-source releases.
3.2.4.6	Intentional releases from a secondary containment (e.g., standby gas treatment systems (SGTS) at BWR reactors) or annulus ventilation systems in dual containment structures should be treated as a ground-level release or an elevated stack release, as appropriate. The diffuse area source model may be appropriate for time intervals for which the secondary containment or annulus ventilation system is not capable of maintaining the requisite negative pressure differential specified in technical specifications or in the FSAR. Secondary containment bypass leakage (i.e., leakage from the primary containment that bypasses the secondary containment and is not collected by the SGTS) should be treated as a ground-level release or an elevated stack release, as appropriate.	Not Applicable	No secondary containment sources exist.

Conformance with Regulatory Guide 1.194 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
3.2.4.7	A second possible application of the diffuse area source model is determining a χ/Q value for multiple (i.e., 3 or more) roof vents. This treatment would be appropriate for configurations in which (1) the vents are in a close arrangement, (2) no individual vent is significantly7 closer to the control room intake than the center of the area source, (3) the release rate from each vent is approximately the same, and (4) no credit is taken for plume rise. The distance to the receptor is measured from the closest point on the perimeter of the assumed area source. For assumed areas that are not circular, the area width is measured perpendicular to the line of sight from the center of the assumed source to the control room intake. The initial diffusion coefficient σ_{Yo} is found by Equation 3; σ_{Zo} is assumed to be 0.0.	Not Applicable	The analysis considers all releases to be point-source releases.

Conformance with Regulatory Guide 1.194 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
3.2.4.8	A third possible application of the diffuse area source model is determining a χ/Q value for large louvered panels or large openings (e.g., railway doors on BWR Mark I plants) on vertical walls. This treatment would be appropriate for a louvered panel or opening when (1) the release rate from the building interior is essentially equally dispersed over the entire surface of the panel or opening and (2) assumptions of mixing, dilution, and transport within the building necessary to meet condition 1 are supported by the interior building arrangement. The staff has traditionally not allowed credit for mixing and holdup in turbine buildings because of the buoyant nature of steam releases and the typical presence of high volume roof exhaust ventilators. The distance to the receptor and the release height is measured from the center of the louvered panel or opening. Initial diffusion coefficients are found using Equations 3 and 4 assuming the width and height is that of the panel or opening rather than that of the building. If the area source and the intake are on the same building surface such that wind flows along the building surface would transport the release to the intake, the initial dispersion coefficient will need to be adjusted. If the included angle between the source-receptor line of sight and the vertical axis of the assumed source is less than 45 degrees, σ_{Yo} should be set to 0.0. If the included angle between the source-receptor line of sight and the horizontal axis of the assumed source is less than 45 degrees, σ_{Zo} should be set to 0.0.	Not Applicable	The analysis considers all releases to be point-source releases.

Conformance with Regulatory Guide 1.194 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
3.3	This section of the guide provides guidance to the meteorological analyst in applying models for determining χ/Q values that are appropriate for the as-built configuration of control room intakes. Radioactive materials released during an accident can enter the control room envelope via several potential pathways. These pathways may be intentional (e.g., ventilation system outside air intakes) and unintentional infiltration paths (e.g., doorways, envelope penetrations, leakage in ventilation system components). The applicable pathways will vary from site to site depending on the arrangement of the control room envelope in relation to other site buildings, the pressure differentials between these buildings and the control room, the configuration of control room ventilation systems, and the classification of the control room dose control (e.g., zone isolation with filtered pressurization, zone isolation with no pressurization). It may be necessary to determine χ/Q values for each potential pathway. However, the selection of one or more bounding intakes for the χ/Q evaluation may be sufficient to establish compliance with regulatory guidelines.	Conform	The χ/Q values for the control room intake were calculated and are used for both the control room intake and unintentional infiltration paths.
3.3.1	All control room ventilation systems draw makeup air from the environment during normal operations and many draw air from the environment for the purpose of supplying filtered pressurization air. The configuration of these systems may change between normal and emergency modes. In some configurations, normal ventilation outside air intakes isolate and different intakes open to supply pressurization air. Some intake dampers may have failure modes related to loss of ac power or single failures. These considerations should be evaluated in identifying the control room outside air intakes for which χ/Q values should be calculated.	Conform	The analysis considers the Control Room air intakes (vents 302 and 303) which supply the Control Room ventilation system. The normal ventilation intake has redundant safety related motor operated dampers that can isolate the flow path. Since redundant dampers are used, failure of one will not prevent the normal ventilation intake from being isolated.

Conformance with Regulatory Guide 1.194 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
3.3.2	This section applies to control room ventilation system configurations that have two outside air intakes, each of which meets applicable design criteria of an engineered safeguards feature (ESF), including single-failure criterion, missile protection, seismic criteria, and operability under loss-of-offsite AC power conditions. Operability requirements should be provided in technical specifications. The outside air intakes should be located with the intent of providing a low contamination intake regardless of wind direction. The assurance of a low contamination outside air intake depends on release point configuration, building wake effects, terrain, and the possibility of wind stagnation or wind direction reversals. The two intakes should not be within the same wind direction window, defined as a wedge centered on the line of sight between the source and the receptor with the vertex located on the release point. If ARCON96 is used, the wedge angle is 90° (i.e., 45 degrees on either side of the line of sight). If the methods of Regulatory Position 4 are used, the size of the wedge is as given in Table 2. Figure 3 illustrates four examples of the interplay between control room intakes, release points, and wind direction windows. In addition, the analyst should consider χ/Q values for infiltration pathways as discussed in Regulatory Position 3.3.3.	Not Applicable	The Control Room has only one air intake location.
3.3.2	The methods of this regulatory position involve identification of the limiting and favorable intakes with regard to their χ/Q value. Because of the interplay of building wake, plume rise, wind direction frequency, intake flow rate, and other parameters, it may not be possible to identify the limiting or favorable intake by observation. In these situations, χ/Q values should be calculated for each release point-intake combination and the limiting and favorable intakes identified on the basis of these values.	Not Applicable	The Control Room has only one air intake location.

Conformance with Regulatory Guide 1.194 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
3.3.2.1	If both of the dual intakes are located within the same wind direction window, both intakes could be contaminated (See Figure 3(a)). In this case, the χ/Q values for each air intake should be calculated using ARCON96 as described in other sections of this guide and an effective χ/Q value calculated. Equation 5a should be used if the intake flow rates are equal. If the intake flow rates are not equal, but the imbalance does not shift between intakes, Equation 5b should be used. If the flow rate imbalance can shift between intakes, Equation 5c should be used. This calculation is repeated for each averaging time interval. $\overline{\chi/Q} = \frac{1}{2}\left[(\chi/Q)_1 + (\chi/Q)_2\right]$ (5a) $\overline{\chi/Q} = \frac{F_1 \cdot (\chi/Q)_1 + F_2 \cdot (\chi/Q)_2}{F_1 + F_2}$ (5b) $\overline{\chi/Q} = \frac{\max(F_1,F_2)\cdot\max\left[(\chi/Q)_1,(\chi/Q)_2\right] + \min(F_1,F_2)\cdot\min\left[(\chi/Q)_1,(\chi/Q)_2\right]}{F_1 + F_2}$ (5c) *Where:* $\overline{\chi/Q}$ = Effective χ/Q, $s \cdot m^{-3}$ $(\chi/Q)_1$, $(\chi/Q)_2$ = χ/Q value for outside air intakes 1 and 2, $s \cdot m^{-3}$ F_1, F_2 = Flow rate for outside air intakes 1 and 2, cfm	Not Applicable	The Control Room has only one air intake location.

Conformance with Regulatory Guide 1.194 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
3.3.2.2	If the dual outside air intakes are not in the same wind direction window but cannot be isolated by design, the χ/Q values for the limiting outside air intake should be calculated for each time interval as described elsewhere in this guide. Equation 6a should be used if the intake flow rates are equal. If the intake flow rates are not equal, but the imbalance does not shift between intakes, Equation 6b should be used. If the flow rate imbalance can shift between intakes, Equation 6c should be used. $\overline{\chi/Q} = \frac{1}{2}\max[(\chi/Q)_1,(\chi/Q)_2]$ (6a) $\overline{\chi/Q} = \frac{\max[F_1\cdot(\chi/Q)_1,F_2\cdot(\chi/Q)_2]}{F_1+F_2}$ (6b) $\overline{\chi/Q} = \frac{\max(F_1,F_2)\cdot\max[(\chi/Q)_1,(\chi/Q)_2]}{F_1+F_2}$ (6c)	Not Applicable	The Control Room has only one air intake location.

Conformance with Regulatory Guide 1.194 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
3.3.2.3	If the ventilation system design allows the operator to manually select the least contaminated outside air intake as a source of outside air makeup and close the other intake, the χ/Q values for each of the outside air intakes should be calculated for each time interval as described elsewhere in this guide. The χ/Q value for the limiting intake should be used for the time interval prior to intake isolation. This χ/Q value may be reduced by a factor of 2 to account for dilution by the flow from the other intake (see Equation 6a).8 The χ/Q values for the favorable intake are used for the subsequent time intervals. The χ/Q values for the favorable intake may be reduced by a factor of 4 to account for the dual inlet and the expectation that the operator will make the proper intake selection. This protocol should be used only if the dual intakes are in different wind direction windows and if there are redundant, ESF-grade radiation monitors within each intake, with control room indication and alarm, to monitor the intakes. The requisite steps to select the least contaminated outside air intake, and provisions for monitoring to ensure the least contaminated intake is in use throughout the event, should be addressed in procedures and in operator training.	Not Applicable	The Control Room has only one air intake location.
3.3.2.3	A conservative delay time should be assumed for the operator to complete the necessary actions. This delay period should consider: (1) the time for the operator to recognize the radiation monitor alarm and determine its validity (as provided for in the alarm response procedure), (2) delays associated with other accident response actions competing for the operator's attention, (3) the time needed to complete the actions, and (4) diesel generator sequencing time, if applicable. If actions are required outside the control room, delays associated with transit to the local control stations (including those delays caused by worker radiological protection controls associated with accident dose rates), and the availability of personnel should be considered.	Not Applicable	The Control Room has only one air intake location.

Conformance with Regulatory Guide 1.194 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
3.3.2.4	If the ventilation system design provides for automatic selection of the least contaminated outside air intake, the χ/Q values for the favorable intake should be calculated for each time interval as described elsewhere in this guide. The χ/Q values may be reduced by a factor of 10 to account for the ability to automatically select a "clean" intake. This protocol should be used only if the dual intakes are in different wind direction windows, there are redundant ESF grade radiation monitors within each intake, and an ESF-grade control logic and actuation circuitry is provided for the automatic selection of a clean intake throughout the event.	Not Applicable	The Control Room has only one air intake location.
3.3.3	Infiltration of contaminated air to a control room can be minimized by proper design and maintenance of the control room envelope (CRE). However, infiltration is always a possibility and the location and significance of these leakage pathways may warrant determination of χ/Q values. An unfiltered inleakage path of 100 cfm can admit the same quantity of radioactive material as a pressurization air intake having a flow of 2000 cfm through a 95 percent efficient filter. The situation can be further compounded if the χ/Q for the unfiltered pathway is more limiting than that for the control room outside air intake.	For information	The paragraph emphasizes the importance of infiltration.
3.3.3	The infiltration paths actually applicable to a particular facility will be identified via inleakage testing or CRE inspections and surveillances. Refer to Table H-1, "Determination of Vulnerability Suspectibility," of NEI 99-03, "Control Room Habitability Guidance" (Ref. 16), for further guidance on infiltration pathways.	Conform	Per Technical Specification SR 3.7.10.4, unfiltered air inleakge testing of the CRE and CBE boundaries is performed in accordance with the Control Room Envelope Habitability Program.

Conformance with Regulatory Guide 1.194 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
3.3.3	A 95th-percentile χ/Q value should be determined for each time interval for any infiltration path that could result in a significant intake of contaminated air into the CRE. Because of the interplay of source-to-receptor distance and direction, infiltration path flow rate, whether the path is filtered or unfiltered, and other considerations, it may not be possible to identify the potential impact of an infiltration path by observation. In these situations, χ/Q values should be calculated for each pathway and the limiting χ/Q value(s) identified. If there is sufficient margin available, it may be possible to calculate χ/Q values assuming the shortest distance between the release point and any identified point of infiltration on the outside of the CRE.	Conform	The primary contribution of in-leakage is primarily due to outside air that enters through the control building intake and is able to leak past filtration equipment while in the HVAC ducts. Thus, the appropriate χ/Q value to use for this parameter is the χ/Q corresponding to the HVAC intake louvers that provide outside air to the control room. The χ/Q values for the control room intake were calculated and are used for both the control room intake and infiltration paths.
3.4	When the combinations of release points and intakes have been identified, the direction and distance between the release point and the intake should be determined. Wind direction data are recorded as the direction from which the wind blows (e.g., a north wind blows from the north, a wind blowing out of the west is recorded with a direction of 270 degrees). The direction input to ARCON96 is the wind direction that would carry the plume from the release point to the intake.9 For example, an analyst standing at the intake facing west to the release point, would enter 270 degrees; an analyst facing north, would enter 360 degrees, etc.	Conform	The analysis uses receptor-to-source orientation inputs that conform to the coordinate system defined by RG 1.194.

Conformance with Regulatory Guide 1.194 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
3.4	The source-to-receptor distance is the shortest horizontal distance between the release point and the intake. ARCON96 will use this distance and the elevations of the source and receptor to calculate the slant path. For an area source such as building surface, the shortest horizontal distance from the building surface to the control room intake is used as the source-to-receptor distance. For releases within building complexes, the shortest horizontal distance between the release point and the intake could be through intervening buildings. In these cases, it is acceptable to take the length of the shortest path (e.g., "taut string length") around or over the intervening building as the source-to-receptor distance. If the distance to the receptor is less than about 10 meters, the ARCON96 code and the procedures in Regulatory Position 4 should not be used to assess χ/Q values. These situations will need to be addressed on a case-by-case basis.	Conform	The analysis calculates distances between release sources and the Control Room air intake along the shortest path ("taut string length"). All distances are longer than 10 meters.
4	This regulatory position addresses alternative methods for determining χ/Q values for control room radiological habitability assessments. The methods in Regulatory Positions 4.1 to 4.3 are based on Murphy-Campe (Ref. 2) and the Standard Review Plan Chapter 6.4 (Ref. 3).	For information	This statement does not require conformance.

Conformance with Regulatory Guide 1.194 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
4.1	The 0-8 hour 95th-percentile[10] χ/Q value for a single point source on the surface of the containment or other building and a single point receptor with a difference in elevation less than 30 percent of the building height may be estimated using Equation 7. $$\frac{\chi}{Q} = \frac{1}{3\pi U \sigma_y \sigma_z} \quad (7)$$ *Where:* χ/Q = Relative concentration at plume centerline for time interval 0-8 hours, $s \cdot m^{-3}$ 3 = Wake factor U = Wind speed at 10 meters, $m \cdot s^{-1}$ σ_y, σ_z = Standard deviation, in meters, of the gas concentration in the horizontal and vertical cross wind directions evaluated at distance x and by stability class	Not Applicable	All χ/Q values have been calculated using the ARCON96 code. No alternative calculation methods are necessary.
4.2	Equation 8 may be used when the activity is assumed to leak from many points on the surface of a building such as the containment in conjunction with a single point receptor. This equation is also appropriate for point source-point receptors where the difference in elevation between the source and the receptor is greater than 30 percent of the height of the upwind building, typically the containment, which creates the most significant building wake impact. The equation is also applicable to a point source and volume receptor (e.g., an isolated control room with infiltration occurring at many locations). $$\frac{\chi}{Q} = \left[U\left(\pi\sigma_y\sigma_z + \frac{A}{K+2}\right)\right]^{-1} \quad (8)$$ *Where:* χ/Q = Relative concentration at plume centerline for time interval 0-8 hours, $s \cdot m^{-3}$ U = Wind speed at 10 meters, $m \cdot s^{-1}$ σ_y, σ_z = Standard deviation, in meters, of the gas concentration in the horizontal and vertical cross wind directions evaluated at distance x and by stability class $K = \frac{3}{(s/d)^{1.4}}$	Not Applicable	All χ/Q values have been calculated using the ARCON96 code. No alternative calculation methods are necessary.

Conformance with Regulatory Guide 1.194 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
	s = Shortest distance between building surface and receptor location, m d = Diameter or width of building, m A = Cross-section area of building, m^2 The reference to "building" in the definitions of s, d, and A is to the diffuse source (e.g., containment). If the equation is used with a point source, the reference is to the building that has the greatest impact on the building wake. The values of the parameters σy, σz and U should be determined on the basis of the values of the site meteorological data. Some early analyses may have been based on generic meteorology conditions (e.g., F stability with wind speeds of 1.0 m·s-1). If these early analyses are to be updated, the staff recommends that the ARCON96 code be used. If the ARCON96 code is not used, site-specific hourly meteorological data should be used to determine the 95th-percentile χ/Q value. Figures 4 and 5 provide sigma values by stability category for distances greater than 10 meters. The data on these graphs should not be extrapolated for distances less than 10 meters.		
4.3	Equations 7 and 8 of this guide may be used in conjunction with the procedures in Regulatory Position 3.3.2 to determine χ/Q values for control room designs having two or more control room outside air intakes, each of which meets the requirements of an engineered safety feature (ESF) including, as applicable, single-failure criteria for active components, seismic criteria, and missile criteria. If Equation 8 of this guide is used, the parameter K should be set to 0.0. In a change from previous practice, the staff no longer finds Equation 7 of Reference 2 to be acceptable for use in new applications.	Not Applicable	All χ/Q values have been calculated using the ARCON96 code. No alternative calculation methods are necessary.
4.4	Equations 7 and 8 are used to determine χ/Q values for the first time interval of 0-8 hours. The χ/Q values for other time intervals are obtained by adjusting for long-term meteorological averaging of wind speed and wind direction.[11] This is accomplished by multiplying the 0-8 hour time interval χ/Q value by a correction factor for wind speed and a correction factor for wind direction.	Not Applicable	All χ/Q values have been calculated using the ARCON96 code. No alternative calculation methods are necessary.

Conformance with Regulatory Guide 1.194 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
4.4.1	This correction is defined as the ratio of the wind speed used to determine the 0-8 hour χ/Q value to the wind speed appropriate for each of the other time intervals. Column 2 of Table 1 tabulates the wind speed percentiles that correspond to each of these intervals. The hourly data should be arranged in order of increasing wind speed and the wind speed percentiles determined (i.e., the lowest wind speeds associated with the lowest percentiles). Include only the wind speed data associated with wind directions from sectors that result in receptor contamination. Table 2 tabulates the size of the minimum wind direction window to be used. From this ranking, identify the wind speed value for each interval that is not exceeded more than the stated percentage of the time. Divide this wind speed value into the 5th-percentile wind speed used to determine the 0-8 hour χ/Q to obtain the χ/Q correction factor for wind speed. The values shown in Column 1 of Table 1 are representative correction factors that may be used if hourly observation meteorological data are not available. [Refer to the RG for the Tables]	Not Applicable	All χ/Q values have been calculated using the ARCON96 code. No alternative calculation methods are necessary.
4.4.2	The average wind direction frequency F is obtained by summing the annual average wind direction frequencies within the minimum window. Table 2 tabulates the size of the minimum wind direction window to be used. Column 2 of Table 3 is used to determine the χ/Q correction factor for wind direction for each time interval. Column 1 is used when F has not been determined. [Refer to the RG for the Table]	Not Applicable	All χ/Q values have been calculated using the ARCON96 code. No alternative calculation methods are necessary.
5	The alternative method in this section may be used to model the release to the environment as an instantaneous puff release. One hundred percent of the radionuclides must be released directly to the environment over a period no longer than about 1 minute for a release to qualify as a puff release. Releases to enclosed buildings, intermittent releases that occur over a period longer than about 1 minute (e.g., releases from relief valves, atmospheric dumps), and releases that occur over a period longer than about 1 minute should be treated as continuous point source releases. The diffusion equation for an instantaneous puff ground level release, with no puff rise and no	Not Applicable	All χ/Q values have been calculated using the ARCON96 code. No alternative calculation methods are necessary.

Conformance with Regulatory Guide 1.194 (cont.)

RG Section	RG Position	Analysis	Comments
	crosswind offset (i.e., center of puff is assumed to pass over control room intake), integrated over the duration of the puff passage is: $$\frac{\chi}{Q}(x,u,k,h) = \frac{\int_0^T \frac{2}{(\sigma_z^2(x,k)+\sigma_I^2)^{1/2}(2\pi)^{3/2}(\sigma_{x,y}^2(x,k)+\sigma^2)} \exp\left[-\frac{1}{2}\left(\frac{(x-u\cdot t)^2}{(\sigma_{x,y}^2(x,k)+\sigma_I^2)} + \frac{h^2}{(\sigma_z^2(x,k)+\sigma_I^2)}\right)\right]\cdot F(t)\,dt}{\int_0^T F(t)\,dt} \quad (10)$$ Where: $\frac{\chi}{Q}(x,u,k,h)$ = Effective puff relative concentration, s m^{-3} χ = Integrated concentration at control room intake, Ci·m^{-3}·s^{-1} Q_i = Release quantity, for nuclide i, Ci x = Release point to receptor distance, m u = Wind speed, m/sec. Assume 1.0 m·s^{-1} k = Stability Class. Assume F. h = Difference in elevation between the physical release point and the control room intake, m. If the control room intakes is at a higher elevation than the release point and the puff is buoyant, assume h = 0. T = Time for trailing edge of puff to pass control room intake, sec. $= \frac{x + 3\cdot\left[\sigma_{x,y}(x,k)+\sigma_I\right]}{u}$ F = Control room total intake flow rate, cfm. (If the control room intake flow rate is constant over the period 0 to T seconds, the F(t) terms can be omitted from Equation 10.)		
	$\sigma_{x,y}(x,k)$ = Standard deviation, m, of the puff in the horizontal along the wind direction and cross-wind directions at the receptor location. Use Figure 4 with the distance x and stability class k to determine $\sigma_{x,y}$ at the receptor, e.g., $\sigma_{x,y} = \sigma_y$. $\sigma_z(x,k)$ = Standard deviation, m, of the puff in the vertical cross-wind direction at the receptor location. Use Figure 5 with the distance x and stability class k to determine σ_z at the receptor. σ_I = Initial standard deviation, m $= \left[\frac{2\cdot V}{(2\pi)^{3/2}}\right]^{1/3}$ V = Initial puff volume (expanded to standard atmospheric conditions), m^3 (The puff dimensions that would exist when the puff is at the control room intake are assumed to exist during the entire puff transit.) Equation 10 provides the effective relative concentration for the puff. This value can be input to dose assessment codes such as RADTRAD or HABIT as any value of χ/Q would be if the intake flows, release duration, and release rates are modeled consistent with the inputs to Equation 10.		

Conformance with Regulatory Guide 1.194 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
6	An applicant or licensee may propose adjustments to the release height for plume rise that are due to buoyancy or mechanical jet on a case-by-case basis. In order to credit these adjustments, the applicant or licensee must be able to demonstrate that the assumed buoyancy or vertical velocity of the effluent plumes will be maintained throughout the time intervals that plume rise is credited. Such justifications need to consider the availability of AC power, failure modes of dampers and ductwork, time-dependent release stream temperatures and pressures, and 95th-percentile wind speeds and ambient temperatures.[13] Plume rise may be considered for freestanding stacks and for vents located on plant buildings. However, plume rise may not be used in demonstrating that a particular stack meets the 2-1/2 times the adjacent structure height criterion in Regulatory Position 3.2.2. A mixed-mode release model, such as that in Regulatory Guide 1.111 (Ref. 10), should not be used for design basis assessments.	Not Applicable	The analysis does not credit plume rise.
6	The plume rise may be determined through the use of the following set of equations (Ref. 17). The plume rise for plant vents is determined using Equation 11. The distance x is entered as the horizontal distance between the vent and the control room outside air intake.	Not Applicable	The analysis does not credit plume rise.

Conformance with Regulatory Guide 1.194 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
6	The plume rise for isolated, free-standing stacks is calculated using Equations 11, 12, and 13. The distance x in Equation 11 should be based on the downwind location corresponding to the maximum χ/Q value. See Regulatory Position 3.2.2. The plume rises calculated using Equations 12 and 13 should be compared and the larger plume rise identified. The result of this comparison is then compared to the plume rise determined using Equation 11 and the smaller plume rise selected for use. $\Delta h = \left[\frac{3}{\beta_1^2}\cdot\frac{F_m}{U^2}\cdot x + \frac{3}{2\beta_1^2}\cdot\frac{F_b}{U^3}\cdot x^2\right]^{1/3}$ (11) $\Delta h = 2.6\cdot\left(\frac{F_b}{Us}\right)^{1/3}$ (12) $\Delta h = 2.44\cdot\left(\frac{F_m}{s}\right)^{1/4}$ (13) Where: Δh = Plume rise, m F_m = Momentum flux parameter, $m^4 \cdot s^{-2}$ $= \frac{\rho_o V_o w_o}{\pi \rho_a}$ β_1 = Dimensionless entrainment constant for momentum = 0.6 U = Wind speed at release height, $m \cdot s^{-1}$ x = Distance from release point to receptor, m (see text) F_o = Buoyancy flux parameter, $m^4 \cdot s^{-3}$ $= \frac{g(\rho_a - \rho_o)V_o}{\pi \rho_a}$ w_o = Effluent exit velocity, $m \cdot s^{-1}$ V_o = Volumetric release rate, $m^3 \cdot s^{-1}$ ρ_o = Effluent density after expansion to atmospheric pressure, $kg \cdot m^{-3}$ ρ_a = Density of air, $kg \cdot m^{-3}$ s = 0.0001 s^{-2} for A, B, C, and D stability; 0.00049 s^{-2} for E stability; 0.0013 s^{-2} for F stability; 0.002 s^{-2} for G stability g = Gravitational acceleration, 9.8 $m \cdot s^{-2}$	Not Applicable	The analysis does not credit plume rise.

Conformance with Regulatory Guide 1.194 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
6	Although ARCON96 processes ambient meteorological conditions on an hour-by-hour basis, the code cannot vary the other parameters that enter into a plume rise determination. For example, wind speed and stability class are varied hour by hour, but the density of air, the density of the effluent stream, and the vertical velocity are not varied hour-by-hour. As such, the analyst should ensure that these parameters are bounding for the entire period of the χ/Q assessment or use individual time intervals to model the time-variant parameters. An alternative approach would be to calculate the plume rise for each hour independently of ARCON96 and to select a plume rise that is exceeded more than 95 percent of the time. This rise is then added to the stack height as input to ARCON96.	Not Applicable	The analysis does not credit plume rise.
6	In lieu of mechanistically addressing the amount of buoyant plume rise associated with energetic releases from steam relief valves or atmospheric dump valves, the ground level χ/Q value calculated with ARCON96 (on the basis of the physical height of the release point) may be reduced[14] by a factor of 5. This reduction may be taken only if (1) the release point is uncapped and vertically oriented and (2) the time-dependent vertical velocity exceeds the 95th-percentile wind speed[13] (at the release point height) by a factor of 5.	Not Applicable	The analysis does not credit plume rise.

Conformance with Regulatory Guide 1.194 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
7	The methods and parameters provided in this guide are acceptable for use for design basis control room habitability radiological assessments provided that all stated prerequisites and conditions are met. The staff believes that use of the guidance in this guide will result in χ/Q values that are acceptably conservative. However, there may be circumstances in which these methods and parameters may not be advantageous for a particular plant configuration and site meteorological regimes and may lead to results that are deemed to be unnecessarily conservative. Licensees and applicants may opt to propose alternative methods and parameters such as those that are based in part on data obtained from site-specific experimental measurements. Data based on wind tunnel tests should be accompanied with an evaluation of the representativeness of the experiment results to the particular plant configuration and site meteorological regimes. These proposed alternatives, with supporting data, will be considered by the staff on a case-by-case basis. The staff recommends that licensees considering an experimental program request a meeting with the staff in advance of starting the program. The intent of this recommendation is to allow the staff and the licensee (or applicant) to discuss the proposed program, prior to resource expenditure, and for the staff to provide a preliminary assessment of the proposal. The staff's approval of the proposed alternative methods and parameters will not be granted, however, until the licensee or applicant completes the experimental program and dockets the proposal with supporting analyses and data for formal staff review. An acceptable experimental program should incorporate the following standards:	Not Applicable	The analysis is performed using the ARCON96 code and follows RG 1.194 recommendations. No alternative methods are used in the analysis.
7.1	The experimental program should be appropriately structured so as to provide data of appropriate quantity and quality to support data analysis and conclusions drawn from that data. The program should be developed by personnel who have educational and work experience credentials in air dispersion meteorology and modeling.	Not Applicable	The analysis is performed using the ARCON96 code and follows RG 1.194 recommendations. No alternative methods are used in the analysis.

Conformance with Regulatory Guide 1.194 (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
7.2	The experimental program should encompass a sufficient range of meteorological conditions applicable to the particular site so as to ensure that the data obtained address the site-specific meteorological regimes and the site-specific release point/receptor configurations that impact the control room χ/Q values. Meteorological conditions observed at the particular site with a frequency of 5 percent or greater in a year should be addressed. Parameters derived from statistical analyses on the experimental data should represent the 95th-percentile confidence level.	Not Applicable	The analysis is performed using the ARCON96 code and follows RG 1.194 recommendations. No alternative methods are used in the analysis.
7.3	The experimental program, including data reduction and analysis, should incorporate applicable quality control criteria of Appendix B to 10 CFR Part 50. The products of the experimental program should be verified and validated.	Not Applicable	The analysis is performed using the ARCON96 code and follows RG 1.194 recommendations. No alternative methods are used in the analysis.

8 NRC REGULATORY ISSUE SUMMARY 2006-04 TABLE

NRC REGULATORY ISSUE SUMMARY 2006-04 COMPARISON

Table 1 Conformance with NRC Regulatory Issue Summary 2006-04

Issue Number	NRC RIS 2006-04 Position	Analysis	Comments
1	Level of Detail Contained in LARs An AST amendment request should describe the licensee's analyses of the radiological and non-radiological impacts and provide a justification for the proposed modification in sufficient detail to support review by the NRC staff. For example, the AST amendment request should (1) provide justification for each individual proposed change to the technical specifications (TS), (2) identify and justify each change to the licensing basis accident analyses, and (3) contain enough details (e.g., assumptions, computer analyses input and output) to allow the NRC staff to confirm the dose analyses results in independent calculations. The provision of sufficient detail is necessary for the NRC staff to be able to conclude, with reasonable assurance, whether the licensee's analyses and changes are acceptable. For a previous NRC staff discussion on the level of detail necessary for review, see RIS 2001-19, "Deficiencies in the Documentation of Design Basis Radiological Analyses Submitted in Conjunction with License Amendment Requests" (Ref. 1). In response to RAIs, some licensees have made changes to originally proposed LARs and their supporting analyses. In some cases, these changes were extensive or involved multiple re-analyses and supplements. Because of the depth and scope of many AST submittals, multiple changes to the original submittal (particularly those with multiple supplements that revise portions of previous supplements) can increase the chance of NRC staff using information that has been superseded during the review. For these cases, NRC staff recommends that licensees identify the most current analyses, assumptions, and TS changes in their submittal and supplements to the submittal.	Conforms	The submittal is modeled after previous NRC-accepted submittals. Included is justification for each proposed change to the TS, identification of changes to licensing basis analyses, and sufficient analysis detail to allow for result verification through independent calculations.

Table 1 (cont.)	Conformance with NRC Regulatory Issue Summary 2006-04		
Issue Number	**NRC RIS 2006-04 Position**	**Analysis**	**Comments**
2	Main Steam Isolation Valve (MSIV) Leakage and Fission Product Deposition in Piping For calculation of aerosol settling velocity in the main steamline (MSL) piping of boiling water reactors, some LARs reference Accident Evaluation Report (AEB) 98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application Using the Revised (NUREG-1465) Source Term" (Ref. 2). This is acceptable. However, it is important to note that the report was written based on the parameters of a particular plant and, therefore, the removal rate constant is specific to that plant. Any licensee who chooses to reference these AEB 98-03 assumptions should provide appropriate justification that the assumptions are applicable to their particular design. Both NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" (Ref. 3) and Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants" (Ref. 4) define AST as a fission product release from the reactor core into the containment. Neither provides sufficient information regarding the amount and composition of fission products in the reactor vessel or the attached piping. As indicated in Appendix A to RG 1.183, Regulatory Position 6.0, the NRC staff accepts the practice of treating fission product concentration in containment (more specifically in the drywell) as representative of that in the vicinity of the MSIV. Some AST amendment requests have reduced the drywell activity levels by assuming mixing with the free air volume of the wetwell. If appropriate justification is provided, the suppression pool free air volume may be included, provided there is a mechanism to ensure mixing between the drywell and wetwell. For example, the NRC staff would expect to see thermal and hydraulic analyses in support of crediting mass exchange between the wetwell and drywell airspace for time periods associated with fission product releases.	Not Applicable	MSIV leakage is not applicable to the WCGS design.

Table 1 (cont.)	Conformance with NRC Regulatory Issue Summary 2006-04		
Issue Number	**NRC RIS 2006-04 Position**	**Analysis**	**Comments**
2	The size distribution of airborne particles in the vicinity of the MSIV is, in general, different from that in the containment. Since the piping is attached to the source of fission product releases, the agglomeration process of highly concentrated but small aerosols may substantially differ from that in containment. Modeling of MSL piping may include volumes between the reactor pressure vessel and the inboard MSIV (inboard volume), between the inboard and outboard valves (in-between volume), and outside of the outboard valve (outboard volume). Since a majority of large (i.e., heavier) particles deposit in the inboard volume, the distribution of the aerosol that leaks to the subsequent volume is smaller (i.e., lighter) particles. This particle behavior leads to the conclusion that the choice of an effective settling velocity in any volume should account for the distribution of particle sizes in that volume. The steam flow rate during the accident also affects the removal of particles and should be accounted for in the analysis. For aerosol settling, only horizontal sections of piping should be credited. The effective settling area should be calculated as the length of horizontal piping multiplied by the pipe diameter. Deposition of gaseous iodine (elemental and organic) in the piping is a frequent point of contention between licensees and NRC staff. Some licensees claim that because of chemical adsorption, a large portion of iodine is deposited on the piping surface. However, this deposition is strongly dependent upon the thermal and hydraulic conditions in the piping. Given the large uncertainty associated with iodine behavior in piping, deposition of gaseous iodine in piping should be omitted unless appropriate justification is provided (including providing estimates of the thermal and hydraulic conditions in the piping).		

Table 1 (cont.)	Conformance with NRC Regulatory Issue Summary 2006-04		
Issue Number	**NRC RIS 2006-04 Position**	**Analysis**	**Comments**
3	Control Room Habitability When implementing an AST, some licensees have proposed that certain engineered safety features (ESF) ventilation systems not be credited as a mitigation feature in response to an accident. In some cases, the licensee's revised design basis analysis introduced the assumption that normal (non-ESF) ventilation systems are operating during all or part of an accident scenario. Such an assumption is inappropriate unless the non-ESF system meets certain qualities, attributes, and performance criteria as described in RG 1.183, Regulatory Positions 4.2.4 and 5.1.2. For example, credit for the operation of non-ESF ventilation systems should not be assumed unless they have a source of emergency power. In addition, the operation of ventilation systems establishes certain building or area pressures based upon their flowrates. These pressures affect leakage and infiltration rates which ultimately affect operator dose. Therefore, to credit the use of these systems, licensees should incorporate the systems into the ventilation filter testing program in Section 5 of the TS. In summary, use of non-ESF ventilation systems during a DBA should not be assumed unless the systems have emergency power and are part of the ventilation filter testing program in Section 5 of the TS.	Conforms	Radiation monitors are not credited for actuation of emergency control room HVAC mode. Therefore, events which do not result in a safety injection signal (rod ejection primary-to-secondary leakage, locked RCP rotor, loss of AC power, fuel handling accident, letdown line break, tank ruptures) are assumed to stay in normal control room HVAC mode. The normal HVAC lineup flow rate is greater than the ESF ventilation system flow rate. Likewise, the normal ventilation system does not credit filtration. Therefore, the resulting dose to personnel within the control room will be greater if it is assumed that the ESF ventilation system is not actuated at the event initiation due to a loss of offsite power.

Table 1 (cont.)	Conformance with NRC Regulatory Issue Summary 2006-04		
Issue Number	**NRC RIS 2006-04 Position**	**Analysis**	**Comments**
	Generic Letter (G L) 2003-01, "Control Room Habitability" (Ref. 5) requested licensees to confirm the ability of their facility's control room to meet applicable habitability regulatory requirements. In addition, licensees were requested to confirm that control room habitability systems were designed, constructed, configured, operated and maintained in accordance with the facility's design and licensing bases. The GL placed emphasis on licensees confirming that the most limiting unfiltered inleakage into the control room envelope (CRE) was not greater than the value assumed in the DBA analyses. The tests, measurements, and analyses which were performed for this confirmation were to be described in the response to the GL. Some AST amendment requests proposed operating schemes for the control room and other ventilation systems which affect areas adjacent to the CRE and are different from the manner of operation and performance described in the response to the GL without providing sufficient justification for the proposed changes in the operating scheme. In some cases, licensees proposed new modes of operation that lacked confirmation of the CRE inleakage characteristics. Measurements of these characteristics are important to confirm inleakage assumptions used in the analyses for an AST amendment, even for those situations in which the air in the control room would appear to be stagnant.		Acceptable control room doses have been calculated with a maximum unfiltered inleakage of 400 cfm to the control building and 50 cfm to the control room. Per Technical Specification SR 3.7.10.4, unfiltered air inleakage testing of the CRE and CBE boundaries is performed in accordance with the Control Room Envelope Habitability Program.
4	Atmospheric Dispersion Licensees may continue to use atmospheric relative concentration (χ/Q) values and methodologies from their existing licensing-basis analyses when appropriate. Licensees also have the option to adopt the generally less conservative (more realistic) updated NRC staff guidance on determining χ/Q values in support of design basis control room radiological habitability assessments provided in RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants" (Ref. 6). Regulatory positions on χ/Q values for offsite (i.e., exclusion area boundary and low population zone) accident radiological consequence assessments are provided in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants" (Ref. 7).	Conform	The re-calculation of atmospheric dispersion factors was performed according to the guidance of RG 1.194 and RG 1.145.

Table 1 (cont.)	Conformance with NRC Regulatory Issue Summary 2006-04		
Issue Number	**NRC RIS 2006-04 Position**	**Analysis**	**Comments**
	Based on submittal reviews, the NRC staff identified the following areas of improvement for licensee submittals that propose revision of the design basis atmospheric dispersion analyses for implementing AST. They should include the following information: • A site plan showing true North and indicating locations of all potential accident release pathways and control room intake and unfiltered inleakage pathways (whether assumed or identified during inleakage testing). • Justification for using control room intake χ/Q values for modeling the unfiltered inleakage, if applicable.	Provided in Comment Response	The following items are being provided: 1. A site plan is provided in Figure 8-1 below. 2. Justification for using control room intake χ/Q values for modeling the unfiltered in-leakage is provided in Note 1 below. Additionally, a detailed drawing for the site plan (8025-C-KG1202 Revision 4) is provided in Enclosure VII CD-ROM. 3. A copy of meteorological data inputs for PAVAN and ARCON96 are contained in Enclosure VII CD-ROM (includes Meteorological Data Used to Determine Offsite, Control Room and TSC Atmospheric Dispersion Factors). 4. A copy of program outputs are provided for PAVAN (Offsite) Output, Control Room Output, and TSC Output in Enclosure VII CD-ROM. 5. A discussion of assumptions for ARCON96 analysis is provided in Note 2 below. 6. A discussion of assumptions for PAVAN analysis is provided in Note 3 below.

Table 1 (cont.) **Conformance with NRC Regulatory Issue Summary 2006-04**

Issue Number	NRC RIS 2006-04 Position	Analysis	Comments

Note 1: Justification for using control room intake χ/Q values for modeling the unfiltered inleakage

With regards to unfiltered in-leakage, the control room is maintained at a higher pressure than the surrounding rooms in the control building. Therefore, the primary contribution of in-leakage will not be from leakage through the walls from the surrounding control building; but rather, in-leakage will be primarily due to outside air that enters through the control building intake and is able to leak past filtration equipment while in the HVAC ducts. Thus, the appropriate χ/Q value to use for this parameter is the χ/Q corresponding to the HVAC intake louvers that provide outside air to the control room.

Note 2: Assumptions for ARCON96 Analysis

1. Ground level release is assumed for all cases.
2. The source type for all cases is assumed to be a point source.
3. The shortest source-to-receptor distances are determined from the closest point on the perimeter of the source to the closest point on the perimeter of the receptor for all cases. In the case of the unit vent stack, one corner of the rectangular-shaped unit vent stack is assumed to point in the direction of the control room air intake to yield a conservatively short distance from the unit vent stack to the control room air intake. Similarly, it is also assumed that another corner of the unit vent stack points in the direction of the TSC air intake to yield a conservatively short distance from the unit vent stack to the TSC air intake.
4. Paths that traverse the perimeter of the reactor building cylinder conservatively ignore the buttresses.
5. The MSSV and ARV vents are treated as a single point source. The closest vent is chosen to represent the source for the purposes of calculating distance to the receiver location. Because the MSSV and ARV vents have different discharge elevations, this results in different release heights being used for control room and TSC calculations.
6. For the control room calculation, it is assumed that the MSSV/ARV release point exists at a distance west of the reactor building centerline equal to the most westward point that exists at the exit of the MSSV vents. The release point is further assumed to be the same distance north of the reactor building centerline as the receiver location. This yields a conservatively short distance between the MSSV/ARV release point and the control room air intake.

Table 1 (cont.)	Conformance with NRC Regulatory Issue Summary 2006-04		
Issue Number	**NRC RIS 2006-04 Position**	**Analysis**	**Comments**

Note 3: Assumptions for PAVAN Analysis

1. A release of radioactive materials during an accident from one or more potential release sources is treated as a single ground-level release source (i.e. no multiple releases at the same time) for all cases. The PAVAN code cannot handle multiple release sources.
2. The distance from the release point to the EAB is 1200 meters for all directions regardless of the release location. Likewise, the distance from the release point to the LPZ boundary is 4023 meters for all directions. This assumption is based on the fact that only these two distances were defined in the Wolf Creek USAR.
3. A building wake effect is included in the calculation.
   - For a release from potential sources such as equipment hatch, unit vent stack, MSSVs/ARVs vent, or TDAFW exhaust vent, the cross-sectional area of the containment building is used for calculation of the building wake term. These release sources are located close to the containment building. Wake effects from other smaller adjacent buildings are ignored.
   - For a release from the RWST, the cross-sectional area of the RWST is used for the building wake term. The RWST is located more than 100 feet from the closest containment building wall. The wake effects due to the containment building for certain wind sectors and other adjacent buildings are conservatively ignored.
4. A straight trajectory of plume movement from the release point to the EAB and LPZ distances is assumed as part of the PAVAN model. The straight trajectory assumption is then supplemented by the use of the terrain correction factors. The terrain correction factors account for the possibility of non-straight trajectory due to temporal and spatial variations in airflow in the EAB and LPZ areas that do not reflect in the meteorological data collected at a single onsite station that is used in this calculation.
   - In the USAR section on the long-term diffusion estimate, terrain/recirculation correction factors for the 16 wind sectors were calculated based on 1973-1974 data and listed in Table 2.3-61. The maximum value for EAB in the W sector and the maximum value for LPZ in the WSW sector were conservatively applied to all wind sectors in the PAVAN calculation.
5. Plume meander in the horizontal direction under low wind conditions is a physical phenomenon that is internally accounted for in the PAVAN calculation for wind speed less than 6 m/s and atmospheric stability conditions being neutral (class D) or stable (class E, F, or G). (This is a code model assumption.)

Table 1 (cont.)	Conformance with NRC Regulatory Issue Summary 2006-04		
Issue Number	**NRC RIS 2006-04 Position**	**Analysis**	**Comments**

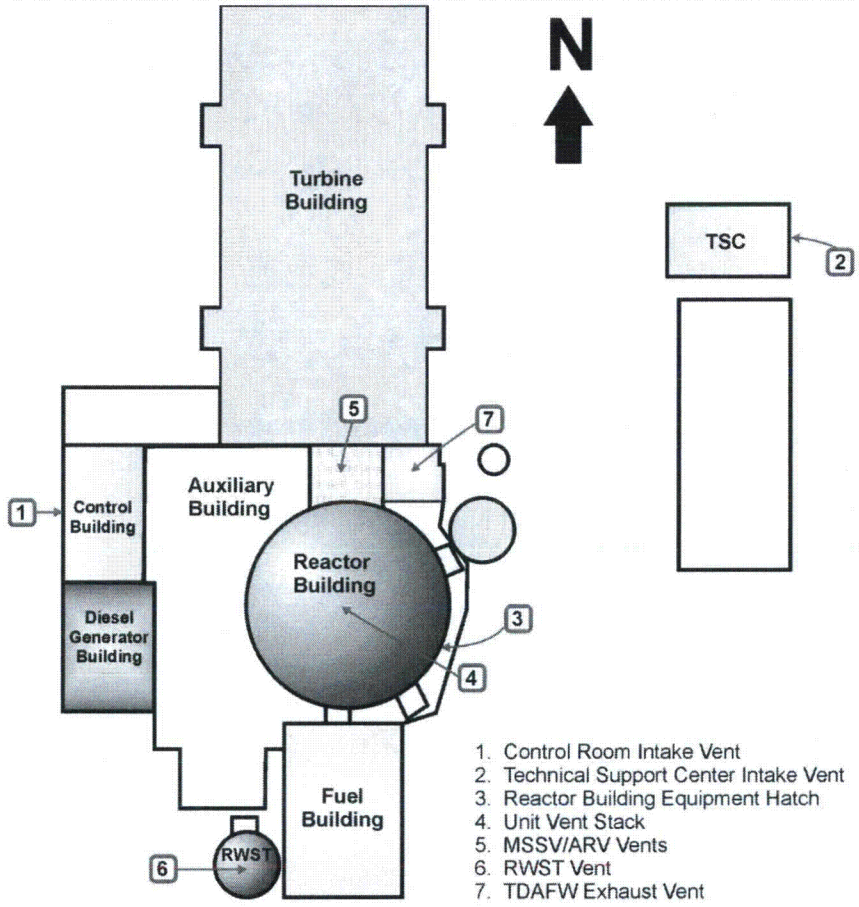


Figure 8-1 Site Plan

Figure 8-1 shows locations of potential release pathways and control room intake. The unfiltered inleakage pathway is the west wall of the control building (identified by number 1 in Figure 8-1).

Table 1 (cont.)	Conformance with NRC Regulatory Issue Summary 2006-04		
Issue Number	**NRC RIS 2006-04 Position**	**Analysis**	**Comments**
	• A copy of the meteorological data inputs and program outputs along with a discussion of assumptions and potential deviations from staff guidelines. Meteorological data input files should be checked to ensure quality (e.g., compared against historical or other data and against the raw data to ensure that the electronic file has been properly formatted, any unit conversions are correct, and invalid data are properly identified).	Conform	The electronic met data files have been properly formatted and checked for correctness. Invalid data were properly identified and marked with fields of "nines" for the entire line entries to make sure that the line would not be read as valid data.
	When running the control room atmospheric dispersion model ARCON96, two or more files of meteorological data representative of each potential release height should be used if χ/Q values are being calculated for both ground-level and elevated releases (see RG 1.23, "Onsite Meteorological Programs," Regulatory Position 2 (Ref. 8) and Table A-2 in Appendix A to RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants"). In addition, licensees should be aware that (1) two levels of wind speed and direction data should always be provided as input to each data file, (2) fields of "nines" (e.g., 9999) should be used to indicate invalid or missing data, and (3) valid wind direction data should range from 1° to 360°.	Conform	Only a ground level release is calculated for all releases to the control room and the TSC. The meteorological data taken at standard 10 m and 60 m heights from the same tower is justified to represent each release height which ranges from 17.37 m to 66.25 m. The meteorological data used is representative of the release height.
	Licensees should also provide detailed engineering information when applying the default plume rise adjustment cited in RG 1.194 to control room χ/Q values to account for buoyancy or mechanical jets of high energy releases. This information should demonstrate that the minimum effluent velocity during any time of the release over which the adjustment is being applied is greater than the 95th percentile wind speed at the height of release.	Not applicable	No elevated release is considered in this calculation. The plume rise adjustment is not applicable to this calculation.

Table 1 (cont.)	Conformance with NRC Regulatory Issue Summary 2006-04		
Issue Number	**NRC RIS 2006-04 Position**	**Analysis**	**Comments**
	When running the offsite atmospheric dispersion model PAVAN, two or more files of meteorological data representative of each potential release height should be used if χ/Q values are being calculated for pathways with significantly different release heights (e.g., ground level versus elevated stack). The joint frequency distributions of wind speed, wind direction, and atmospheric stability data used as input to PAVAN should have a large number of wind speed categories at the lower wind speeds in order to produce the best results (e.g., Section 4.6 of NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations" (Ref. 9), suggests wind speed categories of calm, 0.5, 0.75, 1.0, 1.25, 1.5, 2.0, 3.0, 4.0 5.0, 6.0, 8.0 and 10.0 meters per second).	Conform	Only a ground level release is calculated for site boundary χ/Q. No stack release is considered in this calculation. The meteorological data used is representative of the ground-level release. In the joint frequency distribution, the wind speed data are classified into 14 categories, the maximum number allowed by the PAVAN code, as follows: 0.34 (calm), 0.5, 0.75, 1.0, 1.25, 1.5, 2.0, 3.0, 4.0, 5.0, 6.0, 8.0, 10.0, and 44.7 meters/second.
5	Modeling of ESF Leakage ESF systems that recirculate sump water outside the primary containment may leak during their intended operation. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components. This release source may also include leakage through valves isolating interfacing systems (e.g., refueling water storage tank). Appendix A to RG 1.183, Regulatory Position 5, states that "the radiological consequences from the postulated [ESF] leakage should be analyzed and combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the [loss-of-coolant accident] LOCA." The allowable ESF leakage is typically contained in the plant's TS or procedures. The ESF leakage at accident conditions may differ from the ESF leakage at normal operating conditions. Licensees should account for ESF leakage at accident conditions in their dose analyses so as not to underestimate the release rate. In Appendix A to RG 1.183, Regulatory Position 5.5, the NRC staff provided a conservative value of 10 percent as the assumed amount of iodine that may become airborne from ESF leakage that is less than 212°F. The NRC staff structured this regulatory position to be deterministic and conservative. The 10 percent value also compensates for the lack of	Conforms	The LOCA dose analysis models ESF leakage in accordance with Regulatory Position 5 in Appendix A of RG 1.183. (Refer to Table B in Section 5 - RG 1.183 Conformance Table)

Table 1 (cont.)	Conformance with NRC Regulatory Issue Summary 2006-04		
Issue Number	**NRC RIS 2006-04 Position**	**Analysis**	**Comments**
	research concerning iodine speciation beyond the containment and the uncertainties of applying laboratory data to the post-accident environment of the plant. Regulatory Position 5.5 states that a smaller flash fraction could be justified. Some licensees have referenced NUREG/CR-5950, "Iodine Evolution and pH Control" (Ref. 10) to justify a smaller flash fraction. However, NUREG/ CR-5950 was developed for very specific laboratory conditions and the results have a degree of uncertainty. The mechanism for release of the fluid is also uncertain. Leaked fluid may spray onto surfaces and evaporate, or be sprayed in fine droplets into the air. A value of less than 10 percent can be justified by including considerations for plant-specific variables, including the post-accident environment (e.g., impurities in the water or the presence of organic substances) and the uncertainties in the application of research situations to plant environments. Figure 3.1 in NUREG/CR-5950 can be used to quantify the amount of elemental iodine as a function of the sump water pH and the concentration of iodine in the solution. In some cases, however, licensees have misapplied this figure. Rather than using the total concentration of iodine (i.e., stable and radioactive), licensees based their assessment on only the radioactive iodine in the sump water. By using only the radioactive iodine, licensees have underestimated how much iodine evolves during post-accident conditions.		

Table 1 (cont.)	Conformance with NRC Regulatory Issue Summary 2006-04		
Issue Number	**NRC RIS 2006-04 Position**	**Analysis**	**Comments**
6	Release Pathways Changes to the plant configuration associated with an LAR (e.g., an "open" containment during refueling) may require a re-analysis of the design basis dose calculations. A request for TS modifications allowing containment penetrations (i.e., personnel air lock, equipment hatch) to be open during refueling cannot rely on the current dose analysis if this analysis has not already considered these release pathways. RG 1.194, Regulatory Position 3.2.4.2 supports review of penetration pathways, by stating that "leakage is more likely to occur at a penetration, [and that the] analysts must consider the potential impact of leakage from building penetrations exposed to the environment." Therefore, releases from personnel air locks and equipment hatches exposed to the environment and containment purge releases prior to containment isolation need to be addressed. Some licensees have identified unique release pathways that had not been previously considered. For example, a recent submittal noted that containment hatches and containment plugs may be removed during refueling. The removal of these barriers creates new release pathways. Licensees are responsible for identifying all release pathways and for considering these pathways in their AST analyses, consistent with any proposed modification.	Not Applicable	There were no changes in plant configuration which resulted in additional release pathways modeled in the dose analyses.
7	Primary to Secondary Leakage Some analysis parameters can be affected by density changes that occur in the process steam. The NRC staff continues to find errors in LAR submittals concerning the modeling of primary to secondary leakage during a postulated accident. This issue is discussed in Information Notice (IN) 88-31, "Steam Generator Tube Rupture Analysis Deficiency," (Ref. 11) and Item 3.f in RIS 2001-19. An acceptable methodology for modeling this leakage is provided in Appendix F to RG 1.183, Regulatory Position 5.2.	Conforms	The density of 1.0 gm/cc (62.4 lbm/ft^3) was used in the dose analyses, consistent with Appendix F to RG 1.183, Regulatory Position 5.2.

Table 1 (cont.)	Conformance with NRC Regulatory Issue Summary 2006-04		
Issue Number	**NRC RIS 2006-04 Position**	**Analysis**	**Comments**
8	Elemental Iodine Decontamination Factor (DF) Appendix B to RG 1.183, provides assumptions for evaluating the radiological consequences of a fuel handling accident. If the water depth above the damaged fuel is 23 feet or greater, Regulatory Position 2 states that "the decontamination factors for the elemental and organic [iodine] species are 500 and 1, respectively, giving an overall effective decontamination factor of 200." However, an overall DF of 200 is achieved when the DF for elemental iodine is 285, not 500.	Conforms	An overall DF of 200 for iodine was used in the fuel handling accident analysis.
9	Isotopes Used in Dose Assessments For some accidents (e.g., main steamline break and rod drop), licensees have excluded noble gas and cesium isotopes from the dose assessment. The inclusion of these isotopes should be addressed in the dose assessments for AST implementation.	Conforms	The dose consequences from noble gas and alkali metals (mainly cesium) were included in the dose analyses.
10	Definition of Dose Equivalent ^{131}I In the conversion to an AST, licensees have proposed a modification to the TS definition of dose equivalent 131I. Some have modified the definition to base it upon the thyroid dose conversion factors of International Commission on Radiation Protection (ICRP) Publication 2, "Report of Committee II on Permissible Dose for Internal Radiation" (Ref. 12) or ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers" (Ref. 13). Others have proposed a definition which is a combination of different iodine dose conversion factors, (e.g., RG 1.109, Revision 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR [Part] 50, Appendix I" (Ref. 14), ICRP Publication 2, Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (Ref. 15). Although different references are available for dose conversion factors, the TS definition should be based on the same dose conversion factors that are used in the determination of the reactor coolant dose equivalent iodine curie content for the main steamline break and steam generator tube rupture accident analyses.	Conforms	The Technical Specification definition is being revised to use only the thyroid dose conversion factors from Federal Guidance Report 11, which is consistent with the dose conversion factors used to define dose equivalent ^{131}I in the main steamline break and steam generator tube rupture analyses.

Table 1 (cont.)	Conformance with NRC Regulatory Issue Summary 2006-04		
Issue Number	**NRC RIS 2006-04 Position**	**Analysis**	**Comments**
11	Acceptance Criteria for Off-Gas or Waste Gas System Release As part of full AST implementation, some licensees have included an accident involving a release from their off-gas or waste gas system. For this accident, they have proposed acceptance criteria of 500 millirem (mrem) total effective dose equivalent (TEDE). The acceptance criteria for this event is that associated with the dose to an individual member of the public as described in 10 CFR Part 20, "Standards for Protection Against Radiation." When the NRC revised 10 CFR Part 20 to incorporate a TEDE dose, the offsite dose to an individual member of the public was changed from 500 mrem whole body to 100 mrem TEDE. Therefore, any licensee who chooses to implement AST for an off-gas or waste gas system release should base its acceptance criteria on 100 mrem TEDE. Licensees may also choose not to implement AST for this accident and continue with their existing analysis and acceptance criteria of 500 mrem whole body.	Conforms	AST is being implemented for the waste gas system release and an acceptance criteria of 100 mrem TEDE was used.

Table 1 (cont.)	Conformance with NRC Regulatory Issue Summary 2006-04		
Issue Number	**NRC RIS 2006-04 Position**	**Analysis**	**Comments**
12	Containment Spray Mixing Some plants with mechanical means for mixing containment air have assumed that the containment fans intake air solely from a sprayed area and discharge it solely to an unsprayed region or vice versa. Without additional analysis, test measurements or further justification, it should be assumed that the intake of air by containment ventilation systems is supplied proportionally to the sprayed and unsprayed volumes in containment.	Conforms	The containment building is modeled as two discrete volumes: sprayed and unsprayed. The volumes are conservatively assumed to be mixed only by the containment fan coolers. The containment cooling system provides cooling by recirculation of the containment air across air-to-water heat exchangers. The bulk of this cooled air is supplied to the lower regions of the steam generator compartments. The remaining air is supplied to the instrument tunnel and at each level (operating floor and below) of the containment outside the secondary shield wall. The air supplied to each steam generator compartment is drawn upwards through the compartments by the hydrogen mixing fans and discharged into the upper elevations of the containment. The volume of air recirculated in one hour by the combined air flows of one train of the containment coolers is approximately three times the containment free volume. These air flow patterns and recirculation volumes provide adequate circulation and, therefore, sufficient post accident mixing of the containment atmosphere.

9 PROPOSED TECHNICAL SPECIFICATION MARKUPS

1.1 Definitions (continued)

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 per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using thyroid dose conversion factors from~~:~~ ~~1) Table III of TID-14844, AEC, 1962, Calculation of Distance Factors for Power and Test Reactor Sites," or~~ ~~2) Table E-7 of Regulatory Guide 1.109, Revision 1, NRC, 1977, or~~

(continued)

1.1 Definitions (continued)

DOSE EQUIVALENT I-131 (continued)

~~3) ICRP 30, 1979, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity," or~~

~~4)~~ Table 2.1 of EPA Federal Guidance Report No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

DOSE EQUIVALENT XE-133

DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-87, Kr-88, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using the effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil." ~~or using the dose conversion factors from Table B-1 of Regulatory Guide 1.109, Revision 1, NRC, 1977~~.

ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME

The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal

(continued)

recently

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. ----------NOTE---------- Only applicable during ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment. ----------	B.1 Place and maintain containment purge supply and exhaust valves in closed position.	Immediately
	OR	
One or more Functions with one or more channels or trains inoperable.	B.2 Enter applicable Conditions and Required Actions of LCO 3.9.4, "Containment Penetrations," for containment purge supply and exhaust valves made inoperable by isolation instrumentation.	Immediately

SURVEILLANCE REQUIREMENTS

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

SURVEILLANCE		FREQUENCY
SR 3.3.6.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.6.2	----------NOTE---------- The continuity check may be excluded. ----------	
	Perform ACTUATION LOGIC TEST.	31 days on a STAGGERED TEST BASIS

(continued)

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

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1.	Manual Initiation	1,2,3,4, (a)~~,(b)~~	2	SR 3.3.6.4	NA
2.	Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	1,2,3,4, (a)~~,(b)~~	2 trains	SR 3.3.6.2 SR 3.3.6.6	NA
3.	Containment Atmosphere - Gaseous Radioactivity	1,2,3,4, (a)~~,(b)~~	1	SR 3.3.6.1 SR 3.3.6.3 SR 3.3.6.5	~~(c)~~ (b)
4.	Containment Isolation - Phase A	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 3.a, for all initiation functions and requirements.			

(a) ~~(a)~~ ~~During CORE ALTERATIONS.~~ recently
~~(b)~~ During movement of irradiated fuel assemblies within containment.
~~(c)~~ Trip setpoint concentration value ($\mu Ci/cm^3$) is to be established such that the actual submersion rate would not exceed 9 mR/h in the containment building.
(b)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time for Condition A, B or C not met in MODE 1, 2, 3, or 4.	D .1 Be in MODE 3. AND D .2 Be in MODE 5.	6 hours 36 hours
E. Required Action and associated Completion Time for Condition A, B or C not met during movement of recently irradiated fuel assemblies.	~~E.1~~ ~~Suspend CORE ALTERATIONS.~~ ~~AND~~ 1 E .~~2~~ Suspend movement of recently irradiated fuel assemblies.	~~Immediately~~ Immediately

SURVEILLANCE REQUIREMENTS

----------NOTE----------
Refer to Table 3.3.7-1 to determine which SRs apply for each CREVS Actuation Function.

SURVEILLANCE		FREQUENCY
SR 3.3.7.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.7.2	Perform COT.	92 days

(continued)

Table 3.3.7-1 (page 1 of 1)
CREVS Actuation Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1.	Manual Initiation	1, 2, 3, 4, and (a)	2	SR 3.3.7.4	NA
2.	Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	1, 2, 3, 4, and (a)	2 trains	SR 3.3.7.3	NA
3.	Control Room Radiation-Control Room Air Intakes	1, 2, 3, 4, and (a)	2	SR 3.3.7.1 SR 3.3.7.2 SR 3.3.7.5	(b)
4.	Containment Isolation - Phase A	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 3.a, for all initiation functions and requirements.			

recently

(a) During movement of irradiated fuel assemblies.
(b) Trip Setpoint concentration value ($\mu Ci/cm^3$) is to be established such that the actual submersion dose rate would not exceed 2 mR/hr in the control room.

recently

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time for Condition A, B or C not met during movement of irradiated fuel assemblies in the fuel building.	D.1 Suspend movement of irradiated fuel assemblies in the fuel building.	Immediately

SURVEILLANCE REQUIREMENTS

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

SURVEILLANCE		FREQUENCY
SR 3.3.8.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.8.2	Perform COT.	92 days
SR 3.3.8.3	----------NOTE---------- The continuity check may be excluded. --- Perform ACTUATION LOGIC TEST.	31 days on a STAGGERED TEST BASIS

(continued)

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

	FUNCTION	APPLICABLE MODES OR SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1.	Manual Initiation	(a)	2	SR 3.3.8.4	NA
2.	Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	(a)	2 trains	SR 3.3.8.3	NA
3.	Fuel Building Exhaust Radiation - Gaseous	(a)	2	SR 3.3.8.1 SR 3.3.8.2 SR 3.3.8.5	(b)

recently

(a) During movement of irradiated fuel assemblies in the fuel building.
(b) Trip Setpoint concentration value ($\mu Ci/cm^3$) is to be established such that the actual submersion dose rate would not exceed 4 mR/hr in the fuel building.

3.7 PLANT SYSTEMS

3.7.10 Control Room Emergency Ventilation System (CREVS)

LCO 3.7.10 Two CREVS trains shall be OPERABLE.

----------NOTE----------
The control room envelope (CRE) and control building envelope (CBE) boundaries may be opened intermittently under administrative controls.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One CREVS train inoperable for reasons other than Condition B.	A.1	Restore CREVS train to OPERABLE status.	7 days
B. One or more CREVS trains inoperable due to an inoperable CRE boundary or an inoperable CBE boundary in MODES 1, 2, 3, or 4.	B.1	Initiate action to implement mitigating actions.	Immediately
	AND		
	B.2	Verify mitigating actions to ensure CRE occupant radiological exposures will not exceed limits and CRE occupants are protected from chemical and smoke hazards.	24 hours
	AND		
	B.3	Restore CRE boundary and CBE boundary to OPERABLE status.	90 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours
D. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies.	D.1 Place OPERABLE CREVS train in CRVIS mode. OR ~~D.2.1~~ ~~Suspend CORE ALTERATIONS.~~ ~~AND~~ D.2~~.2~~ Suspend movement of irradiated fuel assemblies.	Immediately ~~Immediately~~ Immediately
E. Two CREVS trains inoperable during movement of irradiated fuel assemblies. OR One or more CREVS trains inoperable due to an inoperable CRE boundary or an inoperable CBE boundary during movement of irradiated fuel assemblies.	~~E.1~~ ~~Suspend CORE ALTERATIONS.~~ ~~AND~~ E.~~2~~ Suspend movement of irradiated fuel assemblies.	~~Immediately~~ Immediately

recently

1

recently

(continued)

3.7 PLANT SYSTEMS

3.7.11 Control Room Air Conditioning System (CRACS)

LCO 3.7.11 Two CRACS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4, 5, and 6,
During movement of recently irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CRACS train inoperable.	A.1 Restore CRACS train to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met in MODE 5 or 6, or during movement of irradiated fuel assemblies .	C.1 Place OPERABLE CRACS train in operation.	Immediately
	OR	
	~~C.2.1~~ ~~Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
recently	C.2~~.2~~ Suspend movement of irradiated fuel assemblies.	Immediately
D. Two CRACS trains inoperable in MODE 5 or 6, or during movement of irradiated fuel assemblies.	~~D.1~~ ~~Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~ 1	
recently	D.~~2~~ Suspend movement of irradiated fuel assemblies.	Immediately
E. Two CRACS trains inoperable in MODE 1, 2, 3, or 4.	E.1 Enter LCO 3.0.3.	Immediately

3.7 PLANT SYSTEMS

3.7.13 Emergency Exhaust System (EES)

LCO 3.7.13 Two EES trains shall be OPERABLE.

----------NOTE----------
The auxiliary building or fuel building boundary may be opened intermittently under administrative controls.

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

----------NOTE----------
The SIS mode of operation is required only in MODES 1, 2, 3, and 4. The FBVIS mode of operation is required only during movement of recently irradiated fuel assemblies in the fuel building.

ACTIONS

----------NOTE----------
LCO 3.0.3 is not applicable to the FBVIS mode of operation.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One EES train inoperable.	A.1 Restore EES train to OPERABLE status.	7 days
B. Two EES trains inoperable due to inoperable auxiliary building boundary in MODE 1, 2, 3, or 4.	B.1 Restore auxiliary building boundary to OPERABLE status.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4. OR Two EES trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B.	C.1 AND C.2	Be in MODE 3. Be in MODE 5.	6 hours 36 hours
D. Required Action and associated Completion Time of Condition A not met during movement of recently irradiated fuel assemblies in the fuel building.	D.1 OR D.2	Place OPERABLE EES train in operation in FBVIS mode. Suspend movement of recently irradiated fuel assemblies in the fuel building.	Immediately Immediately
E. Two EES trains inoperable due to inoperable fuel building boundary during movement of recently irradiated fuel assemblies in the fuel building.	E.1	Restore fuel building boundary to OPERABLE status.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and associated Completion Time of Condition E not met. OR recently Two EES trains inoperable during movement of irradiated fuel assemblies in the fuel building for reasons other than Condition E.	F.1 Suspend movement of irradiated fuel assemblies in the fuel building. recently	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Operate each EES train for ≥ 10 continuous hours with the heaters operating.	31 days
SR 3.7.13.2	Perform required EES filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.13.3	Verify each EES train actuates on an actual or simulated actuation signal.	18 months

(continued)

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, or if open, capable of being closed;

b. One door in the emergency air lock closed and one door in the personnel air lock capable of being closed; and

---NOTE---
An emergency personnel escape air lock temporary closure device is an acceptable replacement for an emergency air lock door.
---

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 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 CORE ALTERATIONS,~~
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 CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~ 1	
	A.~~2~~ Suspend movement of recently irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify each required containment penetration is in the required status.	7 days
SR 3.9.4.2	----------NOTE---------- Only required for an open equipment hatch. ---------- Verify the capability to install the equipment hatch.	7 days
SR 3.9.4.3	Verify each required containment purge isolation valve actuates to the isolation position on an actual or simulated actuation signal.	18 months

5.5 Programs and Manuals

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

The program shall include:

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

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of ≥ ~~0.5~~ 0.1 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and

c. A surveillance program to ensure that the quantity of radioactivity contained in the following outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

   a. Reactor Makeup Water Storage Tank
   b. Refueling Water Storage Tank
   c. Condensate Storage Tank, and
   d. Outside Temporary tanks, excluding demineralizer vessels and the liner being used to solidify radioactive waste.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.13 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

(continued)

10 RETYPED TECHNICAL SPECIFICATION PAGES

1.1 Definitions (continued)

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 per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using thyroid dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

(continued)

1.1 Definitions (continued)

DOSE EQUIVALENT XE-133	DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-87, Kr-88, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using the effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."
ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.
LEAKAGE	LEAKAGE shall be: a. <u>Identified LEAKAGE</u> 1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank; 2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or

(continued)

1.1 Definitions (continued)

LEAKAGE (continued)

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE);

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE;

c. Pressure Boundary LEAKAGE

LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

MASTER RELAY TEST

A MASTER RELAY TEST shall consist of energizing all master relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required master relay. The MASTER RELAY TEST shall include a continuity check of each associated required slave relay. The MASTER RELAY TEST may be performed by means of any series of sequential, overlapping, or total steps.

MODE

A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

OPERABLE--OPERABILITY

A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. ----------NOTE---------- Only applicable during movement of recently irradiated fuel assemblies within containment. ----------	B.1 Place and maintain containment purge supply and exhaust valves in closed position.	Immediately
	OR	
One or more Functions with one or more channels or trains inoperable.	B.2 Enter applicable Conditions and Required Actions of LCO 3.9.4, "Containment Penetrations," for containment purge supply and exhaust valves made inoperable by isolation instrumentation.	Immediately

SURVEILLANCE REQUIREMENTS

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

SURVEILLANCE		FREQUENCY
SR 3.3.6.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.6.2	----------NOTE---------- The continuity check may be excluded. ----------	
	Perform ACTUATION LOGIC TEST.	31 days on a STAGGERED TEST BASIS

(continued)

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

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1.	Manual Initiation	1,2,3,4, (a)	2	SR 3.3.6.4	NA
2.	Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	1,2,3,4, (a)	2 trains	SR 3.3.6.2 SR 3.3.6.6	NA
3.	Containment Atmosphere - Gaseous Radioactivity	1,2,3,4, (a)	1	SR 3.3.6.1 SR 3.3.6.3 SR 3.3.6.5	(b)
4.	Containment Isolation - Phase A	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 3.a, for all initiation functions and requirements.			

(a) During movement of recently irradiated fuel assemblies within containment.
(b) Trip setpoint concentration value ($\mu Ci/cm^3$) is to be established such that the actual submersion rate would not exceed 9 mR/h in the containment building.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time for Condition A, B or C not met in MODE 1, 2, 3, or 4.	D .1 Be in MODE 3. AND D .2 Be in MODE 5.	6 hours 36 hours
E. Required Action and associated Completion Time for Condition A, B or C not met during movement of recently irradiated fuel assemblies.	E.1 Suspend movement of recently irradiated fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

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

Refer to Table 3.3.7-1 to determine which SRs apply for each CREVS Actuation Function.

SURVEILLANCE		FREQUENCY
SR 3.3.7.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.7.2	Perform COT.	92 days

(continued)

Table 3.3.7-1 (page 1 of 1)
CREVS Actuation Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1.	Manual Initiation	1, 2, 3, 4, and (a)	2	SR 3.3.7.4	NA
2.	Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	1, 2, 3, 4, and (a)	2 trains	SR 3.3.7.3	NA
3.	Control Room Radiation-Control Room Air Intakes	1, 2, 3, 4, and (a)	2	SR 3.3.7.1 SR 3.3.7.2 SR 3.3.7.5	(b)
4.	Containment Isolation - Phase A	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 3.a, for all initiation functions and requirements.			

(a) During movement of recently irradiated fuel assemblies.
(b) Trip Setpoint concentration value ($\mu Ci/cm^3$) is to be established such that the actual submersion dose rate would not exceed 2 mR/hr in the control room.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time for Condition A, B or C not met during movement of recently irradiated fuel assemblies in the fuel building.	D.1 Suspend movement of recently irradiated fuel assemblies in the fuel building.	Immediately

SURVEILLANCE REQUIREMENTS

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

SURVEILLANCE		FREQUENCY
SR 3.3.8.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.8.2	Perform COT.	92 days
SR 3.3.8.3	----------NOTE---------- The continuity check may be excluded. --- Perform ACTUATION LOGIC TEST.	31 days on a STAGGERED TEST BASIS

(continued)

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

	FUNCTION	APPLICABLE MODES OR SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1.	Manual Initiation	(a)	2	SR 3.3.8.4	NA
2.	Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	(a)	2 trains	SR 3.3.8.3	NA
3.	Fuel Building Exhaust Radiation - Gaseous	(a)	2	SR 3.3.8.1 SR 3.3.8.2 SR 3.3.8.5	(b)

(a) During movement of recently irradiated fuel assemblies in the fuel building.
(b) Trip Setpoint concentration value ($\mu Ci/cm^3$) is to be established such that the actual submersion dose rate would not exceed 4 mR/hr in the fuel building.

3.7 PLANT SYSTEMS

3.7.10 Control Room Emergency Ventilation System (CREVS)

LCO 3.7.10 Two CREVS trains shall be OPERABLE.

----------NOTE----------
The control room envelope (CRE) and control building envelope (CBE) boundaries may be opened intermittently under administrative controls.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One CREVS train inoperable for reasons other than Condition B.	A.1	Restore CREVS train to OPERABLE status.	7 days
B. One or more CREVS trains inoperable due to an inoperable CRE boundary or an inoperable CBE boundary in MODES 1, 2, 3, or 4.	B.1	Initiate action to implement mitigating actions.	Immediately
	AND		
	B.2	Verify mitigating actions to ensure CRE occupant radiological exposures will not exceed limits and CRE occupants are protected from chemical and smoke hazards.	24 hours
	AND		
	B.3	Restore CRE boundary and CBE boundary to OPERABLE status.	90 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.	C.1	Be in MODE 3.	6 hours
	AND		
	C.2	Be in MODE 5.	36 hours
D. Required Action and associated Completion Time of Condition A not met during movement of recently irradiated fuel assemblies.	D.1	Place OPERABLE CREVS train in CRVIS mode.	Immediately
	OR		
	D.2	Suspend movement of recently irradiated fuel assemblies.	Immediately
E. Two CREVS trains inoperable during movement of recently irradiated fuel assemblies. OR One or more CREVS trains inoperable due to an inoperable CRE boundary or an inoperable CBE boundary during movement of recently irradiated fuel assemblies.	E.1	Suspend movement of recently irradiated fuel assemblies.	Immediately

(continued)

3.7 PLANT SYSTEMS

3.7.11 Control Room Air Conditioning System (CRACS)

LCO 3.7.11 Two CRACS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4, 5, and 6,
During movement of recently irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One CRACS train inoperable.	A.1	Restore CRACS train to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 5.	36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met in MODE 5 or 6, or during movement of recently irradiated fuel assemblies.	C.1	Place OPERABLE CRACS train in operation.	Immediately
	OR		
	C.2	Suspend movement of recently irradiated fuel assemblies.	Immediately
D. Two CRACS trains inoperable in MODE 5 or 6, or during movement of recently irradiated fuel assemblies.	D.1	Suspend movement of recently irradiated fuel assemblies.	Immediately
E. Two CRACS trains inoperable in MODE 1, 2, 3, or 4.	E.1	Enter LCO 3.0.3.	Immediately

3.7 PLANT SYSTEMS

3.7.13 Emergency Exhaust System (EES)

LCO 3.7.13 Two EES trains shall be OPERABLE.

----------NOTE----------
The auxiliary building or fuel building boundary may be opened intermittently under administrative controls.

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

----------NOTE----------
The SIS mode of operation is required only in MODES 1, 2, 3, and 4. The FBVIS mode of operation is required only during movement of recently irradiated fuel assemblies in the fuel building.

ACTIONS

----------NOTE----------
LCO 3.0.3 is not applicable to the FBVIS mode of operation.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One EES train inoperable.	A.1 Restore EES train to OPERABLE status.	7 days
B. Two EES trains inoperable due to inoperable auxiliary building boundary in MODE 1, 2, 3, or 4.	B.1 Restore auxiliary building boundary to OPERABLE status.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4. OR Two EES trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B.	C.1 AND C.2	Be in MODE 3. Be in MODE 5.	6 hours 36 hours
D. Required Action and associated Completion Time of Condition A not met during movement of recently irradiated fuel assemblies in the fuel building.	D.1 OR D.2	Place OPERABLE EES train in operation in FBVIS mode. Suspend movement of recently irradiated fuel assemblies in the fuel building.	Immediately Immediately
E. Two EES trains inoperable due to inoperable fuel building boundary during movement of recently irradiated fuel assemblies in the fuel building.	E.1	Restore fuel building boundary to OPERABLE status.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and associated Completion Time of Condition E not met. OR Two EES trains inoperable during movement of recently irradiated fuel assemblies in the fuel building for reasons other than Condition E.	F.1 Suspend movement of recently irradiated fuel assemblies in the fuel building.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Operate each EES train for ≥ 10 continuous hours with the heaters operating.	31 days
SR 3.7.13.2	Perform required EES filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.13.3	Verify each EES train actuates on an actual or simulated actuation signal.	18 months

(continued)

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, or if open, capable of being closed;

b. One door in the emergency air lock closed and one door in the personnel air lock capable of being closed; and

----------NOTE----------
An emergency personnel escape air lock temporary closure device is an acceptable replacement for an emergency air lock door.
---

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 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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify each required containment penetration is in the required status.	7 days
SR 3.9.4.2	----------NOTE---------- Only required for an open equipment hatch. ---------- Verify the capability to install the equipment hatch.	7 days
SR 3.9.4.3	Verify each required containment purge isolation valve actuates to the isolation position on an actual or simulated actuation signal.	18 months

5.5 Programs and Manuals

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

The program shall include:

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

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of ≥ 0.1 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and

c. A surveillance program to ensure that the quantity of radioactivity contained in the following outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

   a. Reactor Makeup Water Storage Tank
   b. Refueling Water Storage Tank
   c. Condensate Storage Tank, and
   d. Outside Temporary tanks, excluding demineralizer vessels and the liner being used to solidify radioactive waste.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.13 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

(continued)

11 PROPOSED BASES MARKUPS (For Information Only)

B 2.0 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, the continued integrity of the RCS is ensured. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor pressure coolant 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.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and

BASES

SAFETY LIMIT VIOLATIONS

The following SL violations are applicable to the RCS pressure SL.

2.2.2.1

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,"~~ limits (Ref. 4). 50.67

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.

2.2.2.2

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. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.

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

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) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish 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 50.67 ~~100, "Reactor Site Criteria,"~~ limits (Ref. 4). For the boron dilution accident, if the LCO is violated, the minimum required time assumed for operator action to terminate dilution may no longer be sufficient. The required SDM limit is specified in the COLR.

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, SDM is ensured by complying with LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits."

The risk assessments of LCO 3.0.4b. may only be utilized for systems and components, not Criterion 2 values or parameters such as SDM. Therefore, a risk assessment per LCO 3.0.4b. to allow MODE changes with single or multiple system/equipment inoperabilities may not be used to allow a MODE change into or within this LCO while not meeting the SDM limits, even if risk assessment specifically includes consideration of SDM.

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 possible, the borated water source should be a highly concentrated solution, such as that normally found in the boric acid storage tank, or the refueling water storage tank. The operator should borate with the best source available for the plant conditions.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

In MODES 1 and 2, SDM is verified by observing that the requirements of LCO 3.1.5 and LCO 3.1.6 are met. In the event that a rod is known to be untrippable, however, SDM verification must account for the worth of the untrippable rod as well as another rod of maximum worth.

In MODES 2 (with $k_{eff} < 1.0$), 3, 4, and 5, the SDM is verified by performing a reactivity balance calculation, considering the listed reactivity effects:

a. RCS boron concentration;

b. Control and shutdown rod position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; and

g. Isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation; when the reactor is subcritical, 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. 10 CFR 50, Appendix A, GDC 26.

2. USAR, Section 15.1.5.

3. USAR, Section 15.4.6.

4. 10 CFR ~~100~~. 50.67

B 3.3 INSTRUMENTATION

B 3.3.1 Reactor Trip System (RTS) Instrumentation

BASES

BACKGROUND

The RTS initiates a unit shutdown, based on the values of selected unit parameters, to protect against violating the core fuel design limits and Reactor Coolant System (RCS) pressure boundary during Anticipated Operational Occurrences (AOOs) and to assist the Engineered Safety Features (ESF) Systems in mitigating accidents.

The protection and monitoring systems have been designed to assure safe operation of the reactor. This is achieved by specifying Limiting Safety System Settings (LSSS) in terms of parameters directly monitored by the RTS, as well as specifying LCOs on other reactor system parameters and equipment performance.

The LSSS, defined in this specification as the Allowable Values, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits during Design Basis Accidents (DBAs).

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 (DNBR) limit;

2. Fuel centerline melt shall not occur; and

3. The RCS pressure Safety Limit of 2735 psig shall not be exceeded.

Operation within the limits of Specification 2.0, "Safety Limits (SLs)," also maintains the above values and assures that offsite dose will be within the 10 CFR 20 and 10 CFR ~~100~~ 50.67 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~~ limits. Different accident categories are allowed a different fraction of these limits, based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

e. Safety Injection - Steam Line Pressure – Low (continued)

environment instrument uncertainties. The Trip Setpoint is ≥ 615 psig.

This Function is anticipatory in nature and has a lead/lag ratio of 50/5.

Steam Line Pressure - Low must be OPERABLE in MODES 1, 2, and 3 (above P-11 and below P-11 unless the Safe Injection - Steam Line Pressure - Low Function is blocked) when a secondary side break or stuck open valve could result in the rapid depressurization of the steam lines. This signal may be manually blocked by the operator below the P-11 setpoint. Below P-11, feed line break is not a concern. Inside containment SLB will be terminated by automatic SI actuation via Containment Pressure - High 1, and outside containment SLB will be terminated by the Steam Line Pressure - Negative Rate - High signal for steam line isolation. This Function is not required to be OPERABLE in MODE 4, 5, or 6 because there is insufficient energy in the secondary side of the unit to have a significant effect on required plant equipment.

2. Containment Spray

Containment Spray provides three primary functions:

1. Lowers containment pressure and temperature after an HELB in containment;

2. Reduces the amount of radioactive iodine and particulates in the containment atmosphere; and

3. Adjusts the pH of the water in the containment recirculation sumps after a large break LOCA.

These functions are necessary to:

- Ensure the pressure boundary integrity of the containment structure;
- Limit the release of radioactive iodine and particulates to the environment; and

BASES

LCO

12. Steam Generator Water Level (Wide Range) (continued)

used to determine whether adequate core cooling is provided in the absence of wide range level indication for a steam generator. The design limitation of having one wide range level indicator in conjunction with one AFW flow indicator per steam generator is consistent with NUREG-0737, Item II.E.1.2 (Reference 8). Wide range steam generator level is not a Type A variable.

SG Water Level (Wide Range) is used to:

- verify that the intact SGs are an adequate heat sink for the reactor;
- determine the nature of the accident in progress (e.g., verify SGTR ~~overfill~~); and
- verify unit conditions for termination of SI during secondary unit HELBs outside containment.

13. Steam Generator Water Level (Narrow Range)

Steam Generator Water Level (Narrow Range) is a Type A, Category 1 variable for Steam Generator Tube Rupture event diagnosis and SI termination.

SG Water Level (Narrow Range) is used to:

- identify the affected SG following a tube rupture;
- determine the nature of the accident in progress (e.g., verify an SGTR); and
- verify unit conditions for termination of SI during secondary unit HELBs outside containment.

14, 15, 16, 17. Core Exit Temperature

Core exit temperature is a Category 1 variable which provides for verification and long term surveillance of core cooling.

An evaluation was made in support of Reference 2 of the minimum number of valid core exit thermocouples (CET) necessary for measuring core cooling. The evaluation determined the reduced complement of CETs necessary to detect initial core recovery and

B 3.3 INSTRUMENTATION

B 3.3.6 Containment Purge Isolation Instrumentation

BASES

BACKGROUND

Containment purge isolation instrumentation closes the containment isolation valves in the Mini Purge System and the Shutdown Purge System. This action isolates the containment atmosphere from the environment to minimize releases of radioactivity in the event of an accident. The Mini Purge System may be in use during reactor operation and the Shutdown Purge System will be in use with the reactor shutdown.

Containment Purge Isolation initiates on an automatic or manual safety injection (SI) signal through the Containment Isolation - Phase A Function, or by manual actuation of Phase A Isolation. The Bases for LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," discuss these modes of initiation.

Four safety related radiation monitoring channels are also provided as input to the containment purge isolation. The requirement for one Containment Atmosphere Gaseous Radioactivity monitor (GT RE-31 or 32) is addressed by Function 3 of Table 3.3.6-1. The requirement for one Containment Purge Radiation monitor (GT RE-22 or 33) is addressed in the Offsite Dose Calculation Manual (ODCM). Since the containment atmosphere monitors constitute a sampling system, various components such as sample line valves and sample pumps are required to support monitor OPERABILITY. Containment Purge Isolation can also be initiated by manual push buttons in the control room.

Each of the purge systems has inner and outer containment isolation valves in its supply and exhaust ducts. A high radiation signal from any one of the four radiation monitoring channels initiates containment purge isolation, which closes both inner and outer Containment Isolation Valves in the Mini Purge System and the Shutdown Purge System. These systems are described in the Bases for LCO 3.6.3, "Containment Isolation Valves."

APPLICABLE SAFETY ANALYSES

The safety analyses assume that the containment remains intact with penetrations unnecessary for core cooling isolated early in the event. The isolation of the purge valves has not been analyzed mechanistically in the dose calculations, other than for the ~~fuel handling accident inside containment~~ loss of coolant accident (isolation within ~~25 seconds of activity reaching the purge~~ 10 seconds of accident initiation).

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~exhaust ductwork)~~. The containment atmosphere radiation monitors act as backup to the Phase A Isolation signal to ensure closing of the containment purge valves. They are also the primary means for automatically isolating containment in the event of a fuel handling accident during shutdown. Containment isolation in turn ensures meeting the containment leakage rate assumptions of the safety analyses, and ensures that the calculated accidental offsite radiological doses are below 10 CFR ~~100~~ (Ref. 1) limits.

50.67

The containment purge isolation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

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

The fuel handling accident analysis does not credit containment isolation for dose mitigation after 76 hours of decay time. The fuel handling accident analysis does not address fuel movement prior to 76 hours.

1. Manual Initiation

The LCO requires two channels OPERABLE. The operator can initiate Containment Purge Isolation at any time by using either of two push buttons in the control room. Either push button actuates both trains. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

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

Each channel consists of one push button and the interconnecting wiring to the actuation logic cabinet.

2. Automatic Actuation Logic and Actuation Relays (BOP ESFAS)

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

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for BOP ESFAS in the Bases Background for 3.3.2.

BASES

LCO (continued)

3. Containment Atmosphere - Gaseous Radioactivity

The LCO specifies one required channel of radiation monitors (Containment Atmosphere Gaseous Radioactivity monitors, GT RE-31 or 32) to ensure that the radiation monitoring instrumentation necessary to initiate Containment Purge Isolation remains OPERABLE.

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

4. Containment Isolation - Phase A

Containment Purge Isolation is also initiated by all Table 3.3.2-1 Functions that initiate Phase A. Therefore, the requirements are not repeated in Table 3.3.6-1. Instead, refer to LCO 3.3.2, Function 3.a, for all initiating Functions and requirements.

APPLICABILITY

The Manual Initiation, Automatic Actuation Logic and Actuation Relays (BOP ESFAS), Containment Isolation - Phase A, and Containment Atmosphere - Gaseous Radioactivity Functions are required OPERABLE in MODES 1, 2, 3, and 4. The Containment Atmosphere - Gaseous Radioactivity, Manual Initiation, and BOP ESFAS Logic Function are also required OPERABLE, during ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment. Under these conditions, the potential exists for an accident that could release fission product radioactivity into containment. Therefore, the containment purge isolation instrumentation must be OPERABLE in these MODES.

recently

(i.e., fuel that has occupied part of a critical reactor core within the previous 76 hours)

significant

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

ACTIONS

The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by unit specific calibration procedures. Typically, the drift is found

BASES

ACTIONS

B.1 and B.2 (continued)

A Note states that Condition B is applicable ~~during CORE ALTERATIONS and~~ during movement of recently irradiated fuel assemblies within containment.

SURVEILLANCE REQUIREMENTS

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

SR 3.3.6.1

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

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

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

SR 3.3.6.2

SR 3.3.6.2 is the performance of an ACTUATION LOGIC TEST using the BOP ESFAS automatic tester. The continuity check does not have to be performed, as explained in the Note. This SR is applied to the balance of plant actuation logic and relays that do not have circuits installed to perform the continuity check. This test is required every 31 days on a

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.5

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

SR 3.3.6.6

SR 3.3.6.6 is the performance of the required response time verification every 18 months on a STAGGERED TEST BASIS. Response time verification acceptance criteria for the containment purge isolation instrumentation is ≤ 2 seconds. This response time acceptance criteria does not include valve closure time. Each verification shall include at least one train such that both trains are verified at least once per 36 months.

REFERENCES

1. 10 CFR ~~100.11~~. 50.67

2. NUREG-1366, July 22, 1993.

B 3.3 INSTRUMENTATION

B 3.3.7 Control Room Emergency Ventilation System (CREVS) Actuation Instrumentation

BASES

BACKGROUND

The CREVS provides an enclosed control room environment from which the unit can be operated following an uncontrolled release of radioactivity. During normal operation, the Control Building Ventilation System provides control room ventilation. Upon receipt of an actuation signal, the CREVS initiates filtered ventilation and pressurization of the control room. This system is described in the Bases for LCO 3.7.10, "Control Room Emergency Ventilation System (CREVS)."

The actuation instrumentation consists of two radiation monitors in the control room air intake and four radiation monitors in the containment purge isolation system (refer to B 3.3.6 "Containment Purge Isolation Instrumentation" Background). A high radiation signal from any of these gaseous detectors will initiate both trains of the CREVS. The Containment Purge Isolation Instrumentation, however, is not listed as a Function for CREVS because appropriate actions are taken in LCO 3.3.6. The control room operator can also initiate CREVS trains by manual push buttons in the control room. The CREVS is also actuated by a Phase A Isolation signal and a Fuel Building Ventilation Isolation signal. The Phase A Isolation Function is discussed in LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation." The Fuel Building Ventilation Isolation is not listed as a Function for CREVS because appropriate actions are taken in LCO 3.3.8, "EES Actuation Instrumentation."

APPLICABLE SAFETY ANALYSES

The control room must be kept habitable for the operators stationed there during accident recovery and post accident operations.

The CREVS acts to terminate the supply of unfiltered outside air to the control room, initiate filtration, and pressurize the control room. These actions are necessary to ensure the control room is kept habitable for the operators stationed there during accident recovery and post accident operations by minimizing the radiation exposure of control room personnel.

In MODES 1, 2, 3 and 4, the radiation monitor actuation of the CREVS is a backup for the Phase A Isolation signal actuation. This ensures initiation of the CREVS during a loss of coolant accident or steam generator tube rupture.

, main steamline break, rod ejection with releases into containment,

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~During movement of irradiated fuel assemblies, the radiation monitor actuation of the CREVS is the primary means to ensure control room habitability in the event of a fuel handling accident.~~ No control room habitability mitigation is required for the waste gas decay tank rupture accident.

locked RCP rotor, loss of AC power, letdown line break, and rod ejection with no release into containment (Ref. 2). In addition, the fuel handling accident analysis also does not require control room habitability mitigation after a decay time of 76 hours (Ref. 2). The fuel handling accident analysis does not address fuel movement prior to 76 hours.

The CREVS actuation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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

1. Manual Initiation

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

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

Each channel consists of one push button and the interconnecting wiring to the actuation logic cabinet.

2. Automatic Actuation Logic and Actuation Relays (BOP ESFAS)

The LCO requires two trains of Actuation Logic and Relays OPERABLE to ensure that no single random failure can prevent automatic actuation of a control room ventilation isolation signal (CRVIS).

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for BOP ESFAS in the Bases Background for 3.3.2.

BASES

LCO (continued)

3. Control Room Radiation

The LCO specifies two required Control Room Air Gaseous Intake Radiation Monitors (GK RE-04 and -05) to ensure that the radiation monitoring instrumentation necessary to initiate a CRIVS remains OPERABLE.

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

4. Containment Isolation - Phase A

Control Room Ventilation Isolation is also initiated by all Table 3.3.2-1 Functions that initiate Phase A. Therefore, the requirements are not repeated in Table 3.3.7-1. Instead, refer to LCO 3.3.2, Function 3.a, for all initiating Functions and requirements.

APPLICABILITY

All CREVS Functions must be OPERABLE in MODES 1, 2, 3, and 4. The Manual Initiation, Automatic Actuation Logic and Actuation Relay (BOP ESFAS), and Control Room Radiation Functions are also required OPERABLE during movement of irradiated fuel assemblies.

ACTIONS

The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by the unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a COT, when the process instrumentation is set up for adjustment to bring it within specification. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

BASES

ACTIONS (continued)

E.1 ~~and E.2~~

recently

Condition E applies when the Required Action and associated Completion Time for Conditions A, B or C have not been met when irradiated fuel assemblies are being moved. Movement of irradiated fuel assemblies ~~and CORE ALTERATIONS~~ must be suspended immediately to reduce the risk of accidents that would require CREVS actuation. This does not preclude movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.7-1 determines which SRs apply to which CREVS Actuation Functions.

SR 3.3.7.1

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

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

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

SR 3.3.7.2

A COT is performed once every 92 days on each required channel to ensure the entire channel will perform the intended function. This test

BASES

REFERENCES 1. USAR Section 7.3.4 and Table 7.3-8.

2. USAR, Chapter 15.

B 3.3 INSTRUMENTATION

B 3.3.8 Emergency Exhaust System (EES) Actuation Instrumentation

BASES

involving handling recently irradiated fuel

BACKGROUND

The EES ensures that radioactive materials in the fuel building atmosphere following a fuel handling accident are filtered and absorbed prior to exhausting to the environment. The EES is described in the Bases for LCO 3.7.13, "Emergency Exhaust System." The system initiates filtered exhaust from the fuel building following receipt of a fuel building ventilation isolation signal (FBVIS), initiated manually or automatically upon a high radiation signal (gaseous). Initiation may also be performed manually as needed from the main control room and the fuel building.

High gaseous radiation, monitored by two monitors (GGRE-27 and 28), provides a FBVIS. Both EES trains are initiated by high radiation detected by either channel. Each channel contains a gaseous monitor. High radiation detected by either monitor initiates fuel building isolation and starts the EES in the FBVIS mode of operation. These actions function to prevent exfiltration of contaminated air by initiating filtered exhaust, which imposes a negative pressure on the fuel building. Since the radiation monitors include an air sampling system, various components such as sample line valves, sample line heaters, and sample pumps are required to support monitor OPERABILITY.

APPLICABLE SAFETY ANALYSES

The EES ensures that radioactive materials in the fuel building atmosphere following a fuel handling accident are filtered and absorbed prior to being exhausted to the environment. This action reduces the radioactive content in the fuel building exhaust following a fuel handling accident so that offsite doses remain well within the limits specified in 10 CFR ~~100~~ 50.67 (Ref. 1).

The EES actuation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

However, the fuel handling accident analysis does not credit the EES as the dose limits are met without EES actuation after 76 hours decay time (Ref. 4). The fuel handling accident analysis does not address fuel movement prior to 76 hours.

LCO

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

BASES

recently

involving handling recently irradiated fuel.

APPLICABILITY

The manual and automatic EES initiation must be OPERABLE when moving irradiated fuel assemblies in the fuel building, to ensure the EES operates to remove fission products associated with a fuel handling accident.

Due to radioactive decay, the EES Actuation Instrumentation 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 76 hours).

recently

the limiting

High radiation initiation of the FBVIS must be OPERABLE during movement of irradiated fuel assemblies in the fuel building to ensure automatic initiation of the EES when the potential for a fuel handling accident exists.

involving handling recently irradiated fuel

While in any MODE without fuel handling in progress, the EES instrumentation need not be OPERABLE since a fuel handling accident cannot occur.

ACTIONS

The most common cause of channel inoperability is the failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a COT, when the process instrumentation is set up for adjustment to bring it within specification. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

recently

recently

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.

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

Placing a EES train(s) in the FBVIS mode of operation isolates normal air discharge from the fuel building and initiates filtered exhaust, imposing a negative pressure on the fuel building. Further discussion of the FBVIS mode of operation may be found in the Bases for LCO 3.7.13, "Emergency Exhaust System," and in Reference 3.

BASES

ACTIONS

C.1.1, C.1.2, and C.2 (continued)

Alternatively, both trains may be place in the FBVIS mode within 1 hour. This ensures the EES function is performed even in the presence of a single failure.

D.1

recently

Condition D applies when the Required Action and associated Completion Time for Conditions A, B, or C have not been met and irradiated fuel assemblies are being moved in the fuel building. Movement of irradiated fuel assemblies in the fuel building must be suspended immediately to eliminate the potential for events that could require EES actuation. This does not preclude movement of a fuel assembly to a safe position.

SURVEILLANCE REQUIREMENTS

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

SR 3.3.8.1

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

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

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.8.5

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measure parameter within the necessary range and accuracy. The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

REFERENCES

1. 10 CFR ~~100.11~~. 50.67
2. Calculation J-G-SA02.
3. USAR Section 7.3.3 and Table 7.3-5.
4. USAR Section 15.7.4.

BASES

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address RCS operational LEAKAGE. However, the other forms of operational LEAKAGE are related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analyses for events resulting in steam discharge to the atmosphere assume that primary to secondary LEAKAGE from all steam generators (SGs) is one gallon per minute ~~or increases to one gallon per minute~~ as a result of accident induced conditions. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. Other accidents or transients involving secondary steam release to the atmosphere, include the steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The USAR (Ref. 3) analysis for SGTR assumes ~~the~~ some of contaminated secondary fluid is released via atmospheric relief valves.

The safety analysis for the SLB accident assumes the entire 1 gpm primary to secondary LEAKAGE is through the affected generator as an ~~initial condition~~ involving secondary steam release to the atmosphere. The dose consequences resulting from the SLB and ~~SGTR~~ accidents are ~~well~~ within the limits defined in 10 CFR ~~100~~ 50.67 (Ref. 5) ~~(i.e., a small fraction of these limits).~~

The safety analysis for RCS main loop piping for GDC-4 (Ref. 1) assumes 1 gpm unidentified leakage and monitoring per Regulatory Guide 1.45 (Ref. 2) are maintained (Ref. 4).

The RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) and meet the appropriate acceptance criteria in the Standard Review Plan (Ref. 8).

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.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.2 (continued)

secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.17, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 7. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.The Surveillance is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

The Surveillance Frequency of 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 7).

REFERENCES

1. 10 CFR 50, Appendix A, GDC 4 and 30.
2. Regulatory Guide 1.45, May 1973.
3. USAR, ~~Section 15.6.3~~. Chapter 15.
4. NUREG-1061, Volume 3, November 1984.
5. 10 CFR ~~100~~. 50.67
6. NEI 97-06, "Steam Generator Guidelines."
7. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."
8. Standard Review Plan (SRP), Section 15.0.1.

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 exclusion area boundary can receive for any 2 hours following an accident, or at the low population zone outer boundary for the radiological release duration, is specified in ~~10 CFR 100.11~~ 50.67 (Ref. 1). Doses to control room ~~operators~~ must be limited per GDC 19. The limits on specific activity ensure that the offsite and control room doses are appropriately limited 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 dose consequences in the event of a steam line break (SLB) or steam generator tube rupture (SGTR) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133. The allowable levels are intended to ensure that offsite and control room doses meet the appropriate acceptance criteria in the Standard Review Plan (Ref. 2).

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensure that the resulting offsite and control room doses meet the appropriate Standard Review Plan acceptance criteria following a SLB or SGTR accident. The safety analyses (Refs. 3 and 4) assume the specific activity of the reactor coolant is at or more conservative than the LCO limits, and ~~an existing~~ reactor coolant steam generator (SG) tube leakage rate of 1 gpm exist~~s~~. The safety analyses assume the specific activity of the ~~se~~condary coolant is at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 or results from accident induced conditions from LCO 3.7.18, "Secondary Specific Activity."

The analysis for the SLB and SGTR accidents establish the acceptance limits for RCS specific activity. Reference to these analyses is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The analyses consider 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, by a factor of 500 for SLB and 335 for SGTR, the rate of release of iodine from the fuel rods containing cladding defects to the primary coolant immediately after a SLB or SGTR~~, respectively~~. The second case assumes the initial reactor coolant iodine activity at

BASES

APPLICABLE SAFETY ANALYSES (continued)

60 μCi/gm DOSE EQUIVALENT I-131 due to a pre-accident iodine spike caused by an RCS transient. In both cases, the noble gas specific activity is assumed to be equal to or greater than 500 μCi/gm DOSE EQUIVALENT XE-133.

The SGTR analysis also assumes a loss of offsite power at the same time as the reactor trip. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from ~~a low pressurizer pressure signal in the analysis of an SGTR with a failed atmospheric relief valve on the faulted steam generator. In the analysis of an SGTR with a failed AFW flow control valve on the faulted steam generator, reactor trip and safety injection are assumed to occur at the time of the tube rupture to maximize the potential for overfilling the ruptured steam generator.~~ an Overtemperature ΔT signal

The 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 atmospheric relief valves. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends and the Residual Heat Removal (RHR) System is placed into service.

The SLB radiological analysis assumes that offsite power is lost at the same time as the pipe break occurs outside containment. Reactor trip occurs after the generation of an SI signal on low steamline pressure. The affected SG blows down completely and steam is vented directly to the atmosphere. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends and the RHR System is placed in service.

Operation with iodine specific activity levels greater than the LCO limit is permissible if the activity levels do not exceed 60 μCi/gm for more than 48 hours.

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).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1 (continued)

The Note modifies this SR to allow entry into and operating in MODE 4, MODE 3, and MODE 2 prior to performing the SR. This allows the Surveillance to be performed in those MODES, prior to entering MODE 1.

SR 3.4.16.2

This Surveillance is performed to ensure iodine specific activity remains within the LCO limit during normal operation and following fast power changes when iodine spiking is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level, considering noble gas 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 iodine spiking information; samples at other times would provide inaccurate results.

The Note modifies this SR to allow entry into and operation in MODE 4, MODE 3, and MODE 2 prior to performing the SR. This allows the Surveillance to be performed in those MODES, prior to entering MODE 1.

REFERENCES

1. 10 CFR ~~100.11, 1973~~. 50.67
2. Standard Review Plan (SRP), Section ~~15.1.5 Appendix A (SLB) and Section 15.6.3 (SGTR).~~ 15.0.1.
3. USAR Section 15.1.5.
4. USAR, Section 15.6.3.

BASES

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of an SGTR event assumes a bounding primary to secondary LEAKAGE rate ~~equal to the operational LEAKAGE rate limits in LCO 3.4.13, "RCS Operational LEAKAGE,"~~ from all SGs of 1 gpm, plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for an SGTR assumes some of the contaminated secondary fluid is released to the atmosphere via SG atmospheric relief valves and safety valves.

The analysis for design basis accidents and transients other than an SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) In these analyses, the ~~steam~~ discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of 1 gallon per minute ~~or is assumed to increase to 1 gallon per minute~~ as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.16, "RCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR ~~100~~ (Ref. 3) ~~or the NRC approved licensing basis (e.g., a small fraction of these limits)~~ and meet the appropriate acceptance criteria in the Standard Review Plan (Ref. 8).

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). 50.67

LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the plugging criteria be plugged in accordance with the Steam Generator Program.

During a SG inspection, any inspected tube that satisfies the Steam Generator Program plugging criteria is removed from service by plugging. If a tube was determined to satisfy the plugging criteria but was not plugged, the tube may still have tube integrity.

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. For Refueling Outage 18 and the subsequent operating cycle, a one-time alternate plugging criterion for the portion of the tube below 15.2 inches from the top of the tubesheet is specified in TS 5.5.9c.1. (Ref. 7) The tube-to-tubesheet weld is not considered part of the tube.

BASES

LCO (continued)

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed 1 gpm ~~per SG~~ total to all SGs. The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident.

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.13, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to an SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

APPLICABILITY

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODE 1, 2, 3, or 4.

RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.

ACTIONS

The ACTIONS are modified by a Note clarifying that the Conditions may be entered independently for each SG tube. This is acceptable because the Required Actions provide appropriate compensatory actions for each affected SG tube. Complying with the Required Actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Condition entry and application of associated Required Actions.

A.1 and A.2

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube plugging criteria but were not plugged in accordance with the Steam Generator Program as required by SR 3.4.17.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.17.2 (continued)

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

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."

2. 10 CFR 50 Appendix A, GDC 19.

3. 10 CFR ~~100~~. 50.67

4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.

5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.

6. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."

7. License Amendment No. 186, October 19, 2009.

8. Standard Review Plan (SRP), Section 15.0.1.

BASES

BACKGROUND (continued)

2. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves"

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks";

c. All equipment hatches are closed and sealed; and

d. The sealing mechanism associated with a penetration (e.g. welds, bellows, or O-rings) is OPERABLE.

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a loss of coolant accident (LOCA) and a steam line break (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or Rod Ejection Accident (REA). In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.20% of containment air weight per day for the first 24 hours and 0.10% thereafter (Ref. 3). This leakage rate, used to evaluate offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting design bases LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.20% of containment air weight per day in the safety analysis at P_a = 48 psig (Ref. 3).

and control room

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

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

LCO

Containment OPERABILITY is maintained by limiting leakage to $\leq$ 1.0 L_a, except prior to the first startup after performing a required Containment

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks

BASES

BACKGROUND

Containment air locks form part of the containment pressure boundary and provide a means for personnel access during all MODES of operation.

The personnel air lock is nominally a right circular cylinder, approximately 10 ft in diameter, with a door at each end. The emergency air lock is approximately 5 ft inside diameter with a 2 ft 6 inch door at each end. On both air locks, doors are interlocked to prevent simultaneous opening. During periods when containment is not required to be OPERABLE, 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. Each air lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. As such, closure of a single door supports containment OPERABILITY. Each of the doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door).

Each air lock is provided with limit switches on both doors that provide local indication of door position.

The containment air locks form part of the containment pressure boundary. As such, air lock integrity and leak tightness is essential for maintaining the containment leakage rate within limit in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the safety analyses.

APPLICABLE SAFETY ANALYSES

The DBA that results in a release of radioactive material within containment is a loss of coolant accident (Ref. 2). In the analysis of this accident, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.20% of containment air weight per day (Ref. 2). This leakage rate is defined in 10 CFR 50, Appendix J, Option B, (Ref. 1), as L_a = 0.20% of containment air weight per day, the maximum allowable

BASES

BACKGROUND (continued)

within containment prior to and during personnel access. The supply and exhaust lines each contain two isolation valves. Because of their large size, the 36 inch containment purge supply and exhaust valves are not qualified for automatic closure from their open position under DBA conditions. Therefore, the 36 inch containment purge supply and exhaust isolation valves are normally maintained closed and blind flange installed or sealed closed in MODES 1, 2, 3, and 4 to ensure the containment boundary is maintained.

Mini-Purge System (18 inch purge valves)

The Mini-purge System operates to:

a. Reduce the concentration of noble gases within containment prior to and during personnel access, and

b. Equalize containment internal and external pressures.

Since the 18 inch valves used in the Mini-purge System are designed to meet the requirements for automatic containment isolation valves, these valves may be opened as needed, for a limited time as specified in procedures, in MODES 1, 2, 3, and 4.

APPLICABLE SAFETY ANALYSES

The containment isolation valve LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during major accidents. As part of the containment boundary, containment isolation valve OPERABILITY supports leak tightness of the containment. Therefore, the safety analyses of any event requiring isolation of containment is applicable to this LCO.

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident (LOCA) and a rod ejection accident (Ref. 1). In the analyses for each of these accidents, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through containment isolation valves (including containment purge valves) are minimized. The safety analyses assume that the 36 inch shutdown purge valves are closed at event initiation.

The DBA analysis assumes that, ~~within 60 seconds~~ after the accident, isolation of the containment is complete and leakage terminated except for the design leakage rate, L_a. ~~The containment isolation total response time of 60 seconds includes signal delay, diesel generator startup (for loss of offsite power), and containment isolation valve stroke times.~~

BASES

and rod ejection e

APPLICABLE SAFETY ANALYSES (continued)

The LOCA offsite dose analysis assumes leakage from the containment at a maximum leak rate of 0.20 percent of the containment volume per day for the first 24 hours, and at 0.10 percent of the containment volume per day for the duration of the accident.

The single failure criterion required to be imposed in the conduct of plant safety analyses was considered in the original design of the 18 inch containment mini-purge valves. Two valves in series on each purge line provide assurance that both the supply and exhaust lines could be isolated even if a single failure occurred. The inboard and outboard isolation valves on each line are provided with independent electrical power sources to solenoids that open the pneumatically operated spring closed actuators. The actuators fail closed on the loss of power or air. This arrangement was designed to preclude common mode failures from disabling both valves on a purge line.

The 36 inch purge valves may be unable to close against the buildup of pressure following a LOCA. Therefore, each of the purge valves is required to remain sealed closed or closed and blind flange installed during MODES 1, 2, 3, and 4. The Containment Shutdown Purge System valve design precludes a single failure from compromising the containment boundary as long as the system is operated in accordance with the subject LCO.

The containment isolation valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Containment isolation valves form a part of the containment boundary. The containment isolation valves' safety function is related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during a DBA.

The automatic power operated isolation valves are required to have isolation times within limits and to actuate on an automatic isolation signal. The 36 inch containment purge supply and exhaust valves must be maintained sealed closed or closed and blind flange installed. The valves covered by this LCO are listed along with their associated stroke times in the USAR (Ref. 2).

The normally closed containment isolation valves are considered OPERABLE when manual valves are closed, automatic valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact. These passive isolation valves/devices are those listed in Reference 2.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Containment Spray and Cooling Systems

BASES

BACKGROUND

The Containment Spray and Containment Cooling system provides containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the ~~iodine~~ fission product removal capability of the spray reduces the release of fission product radioactivity from containment to the environment, in the event of a Design Basis Accident (DBA), to within limits. The Containment Spray and Containment Cooling system is designed to meet the requirements of 10 CFR 50, Appendix A, GDC 38, "Containment Heat Removal," GDC 39, "Inspection of Containment Heat Removal Systems," GDC 40, "Testing of Containment Heat Removal Systems," GDC 41, "Containment Atmosphere Cleanup," GDC 42, "Inspection of Containment Atmosphere Cleanup Systems," and GDC 43, "Testing of Containment Atmosphere Cleanup Systems," and GDC 50, "Containment Design Bases" (Ref. 1).

The Containment Cooling System and Containment Spray System are Engineered Safety Feature (ESF) systems. They are designed to ensure that the heat removal capability required during the post accident period can be attained. The Containment Spray System and the Containment Cooling System provides a redundant method to limit and maintain post accident conditions to less than the containment design values.

Containment Spray System

The Containment Spray System consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate ESF bus. The refueling water storage tank (RWST) supplies borated water to the Containment Spray System during the injection phase of operation. In the recirculation mode of operation, containment spray pump suction is transferred from the RWST to the containment recirculation sumps.

The Containment Spray System provides a spray of borated water mixed with sodium hydroxide (NaOH) from the Spray Additive System into the upper regions of containment to reduce the containment pressure and temperature and to reduce fission products from the containment atmosphere during a DBA. The RWST solution temperature is an important factor in determining the heat removal capability of the Containment Spray System during the injection phase. In the recirculation mode of operation, heat is removed from the containment recirculation sump water by the residual heat removal heat exchangers. Each train of the Containment Spray System provides adequate spray

BASES

APPLICABLE SAFETY ANALYSES (continued)

The modeled Containment Spray System actuation from the containment analysis is based on a response time associated with exceeding the containment High-3 pressure setpoint to achieving full flow through the containment spray nozzles.

The Containment Spray System total response time includes diesel generator (DG) startup (for loss of offsite power), sequenced loading of equipment, containment spray pump startup, and spray line filling (Ref. 4).

Containment cooling train performance for post accident conditions is given in Reference 4. The result of the analysis is that each train can provide 100% of the required peak cooling capacity during the post accident condition. The train post accident cooling capacity under varying containment ambient conditions, required to perform the accident analyses, is also shown in Reference 4.

The modeled Containment Cooling System actuation from the containment analysis is based upon a response time associated with receipt of an SI signal to achieving full Containment Cooling System air and safety grade cooling water flow. The Containment Cooling System total response time of 70 seconds, includes signal delay, DG startup (for loss of offsite power), and Essential Service Water pump startup times and line refill (Ref. 4).

The Containment Spray System and the Containment Cooling System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

During a DBA, a minimum of one containment cooling train and one containment spray train is required to maintain the containment peak pressure and temperature below the design limits (Ref. 3). Additionally, one containment spray train is also required to remove ~~iodine~~ fission products from the containment ~~atmosphere and maintain~~ concentrations below those assumed in the safety analysis. With the Spray Additive System inoperable, a containment spray train is still available and would remove some ~~iodine~~ fission products from the containment atmosphere in the event of a DBA. To ensure that these requirements are met, two containment spray trains and two containment cooling trains must be OPERABLE. Therefore, in the event of an accident, at least one train in each system operates, assuming the worst case single active failure occurs.

Each Containment Spray System typically includes a spray pump, spray headers, eductor, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an ESF actuation signal and manually transferring to the containment sump.

A containment cooling train typically includes cooling coils, dampers, two fans, instruments, and controls to ensure an OPERABLE flow path.

BASES

APPLICABLE SAFETY ANALYSES

The Spray Additive System is essential to the removal of airborne iodine within containment following a DBA. and the retention of that iodine in the containment sump

Following the assumed release of radioactive materials into containment, the containment is assumed to leak at its design value volume following the accident. The analysis assumes that 85% of the containment free volume is covered by the spray (Ref. 1).

The DBA analysis credit for iodine removal by Containment Spray System is taken, starting at ~~time zero~~ two minutes and continuing until ~~a decontamination factor of 100~~ five hours.

The DBA analyses assume that one train of the Containment Spray System/Spray Additive System is inoperable and that the entire spray additive tank volume is added to the remaining Containment Spray System flow path.

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

LCO

The Spray Additive System is necessary to reduce the release of radioactive material to the environment in the event of a DBA. To be considered OPERABLE, the volume and concentration of the spray additive solution must be sufficient to provide NaOH injection into the spray flow to raise the average long term containment sump solution pH to a level conducive to iodine removal, ~~namely, to between 8.5 and 11.0.~~ and retention This pH range maximizes the effectiveness of the iodine removal mechanism without introducing conditions that may induce caustic stress corrosion cracking of mechanical system components.

The Spray Additive System typically includes a spray additive tank, eductors, valves, instrumentation, and connected piping to ensure OPERABLE flow paths. In addition, it is essential that valves in the Spray Additive System flow paths are properly positioned and that automatic valves are capable of activating to their correct positions.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment requiring the operation of the Spray Additive System. The Spray Additive System assists in reducing the iodine fission product inventory prior to release to the environment.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Thus, the Spray Additive System is not required to be OPERABLE in MODE 5 or 6.

BASES

ACTIONS

A.1

If the Spray Additive System is inoperable, it must be restored to OPERABLE within 72 hours. The pH adjustment of the Containment Spray System flow for corrosion protection and iodine removal enhancement is reduced in this condition. The Containment Spray System would still be available and would remove some ~~iodine~~ fission products from the containment atmosphere in the event of a DBA. The 72 hour Completion Time takes into account the redundant flow path capabilities and the low probability of the worst case DBA occurring during this period.

B.1 and B.2

If the Spray Additive System cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant 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 plant systems. The extended interval to reach MODE 5 allows 48 hours for restoration of the Spray Additive System in MODE 3 and 36 hours to reach MODE 5. This is reasonable when considering the reduced pressure and temperature conditions in MODE 3 for the release of radioactive material from the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1

Verifying the correct alignment of Spray Additive System manual, power operated, and automatic valves in the spray additive flow path provides assurance that the system is able to provide additive to the Containment Spray System in the event of a DBA. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. This SR does not apply to manual vent/drain valves, and to valves that cannot be inadvertently misaligned such as check valves. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown (which may include the use of local or remote indicators), that those valves outside containment and capable of potentially being mispositioned are in the correct position. The 31 day Frequency is based on engineering judgement, is consistent with administrative controls governing valve operation, and ensures correct valve positions.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.7.2

and retention

To provide effective iodine removal, the containment spray must be an alkaline solution. Since the RWST contents are normally acidic, the volume of the spray additive tank must provide a sufficient volume of spray additive to adjust pH for all water injected. This SR is performed to verify the availability of sufficient NaOH solution in the Spray Additive System. The spray additive tank site glass (ENLG0022) is utilized for meeting the SR since the control room level indicators do not provide conservative indication (Ref. 2). The 184 day Frequency was developed based on the low probability of an undetected change in tank volume occurring during the SR interval (the tank is isolated during normal unit operations). (Ref. 3).

SR 3.6.7.3

This SR provides verification of the NaOH concentration in the spray additive tank and is sufficient to ensure that the spray solution being injected into containment is at the correct pH level. The 184 day Frequency is sufficient to ensure that the concentration level of NaOH in the spray additive tank remains within the established limits. This is based on the low likelihood of an uncontrolled change in concentration (the tank is normally isolated) and the probability that any substantial variance in tank volume will be detected.

SR 3.6.7.4

This SR provides verification that each automatic valve in the Spray Additive System flow path actuates to its correct position upon receipt of an actual or simulated actuation of a containment High-3 pressure 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 this Surveillance under the conditions that apply during a plant 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 concluded to be acceptable from a reliability standpoint.

SR 3.6.7.5

To ensure correct pH level is established in the borated water solution provided by the Containment Spray System, the flow rate in the Spray Additive System is verified once every 5 years. Flow of ≥ 52 gpm through the eductor test loops (supplied from the RWST) is throttled to 17 psig at

BASES

LCO (continued)

sources must be removed from the actuation solenoids on all four MSIVs and a drain or vent path must be available from the lower piston chamber. Note 2 allows one or more MSIV bypass valve to be inoperable when closed and de-activated, closed and isolated by a closed manual valve, or isolated by two closed manual valves. When one or more MSIV bypass valves are closed and de-activated, closed and isolated by a closed manual valve, or isolated by two closed manual valves, they are performing their specified safety function. When the valve is de-activated, power and air are removed from both actuation solenoid valves and the valve is spring closed. Requiring the MSIV bypass valve to be closed and isolated by a closed manual valve or isolated by two closed manual valves also provides the dual assurance that the specified safety function is being performed.

This LCO provides assurance that the MSIVs and MSIV bypass valves will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 50.67 10 CFR ~~100~~ (Ref. 4) limits or the NRC staff approved licensing basis.

APPLICABILITY

The MSIVs and MSIV bypass valves must be OPERABLE in MODE 1, and in MODES 2 and 3 due to significant mass and energy in the RCS and steam generators. When the MSIVs and MSIV bypass valves are closed, they are already performing the safety function. The MSIV actuator trains must be OPERABLE in MODES 1, 2, and 3 to support operation of the MSIV.

In MODE 4, 5 or 6, the steam generator energy is low. Therefore, the MSIVs and MSIV bypass valves are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

ACTIONS

The LCO specifies OPERABILITY requirements for the MSIVs as well as for their associated actuator trains. The Conditions and Required Actions for TS 3.7.2 separately address inoperability of the MSIV actuator trains and inoperability of the MSIVs themselves.

A.1

With a single actuator train inoperable on one MSIV, action must be taken to restore the inoperable actuator train to OPERABLE status within 7 days. The 7-day Completion Time is reasonable in light of the dual-redundant actuator train design such that with one actuator train inoperable, the affected MSIV is still capable of closing on demand via the

BASES

REFERENCES

1. USAR, Section 10.3.
2. USAR, Section 6.2.
3. USAR, Section 15.1.5.
4. 10 CFR ~~100.11~~. 50.67

BASES

APPLICABLE SAFETY ANALYSES (continued)

In the accident analysis presented in Reference 2, the ARVs are assumed to be used by the operator to cool down the unit to RHR entry conditions for accidents accompanied by a loss of offsite power. The main steam safety valves (MSSVs) are assumed to operate automatically to relieve steam and maintain the steam generator pressure below the design value. For the recovery from a steam generator tube rupture (SGTR) event in Reference 3, the operator is required to perform a RCS cooldown using three ~~two~~ intact steam generators to establish adequate subcooling as a necessary step to terminate the primary to secondary break flow into the ruptured steam generator. ~~For SG overfill resulting from SGTR, RCS cooldown to RHR entry conditions using intact SG ARVs is necessary to terminate primary to secondary break flow.~~ The time required to terminate the primary to secondary break flow for an SGTR is more critical than the time required to cool down to RHR conditions for this event and also for other accidents. Thus, the SGTR is the limiting event for the ARVs. The number of ARVs required to be OPERABLE to satisfy the SGTR accident analysis requirements is four. ~~If a single failure of one occurs and another is associated with the ruptured SG, two ARVs would remain OPERABLE for heat removal and RCS cooldown, as discussed in Reference 3.~~

The ARVs are equipped with block valves in the event an ARV spuriously fails open or fails to close during use.

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

LCO

Four ARV lines are required to be OPERABLE. One ARV line is required from each of four steam generators to ensure that at least two ARV lines are available to conduct a RCS cooldown following an SGTR, in which one steam generator becomes unavailable due to a SGTR, accompanied by a single, active failure of a second ARV line on an unaffected steam generator. The block valves must be OPERABLE to isolate a failed open ARV line.

Failure to meet the LCO can result in the inability to achieve subcooling, consistent with the assumptions used in the steam generator tube rupture analysis, to facilitate equalizing pressures between the Reactor Coolant System and the ruptured steam generator. ~~Failure to meet the LCO can also impact the recovery capability following a SG overfill scenario.~~

An ARV is considered OPERABLE when it is capable of providing controlled relief of the main steam flow and capable of fully opening and closing on demand and not experiencing excessive seat leakage. Excessive seat leakage, although not associated with a specific

BASES

BACKGROUND (continued)

Outside air is filtered, diluted with air from the electrical equipment and cable spreading rooms, and added to the air being recirculated from the CRE. Pressurization of the CRE prevents infiltration of unfiltered air from the surrounding areas of the building.

The air entering the CBE during normal operation is continuously monitored by radiation and smoke detectors. A high radiation signal initiates the CRVIS; the smoke detectors provide an alarm in the control room. A CRVIS is initiated by the radiation monitors (GKRE0004 and GKRE0005), fuel building ventilation isolation signal, containment isolation phase A, containment atmosphere radiation monitors (GTRE0031 and GTRE0032), containment purge exhaust radiation monitors (GTRE0022 and GTRE0033), or manually.

A single CREVS train operating in the CREVS alignment established by surveillance procedures will pressurize the control room to ≥ 0.25 inches water gauge. The CREVS operation in maintaining the CRE habitable is discussed in the USAR, Section 6.4 and 9.4 (Ref. 1).

Either of the pressurization and recirculation trains provide the required filtration and pressurization to the CRE. Normally open isolation dampers are arranged in series pairs so that the failure of one damper to shut will not result in a breach of isolation. The CREVS is designed in accordance with Seismic Category I requirements.

The CREVS is designed to maintain a habitable environment in the CRE for 30 days of continuous occupancy after a Design Basis Accident (DBA) (Ref. 2) without exceeding a 5 rem ~~whole body dose or its equivalent to any part of the body~~ total effective dose equivalent (Ref. ~~2~~ 9).

By operation of the control room pressurization trains and the control room filtration units, the CREVS pressurizes, recirculates and filters air within the CRE as well as the CBE that generally surrounds the CRE. The boundaries of these two distinct but related volumes are credited in the analysis of record for limiting the inleakage of unfiltered outside air.

The station CRE design is unique. The Control Building by and large surrounds the CRE. The Control Building is also designed to be at a positive pressure with respect to its surrounding environment although not positive with respect to the CRE. In the emergency pressurization and filtration mode, the control room air volume receives air through a filtration system that takes a suction on the Control Building. The Control Building in turn receives filtered air from the outside environment.

BASES

BACKGROUND (continued)

The CRE is the area within the confines of the CRE boundary that contains the spaces that control room occupants inhabit to control the unit during normal and accident conditions. This area encompasses the control room, and may encompass other non-critical areas to which frequent personnel access or continuous occupancy is not necessary in the event of an accident. The CRE is protected during normal operation, natural events, and accident conditions. The CRE boundary is the combination of walls, floor, roof, ducting, doors, penetrations and equipment that physically form the CRE. The CRE boundary must be maintained to ensure that the inleakage of unfiltered air into the CRE will not exceed the inleakage assumed in the licensing basis analysis of design basis accident (DBA) consequences to CRE occupants. The CRE and its boundary are defined in the Control Room Envelope Habitability Program.

The CBE is an area that largely surrounds the CRE. Occupancy of the CBE is not required to control the unit during normal and accident conditions. The CBE boundary is the combination of walls, floor, roof, ducting, doors, penetrations and equipment that physically form the CBE. The CBE boundary must be maintained to ensure that the inleakage of unfiltered air into the CBE will not exceed the inleakage assumed in the licensing basis analysis of DBA consequences to CRE occupants. The CBE and its boundary are defined in the Control Room Envelope Habitability Program.

APPLICABLE SAFETY ANALYSES

The CREVS components are arranged in redundant, safety related ventilation trains. The location of components and ducting within the CRE ensures an adequate supply of filtered air to all areas requiring access. The CREVS provides airborne radiological protection for the CRE occupants, as demonstrated by the CRE occupant dose analyses for the most limiting design basis accident, fission product release presented in the USAR, Chapter 15~~, Appendix 15A~~ (Ref. 2).

The fuel handling accident analysis does not credit the CREVS for dose mitigation after 76 hours decay time (Ref. 2). The fuel handling accident analysis does not address fuel movement prior to 76 hours.

The CREVS provides protection from smoke and hazardous chemicals to the CRE occupants. The analysis of hazardous chemical releases (Ref. 7) determined that hazardous chemicals are not stored or used onsite in quantities sufficient to necessitate CRE protection as required by Regulatory Guide 1.78 (Ref. 8). The evaluation of a smoke challenge demonstrates that it will not result in the inability of the CRE occupants to control the reactor either from the control room or from the remote shutdown panels (Ref. 1). The analysis for smoke and hazardous chemicals has determined no CREVS actuation for such events.

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

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

BASES

LCO

Two independent and redundant CREVS trains are required to be OPERABLE to ensure that at least one is available if a single active failure disables the other train. Total system failure, such as from a loss of both ventilation trains or from an inoperable CRE or CBE boundary, could result in exceeding a dose of 5 rem ~~whole body or its equivalent to any part of the body~~ to the CRE occupants in the event of a large radioactive release.

total effective dose equivalent

Each CREVS train is considered OPERABLE when the individual components necessary to limit CRE occupant exposure are OPERABLE in both trains. A CREVS train is OPERABLE when the associated:

a. Recirculation and pressurization fans are OPERABLE;

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

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

d. Control Room Air Conditioning flow path integrity is maintained.

In order for the CREVS trains to be considered OPERABLE, the CRE and CBE boundaries must be maintained such that the CRE occupant dose from a large radioactive release does not exceed the calculated dose in the licensing basis consequence analyses for DBA's, and that CRE occupants are protected from hazardous chemicals and smoke.

The LCO is modified by a Note allowing the CRE and CBE boundaries to be opened intermittently under administrative controls. This Note only applies to openings in the CRE and CBE boundaries that can be rapidly restored to intended design condition, such as doors, hatches, floor plugs, and access panels. 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 should be proceduralized and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the CRE. This individual will have a method to rapidly close the opening and thereby restore the affected envelope boundary to a condition equivalent to the design condition when a need for CRE isolation is indicated.

Note that the Control Room Air Conditioning System (CRACS) forms a subsystem to the CREVS. The CREVS remains capable of performing its safety function provided the CRACS air flow path is intact and air circulation can be maintained. Isolation or breach of the CRACS air flow path can also render the CREVS flow path inoperable. In these situations LCOs 3.7.10 and 3.7.11 may be applicable.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, and during movement of irradiated fuel assemblies, the CREVS must be OPERABLE to ensure that the CRE will remain habitable during and following a DBA.

recently

During movement of irradiated fuel assemblies, the CREVS must be OPERABLE to cope with the release from a design basis fuel handling accident.

involving handling recently irradiated fuel.

The CREVS is only required to be OPERABLE during fueling handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 76 hours), due to radioactive decay.

ACTIONS

A.1

When one CREVS train is inoperable for reasons other than an inoperable CRE or CBE boundary, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CREVS train is adequate to perform the CRE occupant protection function. However, the overall reliability is reduced because a failure in the OPERABLE CREVS train could result in loss of CREVS 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.

B.1, B.2, and B.3

If the unfiltered inleakage of potentially contaminated air past a CRE or CBE boundary credited in the accident analysis and into the CRE can result in CRE occupant radiological dose greater than the calculated dose of the licensing basis analyses of DBA consequences (allowed to be up to 5 rem ~~whole body or its equivalent to any part of the body~~), or inadequate protection of CRE occupants from hazardous chemicals or smoke, the CRE or CBE boundary is inoperable. Actions must be taken to restore the CRE or CBE boundary to OPERABLE status within 90 days.

total effective dose equivalent

During the period that the CRE or CBE boundary is considered inoperable, action must be initiated to implement mitigating actions to lessen the effect on CRE occupants from the potential hazards of a radiological or chemical event or a challenge from smoke. Actions must be taken within 24 hours to verify that in the event of a DBA, the mitigating actions will ensure that CRE occupant radiological exposures will not exceed the ~~calculated dose~~ of the licensing basis analyses of DBA consequences, and that CRE occupants are protected from hazardous

5 rem TEDE limit

BASES

ACTIONS

B.1, B.2, and B.3 (continued)

chemicals and smoke. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable CBP boundary) should be preplanned for implementation upon entry into the condition, regardless of whether entry is intentional or unintentional.

The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of mitigating actions. The 90 day Completion Time is reasonable based on the determination that the mitigating actions will ensure protection of CRE occupants within analyzed limits while limiting the probability that CRE occupants will have to implement protective measures that may adversely affect their ability to control the reactor and maintain it in a safe shutdown condition in the event of a DBA. In addition, the 90 day Completion Time is a reasonable time to diagnose, plan and possibly repair, and test most conditions adversely affecting the CRE or CBE boundary.

C.1 and C.2

In MODE 1, 2, 3, or 4, if the inoperable CREVS train or the inoperable CRE or CBE 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~~, D.2.1,~~ and D.2~~.2~~

During movement of recently irradiated fuel assemblies, if the inoperable CREVS train cannot be restored to OPERABLE status within the required Completion Time, action must be taken to immediately place the OPERABLE CREVS train in the CRVIS mode. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure would be readily detected.

BASES

ACTIONS

D.1~~, D.2.1,~~ and D.2~~.2~~ (continued)

An alternative to Required Action D.1 is to immediately suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

E.1~~ and E.2~~

During movement of recently irradiated fuel assemblies, with two CREVS trains inoperable or with one or more CREVS trains inoperable due to an inoperable CRE or CBE boundary, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

F.1

If both CREVS trains are inoperable in MODE 1, 2, 3, or 4, for reasons other than an inoperable CRE and CBE boundary (i.e., Condition B), the CREVS 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 this system are not too severe, testing each train once every month, by initiating from the control room, flow through the HEPA filters and charcoal adsorber of both the filtration and pressurization systems, provides an adequate check of this system. Monthly heater operations dry out any moisture accumulated in the charcoal from humidity in the ambient air. Each pressurization system train must be operated for ≥ 10 continuous hours with the heaters energized. Each filtration system train need only be operated for ≥ 15 minutes to demonstrate the function of the system. The 31 day Frequency is based on the reliability of the equipment and the two train redundancy.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.10.2

This SR verifies that the required CREVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The CREVS filter tests use the procedure guidance in Regulatory Guide 1.52, Rev. 2 (Ref. 3) in accordance with the VFTP. The VFTP includes testing the performance of the HEPA filter, charcoal absorber efficiency, minimum 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

This SR verifies that each CREVS train starts and operates on an actual or simulated CRVIS. The actuation signal includes Control Room Ventilation or High Gaseous Radioactivity. The CREVS train automatically switches on an actual or simulated CRVIS into a CRVIS mode of operation with flow through the HEPA filters and charcoal adsorber banks. The Frequency of 18 months is consistent with a typical operating cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.10.4

This SR verifies the OPERABILITY of the CRE and CBE boundaries credited in the accident analysis by testing for unfiltered air inleakage past the credited envelope boundaries and into the CRE. The details of the testing are specified in the Control Room Envelope Habitability Program.

The CRE is considered habitable when the radiological dose to CRE occupants calculated in the licensing basis analyses ~~of DBA~~ consequences is no more than 5 rem ~~whole body or its equivalent to any part of the body~~ total effective dose equivalent and the CRE occupants are protected from hazardous chemicals and smoke. For WCGS, there is no CREVS actuation for hazardous chemical releases or smoke and there are no Surveillance Requirements that verify OPERABILITY for hazardous chemicals or smoke. This SR verifies that the unfiltered air inleakage into the CRE and CBE boundaries is no greater than the flow rate assumed in the licensing basis analyses of DBA consequences. When unfiltered air inleakage is greater than the assumed flow rate, Condition B must be entered. Required Action B.3 allows time to restore the CRE or CBE

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.10.4 (continued)

boundary to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section C.2.7.3, (Ref. 4) which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F (Ref. 5). These compensatory measures may also be used as mitigating actions as required by Required Action B.2. Temporary analytical methods may also be used as compensatory measures to restore OPERABILITY (Ref. 6). Options for restoring the CRE or CBE boundary to OPERABLE status include changing the licensing basis DBA consequence analysis, repairing the boundary, or a combination of these actions. Depending upon the nature of the problem and the corrective action, a full scope inleakage test may not be necessary to establish that the envelope boundary has been restored to OPERABLE status.

REFERENCES

1. USAR, Section 6.4 and 9.4.

2. USAR, Chapter 15~~, Appendix 15A~~.

3. Regulatory Guide 1.52, Rev. 2.

4. Regulatory Guide 1.196.

5. NEI 99-03, "Control Room Habitability Assessment," June 2001.

6. Letter from Eric J. Leeds (NRC) to James W. Davis (NEI) dated January 30, 2004, "NEI Draft White Paper, Use of Generic Letter 91-18 Process and Alternative Source Terms in the Context of Control Room Habitability." (ADAMS Accession No. ML040300694).

7. USAR Section 2.2.

8. Regulatory Guide 1.78, Rev. 0.

9. 10 CFR 50, Appendix A, General Design Criterion 19.

BASES

LCO (continued)

The CRACS is considered to be OPERABLE when the individual components necessary to maintain the control room temperature are OPERABLE in both trains. These components include the refrigeration compressors, heat exchangers, cooling coils, fans, and associated temperature control instrumentation. In addition, the CRACS must be OPERABLE to the extent that air circulation can be maintained. Isolation or breach of the CRACS air flow path also can render the CREVS flowpath inoperable. In these situations, LCO 3.7.10 would also be applicable.

APPLICABILITY

In MODES 1, 2, 3, 4, 5, and 6, and during movement of recently irradiated fuel assemblies, the CRACS must be OPERABLE to ensure that the control room temperature will not exceed equipment operational requirements.

ACTIONS

The CRACS 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 76 hours), due to radioactive decay.

A.1

With one CRACS train inoperable, action must be taken to restore OPERABLE status within 30 days. In this Condition, the remaining OPERABLE CRACS train is adequate to maintain the control room temperature within limits. However, the overall reliability is reduced because a single failure in the OPERABLE CRACS train could result in loss of CRACS function. The 30 day Completion Time is based on the low probability of an event requiring control room isolation and the consideration that the remaining train can provide the required protection.

B.1 and B.2

In MODE 1, 2, 3, or 4, if the inoperable CRACS train 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~~, C.2.1,~~ and C.2~~.2~~

In MODE 5 or 6, or during movement of recently irradiated fuel, if the inoperable CRACS train cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE CRACS train must be placed in

BASES

ACTIONS

C.1~~, C.2.1,~~ and C.2~~.2~~ (continued)

operation immediately. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that active failures will be readily detected.

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 control room. 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~~ and D.2~~

recently

In MODE 5 or 6, or during movement of irradiated fuel assemblies, with two CRACS trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

E.1

If both CRACS trains are inoperable in MODE 1, 2, 3, or 4, the CRACS 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 the CRACS air conditioning units is adequate to remove the heat load assumed in the control room during design basis accidents. This SR consists of verifying the heat removal capability of the condenser heat exchanger (either through performance testing or inspection), ensuring the proper operation of major components in the refrigeration cycle and verification of unit air

BASES

BACKGROUND (continued)

The Emergency Exhaust System is discussed in the USAR, Sections 6.5.1, 9.4.2, 9.4.3, and ~~15.7.4~~ 15.6.5.4 (Refs. 1, 2, and ~~3~~ 10, respectively) because it may be used for normal, as well as post accident, atmospheric cleanup functions.

APPLICABLE SAFETY ANALYSES

The Emergency Exhaust System design basis is established by the consequences of the limiting Design Basis Accidents (DBAs), which are a loss of coolant accident (LOCA) and a fuel handling accident (FHA). ~~The analysis of the fuel handling accident, given in Reference 3, assumes that all fuel rods in an assembly are damaged, and one of the Emergency Exhaust System filter adsorber unit is operating with a failed heater or humidistat. A reduced efficiency in the removal of organic iodine would occur if the heater failure occurred concurrently with high ambient relative humidity.~~ The FHA analysis does not credit the Emergency Exhaust System for dose mitigation after 76 hours of decay time (Ref. 3.) The fuel handling accident analysis does not address fuel movement prior to 76 hours. The analysis of the LOCA ~~assumes~~ (Ref. 10) that radioactive materials leaked from the ECCS and Containment Spray System ~~during the~~ recirculation mode are filtered and adsorbed by the Emergency Exhaust System. ~~The DBA analysis of the LOCA assumes that only one train of the Emergency Exhaust System is functional due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the one remaining train of this filtration system. The amount of fission products available for release from the fuel handling building is determined for a fuel handling accident and for a LOCA.~~ These ~~assumptions and the~~ analysis follows the guidance provided in Regulatory Guides ~~1.4 (Ref. 5) and 1.25 (Ref. 4)~~ 1.183 (Ref. 4).

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

LCO

Two independent and redundant trains of the Emergency Exhaust System are required to be OPERABLE to ensure that at least one train is available, assuming a single failure that disables the other train, coincident with a loss of offsite power. Total system failure could result in the atmospheric release from the auxiliary building or fuel building exceeding the guideline limits of 10 CFR ~~100~~ 50.67 (Ref. 5) ~~limits~~ in the event of a LOCA or fuel handling accident involving handling recently irradiated fuel.

The Emergency Exhaust System is considered OPERABLE when the individual components necessary to control releases from the auxiliary or fuel building are OPERABLE in both trains. An Emergency Exhaust System train is considered OPERABLE when its associated:

BASES

LCO (continued)

a. Fan is OPERABLE;

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

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

The LCO is modified by a Note allowing the auxiliary or 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 auxiliary building or fuel building isolation is indicated.

APPLICABILITY

In MODE 1, 2, 3, or 4, the Emergency Exhaust System is required to be OPERABLE in the SIS mode of operation to provide fission product removal associated with potential radioactivity leaks during the post-LOCA recirculation phase of ECCS operation.

In MODE 5 or 6, when not moving irradiated fuel the Emergency Exhaust System is not required to be OPERABLE since the ECCS is not required to be OPERABLE.

recently

During movement of irradiated fuel in the fuel handling area, the Emergency Exhaust System is required to be OPERABLE to support the FBVIS mode of operation to alleviate the consequences of a fuel handling accident.

(i.e., fuel that has occupied part of a critical reactor core within the previous 76 hours)

The Applicability is modified by a Note. The Note clarifies the Applicability for the two safety related modes of operation of the Emergency Exhaust System, i.e., the Safety Injection Signal (SIS) mode and the Fuel Building Ventilation Isolation Signal (FBVIS) mode. The SIS mode which aligns the system to the auxiliary building is applicable when the ECCS is required to be OPERABLE. In the FBVIS mode the system is aligned to the fuel building. This mode is applicable while handling irradiated fuel in the fuel building.

recently

BASES

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.

A.1

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

B.1

If the auxiliary building boundary is inoperable such that a train of the Emergency Exhaust System operating in the SIS mode cannot establish or maintain the required negative pressure, action must be taken to restore an OPERABLE auxiliary building boundary within 24 hours. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period and the availability of the Emergency Exhaust System to provide a filtered release (albeit with potential for some unfiltered auxiliary building leakage).

C.1 and C.2

In MODE 1, 2, 3, or 4, when Required Action A.1 or B.1 cannot be completed within the associated Completion Time or when both Emergency Exhaust System trains are inoperable for reasons other than an inoperable auxiliary building boundary (i.e., Condition B), 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 MODE 3 within 6 hours, and in MODE 5 within 36 hours. The Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

BASES

ACTIONS

D.1 and D.2

recently

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

recently irradiated

recently irradiated

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

involving handling recently irradiated fuel.

E.1

If the fuel building boundary is inoperable such that a train of the Emergency Exhaust System operating in the FBVIS mode cannot establish or maintain the required negative pressure, action must be taken to restore an OPERABLE fuel building boundary within 24 hours. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period and the availability of the Emergency Exhaust System to provide a filtered release (albeit with potential for some unfiltered fuel building leakage).

recently

F.1

During movement of irradiated fuel assemblies in the fuel building, when two trains of the Emergency Exhaust System are inoperable for reasons other than an inoperable fuel building boundary (i.e., Condition E), or if Required Action E.1 cannot be completed within the associated Completion Time action must be taken to place the unit in a condition in which the LCO does not apply. Action must be taken immediately to suspend movement of irradiated fuel assemblies in the fuel building. This does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

Standby systems should be checked periodically to ensure that they function properly. As the environmental and normal operating conditions on this system are not severe, testing each train once every month, by initiating from the control room flow through the HEPA filters and charcoal adsorbers, provides an adequate check on this system.

BASES

REFERENCES

1. USAR, Section 6.5.1.
2. USAR, Section 9.4.2 and 9.4.3.
3. USAR, Section 15.7.4.
4. Regulatory Guide ~~1.25, Rev. 0 (Safety Guide 25).~~ 1.183, Rev. 0
5. 10 CFR ~~100~~. 50.67
6. ASTM D 3803-1989.
7. ANSI N510-1980.
8. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.
9. Regulatory Guide 1.52 (Rev. 2).
10. USAR, Section 15.6.5.4

B 3.7 PLANT SYSTEMS

B 3.7.15 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 USAR, Section 9.1.2 (Ref. 1). A description of the Spent Fuel Pool Cooling and Cleanup System is given in the USAR, Section 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in the USAR, Section 15.7.4 (Ref. 3).

APPLICABLE SAFETY ANALYSES

The minimum water level in the fuel storage pool meets the assumptions of the fuel handling accident described in Regulatory Guide ~~1.25~~ 1.183 (Ref. 4). The resultant 2 hour ~~thyroid~~ dose per person at the exclusion area boundary is well within the 10 CFR ~~100~~ 50.67 (Ref. 5) limit~~s~~.

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. This minimum water depth provides the basis for supporting an overall effective decontamination factor of ~~100 for iodine~~ 200. 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 plus the distance from the top of the storage racks to the top of the fuel seated in the storage racks. This results in a minimum water depth of approximately 21 feet 2 inches between the top of the damaged fuel rods to the fuel pool surface. With a minimum depth < 23 ft, the capability of the water to remove iodine from the gas bubbles released from the damaged fuel rods would be affected as the bubble rise time is diminished. The effect of water depths < 23 ft has been reviewed against the pool scrubbing model described in Reference 6, and it has been determined that a pool decontamination factor of > 100 can be supported for a pool depth of 20 ft. Thus, the pool decontamination factor of 100, assumed in Reference 4, remains conservatively bounding and~~

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained within allowable limits.~~

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

LCO

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

APPLICABILITY

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

ACTIONS

A.1

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

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

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

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

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

REFERENCES

1. USAR, Section 9.1.2.

2. USAR, Section 9.1.3.

3. USAR, Section 15.7.4.

4. Regulatory Guide ~~1.25, Rev. 0~~. 1.183, Rev. 0.

5. 10 CFR ~~100.11~~. 50.67

~~6. WCAP-7828, "Radiological Consequences of a Fuel Handling Accident," December 1971.~~

B 3.7 PLANT SYSTEMS

B 3.7.18 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 150 gpd/SG ~~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"). A ~~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).~~

Operating a unit at the allowable secondary coolant specific activity will assure that the potential 2 hour exclusion area boundary (EAB) exposure is limited to ~~a small fraction of the~~ the limits of 10 CFR ~~100~~ 50.67 (Ref. 1) ~~limits~~ and meet the appropriate acceptance criteria in the Standard Review Plan (Ref. 3).

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB), as discussed in the USAR, Chapter 15 (Ref. 2) assumes the initi 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 ~~a small fraction of~~ the unit EAB limits (Ref. ~~1~~ 3) for total effective dose equivalent ~~whole body and thyroid dose rates~~.

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 atmospheric relief valves (ARVs). The Auxiliary Feedwater System supplies the necessary

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 ARVs 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.

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

BASES

ACTIONS

A.1 and A.2 (continued)

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 3.7.18.1

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~~. 50.67
2. USAR, Chapter 15.
3. Standard Review Plan (SRP), Section 15.0.1.

B 3.9 REFUELING OPERATIONS

B 3.9.4 Containment Penetrations

recently

BASES

BACKGROUND

During ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment penetration closure" rather than "containment OPERABILITY." Containment penetration closure means that all potential escape paths are closed or capable of being closed within 2 hours. Since there is no potential for containment pressurization, the 10 CFR 50, Appendix J leakage criteria and tests are not required.

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

50.67

recently

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. If closed, 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 equipment hatch may be open during ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment, provided it can be installed with a minimum of four bolts holding it in place. During shutdown conditions, adequate missile protection for safety related equipment in containment is provided with the equipment hatch held in place with 6 bolts. Administrative controls ensure the equipment hatch is in place during the threat of severe weather that could result in the generation of tornado driven missiles. (Ref. 5).

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during

BASES

BACKGROUND (continued)

MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when containment penetration 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 ~~CORE ALTERATIONS or~~ recently movement of irradiated fuel assemblies within containment, containment penetration closure is required; however, the door interlock mechanism may remain disabled provided one personnel air lock door is capable of being closed and one emergency air lock door is closed. In the case of the emergency air lock door, a temporary closure device is an acceptable replacement for the air lock door (Ref. 1).

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 containment due to a fuel handling accident involving handling recently irradiated fuel during refueling.

The Containment Purge System includes two subsystems. The shutdown purge subsystem includes a 36 inch supply penetration and a 36 inch exhaust penetration. The second subsystem, a mini-purge system, includes an 18 inch supply penetration and an 18 inch exhaust penetration. During MODES 1, 2, 3, and 4, the two valves in each of the shutdown purge supply and exhaust penetrations are secured in the closed position or blind flange installed. The two valves in each of the two minipurge penetrations can be opened intermittently, but are closed automatically by the Engineered Safety Features Actuation System (ESFAS). Neither of the subsystems is subject to a Specification in MODE 5 or MODE 6 excluding ~~CORE ALTERATIONS or~~ recently movement of irradiated fuel in containment.

In MODE 6, large air exchanges are necessary to conduct refueling operations. The normal 36 inch purge system is used for this purpose, and all four valves may be closed by the ESFAS in accordance with LCO 3.3.6, "Containment Purge Isolation Instrumentation," during ~~CORE ALTERATIONS or~~ movement of recently irradiated fuel in containment.

When the minipurge system is not used in MODE 6, all four 18 inch valves are closed.

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at

BASES

recently irradiated

Additionally, the fuel handling accident analysis does not assume containment isolation to mitigate the dose after 76 hours of decay time. The fuel handling accident analysis does not address fuel movement prior to 76 hours.

BACKGROUND (continued)

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 and the emergency personnel escape lock during fuel movements (Ref. 1).

APPLICABLE SAFETY ANALYSES

During ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 2). Fuel handling accident, analyzed in Reference 2, assumes dropping a single irradiated fuel assembly. The time to close containment penetrations under administrative controls is assumed to be not more than a 2 hour period, consistent with the 2 hour period of release assumed in the accident analysis (Ref. 6). The requirements of LCO 3.9.7, "Refueling Pool Water Level," and the minimum decay time of 76 hours prior to ~~CORE ALTERATIONS~~ ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR ~~100~~. ~~Standard Review Plan, Section 15.7.4, Rev. 1 (Ref. 3), defines "well within" 10 CFR 100 to be 25% or less of the 10 CFR 100 values. The~~ acceptance limits for offsite radiation exposure ~~will be~~ 25% of 10 CFR ~~100~~ values.

the movement of irradiated fuel assemblies

50.67

Per Standard Review Plan 15.0.1, the

50.67

are

Containment penetrations satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO limits the consequences of a fuel handling accident ~~in~~ involving handling recently irradiated fuel. 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 penetrations, the personnel airlock, and the equipment hatch (which must be capable of being closed). For the OPERABLE containment purge penetrations, this LCO ensures that each penetration is isolable by the Containment Purge Isolation System to ensure that releases through the valves are terminated~~, such that radiological doses are within the acceptance limit~~.

recently

One door in the emergency air lock must be closed and one door in the personnel air lock must be capable of being closed. Both containment personnel air lock doors may be open during movement of irradiated fuel ~~or CORE ALTERATIONS~~, provided an air lock door is capable of being closed and the water level in the refueling pool is maintained as required. Administrative controls ensure that 1) appropriate personnel are aware of the open status of the containment during movement of

BASES

LCO (continued)

recently

involving handling recently irradiated fuel

irradiated fuel ~~or CORE ALTERATIONS~~, 2) specified individuals are designated and readily available to close the air lock following an evacuation that would occur in the event of a fuel handling accident, and 3) any obstructions (e.g., cables and hoses) that would prevent rapid closure of an open air lock can be quickly removed (Ref. 4). LCO 3.9.4.b is modified by a Note allowing an emergency escape air lock temporary closure device to be an acceptable replacement for an emergency air lock door.

recently

The equipment hatch may be open during movement of irradiated fuel ~~or CORE ALTERATIONS~~ provided the hatch is capable of being closed and the water level in the refueling pool is maintained as required. Administrative controls ensure that 1) appropriate personnel are aware of the open status of the containment during movement of irradiated fuel ~~or CORE ALTERATIONS~~, 2) specified individuals are designated and readily available to close the equipment hatch following an evacuation that would occur in the event of a fuel handling accident, and 3) any obstructions (e.g., cables and hoses) that would prevent rapid closure of the equipment hatch can be quickly removed.

involving handling recently irradiated fuel

recently

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 ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment, and 2) specified individuals are designated and readily available to isolate the flow path within 2 hours in the event of a fuel handling accident.

involving handling recently irradiated fuel.

APPLICABILITY

recently

the limiting

The containment penetration requirements are applicable during ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment because this is when there is a potential for ~~a~~ 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 ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment ~~are~~ is not being conducted, or the fuel assemblies being moved have been outside a critical reactor core for greater than 76 hours, the potential for ~~a~~ fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.

Due to radioactive decay, the containment penetration requirements are only necessary during fueling handling involving recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 76 hours).

ACTIONS

A.1 ~~and A.2~~

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,

BASES

ACTIONS

A.1 and A.2 (continued)

recently

including the containment purge isolation valve not capable of automatic actuation, the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending ~~CORE ALTERATIONS and~~ movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. For the open purge isolation valves, this Surveillance will ensure that each valve is not blocked from closing and each valve operator has motive power by demonstrating that each valve actuates to its isolation position. Containment penetrations that are open under administrative controls are not required to meet the SR during the time the penetrations are open.

recently

involving handling recently irradiated fuel.

The Surveillance is performed every 7 days during ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. A surveillance before the start of refueling operations will provide sufficient surveillance verification during the applicable period for this LCO. As such, this Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in a release of fission product radioactivity to the outside atmosphere.

SR 3.9.4.2

significant

This Surveillance demonstrates that the necessary hardware, tools, and equipment are available to install the equipment hatch. The equipment hatch is provided with a set of hardware, tools, and equipment for moving the hatch from its storage location and installing it in the opening. The required set of hardware, tools, and equipment shall be inspected to ensure that they can perform the required functions.

The Surveillance is performed every 7 days during ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within the containment. The Surveillance interval is selected to be commensurate

recently

B 3.9 REFUELING OPERATIONS

B 3.9.7 Refueling Pool 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 fuel transfer canal, refueling pool, and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to < 25% of ~~10 CFR~~ 50.67 ~~100~~ limits, as provided by the guidance of Reference ~~3 and acceptance in Reference 6~~ 1.

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies, the water level in the refueling pool 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). The reactor is assumed to have been subcritical for 76 hours prior to movement of irradiated fuel in the reactor vessel. A minimum water level of 23 ft ~~(Regulatory Position C.1.c of Ref. 1)~~ allows a decontamination factor ~~of 100~~ 200 (Regulatory Position ~~C.1.g of~~ B.2 Ref. 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 pool water. ~~In addition~~ For the analyses ~~for the accident in the reactor building~~ the dropped assembly is assumed to damage 20 percent of the rods of an additional assembly. ~~The fuel pellet to cladding gap is assumed to contain 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 and a minimum decay time of 76 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained well within allowable limits (Refs. ~~4, 5, and 6~~ 2 and 5).

Refueling pool water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

REFERENCES

1. Regulatory Guide ~~1.25, March 23, 1972~~. 1.183, July 2000.

2. USAR, Section 15.7.4.

3. ~~NUREG-0800, Section 15.7.4, Rev 1, July 1981.~~ Not used.

4. 10 CFR ~~100.11~~. 50.67

5. Malinowski, D. D., Bell, M. J., Duhn, E., and Locante, J., WCAP-828, Radiological Consequences of a Fuel Handling Accident, December 1971.

6. ~~NUREG-0881, Safety Evaluation Report, Wolf Creek Generating Station, April 1982, Section 15.4.6.~~ Not used.

7. NUREG/CR-5009, February 1988.

12 PROPOSED TECHNICAL REQUIREMENTS MANUAL MARKUPS (For Information Only)

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.10.3.1	----------NOTE---------- This surveillance is not required when swapping the contents from one tank to another, when the tank contains less than ~~2.0 x 10^5~~ 5.0×10^4 curies of noble gases (considered as Xe-133 equivalent). ----------	
	Verify quantity of radioactive material in each tank is ≤ ~~2.5 x 10^5~~ 1.0×10^5 Curies of noble gases (considered as Xe-133 equivalent).	Once per 7 days during addition of radioactive material to the tank AND Once within 7 days following addition of radioactive material to the tank

BASES

BACKGROUND (continued)

The Condenser Air Discharge Monitor (GE RE-92) is provided to detect, indicate, and alarm gaseous activity in the Condenser Air Removal System exhaust. The monitor closes the steam generator blowdown isolation valves on high radiation to prevent discharge of radioactive fluid and to limit radioactive contamination of the blowdown demineralizers. (Reference 2.)

The Steam Generator Blowdown Process Radiation Monitor (BM RE-25) continuously monitors the fluid entering the steam generator blowdown filters to detect, alarm, and indicate excessive radioactivity levels in the blowdown system. This monitor acts to terminate blowdown from the steam generators to prevent discharge of radioactive fluid and to limit radioactive contamination of the blowdown deminerlizers. (Reference 3).

The Steam Generator Liquid Radiation Monitor (SJ RE-02) continuously monitors the blowdown from the steam generators, either individually or collectively, to detect, indicate, and alarm primary to secondary system leaks in the steam generators. This monitor closes the steam generator blowdown isolation valves on high radiation to prevent the discharge of radioactive fluid and to limit radioactive contamination of the blowdown demineralizers. (Reference 4).

APPLICABLE SAFETY ANALYSES

The safety analyses for events resulting in steam discharge to the atmosphere assume a 1 gpm primary to secondary LEAKAGE as an initial condition. Primary to secondary LEAKAGE is a factor in the dose release outside containment resulting from a steam line break accident. Other accidents or transients involving secondary steam release to the atmosphere~~,~~ include the steam generator tube rupture, locked RCP rotor, loss of AC power, and rod ejection with no release into containment. The leakage contaminates the secondary fluid.

TR

This TR requires the radiation monitors used to detect and monitor steam generator primary to secondary LEAKAGE be FUNCTIONAL when needed for detection and monitoring. A radiation monitor is FUNCTIONAL if it is directly correlated to gpd leakage, can be monitored, will produce an alarm in the control room, and can detect leak rates greater than 30 gpd at existing Reactor Coolant System (RCS) activity levels. The capability to isolate steam generator blowdown is not required for radiation monitor FUNCTIONALITY as this TR is to ensure the monitor is capable of detecting and monitoring primary to secondary LEAKAGE.

During normal operation, process radiation monitors and radiochemical grab sampling provide indication of primary to secondary LEAKAGE.

B 3.7 PLANT SYSTEMS

TR B 3.7.13 Emergency Exhaust System (EES) for Crane Operation – Fuel Building

BASES

BACKGROUND

A description of the EES is provided in the Bases for Technical Specification 3.7.13, "Emergency Exhaust System," (Ref. 1).

APPLICABLE SAFETY ANALYSES

The EES has a design function to filter radioactive particles which have been released as a result of a fuel handling accident. The OPERABILITY of the EES with respect to a fuel handling accident is addressed by Technical Specification 3.7.13 (Ref. 1). The dose consequences of dropping of a light load (i.e, load ≤ 2250 pounds) into the spent fuel pool storage area, which may result in partial damage to one or more irradiated fuel assembly(s) is less than the dose consequences of a fuel handling accident. Therefore, since the potential for damage exists which would result in the release of radioactive material, it is necessary for operational requirements to be in place to protect against the inadvertent release of radioactive materials to the environment. Heavy loads (i.e., loads in excess of 2250 pounds), with the exception of the spent fuel transfer gates, are prevented from being moved over fuel assemblies in the spent fuel storage facility by crane travel interlocks and physical stops. FUNCTIONALITY requirements for crane travel interlocks and physical stops are specified in TR 3.7.17, "Crane Travel - Spent Fuel Storage Facility," (Ref. 2).

TR

Two independent and redundant trains of the EES are required to be FUNCTIONAL to ensure that at least one train is available, assuming a single failure that disables the other train, coincident with the loss of offsite power. Total system failure could result in the atmospheric release from the fuel building. Such a release is not expected to exceed the guideline limits of 10 CFR ~~100~~ 50.67 for the situation addressed by this TR.

The EES is considered FUNCTIONAL when the individual components necessary to control releases from the fuel building are FUNCTIONAL in both trains. An EES train is considered FUNCTIONAL when its associated:

a. Fan is FUNCTIONAL;

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

B 3.9 REFUELING OPERATIONS

TR B 3.9.7 Refueling Pool Water Level

BASES

BACKGROUND

While the conditions involved in the movement of control rods within the reactor vessel during MODE 6 do not meet the initial conditions associated with a fuel handling accident, maintaining a minimum water level of 23 feet above the top of the irradiated fuel will continue to provide for a decontamination factor of ~~100~~ 200 to be applied to the capture of iodine. This decontamination factor was initially calculated as part of an accident analysis for a dropped fuel assembly. A decontamination factor of ~~100~~ 200 is calculated based upon the assumption that ~~99%~~ 99.5% of the total iodine released from the pellet to cladding gap is retained by the refueling pool water. ~~The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory.~~

Therefore, if activities connected to the movement of control rods within the reactor vessel during MODE 6 results or facilitates in the release of iodine from a failed fuel assembly, there will be a sufficient water level to capture the escaping iodine.

APPLICABLE SAFETY ANALYSES

The reactor water level during refueling operations has not been shown to be significant to public health and safety by either operational experience or PSA (Ref. 1). During the movement of irradiated fuel, refueling pool water level is addressed by Technical Specification 3.9.7, "Refueling Pool Water Level." The activity associated with the movement of control rods under these circumstances is not viewed as an activity which would result in fuel failure, however if the irradiated fuel assembly has already experienced some level of failure, a refueling pool water level of ≥ 23 feet will provide a iodine decontamination factor of ~~100~~ 200 as well as provide shielding for the refueling personnel who will be working directly above the core.

TR

This TR is written to provide assurance that the refueling pool water level during the movement of control rods within the vessel while in MODE 6 is greater than or equal to 23 ft above the top of the irradiated fuel assemblies. This water level is required to capture iodine which may be released as a result of control rod manipulation from an irradiated fuel assembly which is already damaged.

APPLICABILITY

This TR is only applicable in MODE 6 during the movement of control rods within the reactor pressure vessel.

B 3.10 EXPLOSIVE GAS AND STORAGE TANK RADIOACTIVITY MONITORING

TR B 3.10.3 Gas Storage Tanks

BASES

BACKGROUND

The gas storage tanks, commonly known as the gas decay tanks, are components within the Gaseous Radwaste System (GRWS). The GRWS's main flow path is a closed loop comprised of two waste gas compressors, two catalytic hydrogen recombiners, six gas decay tanks for normal power service, and two gas decay tanks for service at shutdown and startup. The tanks are of the vertical-cylindrical type and are constructed of carbon steel (Ref. 1).

The primary source of the radioactive gases for the GRWS is from the volume control tank during purging evolutions. The operation of the GRWS serves to reduce the fission gas concentration in the reactor coolant system, which in turn, reduces the escape of fission gases from the Reactor Coolant System (RCS) during maintenance operations or through equipment leakage. Smaller quantities are received from the recycle evaporator gas stripper, the reactor coolant drain tank, the pressurizer relief tank, and the recycle holdup tanks (Ref. 1).

Operation of the system is such that fission gases are distributed throughout the six normal operation gas decay tanks. Separation of the GRWS gaseous inventory is several tanks assures that the allowable site boundary dose will not be exceeded in the event of a gas decay tank rupture (Ref. 1).

The GRWS also provides the capacity for indefinite holdup of gases generated during reactor shutdown. Nitrogen gas from previous shutdowns is contained in the shutdown gas decay tank for use in stripping hydrogen from the reactor coolant system (Ref. 1).

The tanks included in this TR are those tanks for which the quantity of radioactivity contained is not limited directly or indirectly by another TS or TR. Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting whole body exposure to a member of the public at the nearest site boundary will not exceed ~~0.5~~ 0.1 rem. This is consistent with Reference 2.

Technical Specification 5.5.12, "Explosive Gas and Storage Tank Radioactivity Monitoring," (Ref. 3) requires a surveillance program to ensure that the quantity of radioactivity contained in the gas storage tanks is less than the amount that would result in a whole body exposure of ≥ 0.1 ~~0.5~~ rem to any individual is an unrestricted area, in the event of an uncontrolled release of the tanks contents.

BASES

APPLICABLE SAFETY ANALYSES

The limitations imposed on the radioactive content of each gas storage tank governed by this TR are put in place so as to ensure that the public is not exposed to doses from gaseous effluents in excess of the requirements of 10CFR Part 20 to unrestricted areas. The total dose to the public ensures that the dose limitations of 40CFR Part 190 which have been incorporated in 10CFR Part 20 are met (Ref. 4).

These requirements are not important to dominant risk sequences as defined in Reference 4.

TR

TR 3.10.3 is provided to ensure that the radioactive material contained in each gas storage tank is less than ~~2.5×10^5~~ 1.0×10^5 Curies of noble gases. (Calculated based on Xe-133 equivalent.)

APPLICABILITY

Radioactive content of the gas storage tanks addressed by this TR is required to be monitored and maintained within limits at all times.

ACTIONS

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Requirement may be entered independently for each tank listed in the TR. The Completion Time(s) associated with each tank outside of its requirements will be tracked separately for each tank starting from the time the Condition was entered for that tank.

A.1, A.2, and A.3

Once the quantity of radioactive material is determined to be greater than the limit, addition of radioactive material is required to be suspended immediately and the level reduced to within limits within 48 hours. The immediate Completion Time for the suspension of the addition of radioactive material to the tank is consistent with the required times for actions to be performed without delay and in a controlled manner. The 48 hour Completion Time for the restoration of the tanks radioactive content to within limits is based upon operating experience and is reasonable considering the time it will take to identify the problem and take the proper corrective actions.

Condition A is modified by a Note that requires that Required Action A.3 must be completed whenever Condition A is entered. The Note emphasizes the need to initiate a Condition Report (CR) regardless of whether the gas storage tank(s) is restored to within limits.

BASES

ACTIONS

A.1, A.2, and A.3 (continued)

The initiation of a CR ensures that the event will be included in the next Radioactive Effluent release Report in accordance with Technical Specification 5.6.3.

B.1

In the event that the Required Action and associated Completion Time are not met, Required Action B.1 requires initiation of a CR immediately to address why the gas storage tank(s) was not restored to within limit within the Completion Time. As part of the initiation of the CR, action shall be implemented in a timely manner to place the unit in safe condition as determined by plant management. The CR should provide an accurate description of the problem, the Required Action and associated Completion Time not complied with. The intent of this Required Action is to utilize the corrective action program to assure prompt attention and adequate management oversight to minimize the additional time the tank(s) is not within the limit.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.10.3.1

Demonstrating that the quantity of radioactive material in each tank is within limits at a Frequency of within 7 days following additions and once per 7 days during additions provides adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

The use of Xe-133 equivalent for the curie limit is based on Xe-133 being the predominant moble gas accumulated is the reactor coolant and transferred to the gas storage tanks and is therefore used as the isotope source term.

A Note has been added to allow the exception of the performance of this surveillance as a result of swapping the contents of one gas storage tank to another when the tank being swapped contains less than ~~2.0 x 10^5~~ 1.0×10^5 curies (which is less than the ~~2.5 x 10^5~~ 5.0×10^4 limit). This is acceptable based upon Reference 5, which calculates the maximum increase in curie content as a result of swapping the contents from one tank to the other considering the volume of piping between tanks.

BASES

REFERENCES

1. USAR, Section 11.3.2.1.

2. Standard Review Plan ~~11.3~~, Branch Technical Position ~~ETSB~~ 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," in NUREG-0800, ~~July 1981~~. March 2007

3. Technical Specification 5.5.12, "Explosive Gas and Storage Tank Radioactivity Monitoring Program."

4. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.

5. USAR Change Request 96-074.

13 PROPOSED UPDATED SAFETY ANALYSIS REPORT CHANGES (For Information Only)

c. Meet a specific operator interface, startup, or specific testing requirement.

d. Meet a design classification or code requirement (e.g., be designed to withstand the safe shutdown earthquake).

Items implicit in contemporary design practices (e.g., use of the English system of weights and measures or the exercise of good engineering practice) are not specified as design bases.

Safety design bases are engineering objectives which must be met by safety-related structures, systems, or components.

Safety-related items are defined as those plant features necessary to ensure the following:

a. The integrity of the reactor coolant pressure boundary.

b. The capability to shut down the reactor after a design basis accident and maintain it in a post-accident safe shutdown condition.

c. The capability to prevent or mitigate the consequences of accidents that could potentially result in offsite exposures approaching the guideline exposures of 10 CFR ~~100~~. 50.67

Items which are associated with safety-related equipment, but which in themselves are not absolutely essential to the safety function of the equipment, are not considered safety-related.

Power generation design bases support, either directly or indirectly,the major electrical power generation function of the station. Examples of power generation design bases are the requirements to provide adequate radiation shielding and domestic water for plant personnel.

Sections describing Westinghouse-supplied systems and components do not provide safety design bases or power generation design bases as such. These sections do give functional descriptions and are in compliance with Regulatory Guide 1.70.

TABLE 1.3-4 (Sheet 19)

Regulation (10 CFR)	Compliance
50.63	This regulation pertains to the Station Blackout rule.
50.64	This regulation pertains to non-power reactors only and is not applicable to WCGS.
50.65	This regulation requires the implementation of a program to monitor the effectiveness of maintenance programs by monitoring performance of plant SSCs. Plant procedures implement and control this program.
50.68	This regulation provides the licensee with eight requirements that may be complied with in lieu of compliance with 10CFR70.24 for criticality monitoring. WCGS complies with this regulation.
50.67	
50.70	The Commission has assigned resident inspectors to WCGS and space was provided in conformance with 50.70(b)(1) through (3).
50.71	Records are and will be maintained and reports will be made in accordance with the requirements of sections (a) through (e) of this regulation and the license.
50.72	This regulation provides the immediate notification requirements for operating nuclear power reactors.
50.73	This regulation requires the licensee to submit Licensee Event Reports for certain specific events.
50.74	This regulation requires the licensee to notify the NRC pertaining to a change in Reactor Operator or Senior Reactor Operator status.
50.78	This regulation pertains to holders of construction permits and does not apply to WCGS.
50.80	This regulation provides that licenses may not be transferred without NRC consent. No application for transfer has been made by the WCGS Licensees.
50.81	This regulation permits the creation of mortgages, pledges, and liens on licensed facilities, subject to certain provisions. The regulation prohibits secured creditors from violating the Atomic Energy Act and the Commission's regulations.
50.82	This regulation provides for the termination of licenses. It does not apply to WCGS because no termination of licenses has been requested.

The USAR accident analyses, particularly those in Chapter 6.0 and 15.0, demonstrate that offsite doses resulting from postulated accidents would not exceed the criteria in this section of the regulation.

TABLE 1.3-4 (Sheet 36)

Regulation (10 CFR)	Compliance
100.10	The factors listed related to both the unit design and the site have been provided in the application. Site specifics, including seismology, meteorology, geology, and hydrology, are presented in Chapter 2.0 of the USAR. The exclusion area, low population zone, and population center distance are provided and described. The USAR also describes the characteristics of reactor design and operation.
100.11	Exclusion areas have been established, as described in Section 2.1. The low population zone has been established in accordance with this requirement. ~~The USAR accident analyses, particularly those in Chapters 6.0 and 15.0, demonstrate that offsite doses resulting from postulated accidents would not exceed the criteria in this section of the regulation.~~
Appendix A	Appendix A to 10 CFR Part 100 provides seismic and geologic siting criteria for nuclear power plants. Site suitability was determined at the construction permit stage.

g. Energy Research & Consultants Corporation

This consultant reviewed design and operation of pumps and other rotating equipment, including advising WCGS during the bid evaluation for several pumps, and performing tests necessary to evaluate the auxiliary feedwater pumps.

h. ~~Dr. James Halitsky~~ Fauske and Associates, LLC.

This consultant developed calculations of atmospheric dispersion ~~parameters~~ factors for the control room fresh air intake for use in control room accident dose /TSC calculations. /TSC

i. Energy Incorporated

This consultant was engaged to assist the SNUPPS utilities to develop an independent plant transient and analysis capability using the RETRAN computer code.

j. Essex Corporation

This consultant was engaged to perform an independent design evaluation of the SNUPPS control room, emphasizing human factors considerations.

1.4.7.2 WCGS Specific Consultants

a. Dames & Moore

The independent consulting firm of Dames & Moore was retained to perform site investigations relating to demography, geography and land use, meteorology, hydrology, geology and seismology. Having performed such safety-realted and environmental impact related investigations for over 75 nuclear power plant sites, Dames & Moore is an acknowledged leader in the field of site investigations related to nuclear plant construction.

LIST OF TABLES (Continued)

Insert A

Results at Low Population Zone for Release from RWST

Results at Exclusion Area Boundary for Release from RWST

INSERT A

Table No.	Title
Table 2.3-54a	Joint Frequency Distribution (in percent of total hours) for Stability Class A
Table 2.3-54b	Joint Frequency Distribution (in percent of total hours) for Stability Class B
Table 2.3-54c	Joint Frequency Distribution (in percent of total hours) for Stability Class C
Table 2.3-54d	Joint Frequency Distribution (in percent of total hours) for Stability Class D
Table 2.3-54e	Joint Frequency Distribution (in percent of total hours) for Stability Class E
Table 2.3-54f	Joint Frequency Distribution (in percent of total hours) for Stability Class F
Table 2.3-54g	Joint Frequency Distribution (in percent of total hours) for Stability Class G
Table 2.3-54h	Wind Direction Occurrence Frequency
Table 2.3-54i	Wind Speed Occurrence Frequency

LIST OF TABLES (Continued)

CHAPTER 2 - LIST OF FIGURES

*Refer to Section 1.6 and Table 1.6-3. Controlled drawings were removed from the USAR at Revision 17 and are considered incorporated by reference.

Replace with "Deleted"

$T_A - T_W = 10^{o}C$

$\Delta T/\Delta Z = -.015^{o}C/m$ (C stability)

The mixing depth (H) for this conservative case will not exceed approximately 20 meters. Since an elevated release from the 60-meter vent would not easily penetrate to groundlevel through this inversion layer, X/Q values would generally be lower than the present analyses.

Conclusion

where only a ground-level release is considered

Only for Case~~s 2 and~~ 7 would ~~an~~ analysis which considers the presence of the cooling lake tend to be more conservative than the existing analysis of Sections 2.3.4 and 2.3.5. For Cases 1, 5, 8, and perhaps 6 the existing analysis should be more conservative.

Case 2 is an elevated release.

Case~~s 2 and~~ 7 will differ from the present analyses only in the immediate vicinity of the cooling lake and then only for wind directions which would produce the largest over-water fetch (i.e., N, S, NW, and SSE). From three years of onsite data at 10- and 60-meter wind levels (Tables 2.3-29 and 2.3-30) stable stability classes (E, F, and G) occur approximately ~~20~~ 46 percent of the time and unstable classes (A, B, and C) occur approximately ~~9~~ 23 percent of the time.

It is expected that over long averaging periods the effect of Cases 1 and 8 will tend to balance the effect of Cases ~~2~~ 7 and ~~7~~ 2. ~~The short-term accident analyses presented in Tables 2.3-55 through 2.3-57 show strong stable cases resulting from Case 7. With respect to Case 2, it is expected that the resulting fumigation will not result in a X/Q value which exceeds the X/Q values of a ground level release in a stable atmosphere.~~

A preoperational fog monitoring program was used to evaluate the meteorological impacts of the cooling lake. The purpose of the study was to document the frequency of occurrence of natural fog (as opposed to fogs induced by the operation of the cooling lake) along Highway 75 which is located from 0.5 miles to 2.0 miles west of the cooling lake.

Table 2.3-29 shows that the predominant frequency of light wind (less than 3 meters per second) is from the sectors southeast through south. This corresponds with the Dames & Moore Program FOGALL analyses which shows the maximum increase in cooling lake induced fogging frequency along Highway 75 to occur approximately 3 miles south through 2 miles north of New Strawn, Kansas.

c. Frequency distribution of the relative concentration;

d. Annual average values of relative concentration with direction and distance.

2.3.3.7.3.2 Operational Program

The Operational Meteorlogical Program records and parameters identified in Table 2.3-48. A description of data availability during plant operation is provided in Section 2.3.3.1 above. The Operational Program meets the requirements of Regulatory Guide 1.23

2.3.3.8 Regional Climatological Data

Regional climatological data concerning wind direction and wind speed were based on measurements taken at Chanute Flight Service Station, Kansas, over the period 1955-1964. Regional data concerning temperature, atmospheric water vapor, and precipitation were based on measurements taken by the National Weather Service over the period 1941-1978 at Topeka and Wichita, Kansas. The elevations of the regional measurements are provided in Table 2.3-53.

2.3.4 Short-Term (Accident) Diffusion Estimates

Replace the entire Section 2.3.4 with text of Insert B

~~The objective of this section is to provide conservative estimates of atmospheric diffusion at both the site boundary and at the outer limits of the low population zone (LPZ) for appropriate time periods up to 30 days and to provide the short-term atmospheric dispersion factors (X/Qs) for the postulated accident analyses presented in Chapter 15. The diffusion evaluations for the short-term accident are based on the assumption of a ground-level release (i.e., no reduction in ground concentrations due to elevation of the plume). The plant parameters used in the calculations are presented in Table 2.3-54. Meteorological data used are described in the joint frequency distributions of wind speed, wind direction, and of the WCSA atmospheric stability presented in Section 2.3.2.~~

~~Table 2.3-59d lists the limiting X/Qs for the Wolf Creek site. The detailed procedures used in the calculations are given in Section 2.3.4.2.~~

2.3.4 Short-Term (Accident) Diffusion Estimates

This section presents a re-evaluation of short-term diffusion estimates based on recent 5-year meteorological data (2006-2010) and an updated methodology for atmospheric dispersion estimates. The previous evaluation was based on 3-years of meteorological data taken from 6/1/1973 through 5/31/1975 and 3/5/1979 through 3/4/1980. The objective of this section is to provide conservative estimates of atmospheric diffusion at both the exclusion area boundary (EAB) and at the outer limits of the low population zone (LPZ) as well as at the air intake of the control room and of the technical support center (TSC) for appropriate time periods up to 30 days and to provide the short-term atmospheric dispersion factors (χ/Qs) for the postulated accident analyses presented in Chapter 15. The diffusion evaluations for the short-term accident are based on the assumption of a ground-level release (i.e., no reduction in ground concentrations due to elevation of the plume). The plant parameters used in the evaluations are presented in Table 2.3-54. Five consecutive years of hourly measured site-specific meteorological data from January 1, 2006 to December 31, 2010 were used in the evaluations. Meteorological data used are described in the joint frequency distributions of wind speed, wind direction, and of the atmospheric stability class presented in Table 2.3-54a through 2.3-54i. Five release sources including the unit vent stack, equipment hatch, MSSVs/ARVs vent, TDAFW exhaust vent, and RWST were considered in the evaluations.

The evaluation of EAB and LPZ χ/Qs was performed according to the methodology of Regulatory Guide 1.145 Rev. 1 (Reference 50) using the PAVAN code (Reference 51). The PAVAN code is a computer program specifically developed for the NRC for implementing the Regulatory Guide 1.145 methodology for such calculations. A building wake effect and a site-specific terrain correction factor are included in the calculation. The PAVAN model is based on the straight-line Gaussian Dispersion Model.

Likewise, the evaluation of control room and TSC χ/Qs was performed according to the methodology of Regulatory Guide 1.194 (Reference 52) using the ARCON96 code (reference 53). The ARCON96 code is a computer program specifically developed for the NRC for implementing the Regulatory Guide 1.194 methodology. The ARCON96 model is based on the straight-line Gaussian dispersion model that takes into consideration the building wake effect.

Table 2.3-59d lists the limiting EAB and LPZ χ/Qs for the Wolf Creek site. Tables 2.3-78 and 2.3-79 list the χ/Qs for the control room air intake and for the TSC air intake, respectively.

2.3.4.1 Method for EAB and LPZ

The PAVAN method for determination of the atmospheric dispersion factor χ/Q for site boundary or offsite location is to calculate three different types of χ/Q from the meteorological data input. These three types of χ/Q are (1) the maximum sector χ/Q (described in section 2.3.4.1.1), (2) the overall site χ/Q (described in section 2.3.4.1.2), and the annual average χ/Q (described in section 2.3.4.1.3). Then according to Regulatory Guide 1.145, the 0 to 2-hour time period χ/Q is defined as the larger value between the maximum sector χ/Q value and the overall site value. The χ/Q values for intermediate time periods including 0 to 8 hours, 8 to 24 hours, 1 to 4 days, and 4 to 30 days are determined by logarithmic interpolation between the 0 to 2 hour period χ/Q value and the annual average χ/Q value.

2.3.4.1.1 Maximum Sector χ/Q

For each of the 16 downwind direction sectors, χ/Q values were calculated for each combination of wind speed category and atmospheric stability class at an EAB distance and an LPZ distance. The χ/Q values calculated for each sector are arranged in order from largest to smallest, and the associated cumulative frequency distribution is derived based on the frequency distribution of wind speed and stability class for that sector. The smallest χ/Q value

for the sector has a cumulative frequency of 100 percent for that sector. Then the 0.5 percent χ/Q value for the sector that corresponds to the cumulative frequency of 0.5 percent of the total time of all sectors is determined by logarithmic interpolation. Such 0.5 percent χ/Q values are calculated for all 16 sectors. Then the maximum sector χ/Q value is selected from the maximum value of the 16 sectors.

2.3.4.1.2 Overall Site χ/Q

PAVAN calculates the overall site χ/Q value by combining χ/Q values from all wind sectors into a cumulative frequency distribution for the entire site. Then the overall site χ/Q value is selected from the value that corresponds to 5.0 percent of the total time.

2.3.4.1.3 Annual Average χ/Q

The annual average χ/Q value is calculated in PAVAN by using a long-term continuous release model (as opposed to short-term release) described in Regulatory Guide 1.111. In this model, the plume horizontal dispersion is assumed to be evenly distributed to fill the entire width of the 22.5-degree wind sector.

2.3.4.2 Results of Short-Term Diffusion Estimates for EAB and LPZ

The χ/Q results that account for the building wake effect due to the reactor building are presented in Table 2.3-55 for the EAB and in Table 2.3-56 for the LPZ. These results are applied to most release sources including the unit vent stack, equipment hatch, MSSVs/ARVs vent, and TDAFW exhaust vent. These release sources are located within close proximity of the reactor building and would be impacted by the wake effect of the reactor building. For a release from the RWST, another set of χ/Q calculations that only accounts for the wake effect of the RWST itself is presented in Table 2.3-57 for the EAB and in Table 2.3-58 for the LPZ.

According to Regulatory Guide 1.145, the χ/Q value for use in the dose assessment is the larger of the maximum sector χ/Q value and the overall site value. This is presented in Table 2.3-59d as limiting χ/Q values. These limiting χ/Q values are associated with RWST as the release source. The RWST release source bounds all other sources. The difference between the RWST source and other sources is the building wake effect. The RWST source has less building wake effect than other potential sources.

2.3.4.3 Control Room and Technical Support Center Air Intake

Atmospheric dispersion factors (χ/Q, sec/m^3) were calculated for the air intake of the control room and the technical support center (TSC) from the following release sources: (1) Equipment Hatch, (2) Unit Vent Stack, (3) MSSVs/ARVs Vent, (4) RWST Vent, and (5) Turbine-Driven AFW Exhaust Vent. The calculation was performed with the ARCON96 code (Reference 53). ARCON96 is an NRC accepted methodology for determining atmospheric dispersion factors (χ/Q) in the design basis accident evaluations of control room radiological consequences. ARCON96 is a computer program for calculating atmospheric relative concentrations in plumes in building wakes under a wide range of situations. ARCON96 implements a straight-line Gaussian dispersion model with dispersion coefficients that are modified to account for low wind meander and building wake effects. The contiguous building arrangement is shown in Figure 2.3-20. Hourly, normalized concentrations (χ/Q) were calculated from hourly meteorological data collected during a 5-year period from January 1, 2006 to December 31, 2010. The hourly values are averaged to form χ/Q's for periods ranging from 2 to 720 hours in duration. The calculated values for each period are used to form cumulative frequency distributions and 95th percentile χ/Q values. The input parameters to the ARCON96 code were prepared according to the guidance on the use of ARCON96, as discussed in Regulatory Guide 1.194 (Reference 52)

and are summarized in Table 2.3-54. All release sources were treated as a point source, ground level release in the calculation.

The results for all source-receptor pairs of the 95th percentile χ/Q values averaged over periods of 0-2 hr, 2-8 hr, 8-24 hr, 1-4 days, and 4-30 days are given in Table 2.3-78 for the control room and in Table 2.3-79 for the TSC.

Insert B

2.3.4.1 Diffusion Model for 0-2 Hours

The analytical procedure for evaluating the 0-2 hour accident period is based on a revision of the model described in Regulatory Guide 1.3. The changes reflect variations in atmospheric diffusion factors that occur as a function of wind direction and variable site boundary distance. Allowances are made for meandering plumes during light winds and stable atmospheric conditions. The new approach is described in Regulatory Guide 1.145.

The model is distance and direction dependent. Variability of wind direction frequency was considered in determining the relative concentration (X/Q) values. The hourly X/Q values were determined as described below.

During neutral (D) or stable (E, F, or G) atmospheric stability conditions when the windspeed at the 10-meter level is less than 6 meters per second, horizontal plume meander can be considered. X/Q values were determined through selective use of the following set of equations for ground-level relative concentrations at the plume centerline:

$$X/Q = \frac{1}{\bar{u}_{10}\ (p s_y s_z + A/2)} \qquad [2.3\text{-}27]$$

$$X/Q = \frac{1}{\bar{u}_{10}\ (3 p s_y s_z)} \qquad [2.3\text{-}28]$$

$$X/Q = \frac{1}{\bar{u}_{10} p \sum_y s_z} \qquad [2.3\text{-}29]$$

where:

X/Q = relative concentration, in sec/m3,

= 3.14159,

$\bar{u}_{10}$ = windspeed at 10 meters above plant grade,* in m/sec,

s_y = lateral plume spread, in m, at a given distance and stability based on logarithmic curves in Reg. Guide 1.145,

*The 10-meter level is representative of the depth through which the plume is mixed with building wake effects.

Insert B

z = vertical plume spread, in m, at a given distance and stability based on logarithmic fit of NRC curves in Reg. Guide 1.145,

Sy = lateral plume spread with meander and building wake effects, in m, a function of atmospheric stability, windspeed $\bar{u}_{10}$, and distance. For distances of 800 meters of less, $y = Ms_y$. For distances greater than 800 meters,

$$S\ y = (M - 1)\ s_y 800m + s_y \qquad [2.3\text{-}30]$$

A = the smallest vertical-plane cross-sectional area of the reactor building, in m2

X/Q values were calculated using equations 2.3-27, 2.3-28, and 2.3-29. The values from equations 2.3-27 and 2.3-28 were compared and the higher value selected. This value was compared with the value from Equation 2.3-29 and the lower value of these two was selected as the appropriate X/Q value.

During all other meteorological conditions (unstable (A, B, or C) atmospheric stability and/or 10-meter level wind-speeds of 6 meters per second or more), plume meander was not considered. The appropriate X/Q value was the higher value calculated from Equation 2.3-27 or 2.3-28.

Plume meander was accounted for by modifying the lateral diffusion coefficient s_y in accordance with equation 2.3-30. The meander function (M) is calculated as explained below:

1. For Pasquill Stabilities A-C at all wind speeds or all stabilities when wind speed >6 mps, M = 1;

2. For wind speed <2 mps, M is independent of wind speed and varies in the following manner:

   Stability D, M = 2;
   Stability E, M = 3;
   Stability F, M = 4;
   Stability G, M = 6;

3. For wind speeds greater than 2 mps but less than 6 mps, M is determined from Figure 3 of Regulatory Guide 1.145.

Insert B

An hourly observation is considered to be calm if the wind speed is less than the starting speed (threshold) of the wind instruments. For calm conditions a wind speed is assigned equal to the vane or anemometer starting speed, whichever is higher. A wind direction is assigned in proportion to the directional distribution of the lowest non-calm wind speed group for each atmospheric stability class.

2.3.4.1.1 Exclusion Area Boundary

The sector X/Q values at the exclusion boundary are determined for each sector. These are defined as the X/Q values that are exceeded 0.5 percent of the total time. To extract this value, the hourly /Q values are sorted according to sector and magnitude. A cumulative probability distribution or X/Q values can easily be constructed.

$$P(X/Q) = \frac{\text{rank of } X/Q}{X/Q \text{ population size}} \qquad [2.3\text{-}31]$$

P(X/Q) is the probability of being exceeded. For example, the 10th largest value of a 100-value population has a probability of being exceeded 10/100 or 10 percent. The highest of the 16 sector X/Q values is defined as the maximum sector X/Q value.

2.3.4.1.2 Outer LPZ Boundary

Sector X/Q values are determined for the outer LPZ for 8 and 16 hours and 3 and 26 days. The average /Q values for the various time periods are approximated for each sector by a logarithmic interpolation between the 2-hourly sector[1] X/Q values (same general methods as in Section 2.3.4.1.1) and the annual average X/Q (see Section 2.3.5) at the same point. The highest of the 16 sector X/Q values are identified for each time period.

2.3.4.1.3 Five and Fifty Percent Overall Site X/Q Value

The X/Q values that are exceeded no more than 5 and 50 percent of the total time around the exclusion area boundary and the outer LPZ boundary are determined in a manner similar to the 0.5 percent sector X/Q values. All of the hourly X/Q values were sorted according to magnitude (independent of the direction) and the 5 and 50 percent values chosen from the list. For the same time periods used in Section 2.3.4.1.2, the 5 and 50 percent X/Q values

[1]The X/Q's are based on 1-hour averaged data, but are assumed to apply for 2 hours.

Insert B

are determined by logarithmic interpolation between the maximum annual average X/Q values at the LPZ distance and the LPZ 2-hour 5 and 50 percent X/Q value.

2.3.4.2 Results of Short-Term Diffusion Estimates

Two-hour X/Q values were computed at the exclusion zone boundary (1200 m) and X/Q values for 2-, 8-, 16-, 72-, and 624-hour postulated accident periods were computed at the LPZ (4023 m). The computations were based on onsite meteorological data for three one-year data sets; June 1, 1973 through May 31, 1975, and March 5, 1979 through March 4, 1980. An analysis was also performed for the 3 years of data combined.

Results of the analysis for each data set and the combined three-year period are presented in Tables 2.3-55 through 58. Each table presents the greatest 0.5 percent 0-2 hour X/Q values for each of the 16 sectors at the exclusion zone boundary (1200 m) and the greatest 0.5 percent 2-, 8-, 16-, 72-, and 624-hour X/Q values for each of the 16 sectors at the LPZ (4023 m). The highest sector value for each accident period is asterisked to clarify the maximum sector X/Q value at the exclusion zone boundary and the LPZ for each accident period. Also presented in each table are the greatest 0-2 hour 5 and 50 percent X/Q values at the exclusion zone boundary and the greatest 5 and 50 percent X/Q values for each accident period at the LPZ.

The highest 0.5 percent 2-hour X/Q values at the exclusion zone boundary was 1.5×10^{-4} for all 3 individual years and for the 3 years combined. The highest values occurred in the northwest through north sectors. The maximum sector X/Q at the LPZ from this data set was 5.0×10^{-5} sec/m3 in the northwest sector for the data period March 5, 1979 through March 4, 1980. The highest 5 and 50 percent 2-hour X/Q values resulted from the analysis of the March 5, 1979 through March 4, 1980 data set. The greatest 5 and 50 percent X/Q values were 4.5×10^{-5} sec/m3 and 5.0×10^{-6} sec/m3 for the LPZ and 1.5×10^{-4} sec/m3 and 2.8×10^{-5} sec/m3 for the exclusion zone boundary, respectively.

2.3.4.3 Control Room Intake

The basic model employed for the distribution of relative concentrations (X/Qs) within a building wake at WCGS control room intakes following an accident is given by Reference 17 to be:

$$X/Q = \frac{K_C}{AV} \qquad (1)$$

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Where A = reference cross-sectional building area, m^2

V = reference wind speed, m/sec

K_c = nondimensional concentration coefficient

K_c is a function of nondimensional space coordinates x/L, y/L, and z/L, building configuration, wind direction, and source configuration. The K_c field for a given building configuration, source configuration, and wind direction is considered to be invariant. Accordingly, K_c values determined by wind tunnel tests with a model structure are expected to be the same as those that would be obtained with a geometrically similar building in the full-scale atmosphere in the same wind direction, with a similar leak. The contiguous building arrangement is shown in Figure 2.3-20. The K_c data used in the analysis for low level release are presented in Figure 2.3-25 and were derived from two sets of tests. One used rectangular prisms (Ref. 18), the other used a model of the EBR-II complex (Ref. 17). Both tests were described and portions of the data presented in Reference 35. The Kc data for the unit vent release from the top of the containment were extracted from Figure 10 of Reference 17 and are presented in Table 2.3-78. The value of A used in conjunction with K_c in Figure 2.3-2 and Table 2.3-78 is the WCGS equivalent of the EBR-II area, A = 1.12 D^2 = 2280 m^2 with the diameter of the reactor D = 45.1m.

The value of V used in conjunction with Figure 2.3-25 is the mean velocity of the approach flow at an elevation corresponding to the anemometer elevation of the EBR-II model tests. Reference 3 reports this elevation to be 62 feet or 0.77D above the top of the dome. The WCGS equivalent height becomes 63.4 + 0.77 x 45.1 = 98.1m above ground. The V values were obtained by extrapolating wind speeds at anemometer elevations equivalent to 98.1 meters by the power law.

$$V = u_1 \, (98.1/z_1)^n \qquad (2)$$

Where

u_1 = mean speed at elevation z_1, m/sec

z_1 = anemometer elevation at a given site, m

n = atmospheric stability exponent

Values of n were arbitrarily assumed for the various stability classes as follows:

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Pasquill Stability Class	A	B	C	D	E	F	G
n	0.20	0.25	0.29	0.33	0.40	0.50	0.60

A cumulative frequency distribution was constructed for the X/Q values calculated by equations 1 and 2 above, using 3 years combined onsite meteorological data. The corresponding highest 5 percent, 10 percent, 20 percent, and 40 percent X/Q values are given in Table 2.3-79.

2.3.5 LONG-TERM DIFFUSION ESTIMATES

The objective of Section 2.3.5 is to provide realistic estimates of annual average release atmospheric transport and diffusion characteristics to a distance of 80 km (50 miles) from the plant for annual average release limit calculations and man-rem estimates. The terrain within 50 km (31 miles) of the site is essentially flat becoming gently rolling to 80 km (50 miles). No important ranges of hills or mountains are within the region. No substantial water bodies are present, which are large enough to affect ambient dispersion parameters.

The analyses were based on on-site meteorological data over the periods June 1, 1973 through May 31, 1975 and March 5, 1979 to March 4, 1980.

2.3.5.1 Calculations

Both the PUFF and straight-line Gaussian dispersion models, described in Regulatory Guide 1.111, were used for determination of annual average diffusion estimates.

2.3.5.1.1 PUFF Model

The Equation for the PUFF model, as specified by Regulatory Guide 1.111 is:

$$X/Q = 2[(2p)^{3/2}\sigma_H\sigma_Z]^{-1}\exp\left(-\frac{1}{2}\left(\frac{r^2}{\sigma_H^2}+\frac{h_e^2}{\sigma_z^2}\right)\right) \qquad [2.3\text{-}32]$$

where:

$r^2 = (x - \bar{u}t)^2 + y^2$; and

$\sigma_H = \sigma_y = \sigma_x$

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TABLE 2.3-54

PLANT AND METEOROLOGICAL PARAMETERS
KANSAS GAS & ELECTRIC COMPANY
WOLF CREEK GENERATING STATION

Parameter	Measurement
Height of Containment Building	63.41 m
Plant Vent Height	66.45 m
Area of Reactor Building	2650 m^2
Building Shape Factor	0.5
Stack Diameter	2.11 m
Stack Gas Exit Velocity	10 m/sec
Plant Grade Elevation	1099.5 ft (335.2 m) MSL
Anemometer Starting Speed Threshold	0.33 m/sec
Height of Mixing Layer	870 m above grade
Meteorological Data Period (On Site)	6/1/73 through 5/31/75 and 3/5/79 through 3/4/80

Replace this table with the updated Table 2.3-54 and insert Tables 2.3-54a through 2.3-54i

Table 2.3-54

Plant and Meteorological Parameters for Calculation of Atmospheric Dispersion Factors

Parameter		Release Source				
		Equipment Hatch	**Unit Vent Stack**	**MSSVs/ARVs Vent**	**RWST Vent**	**Turbine-Driven AFW Exhaust Vent**
Meteorological Data Period		1/1/2006 through 12/31/2010				
Height of Lower Wind Speed Instrument		13.47 m				
Height of Upper Wind Speed Instrument		63.47 m				
Release Type		Ground				
Release Height		17.37 m	66.25 m	34.29 m	17.39 m	13.87 m
Building Wake Effect Area	To Control Room	2649 m^2 (reactor building area)			170.94 m^2 (RWST area)	2649 m^2
	To TSC	2649 m^2				
Height of Reactor Building		63.66 m				
Height of RWST		14.02 m				
Direction to Source from Control Room		115°	111°	90	153°	89°
Equivalent Horizontal Distance to Control Room Air Intake		113.17 m	79.01 m	75.66 m	101.71 m	111.95 m
Direction to Source from TSC		222°	232°	243°	222°	241°
Equivalent Horizontal Distance to TSC Air Intake		124.28 m	129.29 m	108.26 m	187.33 m	106.67 m
Control Room Air Intake Height		6.10 m				
TSC Air Intake Height		2.84 m				

Table 2.3-54a Joint Frequency Distribution (in percent of total hours) for Stability Class A

Atmospheric Stability: Class A
Period of Record: January 1, 2006 to December 31, 2010
(based on lower wind speed instrument)

Maximum Wind Speed (m/s)	Wind Direction																
	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	Total
0.34	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
0.50	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
0.75	0.000	0.000	0.000	0.000	0.000	0.002	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.002	0.000	0.005
1.00	0.002	0.000	0.005	0.002	0.000	0.000	0.000	0.000	0.002	0.002	0.000	0.000	0.000	0.000	0.000	0.000	0.015
1.25	0.000	0.002	0.010	0.000	0.002	0.000	0.000	0.005	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.002	0.022
1.50	0.000	0.002	0.002	0.002	0.002	0.010	0.005	0.007	0.002	0.002	0.010	0.005	0.002	0.005	0.002	0.002	0.065
2.00	0.020	0.025	0.012	0.007	0.020	0.007	0.015	0.030	0.005	0.007	0.012	0.017	0.005	0.015	0.007	0.005	0.212
3.00	0.070	0.075	0.105	0.082	0.082	0.050	0.050	0.107	0.057	0.035	0.045	0.042	0.080	0.057	0.047	0.062	1.048
4.00	0.080	0.102	0.152	0.120	0.127	0.142	0.147	0.314	0.195	0.075	0.090	0.055	0.095	0.070	0.067	0.090	1.922
5.00	0.100	0.110	0.070	0.047	0.097	0.135	0.132	0.469	0.377	0.190	0.172	0.072	0.087	0.077	0.077	0.110	2.324
6.00	0.107	0.032	0.035	0.025	0.040	0.092	0.122	0.349	0.579	0.255	0.152	0.037	0.057	0.040	0.100	0.177	2.201
8.00	0.150	0.035	0.010	0.022	0.045	0.050	0.142	0.384	0.826	0.522	0.070	0.040	0.130	0.092	0.207	0.277	3.003
10.00	0.042	0.020	0.000	0.000	0.005	0.005	0.010	0.132	0.412	0.165	0.005	0.017	0.037	0.062	0.122	0.177	1.213
44.70	0.020	0.000	0.000	0.000	0.000	0.000	0.005	0.027	0.115	0.030	0.002	0.012	0.035	0.085	0.050	0.022	0.404
Total	0.59	0.40	0.40	0.31	0.42	0.49	0.63	1.83	2.57	1.28	0.56	0.30	0.53	0.50	0.68	0.93	12.43

Table 2.3-54b Joint Frequency Distribution (in percent of total hours) for Stability Class B

Atmospheric Stability: Class B
Period of Record: January 1, 2006 to December 31, 2010
(based on lower wind speed instrument)

Maximum Wind Speed (m/s)	Wind Direction																
	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	Total
0.34	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
0.50	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
0.75	0.000	0.000	0.000	0.000	0.000	0.002	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.002
1.00	0.000	0.002	0.000	0.000	0.002	0.000	0.000	0.000	0.002	0.002	0.000	0.002	0.002	0.000	0.000	0.000	0.015
1.25	0.002	0.000	0.002	0.002	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.005	0.000	0.000	0.002	0.000	0.015
1.50	0.002	0.002	0.002	0.005	0.007	0.005	0.007	0.002	0.000	0.000	0.005	0.002	0.000	0.000	0.000	0.010	0.052
2.00	0.012	0.010	0.015	0.022	0.012	0.010	0.012	0.022	0.012	0.002	0.005	0.027	0.007	0.027	0.015	0.015	0.230
3.00	0.055	0.070	0.125	0.060	0.062	0.072	0.052	0.122	0.102	0.035	0.035	0.040	0.045	0.020	0.037	0.055	0.988
4.00	0.060	0.090	0.077	0.072	0.082	0.075	0.072	0.155	0.145	0.072	0.037	0.037	0.032	0.030	0.035	0.065	1.138
5.00	0.092	0.065	0.015	0.045	0.040	0.050	0.062	0.140	0.162	0.072	0.060	0.032	0.032	0.017	0.047	0.085	1.018
6.00	0.050	0.030	0.007	0.005	0.027	0.022	0.037	0.055	0.112	0.122	0.032	0.007	0.030	0.020	0.060	0.082	0.701
8.00	0.060	0.020	0.002	0.010	0.005	0.012	0.047	0.087	0.160	0.160	0.020	0.015	0.027	0.050	0.115	0.137	0.928
10.00	0.040	0.005	0.000	0.000	0.000	0.002	0.007	0.030	0.087	0.060	0.002	0.005	0.022	0.037	0.087	0.107	0.494
44.70	0.017	0.000	0.000	0.000	0.002	0.000	0.007	0.015	0.037	0.025	0.000	0.015	0.012	0.047	0.040	0.010	0.230
Total	0.39	0.29	0.25	0.22	0.24	0.25	0.31	0.63	0.82	0.55	0.20	0.19	0.21	0.25	0.44	0.57	5.81

Table 2.3-54c Joint Frequency Distribution (in percent of total hours) for Stability Class C

Atmospheric Stability: Class C
Period of Record: January 1, 2006 to December 31, 2010
(based on lower wind speed instrument)

Maximum Wind Speed (m/s)	Wind Direction																
	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	Total
0.34	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
0.50	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
0.75	0.000	0.000	0.000	0.000	0.000	0.000	0.002	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.002
1.00	0.002	0.002	0.000	0.002	0.000	0.000	0.000	0.000	0.002	0.000	0.000	0.002	0.000	0.000	0.002	0.000	0.015
1.25	0.007	0.007	0.000	0.002	0.002	0.002	0.002	0.012	0.005	0.000	0.000	0.002	0.005	0.002	0.005	0.002	0.060
1.50	0.002	0.010	0.010	0.002	0.010	0.005	0.015	0.010	0.005	0.002	0.000	0.007	0.007	0.015	0.010	0.002	0.115
2.00	0.010	0.020	0.030	0.012	0.030	0.027	0.020	0.037	0.022	0.020	0.010	0.020	0.012	0.017	0.007	0.007	0.305
3.00	0.047	0.100	0.085	0.070	0.097	0.070	0.085	0.187	0.100	0.042	0.040	0.040	0.032	0.045	0.037	0.050	1.128
4.00	0.100	0.075	0.075	0.117	0.080	0.062	0.050	0.145	0.167	0.080	0.052	0.045	0.060	0.035	0.075	0.067	1.285
5.00	0.100	0.047	0.017	0.050	0.050	0.045	0.035	0.072	0.152	0.102	0.027	0.022	0.022	0.032	0.045	0.077	0.899
6.00	0.097	0.015	0.015	0.012	0.040	0.045	0.067	0.082	0.082	0.130	0.025	0.007	0.020	0.032	0.067	0.105	0.844
8.00	0.090	0.032	0.000	0.012	0.022	0.015	0.035	0.095	0.135	0.170	0.017	0.025	0.025	0.060	0.105	0.140	0.978
10.00	0.055	0.002	0.000	0.000	0.005	0.002	0.012	0.035	0.070	0.080	0.012	0.010	0.007	0.050	0.095	0.067	0.504
44.70	0.005	0.000	0.000	0.000	0.002	0.000	0.002	0.010	0.040	0.015	0.002	0.002	0.010	0.022	0.037	0.012	0.162
Total	0.52	0.31	0.23	0.28	0.34	0.27	0.33	0.69	0.78	0.64	0.19	0.18	0.20	0.31	0.49	0.53	6.30

Table 2.3-54d Joint Frequency Distribution (in percent of total hours) for Stability Class D

Atmospheric Stability: Class D
Period of Record: January 1, 2006 to December 31, 2010
(based on lower wind speed instrument)

Maximum Wind Speed (m/s)	Wind Direction																
	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	Total
0.34	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.002
0.50	0.000	0.000	0.000	0.000	0.002	0.002	0.000	0.000	0.002	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.007
0.75	0.005	0.010	0.027	0.005	0.015	0.005	0.005	0.002	0.000	0.002	0.007	0.002	0.002	0.005	0.000	0.002	0.097
1.00	0.010	0.047	0.045	0.022	0.010	0.022	0.012	0.010	0.002	0.002	0.007	0.002	0.000	0.007	0.002	0.007	0.212
1.25	0.022	0.045	0.050	0.025	0.035	0.022	0.027	0.022	0.022	0.010	0.012	0.015	0.007	0.005	0.007	0.025	0.354
1.50	0.020	0.050	0.067	0.045	0.042	0.020	0.032	0.050	0.030	0.015	0.025	0.030	0.010	0.015	0.012	0.030	0.494
2.00	0.085	0.130	0.167	0.155	0.102	0.097	0.140	0.125	0.077	0.060	0.092	0.085	0.045	0.035	0.042	0.070	1.508
3.00	0.295	0.409	0.532	0.384	0.334	0.369	0.290	0.317	0.359	0.212	0.272	0.132	0.135	0.102	0.130	0.305	4.578
4.00	0.494	0.309	0.384	0.474	0.392	0.374	0.414	0.659	0.524	0.404	0.232	0.155	0.172	0.195	0.362	0.474	6.020
5.00	0.542	0.297	0.165	0.252	0.255	0.280	0.429	0.689	0.726	0.474	0.115	0.095	0.192	0.267	0.484	0.544	5.806
6.00	0.492	0.212	0.040	0.115	0.192	0.190	0.295	0.624	0.884	0.389	0.112	0.075	0.145	0.257	0.442	0.609	5.072
8.00	0.794	0.152	0.020	0.077	0.162	0.125	0.217	0.696	1.248	0.534	0.035	0.095	0.205	0.404	0.729	1.101	6.594
10.00	0.297	0.052	0.000	0.005	0.030	0.010	0.042	0.282	0.547	0.242	0.017	0.060	0.077	0.175	0.339	0.507	2.683
44.70	0.097	0.000	0.002	0.000	0.005	0.000	0.020	0.097	0.297	0.077	0.010	0.012	0.020	0.052	0.125	0.102	0.919
Total	3.15	1.72	1.50	1.56	1.58	1.52	1.92	3.57	4.72	2.42	0.94	0.76	1.01	1.52	2.68	3.78	34.35

Table 2.3-54e Joint Frequency Distribution (in percent of total hours) for Stability Class E

Atmospheric Stability: Class E
Period of Record: January 1, 2006 to December 31, 2010
(based on lower wind speed instrument)

Maximum Wind Speed (m/s)	Wind Direction																
	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	Total
0.34	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.002
0.50	0.002	0.000	0.007	0.002	0.002	0.000	0.002	0.005	0.000	0.005	0.000	0.000	0.000	0.000	0.000	0.000	0.027
0.75	0.005	0.015	0.012	0.012	0.007	0.005	0.007	0.002	0.002	0.005	0.000	0.000	0.005	0.005	0.005	0.005	0.095
1.00	0.030	0.035	0.037	0.025	0.010	0.010	0.007	0.017	0.007	0.015	0.000	0.010	0.002	0.005	0.005	0.015	0.232
1.25	0.050	0.042	0.050	0.050	0.045	0.030	0.032	0.002	0.017	0.012	0.020	0.007	0.005	0.000	0.012	0.017	0.394
1.50	0.050	0.077	0.065	0.045	0.070	0.072	0.057	0.022	0.007	0.020	0.037	0.012	0.010	0.007	0.020	0.025	0.599
2.00	0.155	0.117	0.160	0.175	0.157	0.207	0.155	0.097	0.052	0.090	0.155	0.062	0.025	0.022	0.042	0.115	1.787
3.00	0.245	0.187	0.245	0.369	0.432	0.459	0.547	0.599	0.407	0.260	0.414	0.140	0.187	0.140	0.285	0.387	5.301
4.00	0.312	0.125	0.115	0.182	0.282	0.422	0.686	1.051	0.674	0.434	0.202	0.132	0.197	0.225	0.377	0.417	5.833
5.00	0.195	0.065	0.027	0.107	0.175	0.260	0.454	1.113	0.809	0.437	0.072	0.100	0.107	0.167	0.250	0.302	4.640
6.00	0.120	0.027	0.010	0.080	0.110	0.150	0.215	0.764	0.614	0.175	0.040	0.037	0.042	0.042	0.115	0.162	2.703
8.00	0.107	0.035	0.007	0.007	0.032	0.055	0.152	0.923	0.824	0.200	0.020	0.015	0.007	0.015	0.050	0.097	2.548
10.00	0.005	0.002	0.002	0.000	0.002	0.010	0.030	0.267	0.354	0.085	0.012	0.007	0.010	0.002	0.005	0.012	0.809
44.70	0.010	0.000	0.000	0.002	0.000	0.005	0.020	0.065	0.235	0.017	0.002	0.002	0.000	0.000	0.000	0.000	0.359
Total	1.29	0.73	0.74	1.06	1.33	1.68	2.37	4.93	4.00	1.75	0.98	0.53	0.60	0.63	1.17	1.56	25.33

Table 2.3-54f Joint Frequency Distribution (in percent of total hours) for Stability Class F

Atmospheric Stability: Class F
Period of Record: January 1, 2006 to December 31, 2010
(based on lower wind speed instrument)

Maximum Wind Speed (m/s)	Wind Direction																
	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	Total
0.34	0.001	0.002	0.002	0.001	0.001	0.001	0.001	0.001	0.000	0.000	0.001	0.001	0.000	0.000	0.001	0.001	0.012
0.50	0.005	0.005	0.002	0.007	0.000	0.007	0.000	0.000	0.002	0.002	0.002	0.000	0.000	0.000	0.000	0.000	0.035
0.75	0.002	0.017	0.022	0.015	0.002	0.005	0.002	0.007	0.002	0.000	0.010	0.005	0.005	0.005	0.005	0.005	0.112
1.00	0.022	0.032	0.037	0.010	0.027	0.022	0.012	0.010	0.005	0.007	0.007	0.015	0.007	0.010	0.015	0.007	0.250
1.25	0.020	0.047	0.057	0.022	0.027	0.030	0.020	0.020	0.020	0.015	0.015	0.012	0.005	0.010	0.012	0.025	0.359
1.50	0.035	0.072	0.062	0.042	0.075	0.037	0.027	0.042	0.012	0.005	0.022	0.007	0.002	0.015	0.022	0.027	0.509
2.00	0.110	0.125	0.132	0.145	0.197	0.197	0.305	0.090	0.032	0.032	0.057	0.020	0.010	0.030	0.075	0.092	1.650
3.00	0.245	0.162	0.122	0.487	0.447	0.552	0.607	0.389	0.147	0.112	0.112	0.045	0.047	0.037	0.210	0.277	3.999
4.00	0.160	0.082	0.040	0.100	0.145	0.160	0.275	0.437	0.212	0.112	0.032	0.027	0.032	0.020	0.132	0.167	2.134
5.00	0.032	0.007	0.000	0.012	0.017	0.015	0.080	0.207	0.175	0.067	0.010	0.015	0.007	0.005	0.010	0.050	0.711
6.00	0.007	0.000	0.000	0.000	0.010	0.002	0.020	0.100	0.110	0.037	0.000	0.002	0.000	0.000	0.002	0.010	0.302
8.00	0.000	0.000	0.000	0.000	0.000	0.000	0.005	0.060	0.075	0.007	0.002	0.002	0.000	0.000	0.000	0.002	0.155
10.00	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.007	0.027	0.002	0.005	0.000	0.000	0.000	0.000	0.000	0.042
44.70	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.007	0.002	0.000	0.000	0.000	0.000	0.000	0.000	0.010
Total	0.64	0.55	0.48	0.84	0.95	1.03	1.35	1.37	0.83	0.40	0.28	0.15	0.12	0.13	0.48	0.66	10.28

Table 2.3-54g Joint Frequency Distribution (in percent of total hours) for Stability Class G

Atmospheric Stability: Class G
Period of Record: January 1, 2006 to December 31, 2010
(based on lower wind speed instrument)

Maximum Wind Speed (m/s)	Wind Direction																
	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	Total
0.34	0.001	0.001	0.001	0.001	0.001	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.001	0.000	0.001	0.007
0.50	0.007	0.005	0.005	0.000	0.005	0.000	0.000	0.002	0.002	0.002	0.005	0.000	0.002	0.002	0.000	0.000	0.040
0.75	0.012	0.007	0.015	0.010	0.005	0.010	0.007	0.005	0.000	0.002	0.005	0.005	0.002	0.005	0.002	0.012	0.107
1.00	0.010	0.025	0.012	0.012	0.017	0.010	0.012	0.000	0.000	0.010	0.002	0.012	0.007	0.017	0.012	0.017	0.180
1.25	0.020	0.042	0.040	0.022	0.017	0.020	0.015	0.012	0.010	0.007	0.002	0.005	0.002	0.017	0.017	0.045	0.297
1.50	0.035	0.062	0.062	0.012	0.027	0.037	0.020	0.010	0.002	0.007	0.012	0.005	0.010	0.007	0.025	0.032	0.369
2.00	0.082	0.225	0.137	0.050	0.152	0.162	0.115	0.040	0.012	0.015	0.017	0.007	0.005	0.042	0.050	0.102	1.216
3.00	0.182	0.187	0.150	0.280	0.319	0.404	0.309	0.132	0.087	0.015	0.040	0.007	0.002	0.025	0.125	0.172	2.439
4.00	0.040	0.050	0.022	0.072	0.055	0.067	0.027	0.097	0.072	0.005	0.020	0.002	0.002	0.005	0.055	0.092	0.686
5.00	0.015	0.005	0.000	0.002	0.005	0.002	0.007	0.025	0.015	0.005	0.007	0.000	0.000	0.000	0.002	0.012	0.105
6.00	0.000	0.002	0.000	0.000	0.002	0.002	0.002	0.002	0.012	0.002	0.002	0.000	0.000	0.000	0.000	0.002	0.032
8.00	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.010	0.005	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.015
10.00	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.002	0.000	0.000	0.000	0.000	0.000	0.000	0.002
44.70	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
Total	0.40	0.61	0.45	0.46	0.61	0.72	0.52	0.34	0.22	0.08	0.11	0.05	0.04	0.12	0.29	0.49	5.50

Table 2.3-54h Wind Direction Occurrence Frequency
Period of Record: January 1, 2006 to December 31, 2010
(based on lower wind speed instrument)

Wind Direction	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
Frequency	7.0	4.6	4.0	4.7	5.5	6.0	7.4	13.4	13.9	7.1	3.3	2.2	2.7	3.5	6.2	8.5

Total Frequency = 100

Table 2.3-54i Wind Speed Occurrence Frequency
Period of Record: January 1, 2006 to December 31, 2010
(based on lower wind speed instrument)

Maximum Wind Speed (m/s):	0.34	0.50	0.75	1.00	1.25	1.50	2.00	3.00	4.00	5.00	6.00	8.00	10.00	44.70
Wind Speed Frequency:	0.02	0.11	0.42	0.92	1.50	2.20	6.91	19.48	19.02	15.50	11.86	14.22	5.75	2.08

Total Frequency = 100

TABLE 2.3-55

ACCIDENT ATMOSPHERIC RELATIVE CONCENTRATIONS (X/Q)[a]
FOR 3-YEAR DATA PERIOD

Affected Sector	Exclusion Zone Circular (1200 m)	Low Population Zone Circular (4023 m)					Remarks
	Time						
	2-Hr	2-Hr	8-Hr	16-Hr	72-Hr	624-Hr	
NNE	1.0E-04	2.9E-05	1.3E-05	8.3E-06	3.4E-06	9.3E-07	Highest 0.5%
NE	7.6E-05	1.9E-05	7.9E-06	5.1E-06	1.9E-06	4.9E-07	Highest 0.5%
ENE	7.8E-05	2.1E-05	8.4E-06	5.3E-06	2.0E-06	4.7E-07	Highest 0.5%
E	8.0E-05	2.2E-05	9.0E-06	5.8E-06	2.2E-06	5.5E-07	Highest 0.5%
ESE	1.1E-04	3.3E-05	1.3E-05	8.4E-06	3.1E-06	7.4E-07	Highest 0.5%
SE	1.3E-04	4.3E-05	1.7E-05	1.0E-05	3.7E-06	8.5E-07	Highest 0.5%
SSE	8.8E-05	2.7E-05	1.1E-05	7.3E-06	2.8E-06	7.3E-07	Highest 0.5%
S	1.1E-04	3.2E-05	1.3E-05	8.2E-06	3.0E-06	7.4E-07	Highest 0.5%
SSW	1.4E-04	4.3E-05	1.7E-05	1.1E-05	3.8E-06	8.9E-07	Highest 0.5%
SW	1.2E-04	3.8E-05	1.5E-05	9.1E-06	3.2E-06	7.4E-07	Highest 0.5%
WSW	8.0E-05	2.3E-05	9.9E-06	6.5E-06	2.6E-06	7.0E-07	Highest 0.5%
W	1.3E-04	4.2E-05	1.6E-05	1.0E-05	3.7E-06	8.4E-07	Highest 0.5%
WNW	1.3E-04	4.2E-05	1.7E-05	1.0E-05	3.8E-06	8.9E-07	Highest 0.5%
NW	1.5E-04*	4.4E-05*	1.8E-05	1.2E-05	4.7E-06	1.2E-06	Highest 0.5%
NNW	1.5E-04*	4.4E-05*	2.0E-05*	1.3E-05*	5.4E-06*	1.5E-06*	Highest 0.5%
N	1.5E-04*	4.4E-05*	1.9E-05	1.3E-05*	5.4E-06*	1.5E-06*	Highest 0.5%
5%	1.4E-04	4.4E-05	1.4E-05	9.8E-06	4.3E-06	1.3E-06	Highest 5%
50%	2.5E-05	4.4E-06	2.4E-06	2.0E-06	1.3E-06	6.9E-07	Highest 50%

[a]Units sec/m^3.
*Maximum sector values.

Replace this table with the updated Table 2.3-55

Table 2.3-55 Relative Concentration (χ/Q – sec/m^3) Results at Exclusion Area Boundary for Release from Sources near Reactor Building

Downwind Distance							
Sector	(meters)	0-2 hours	0-8 hours	8-24 hours	1-4 days	4-30 days	Annual Average
S	1200.	1.18E-04	5.69E-05	3.95E-05	1.79E-05	5.73E-06	1.43E-06
SSW	1200.	1.38E-04	6.50E-05	4.47E-05	1.98E-05	6.18E-06	1.48E-06
SW	1200.	1.32E-04	6.25E-05	4.29E-05	1.90E-05	5.90E-06	1.41E-06
WSW	1200.	1.27E-04	6.04E-05	4.16E-05	1.85E-05	5.81E-06	1.41E-06
W	1200.	1.32E-04	6.37E-05	4.42E-05	2.00E-05	6.43E-06	1.60E-06
WNW	1200.	1.36E-04	6.59E-05	4.60E-05	2.10E-05	6.84E-06	1.73E-06
NW	1200.	1.29E-04	6.36E-05	4.48E-05	2.09E-05	6.99E-06	1.83E-06
NNW	1200.	1.10E-04	5.73E-05	4.13E-05	2.03E-05	7.35E-06	2.12E-06
N	1200.	8.77E-05	4.54E-05	3.27E-05	1.60E-05	5.75E-06	1.64E-06
NNE	1200.	6.48E-05	3.23E-05	2.28E-05	1.07E-05	3.62E-06	9.60E-07
NE	1200.	6.76E-05	3.20E-05	2.20E-05	9.80E-06	3.06E-06	7.36E-07
ENE	1200.	4.12E-05	1.95E-05	1.34E-05	5.93E-06	1.85E-06	4.43E-07
E	1200.	4.48E-05	2.07E-05	1.41E-05	6.12E-06	1.84E-06	4.25E-07
ESE	1200.	5.26E-05	2.49E-05	1.72E-05	7.63E-06	2.39E-06	5.76E-07
SE	1200.	9.86E-05	4.64E-05	3.19E-05	1.41E-05	4.35E-06	1.04E-06
SSE	1200.	1.26E-04	6.09E-05	4.23E-05	1.91E-05	6.13E-06	1.52E-06
Maximum sector χ/Q		1.38E-04					
Overall site χ/Q		1.31E-04	6.64E-05	4.72E-05	2.25E-05	7.77E-06	2.12E-06

TABLE 2.3-56

ACCIDENT ATMOSPHERIC RELATIVE CONCENTRATIONS (χ/Q)[a]
FOR 6/1/73 TO 5/31/74 DATA PERIOD

Affected Sector	Exclusion Zone Circular (1200 m)	Low Population Zone Circular (4023 m)					Remarks
		Time					
	2-Hr	2-Hr	8-Hr	16-Hr	72-Hr	624-Hr	
NNE	1.2E-04	3.9E-05	1.6E-05	1.0E-05	3.9E-06	9.8E-07	Highest 0.5%
NE	7.6E-05	1.8E-05	7.6E-06	4.9E-06	1.9E-06	5.0E-07	Highest 0.5%
ENE	7.8E-05	2.0E-05	8.2E-06	5.3E-06	2.0E-06	5.0E-07	Highest 0.5%
E	1.0E-04	3.1E-05	1.2E-05	7.4E-06	2.6E-06	6.0E-07	Highest 0.5%
ESE	1.1E-04	3.3E-05	1.3E-05	8.4E-06	3.1E-06	7.4E-07	Highest 0.5%
SE	1.3E-04	4.0E-05	1.6E-05	1.0E-05	3.7E-06	8.7E-07	Highest 0.5%
SSE	8.8E-05	2.7E-05	1.1E-05	7.4E-06	2.9E-06	7.7E-07	Highest 0.5%
S	8.3E-05	2.2E-05	9.5E-06	6.3E-06	2.5E-06	6.9E-07	Highest 0.5%
SSW	1.5E-04*	4.4E-05*	1.8E-05	1.1E-05	4.3E-06	1.1E-06	Highest 0.5%
SW	8.4E-05	2.4E-05	1.0E-05	6.6E-06	2.6E-06	6.7E-07	Highest 0.5%
WSW	7.8E-05	2.1E-05	8.9E-06	5.8E-06	2.3E-06	6.1E-07	Highest 0.5%
W	1.3E-04	4.1E-05	1.6E-05	9.8E-06	3.5E-06	8.0E-07	Highest 0.5%
WNW	8.8E-05	2.7E-05	1.1E-05	7.2E-06	2.8E-06	7.0E-07	Highest 0.5%
NW	1.4E-04	4.4E-05*	1.8E-05	1.2E-05	4.5E-06	1.1E-06	Highest 0.5%
NNW	1.5E-04*	4.4E-05*	2.0E-05*	1.3E-05*	5.6E-06*	1.6E-06*	Highest 0.5%
N	1.5E-04*	4.4E-05*	2.0E-05*	1.3E-05*	5.5E-06	1.6E-06*	Highest 0.5%
5%	1.4E-04	4.4E-05	1.5E-05	1.0E-05	4.5E-06	1.4E-06	Highest 5%
50%	2.5E-05	4.5E-06	2.5E-06	2.1E-06	1.4E-06	7.4E-07	Highest 50%

[a]Units sec/m^3
*Maximum sector values

Rev. 0

Replace this table with the updated Table 2.3-56

Table 2.3-56 Relative Concentration (χ/Q - sec/m^3) Results at Low Population Zone for Release from Sources near Reactor Building							
Downwind Distance							
Sector	(meters)	0-2 hours	0-8 hours	8-24 hours	1-4 days	4-30 days	Annual Average
S	4023.	3.49E-05	1.53E-05	1.01E-05	4.14E-06	1.15E-06	2.38E-07
SSW	4023.	4.50E-05	1.90E-05	1.24E-05	4.87E-06	1.28E-06	2.48E-07
SW	4023.	4.21E-05	1.79E-05	1.17E-05	4.60E-06	1.21E-06	2.36E-07
WSW	4023.	3.89E-05	1.68E-05	1.10E-05	4.42E-06	1.19E-06	2.40E-07
W	4023.	4.26E-05	1.85E-05	1.22E-05	4.92E-06	1.34E-06	2.72E-07
WNW	4023.	4.33E-05	1.90E-05	1.26E-05	5.15E-06	1.43E-06	2.97E-07
NW	4023.	4.10E-05	1.83E-05	1.23E-05	5.12E-06	1.46E-06	3.16E-07
NNW	4023.	3.06E-05	1.47E-05	1.02E-05	4.58E-06	1.46E-06	3.60E-07
N	4023.	2.20E-05	1.07E-05	7.41E-06	3.37E-06	1.09E-06	2.72E-07
NNE	4023.	1.58E-05	7.37E-06	5.04E-06	2.21E-06	6.76E-07	1.59E-07
NE	4023.	1.67E-05	7.43E-06	4.95E-06	2.06E-06	5.84E-07	1.25E-07
ENE	4023.	9.61E-06	4.29E-06	2.87E-06	1.19E-06	3.40E-07	7.30E-08
E	4023.	9.32E-06	4.15E-06	2.77E-06	1.15E-06	3.26E-07	6.96E-08
ESE	4023.	1.11E-05	5.03E-06	3.39E-06	1.44E-06	4.22E-07	9.38E-08
SE	4023.	2.67E-05	1.16E-05	7.64E-06	3.09E-06	8.44E-07	1.72E-07
SSE	4023.	3.88E-05	1.69E-05	1.11E-05	4.51E-06	1.23E-06	2.53E-07
Maximum sector χ/Q		4.50E-05					
Overall Site χ/Q		4.19E-05	1.91E-05	1.29E-05	5.49E-06	1.61E-06	3.60E-07

TABLE 2.3-57

ACCIDENT ATMOSPHERIC RELATIVE CONCENTRATIONS (X/Q)[a]
FOR 6/1/74 TO 5/31/75 DATA PERIOD

Affected Sector	Exclusion Zone Circular (1200 m)	Low Population Zone Circular (4023 m)					Remarks
	Time						
	2-Hr	2-Hr	8-Hr	16-Hr	72-Hr	624-Hr	
NNE	9.3E-05	2.7E-05	1.2E-05	7.8E-06	3.2E-06	8.7E-07	Highest 0.5%
NE	7.7E-05	1.9E-05	7.7E-06	4.9E-06	1.9E-06	4.6E-07	Highest 0.5%
ENE	7.7E-05	1.8E-05	7.0E-06	4.4E-06	1.6E-06	3.7E-07	Highest 0.5%
E	7.8E-05	1.9E-05	7.7E-06	4.9E-06	1.8E-06	4.5E-07	Highest 0.5%
ESE	7.8E-05	1.9E-05	8.1E-06	5.3E-06	2.1E-06	5.6E-07	Highest 0.5%
SE	1.0E-04	3.1E-05	1.2E-05	7.7E-06	2.8E-06	6.5E-07	Highest 0.5%
SSE	1.0E-04	3.1E-05	1.3E-05	8.0E-06	3.0E-06	7.3E-07	Highest 0.5%
S	1.1E-04	3.3E-05	1.3E-05	8.2E-06	3.0E-06	7.0E-07	Highest 0.5%
SSW	1.1E-04	3.4E-05	1.4E-05	8.5E-06	3.1E-06	7.5E-07	Highest 0.5%
SW	8.0E-05	2.2E-05	9.0E-06	5.8E-06	2.2E-06	5.5E-07	Highest 0.5%
WSW	1.1E-04	3.5E-05	1.4E-05	8.6E-06	3.1E-06	7.2E-07	Highest 0.5%
W	1.3E-04	3.9E-05	1.5E-05	9.5E-06	3.4E-06	7.8E-07	Highest 0.5%
WNW	8.8E-05	2.7E-05	1.1E-05	7.0E-06	2.7E-06	6.6E-07	Highest 0.5%
NW	1.5E-04*	4.4E-05*	1.8E-05	1.2E-05	4.5E-06	1.1E-06	Highest 0.5%
NNW	1.5E-04*	4.4E-05*	1.9E-05*	1.3E-05*	5.2E-06*	1.4E-06*	Highest 0.5%
N	1.4E-04	4.3E-05	1.8E-05	1.2E-05	4.8E-06	1.3E-06	Highest 0.5%
5%	1.4E-04	4.3E-05	1.4E-05	9.4E-06	4.1E-06	1.3E-06	Highest 5%
50%	2.1E-05	3.7E-06	2.1E-06	1.7E-06	1.1E-06	6.2E-07	Highest 50%

[a]Units sec/m^3
*Maximum sector values

Rev. 0

Replace this table with the updated Table 2.3-57

Table 2.3-57 Relative Concentration (χ/Q - sec/m^3) Results at Exclusion Area Boundary for Release from RWST							
Downwind Distance							
Sector	(meters)	0-2 hours	0-8 hours	8-24 hours	1-4 days	4-30 days	Annual Average
S	1200.	1.22E-04	5.86E-05	4.05E-05	1.82E-05	5.80E-06	1.43E-06
SSW	1200.	1.40E-04	6.58E-05	4.52E-05	2.00E-05	6.21E-06	1.48E-06
SW	1200.	1.34E-04	6.30E-05	4.32E-05	1.91E-05	5.92E-06	1.41E-06
WSW	1200.	1.28E-04	6.06E-05	4.17E-05	1.86E-05	5.82E-06	1.41E-06
W	1200.	1.34E-04	6.45E-05	4.47E-05	2.02E-05	6.46E-06	1.60E-06
WNW	1200.	1.37E-04	6.67E-05	4.64E-05	2.12E-05	6.87E-06	1.73E-06
NW	1200.	1.30E-04	6.42E-05	4.51E-05	2.10E-05	7.01E-06	1.83E-06
NNW	1200.	1.17E-04	6.02E-05	4.32E-05	2.10E-05	7.49E-06	2.12E-06
N	1200.	9.90E-05	5.03E-05	3.58E-05	1.72E-05	5.97E-06	1.64E-06
NNE	1200.	7.60E-05	3.69E-05	2.57E-05	1.17E-05	3.81E-06	9.60E-07
NE	1200.	6.84E-05	3.23E-05	2.22E-05	9.86E-06	3.07E-06	7.36E-07
ENE	1200.	4.67E-05	2.16E-05	1.47E-05	6.38E-06	1.92E-06	4.43E-07
E	1200.	4.50E-05	2.08E-05	1.42E-05	6.14E-06	1.85E-06	4.25E-07
ESE	1200.	5.28E-05	2.50E-05	1.72E-05	7.66E-06	2.39E-06	5.76E-07
SE	1200.	1.00E-04	4.71E-05	3.23E-05	1.42E-05	4.37E-06	1.04E-06
SSE	1200.	1.28E-04	6.15E-05	4.27E-05	1.93E-05	6.15E-06	1.52E-06
Maximum sector χ/Q		1.40E-04					
Overall site χ/Q		1.35E-04	6.77E-05	4.80E-05	2.28E-05	7.83E-06	2.12E-06

TABLE 2.3-58

ACCIDENT ATMOSPHERIC RELATIVE CONCENTRATIONS (χ/Q)[a]
FOR 3/5/79 TO 3/4/80 DATA PERIOD

Affected Sector	Exclusion Zone Circular (1200 m)	Low Population Zone Circular (4023 m)					Remarks
		Time					
	2-Hr	2-Hr	8-Hr	16-Hr	72-Hr	624-Hr	
NNE	7.8E-05	2.3E-05	1.0E-05	7.1E-06	3.0E-06	8.9E-07	Highest 0.5%
NE	7.6E-05	1.9E-05	7.9E-06	5.1E-06	2.0E-06	5.0E-07	Highest 0.5%
ENE	7.8E-05	2.2E-05	9.0E-06	5.8E-06	2.2E-06	5.4E-07	Highest 0.5%
E	1.2E-04	3.3E-05	1.3E-05	8.5E-06	3.2E-06	7.8E-07	Highest 0.5%
ESE	1.4E-04	4.5E-05	1.8E-05	1.1E-05	3.9E-06	9.0E-07	Highest 0.5%
SE	1.4E-04	4.5E-05	1.8E-05	1.1E-05	3.9E-06	9.0E-07	Highest 0.5%
SSE	8.8E-05	2.7E-05	1.1E-05	7.3E-06	2.8E-06	7.3E-07	Highest 0.5%
S	1.3E-04	4.3E-05	1.7E-05	1.0E-05	3.7E-06	8.5E-07	Highest 0.5%
SSW	1.3E-04	4.2E-05	1.6E-05	1.0E-05	3.7E-06	8.4E-07	Highest 0.5%
SW	1.4E-04	4.3E-05	1.7E-05	1.1E-05	3.8E-06	8.9E-07	Highest 0.5%
WSW	8.4E-05	2.3E-05	1.0E-05	6.9E-06	2.9E-06	8.3E-07	Highest 0.5%
W	1.4E-04	4.3E-05	1.7E-05	1.1E-05	4.0E-06	9.7E-07	Highest 0.5%
WNW	1.4E-04	4.4E-05	1.8E-05	1.2E-05	4.4E-06	1.1E-06	Highest 0.5%
NW	1.5E-04*	5.0E-05*	2.1E-05*	1.4E-05*	5.4E-06	1.4E-06	Highest 0.5%
NNW	1.5E-04*	4.5E-05	2.0E-05	1.3E-05	5.4E-06	1.5E-06	Highest 0.5%
N	1.5E-04*	4.4E-05	2.0E-05	1.3E-05	5.6E-06*	1.6E-06*	Highest 0.5%
5%	1.5E-04	4.5E-05	1.5E-05	1.0E-05	4.6E-06	1.5E-06	Highest 5%
50%	2.8E-05	5.0E-06	2.7E-06	2.2E-06	1.4E-07	7.7E-07	Highest 50%

[a]Units sec/m^3

*Maximum sector values

Replace this table with the updated Table 2.3-58

Table 2.3-58 Relative Concentration (χ/Q - sec/m^3) Results at Low Population Zone for Release from RWST							
Downwind Distance							
Sector	(meters)	0-2 hours	0-8 hours	8-24 hours	1-4 days	4-30 days	Annual Average
S	4023.	3.50E-05	1.53E-05	1.01E-05	4.14E-06	1.15E-06	2.38E-07
SSW	4023.	4.50E-05	1.90E-05	1.24E-05	4.87E-06	1.28E-06	2.48E-07
SW	4023.	4.21E-05	1.79E-05	1.17E-05	4.60E-06	1.21E-06	2.36E-07
WSW	4023.	3.89E-05	1.68E-05	1.10E-05	4.42E-06	1.19E-06	2.40E-07
W	4023.	4.26E-05	1.85E-05	1.22E-05	4.92E-06	1.34E-06	2.72E-07
WNW	4023.	4.33E-05	1.90E-05	1.26E-05	5.15E-06	1.43E-06	2.97E-07
NW	4023.	4.10E-05	1.83E-05	1.23E-05	5.12E-06	1.46E-06	3.16E-07
NNW	4023.	3.17E-05	1.51E-05	1.04E-05	4.68E-06	1.48E-06	3.60E-07
N	4023.	2.24E-05	1.08E-05	7.50E-06	3.40E-06	1.09E-06	2.72E-07
NNE	4023.	1.58E-05	7.37E-06	5.04E-06	2.21E-06	6.76E-07	1.59E-07
NE	4023.	1.67E-05	7.43E-06	4.95E-06	2.06E-06	5.84E-07	1.25E-07
ENE	4023.	9.61E-06	4.29E-06	2.87E-06	1.19E-06	3.40E-07	7.30E-08
E	4023.	9.32E-06	4.15E-06	2.77E-06	1.15E-06	3.26E-07	6.96E-08
ESE	4023.	1.11E-05	5.03E-06	3.39E-06	1.44E-06	4.22E-07	9.38E-08
SE	4023.	2.67E-05	1.16E-05	7.64E-06	3.09E-06	8.44E-07	1.72E-07
SSE	4023.	3.88E-05	1.69E-05	1.11E-05	4.51E-06	1.23E-06	2.53E-07
Maximum sector χ/Q		4.50E-05					
Overall site χ/Q		4.19E-05	1.91E-05	1.29E-05	5.49E-06	1.61E-06	3.60E-07

TABLE 2.3-59d

LIMITING ATMOSPHERIC DISPERSION FACTOR, χ/Q (sec/m^3)

	χ/Q
Site Boundary	
0-2 hr.	1.5E-4
Low Population Zone	
0-8 hr.	1.9E-5
8-24 hr.	1.3E-5
24-96 hr.	5.3E-6
96-720 hr.	1.5E-6

Replace this table with the updated Table 2.3-59d

Table 2.3-59d Limiting Atmospheric Dispersion Factor, χ/Q (sec/m^3)						
	χ/Q values Limiting values for release from any one of the following sources: unit vent stack, equipment hatch, MSSVs/ARVs vent, TDAFW exhaust vent, or RWST					
Location	**0-2 hours**	**2-8 hours[1]**	**0-8 hours**	**8-24 hours**	**1-4 days**	**4-30 days**
EAB[2]	1.40E-4					
LPZ[3]	4.50E-5	2.39E-5	1.91E-5	1.29E-5	5.49E-6	1.61E-6

Notes:

1. This value was not calculated by PAVAN. It was obtained by logarithmic interpolation between the 2-hour average value $(\chi/Q)_{2hr}$ and the annual average value $(\chi/Q)_{1yr}$.
2. Defined as 1200 m distance.
3. Defined as 4023 m distance.

TABLE 2.3-78

VARIATION OF INTAKE K_C WITH WIND DIRECTION

UNIT VENT RELEASE

Wind Direction	Wolf Creek
N	0
NNE	0
NE	0
ENE	0.5
E	1.5
ESE	2.5
SE	1.5
SSE	0.5
S	0
SSW	0
SW	0
WSW	0
W	0
WNW	0
NW	0
NNW	0

Replace this table with the updated Table 2.3-78

Table 2.3-78

Relative Concentration (χ/Q - sec/m^3) at Control Room Air Intake

Equipment Hatch to Control Room Air Intake	
0 to 2 hours	5.31E-04
2 to 8 hours	4.22E-04
8 to 24 hours	1.72E-04
1 to 4 days	1.22E-04
4 to 30 days	9.91E-05
Unit Vent Stack to Control Room Air Intake	
0 to 2 hours	4.41E-04
2 to 8 hours	3.21E-04
8 to 24 hours	1.35E-04
1 to 4 days	8.66E-05
4 to 30 days	7.70E-05
MSSVs/ARVs to Control Room Air Intake	
0 to 2 hours	9.24E-04
2 to 8 hours	6.89E-04
8 to 24 hours	2.94E-04
1 to 4 days	1.88E-04
4 to 30 days	1.56E-04
RWST Vent to Control Room Air Intake	
0 to 2 hours	6.77E-04
2 to 8 hours	5.96E-04
8 to 24 hours	2.42E-04
1 to 4 days	2.04E-04
4 to 30 days	1.59E-04
TDAFW Exhaust to Control Room Air Intake	
0 to 2 hours	5.15E-04
2 to 8 hours	4.08E-04
8 to 24 hours	1.75E-04
1 to 4 days	1.10E-04
4 to 30 days	8.93E-05

TABLE 2.3-79

RELATIVE CONCENTRATION (X/Q) AT CONTROL BUILDING AIR INTAKE*

From Low Level Release

Percentage	Wolf Creek
5	5.33
10	3.62
20	0.66
40	0

For Unit Vent Release

Percentage	Wolf Creek
5	1.14
10	0.68
20	0.17
40	0

*Units for X/Qs are 10^{-4} m/sec^3

Replace this table with the updated Table 2.3-79

Table 2.3-79

Relative Concentration (χ/Q - sec/m^3) at the TSC Air Intake

Equipment Hatch to TSC Air Intake	
0 to 2 hours	3.91E-04
2 to 8 hours	2.66E-04
8 to 24 hours	9.62E-05
1 to 4 days	7.05E-05
4 to 30 days	5.52E-05
Unit Vent Stack to TSC Air Intake	
0 to 2 hours	2.80E-04
2 to 8 hours	1.80E-04
8 to 24 hours	6.44E-05
1 to 4 days	4.42E-05
4 to 30 days	3.22E-05
MSSVs/ARVs to TSC Air Intake	
0 to 2 hours	4.14E-04
2 to 8 hours	2.08E-04
8 to 24 hours	8.22E-05
1 to 4 days	5.21E-05
4 to 30 days	4.09E-05
RWST Vent to TSC Air Intake	
0 to 2 hours	1.87E-04
2 to 8 hours	1.25E-04
8 to 24 hours	4.51E-05
1 to 4 days	3.33E-05
4 to 30 days	2.61E-05
TDAFW Exhaust to TSC Air Intake	
0 to 2 hours	4.83E-04
2 to 8 hours	2.58E-04
8 to 24 hours	9.63E-05
1 to 4 days	6.45E-05
4 to 30 days	4.89E-05

been established one foot above the normal operating level, or at elevation 1,088 feet. Water-level determinations for the cooling lake are presented in USAR Sections 2.4.3.5 and 2.4.11.3.2.

This analysis shows that the average time of contaminant travel to the cooling lake is at least equal to half the expected life of the plant (Table 2.4-37). For this reason, an analysis has also been made for the case of an accidental release toward the end of the life of the plant. Although there are no plans to drain the cooling lake after decommissioning of WCGS, the conservative assumption is made that by the time the contaminants reach the shoreline after such an accident, the cooling lake may have been drained. Thus, consideration was given to contaminant transport down-gradient to the closest discharge point on the tributary to Wolf Creek, approximately 2,450 feet southwest of the radwaste building.

Wells C-20 and C-50 (Table 2.4-29 and Figure 2.4-52) are the nearest wells in the down-gradient direction that were not purchased by the Licensees or inundated by the cooling lake. They are the nearest potable water supplies. These wells are located approximately 10,500 and 13,700 feet, respectively, from the radwaste building. The shallow ground water that flows by these wells in the over-burden soils and the underlying Heumader Shale is physically separated from the plant site by the valleys of Wolf Creek and its tributaries, and by the cooling lake. Ground water coming from the direction of these two wells tends to flow toward the plant and discharge into the intervening streams. For this reason, analysis of ground-water transport from the radwaste tanks to the wells was not performed.

In the analysis which follows, it is shown that, with the exception of tritium concentrations, ground water contaminated at the plant site by accidental radioactive releases will have radionuclide concentrations below the maximum permissible concentrations of 10 CFR 20, Appendix B, Table II, for unrestricted areas by the time the contaminated ground water reaches the nearest surface water (the cooling lake or the Wolf Creek tributary). However, it is noted that tritium is a very weak beta emitter (decay energy for total disintegration = 0.0186 MeV) and also, the tritium-related offsite doses from this postulated accident will be a very small fraction of the 10 CFR Part ~~100~~ 50.67 dose limits. The following analysis also shows that the tritium concentration in the cooling lake and the Wolf Creek tributary will be well below the 10 CFR 20 limits for unrestricted areas. The effects of hydrodynamic dispersion, fluid convection, cation exchange, and radionuclide decay were included in the analysis.

3.0 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

This chapter identifies, describes, and discusses the principal architectural and engineering design features of those structures, components, equipment, and systems which are necessary to assure:

a. The integrity of the reactor coolant pressure boundary

b. The capability to shut down the reactor and maintain it in a post-accident safe shutdown condition

c. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline values of 10 CFR 50.67 ~~100~~.

3.1 CONFORMANCE WITH NRC GENERAL DESIGN CRITERIA

This section briefly discusses the extent to which the design criteria for safety-related plant structures, systems, and components comply with Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, "General Design Criteria for Nuclear Power Plants" (GDC). As presented in this section, each criterion is first quoted and then discussed in enough detail to demonstrate compliance with each criterion. For some criteria, additional information may be required for a complete discussion. In such cases, detailed evaluations of compliance with the various general design criteria are incorporated in more appropriate USAR sections, but are located by reference.

3.1.1 DEFINITION OF SINGLE FAILURE

The single failure criterion is a constraint used in the design of safety systems to improve the reliability of the system to perform its safety function following a design-basis event or design occurrence.

A single failure means an occurrence which results in the loss of the capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electrical systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming that passive components function properly) nor (2) a single failure of a passive component (assuming that active components function properly) results in a loss of the capability of the system to perform its safety functions.

group 2 Class 1E loads simultaneously. The 4160-V and 480-V circuit breakers and the associated equipment will be tested one at a time only while redundant equipment is operational.

The dc system is provided with detectors to indicate and alarm when there is a ground existing on any part of the system. During plant operation, normal maintenance may be performed.

Complete provisions for the testing of Class 1E electric power systems and the standby power supplies (diesel generators) are described in Chapter 8.0.

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."

DISCUSSION

A separate control room is provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain in a safe manner under accident conditions, including LOCAs. Operator action outside of the control room to mitigate the consequences of an accident is permitted. The control room and its post-accident ventilation systems are designed to satisfy seismic Category I requirements, as discussed in Chapter 3.0. Adequate concrete shielding and radiation protection are provided against direct gamma radiation and inhalation doses postulated to result from a ~~TID-14844~~ release of fission products inside the containment structure. The shielding and the control room standby air-conditioning system allow access to and occupancy of the control rooms 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. Refer to Chapter 15.0. Fission product removal is provided

total effective dose equivalent (TEDE)

in the control room recirculation equipment to remove iodine and particulate matter, thereby minimizing the ~~thyroid~~ dose which could result from the accident. The control room habitability features are described in Chapter 6.0.

In the event that the operators are forced to abandon the control room, panel-mounted local instrumentation and controls are provided to achieve and maintain the plant in the hot shutdown condition (see Chapter 7.0). The capability for bringing the plant to a cold shutdown is also provided outside the control room through the use of local controls.

3.1.5 PROTECTION AND REACTIVITY CONTROL SYSTEMS

CRITERION 20 - PROTECTION SYSTEM FUNCTIONS

"The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety."

DISCUSSION

A fully automatic protection system with appropriate redundant channels is provided to cope with transient events where insufficient time is available for manual corrective action. The design basis for all protection systems is in accordance with the intent of IEEE Standards 279-1971 and 379-1972. The reactor protection system automatically initiates a reactor trip when any variable monitored by the system or combination of monitored variables exceeds the normal operating range. Setpoints are designed to provide an envelope of safe operating conditions with adequate margin for uncertainties to ensure that the fuel design limits are not exceeded.

Reactor trip is initiated by removing power to the rod drive mechanisms of all the rod cluster control assemblies. This causes the rods to insert by gravity, thus rapidly reducing the reactor power. The response and adequacy of the protection system have been verified by analysis of anticipated transients.

The engineered safety features actuation system automatically initiates emergency core cooling and other safety functions by sensing accident conditions, using redundant analog channels measuring diverse variables. Manual actuation of safety features

structural and leaktight integrity of their components and to assure the operability and performance of the active components of the systems. All active components of the containment spray system and delivery piping up to the last powered valve before the spray nozzle have the capability to be tested during reactor power operation. In addition, when the unit is shutdown, smoke or air can be blown through the test connections for visual verification of the flow path. All safety-related active components of the containment fan cooling system can be tested to verify operability during reactor power operation. In addition, since the containment fan cooling system is a normally operating system, the performance and operability of portions of the system are continuously verified during normal reactor power operation. The facility design allows, under conditions as close to the design as practicable, the performance of a full operational sequence that brings these systems into operation. More complete discussions of the testing of these systems are in Chapters 6.0, 8.0, and the Technical Specifications.

CRITERION 41 - CONTAINMENT ATMOSPHERE CLEANUP

"Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quantity of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

"Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure."

DISCUSSION

The containment spray system serves to remove radioiodine and other airborne particulate fission products from the containment atmosphere following a LOCA. The system consists of two independent systems, each supplied from separate electrical power busses, as described in Chapter 8.0. Either subsystem alone can provide the fission product removal capacity for which credit is taken in Chapter 15.0, in compliance with Regulatory Guide ~~1.4~~. (See Section 3A for discussion of RG ~~1.4~~) 1.183

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Certain structures, components, and systems of the nuclear plant are considered to serve a safety function because they:

a. Assure the integrity of the reactor coolant pressure boundary.

b. Assure the capability to shut down the reactor and maintain it in a safe condition.

c. Assure the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR ~~100~~. 50.67

d. Contain or may contain radioactive material.

The purpose of this section is to classify structures, systems, and components according to the importance of the item in order to provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. Table 3.2-1 delineates each of the items in the plant which fall under the above-mentioned categories and the respective associated classification that the NRC, ANS, and industrial codes committees have developed. Each of the classification categories in Table 3.2-1 is addressed in the following sections.

For identification of system and subsystem boundaries, Table 3.2-1 is supplemented (i.e., referenced to applicable figures) by piping and instrument diagrams which have been marked to clearly show the limits of the seismic Category I and various quality group classifications on a system. The legend for the piping and instrument diagrams is provided in Figure 1.1-1.

Classification of power supplies, instrumentation and controls, motors, piping and valves, ductwork and dampers, and associated supports, hangers, and restraints is not delineated in Table 3.2-1 because of the extensive listing required. Their classification, however, is consistent with the boundaries shown on the piping and instrumentation drawings. A listing of the piping and instrumentation drawings and their associated USAR figures is found in Table 1.7-2.

TABLE 3.2-3 (Sheet 5)

, except that WCGS is using meteorology as recommended by Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors"

Regulatory Guide 1.29 Position	WCGS
o. Primary and secondary reactor containment.	o. Complies. Note that the WCGS design does not incorporate a secondary containment.
p. Systems1, other than radioactive waste management systems, not covered by items 1.a through 1.o above that contain radioactive material and whose postulated failure would result in conservatively calculated potential offsite doses (using meteorology as prescribed by Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors") that are more than 0.5 rem to the whole body or its equivalent to any part of the body.	p. Complies. Note that Regulatory Guide 1.143 provides guidance on radioactive waste management systems and structural seismic design. Table 3.2-1 indicates those systems for which the D (Augmented) design criteria are applied. The dividing line value of 0.5 rem is inappropriate for the types of failures which the guide addresses. Quality Group D or D (Augmented) is applied to such systems unless their failure would result in offsite doses approaching the guide values of 10 CFR Part 10~~0~~.
q. The Class IE electric systems, including the auxiliary systems for the onsite electric power supplies, that provide the emergency electric power needed for functioning of plant features included in items 1.a through 1.p above.	q. Complies; however in certain cases Class 1E conduits are supported from non-Category I seismic walls. Although not Category I, these reinforced block walls are analyzed for SSE loads in accordance with position Z and are subject to the QA program described in Position 4.

which were the 10 CFR Part 20 limits at that time. Since then, the value was changed from 0.5 rem whole body to 0.1 rem TEDE.

TABLE 3.2-4 (Sheet 5)

Regulatory Guide 1.26 Position	WCGS
c. Systems or portions of systems that are connected to the reactor coolant pressure boundary and are capable of being isolated from that boundary during all modes of normal reactor operation by two valves, each of which is either normally closed or capable of automatic closure.	c. Complies.
d. Systems, other than radioactive waste management systems, not covered by items 2.a through 2.c above that contain radioactive material and whose postulated failure would result in conservatively calculated potential offsite doses (using meteorology as recommended by Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors") that exceed 0.5 rem to the whole body or its equivalent to any part of the body. For those systems located in Seism Category I structures, only single component failures need be assumed However, no credit for automatic isolation from other components in the system or for treatment of released material should be taken unless the isolation or treatment capability is designed	d. Complies. Note that Regulatory Guide 1.143 provides guidance on radioactive waste management system design. Table 3.2-1 indicates those systems to which the D (Augmented) design criteria are applied. The dividing line value of 0.5 rem is inappropriate for the types of failures which the guide addresses. Quality Group D [or D (Augmented)] is applied to such systems unless their failure would result in offsite doses approaching the guideline values of 10 CFR Part 100. Radwaste systems, except for portions of the steam generator blowdown system located in the turbine building, are located within a seismically designed building as permitted by Regulatory Guide 1.143, and only single component failures are considered.

, except that WCGS is using meteorology as recommended by Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors"

which were the 10 CFR Part 20 limits at that time. Since then, the value was changed from 0.5 rem whole body to 0.1 rem TEDE.

3.5.1.5 Missiles Generated by Events Near the Site

As described in Section 2.2.3, there are no postulated explosions or military activities in the site vicinity that could generate missiles.

3.5.1.6 Aircraft Hazards

The Burlington Municipal Airport has been replaced by the Coffey County Airport which opened in 1989 and is not described in Section 2.2.1.3. The hazards associated with the new airport have been evaluated and do not constitute a significant hazard as defined in Standard Review Plan, Section 3.5.1.6. The following evaluation applies to hazards due to the Coffey County Airport.

a. The following small airport is within 5 miles of the station:

1. the Coffey County Airport located 4.5 miles north-northwest of the site. It is classified as a small aircraft airport servicing primarily single and twin engine piston type aircraft.

b. There are no airports between 5 and 10 miles of the station.

c. There are no airports outside 10 miles of the plant with the projected annual number of operations greater than 1000 d (d=miles from site to airport).

d. There is a low-altitude federal airway and a high-altitude jet route passing within 2 miles of the plant which have widths of 8 and 16 nautical miles, respectively. In addition, there is a low level military training route whose centerline passes within 17 miles of the plant. A probability analysis of the aircraft accidents on these routes has indicated that the probability of accidents leading to radiological consequences worse than the exposure guidelines of 10 CFR ~~100~~ 50.67 is less than 10-7 per year. The details of this analyse given in the following.

V-234 is an east-west low-altitude route, and its centerline passes within 3.9 miles of the plant site. It has a width of 8 nautical miles. V-131 is a north-south route passing within 6.1 miles of the plant site. It has a width of 8 nautical miles. Daily traffic on these routes and from direct flights traversing the airspace overlying Wolf Creek is reported to be 161 flights.

J-110 is a high-altitude east-west jet route passing within 0.5 miles of the plant site. It has a width of 16 nautical miles. Daily traffic on this route and from direct flights traversing the airspace overlying Wolf Creek is reported to be 132 flights.

IR-502 is a military low-level training route whose centerline passes within 17 miles of the plant. It has a width of 8 nautical miles, and the annual number of flights on this route is 1,560. Conservatively, the hazard from this route has been accounted for in the analysis with the assumption of an in-flight crash rate 2 times higher than that for general aviation.

The traffic count on low-altitude air routes does not include the aircraft operating under visual flight rules (VFR). It is conservatively assumed that the VFR traffic is equal to the IFR (instrumental flight rule) traffic reported above. The increase of traffic in the future is practically offset by a decrease in accident rates. The area of the safety-related structures in the plant whose damaged would lead to unacceptable radiological consequences is calculated to be 0.008 mi^2.

AIRCRAFT IMPACT PROBABILITY DUE TO AIR TRAFFIC

The probability, PFA, of an aircraft crashing into the plant and leading to radiological consequences in excess of 10 CFR ~~100~~ 50.67 exposure guidelines is calculated as follows:

$$PFAY = C\ A \left(\frac{N_L}{W_L} + \frac{2N_T}{W_T} + \frac{N_H}{W_H}\right)$$

where: C = inflight crash rate per mile for aircraft using airway = 4×10^{-10};

A = area of safety-related structures whose damage would lead to unacceptable radiological consequences = 0.008 mi^2;

N_L = number of aircraft movements on low-altitude federal air routes V-234 and V-131 [161 x 2] = 322/day = 117,530/year;

W_L = width of low-altitude air route = 8 nautical miles = 9.2 miles;

N_T = number of aircraft movements on the training route IR-502 = 1,560/year;

W_T = width of training route IR-502 (plus twice the distance from the airway edge to the site since the site is outside the airway) = 34.0 miles;

N_H = number of aircraft movements on high-altitude federal jet route J-110 = 132/day = 48,180/year; and

W_H = width of jet route J-110 = 18.4 miles.

Using these data, the probability of an aircraft crashing into the plant and causing unacceptable radiological consequences is calculated as 5.0×10^{-8} per year.

AIRCRAFT IMPACT PROBABILITY DUE TO COFFEY COUNTY AIRPORT

The Standard Review Plan (section 2.2) specifies a method for calculating the probability of aircraft impact due to airports located near a nuclear power plant. The probability per year of an aircraft crashing into the site (PA) is calculated by using the following expression:

PA = C x N x A

where:

C = probability per square mile of a crash per aircraft movement N = number of aircraft movements
A = effective plant area (in square miles)

Where multiple aircraft types and/or trajectories are involved this expression may be summed for each separate aircraft type and trajectory considered. Aircraft using the Coffey County Airport are lumped into a single category for the purposes of calculating the effective area as described below. No credit was taken for separate trajectories.

Numerical values for "C" are given in the standard review plan as a function of the distance from the airport to the nuclear power plant. The Coffey County Airport is located approximately 4.5 miles from Wolf Creek. The probability for general aviation aircraft for airports from 4 to 5 miles from nuclear power plants is given as 1.2×10^{-8} per aircraft movement. The safety of general aviation has improved significantly in the intervening years and therefore the SRP crash probabilities are now very conservative. The National Transportation Safety Board publishes the "Annual Review of Aircraft Accident Data-U.S.

General Aviation." These publications were reviewed for the period of 1982 through 1986 (the most recent 5 years of available data) and the number of aircraft accidents occurring within 5 miles of airports has decreased by more than half in this 5 year period alone.

The effective area of the plant "A" is the portion of the plant that is susceptible to impact from a given type of aircraft and could result in radiological consequences greater than 10 CFR ~~100~~ 50.67 guidelines. For the large commercial and military aircraft considered in the impact probability due to air traffic, a value of 0.008 miles^2 was used. However, for the small general aviation aircraft utilizing the Coffey County Airport a significantly smaller value is appropriate. Calculations show than an effective area of 0.0016 miles^2 may conservatively be used for general aviation aircraft with approach speeds of <140 mph and weighing <12,500 pounds. These parameters envelope the expected aircraft usage at Coffey County Airport through the year 2000.

Substituting the values described above along with actual usage levels results in:

$PA = 4.0 \times 10^{-8}$

When combined with the probability originally calculated for air routes of 5×10^{-8} this results in a total probability of aircraft impacts causing significant radiological releases of 9×10^{-8} per year. This result remains below the value of 1×10^{-7} per year given in the SRP as acceptable for siting of nuclear power plants.

Since the aircraft movements at the airports and on the air routes do not pose any undue risk to the safe operation of WCGS Unit No. 1, no design-basis aircraft impact is postulated.

restraint or second check valve will not result in an uncontrolled loss of reactor coolant if either of the two valves in the line closes.

Accordingly, both of the automatic isolation valves are suitably protected and restrained as close to the valves as possible so that a pipe break beyond the restraint will not jeopardize the integrity and operability of the valves. Further, periodic testing capability of the valves to perform their intended function is essential. This criterion takes credit for only one of the two valves performing its intended function. For normally closed isolation or incoming check valves (Cases I and IV in Figure 3.6-2), a LOCA is assumed to occur for pipe breaks on the reactor side of the valve.

Branch lines connected to the reactor coolant loop (RCL) are defined as "large" for the purpose of this criteria and as having an inside diameter greater than 4 inches up to the largest connecting line, generally the pressurizer surge line. Rupture of these lines results in a rapid blowdown from the RCL, and protection is basically provided by the accumulators and the low head safety injection pumps (residual heat removal pumps).

Branch lines connected to the RCL are defined as "small" if they have an inside diameter equal to or less than 4 inches. This size is such that emergency core cooling system analyses, using realistic assumptions, show that no clad damage is expected for a break area of up to 12.5 square inches, corresponding to 4-inch inside diameter piping.

Engineered safety features are provided for core cooling and boration, pressure reduction, and activity confinement in the event of a LOCA or steam or feedwater line break accident to ensure that the public is protected in accordance with 10 CFR ~~100~~ 50.67 guidelines. These safety systems are designed to provide protection for a reactor coolant system pipe rupture of a size up to and including a double-ended severance of a reactor coolant loop.

In order to assure the continued integrity of the vital components and the engineered safety systems, consideration is given to the consequential effects of the pipe break itself to the extent that:

a. The minimum performance capabilities of the engineered safety systems are not reduced below that required to protect against the postulated break.

3.7(N) SEISMIC DESIGN

For the OBE loading condition, the nuclear steam supply system is designed to be capable of continued safe operation. The design for the SSE is intended to ensure:

a. That the integrity of the reactor coolant pressure boundary is not compromised;

b. That the capability to shut down the reactor and maintain it in a safe condition is not compromised; and

c. That the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR ~~100~~ 50.67 is not compromised.

It is necessary to ~~ensure~~ that required critical structures and components do not lose their capability to perform their safety function. Not all critical components have the same functional safety requirements. For example, a safety injection pump must retain its capability to function normally during the SSE. Therefore, the deformation in the pump must be restricted to appropriate limits in order to ensure its ability to function. On the other hand, many components can experience significant permanent deformation without loss of function. Piping and vessels are examples of the latter where the principal requirement is that they retain their contents and allow fluid flow.

The seismic requirements for safety-related instrumentation and electrical equipment are covered in Sections 3.10(N) and (B). The safety class definitions, classification lists, operating condition categories, and the methods used for seismic qualification of mechanical equipment are given in Section 3.2.

3.7(N).1 SEISMIC INPUT

3.7(N).1.1 Design Response Spectra

Refer to Section 3.7(B).1.1.

3.7(N).1.2 Design Time History

Refer to Section 3.7(B).1.2.

welds where the welds are inaccessible to nondestructive examination after construction. Refer to Figure 3.8-29 for typical details for these items.

3.8.1.1.4 Shell Discontinuities

The significant discontinuities in the shell structure are at the wall-to-base-slab connection, the buttresses, and the large penetration openings.

The shell wall interface at the base slab incorporated a straight wall-to-slab joint. Refer to Figure 3.8-10 for details of the lower wall configuration.

Buttresses project out from the exterior surface of the shell wall and dome to provide adequate space for the hoop tendon anchorage and tendon-stressing equipment. The anchorage surfaces of the buttress are normal to the tangent line of the anchored hoop tendons. Details are shown in Figure 3.8-30.

The concrete shell around the equipment hatch opening is thickened by the method shown in Figures 3.8-31 and 3.8-32.

3.8.1.1.5 Special Reinforcing Requirements

Special reinforcing is required in such areas as the major penetrations. Refer to Figures 3.8-31 through 3.8-35 for typical details in these areas.

3.8.1.2 Applicable Codes, Standards, and Specifications

The following codes, regulations, standards, and specifications were utilized in the reactor building design. Subsequent to operation, additional codes have been approved for use and are noted with an asterik.

3.8.1.2.1 Regulations

a. 10 CFR 50, "Licensing of Production and Utilization Facilities"

b. 10 CFR 100, "Reactor Site Criteria"

3.8.1.2.2 Codes

a. American Concrete Institute, Building Code Requirements for Reinforced Concrete (ACI-318-71)

c. 10 CFR 50.67, "Accident Source Term"

categorized according to NUREG-0588, Appendix E, for each of three accident groups. The accident groupings were LOCA, MSLB/MFLB inside containment and steam/feedwater tunnel, and HELB outside containment (except MSLB and MFLB). The equipment located in a harsh environment, for any of the three accident groups, was reviewed under the NUREG-0588 program.

Equipment required as a result of NUREG-0737 was incorporated into the design by the time Table 3.11(B)-3 was being developed. Accordingly, new equipment added to the design was included in the equipment list and reviewed to the same criteria as all other safety-related equipment. USAR Section 18.0 identifies the Wolf Creek design relative to the NUREG requirements.

Table 3.11(B)-10 provides a listing of equipment added as a result of NUREG-0737. It should be noted that for every piece of equipment identified in Table 3.11(B)-10, there is other support equipment that is purchased in bulk quantities that cannot be clearly identified as supporting the NUREG-0737 requirements (e.g., cable, terminal boxes, connectors, hangers, instrument isolation valves, etc.). This generic type equipment is included in the qualification review program. For further information about each of the devices listed in Table 3.11(B)-10, refer to the equipment listing in Table 3.11(B)-3.

3.11(B).1.1.2 Safety-Related System Listing

Safety-related systems are those plant systems necessary to ensure:

a. The integrity of the reactor coolant pressure boundary.

b. The capability to shut down the reactor and maintain it in a safely shutdown condition.

c. The capability to prevent or mitigate the consequences of accidents which could result in offsite exposures comparable to the guidelines of 10 CFR ~~100~~. 50.67

Systems that perform these type functions are those systems required to achieve or support emergency reactor shutdown, containment isolation, reactor core cooling, containment heat removal, core residual heat removal, and prevention of significant release of radioactive material to the environment. A listing of the systems that perform or support these functions is provided in Table 3.11(B)-9. The listing identifies the function that the system performs (or supports) and includes all systems that receive Class 1E power. Systems are listed even if only a portion of the system provides a safety-related function. Multiple

conservative bounding time for termination of accident effects within the containment. The containment pressure-temperature analysis, as reflected in Figures 3.11(B)-3 and 6, indicates that containment conditions return to normal or below normal operating conditions within 30 days. It should also be noted that Regulatory Guide ~~1.4~~, provides criteria for evaluating the offsite radiological conseque 1.183 of a LOCA event for a maximum of 30 days following the accident.

Margins of 1 hour or more for equipment with required operability times of less than 10 hours have generally been used for the WCGS equipment qualification review. However, margins of less than 1 hour have been used when adequate technical justification could be provided.

The Operating Agent concurs with the AIF position on the 1-hour time margin, as stated in a letter to Mr. Harold Denton dated January 4, 1982, in that an arbitrary time margin of 1 hour appears inappropriate and should not be required when adequate technical justification for a shorter period exists.

3.11(B).5.3 Margins

The discussions in Section 3.11(B).1 show that post-accident environmental parameters were conservatively and uniquely determined using plant-specific data. Hence, the guideline generic techniques discussed in NUREG-0588 are not applicable.

The values for margin identified in Section 6.3.1.5 of IEEE 323-1974 were used as acceptance criteria during the NUREG-0588 review. The only regular exception to the IEEE 323-1974 margins was for radiation. As identified in Item 1.4 of NUREG-0588, additional margin need not be added to the radiation parameters if the methods identified in Appendix D of NUREG-0588 are utilized. The methods used to determine the WCGS radiation parameters are consistent with the Appendix D methodology. Hence, the radiation margins required by Section 6.3.1.5 of IEEE 323-1974 were not necessary.

3.11(B).5.4 Aging

During the NUREG-0588 review, two general observations were made concerning equipment aging:

1. Some IEEE 323-1974 equipment underwent accelerated thermal aging based on the Arrhenius method. This approach was considered acceptable.

DISCUSSION:

The recommendations of this regulatory guide are not applicable to a PWR.

REGULATORY GUIDE 1.4 REVISION 2 DATED 6/74

Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors

DISCUSSION: Use of this regulatory guide has been replaced by Regulatory Guide1.183 for alternative source term application.

~~The recommendations of this regulatory guide are met as described in Table 15.6-7.~~

REGULATORY GUIDE 1.5 REVISION 0 DATED 3/71

Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors (Safety Guide 5)

DISCUSSION:

The recommendations of this regulatory guide are not applicable to a PWR.

REGULATORY GUIDE 1.6 REVISION 0 DATED 3/71

Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems (Safety Guide 6)

DISCUSSION:

The recommendations of this regulatory guide are met. Refer to Section 8.1.4.3.

REGULATORY GUIDE 1.7 REVISION 2 DATED 11/78

Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident

DISCUSSION:

The recommendations of this regulatory guide are met as described in Table 6.2.5-6.

10 CFR 50.44 was revised in 2003 and Revision 3 to Regulatory Guide 1.7 was issued in May 2003. The revised 10 CFR 50.44 no longer defines a design-basis LOCA hydrogen release, and eliminates the requirements for hydrogen control systems to mitigate such a release. License Amendment No. 157 was issued by the NRC on January 31, 2005 and deleted the Technical Specification requirements for the hydrogen recombiners and relocated the requirements for the hydrogen monitors.

REGULATORY GUIDE 1.8 DRAFT REVISION 2 DATED 2/79

Personnel Selection and Training

The recommendations of this regulatory guide are met as described in Table 15.7-1.

REGULATORY GUIDE 1.25 REVISION 0 DATED 3/72

Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25)

DISCUSSION: Use of this regulatory guide has been replaced by Regulatory Guide1.183 for alternative source term application.

~~The recommendations of this regulatory guide are met as described in Table 15.7-2.~~

REGULATORY GUIDE 1.26 REVISION 3 DATED 2/76

Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants

DISCUSSION:

The recommendations of this regulatory guide are met as described in Table 3.2-4. As described in Section 3.2, Westinghouse utilizes the safety classes as defined in ANSI N18.2a-1975. Except for the deviation described in section 3.2.3.

REGULATORY GUIDE 1.27 REVISION 2 DATED 1/76

Ultimate Heat Sink for Nuclear Power Plants

DISCUSSION:

The recommendations of this regulatory guide are met as described in Table 9.2-21 and Section 9.2.5.

REGULATORY GUIDE 1.28 REVISION 2 DATED 2/79

Quality Assurance Program Requirements (Design and Construction)

DISCUSSION:

Refer to the SNUPPS Quality Assurance Programs for Design and Construction. This regulatory guide is not applicable to the operating phase.

REGULATORY GUIDE 1.142 REVISION 0 DATED 4/78

Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments)

DISCUSSION:

The recommendations of this regulatory guide, which generally endorses ACI-349-76, have not been applied to the design of safety-related concrete structures of the power block. The procedures and requirements described in ACI 318-71, Building Code Requirements for Reinforced Concrete, along with the exceptions, clarifications, and additions described in Sections 3.8.3, 3.8.4, and 3.8.5, have been used instead.

REGULATORY GUIDE 1.143 REVISION 0 DATED 7/78

Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants

DISCUSSION:

The recommendations of this regulatory guide are met as described in Table 3.2-5.

REGULATORY GUIDE 1.144 REVISION 1 DATED 9/80

Auditing of Quality Assurance Programs for Nuclear Power Plants

DISCUSSION:

The recommendations of this regulatory guide are met, the following is provided as clarification:

Section 4.5.1 of ANSI N45.2.12-1977 states in part, "In the event that corrective action cannot be completed within thirty days, the audited organization's response shall include a scheduled date for the corrective action." WCNOC satisfies this requirement by using the following process:

Audit findings are documented using the WCNOC corrective action process (PIRs). As part of the WCNOC corrective action process PIRs are assigned a scheduled completion date for non-significant or a date for completion of the root cause and planned corrective action for significant PIRs prior to the responsible organization receiving it. This tracking system tracks the status of the finding through to closure.

REGULATORY GUIDE 1.145 REVISION ~~0~~ 1 DATED ~~8/79~~ 11/82 (Reissued 2/83)

Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants

DISCUSSION:

The recommendations of this regulatory guide are met as discussed in Section ~~2.3.~~ 2.3.4.

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirement of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements and frequency specified by ASME Section XI code, Subsection IWE, except where relief has been authorized by the NRC.

DISCUSSION:

The recommendations of this regulatory guide are met as described in Section 6.2.6.

REGULATORY GUIDE 1.166 REVISION 0 DATED 3/97

Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Postearthquate Actions

The recommendations of this regulatory guidance are met, except for the following:

1. WCGS is not subject to the requirements of Appendix S to 10 CFR Part 50. WCGS will continue to use appendix A to 10 CFR Part 100.

REGULATORY GUIDE 1.167 REVISION 0 DATED 3/97

Restart of a Nuclear Power Plant Shut Down by a Seismic Event

DISCUSSION:

The recommendations of this Regulatory Guide are met.

REGULATORY GUIDE 1.181 REVISION 0 DATED 9/99

Content of the Updated Final Safety Analysis Report in Accordance With 10 CFR 50.71(e)

DISCUSSION:

NEI 98-03, Revision 1, "Guidelines for Updating FSARs," was endorsed by this regulatory guide without exception. NEI 98-03, Revision 1, provides guidance for updating the USAR consistent with 10 CFR 50.71(e), and provides for making voluntary modifications to the USAR (i.e., removal, reformatting, and simplification of information, as appropriate) to improve focus, clarity and maintainability.

WCNOC currently utilizes NEI 98-03 as guidance for determining what information in the USAR is to be updated, to what level of detail the update needs to reflect, and what type of information may be removed from the USAR.

(Refer to discussion on Regulatory Guide 1.70, Revision 3, for background information.)

REGULATORY GUIDE 1.187 REVISION 0 DATED 11/00

Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments

Insert Reg Guide 1.183

Insert to Reg Guide 1.183

Regulatory Guide 1.183 Revision 0 Dated 7/00

Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors

DISCUSSION:

The recommendations of this regulatory guide are met as described in Appendix 15B.

DISCUSSION:

NEI 96-07, Revision 1, "Guidelines for 10 CFR 50.59 Implementation," was endorsed by this regulatory guide without exception. NEI 96-07 provides guidance and examples to aid in the application of the revised regulation to plant activities. WCNOC currently uses NEI 96-07 as a basis for the procedures, forms and guidance for implementation of the regulation. However, WCNOC has taken some minor exceptions to the guidance.

First, WCNOC reports on an annual basis evaluations performed and approved by the PSRC within a calendar year regardless of implementation status. This method of reporting is conservative in approach and has been the established format since licensing.

Second, WCNOC has broadened the use of administrative changes to consider the activity rather than the type of document containing the activity. In essence then, Operating procedures may screen out of 50.59 if the change is administrative.

Finally, WCNOC's commitment to Generic Letter 83-11, Supplement 1 are limited therefore, WCNOC will not utilize that allowance to the extent described in NEI 96-07.

REGULATORY GUIDE 1.192 CURRENT REVISION INCORPORATED BY REFERENCE IN 10CFR50.55a(b)

Operation and Maintenance Code Case Acceptability, ASME OM Code

DISCUSSION:

Refer to the discussion on Regulatory Guide 1.147 for ASME OM Code Cases for use in the second interval of the inservice testing program.

REGULATORY GUIDE 1.196 REVISION 0 DATED 5/03

Control Room Habitability at Light -

DISCUSSION:

The Control Room Habitability Program
consistent with the guidance of TSTF-
specific aspects of Regulatory Guide

The recommendations of this regulator
and preventive maintenance of the co

REGULATORY GUIDE 1.194 REVISION 0
DATED 6/2003
Atmospheric Relative Concentrations for Control Room Radiological
Habitability Assessments at Nuclear Plants
DISCUSSION:
The recommendations of this regulatory guide are met as discussed in Section 2.3.4.

REGULATORY GUIDE 1.197 REVISION 0 DATED 5/03

Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors

DISCUSSION:

The Control Room Envelope Habitability Program in Technical Specification 5.5.18 is consistent with the guidance of TSTF-448, Revision 3, which incorportes specific aspects of Regulatory Guide 1.197. An exception to Section C.1 and C.2 is taken in that the Tracer Gas Test based on the Brookhaven National Laboratory Atmospheric Tracer Depletion (ATD) Method is used to determine the unfiltered air inleakage past the control room envelope and control building envelope boundaries.

4.2 FUEL SYSTEM DESIGN

The plant design conditions are divided into four categories in accordance with their anticipated frequency of occurrence and risk to the public: Condition I - Normal Operation; Condition II - Incidents of Moderate Frequency; Condition III - Infrequent Incidents; and Condition IV - Limiting Faults. Chapter 15.0 describes bases and plant operation and events involving each condition.

The reactor is designed so that its components meet the following performance and safety criteria:

a. The mechanical design of the reactor core components and their physical arrangement, together with corrective actions of the reactor control, protection, and emergency cooling systems (when applicable) ensure that:

   1. Fuel damage* is not expected during Condition I and Condition II events. It is not possible, however, to preclude a very small number of rod failures. These are within the capability of the plant cleanup system and are consistent with plant design bases.

   2. The reactor can be brought to a safe state following a Condition III event with only a small fraction of fuel rods damaged** although sufficient fuel damage might occur to preclude immediate resumption of operation.

   3. The reactor can be brought to a safe state and the core can be kept subcritical with acceptable heat transfer geometry following transients arising from Condition IV events.

b. The fuel assemblies are designed to withstand loads induced during shipping, handling, and core loading without exceeding the criteria of Section 4.2.1.5.

c. The fuel assemblies are designed to accept control rod insertions in order to provide the required reactivity control for power operations and reactivity shutdown conditions (if in such locations).

* Fuel damage as used here is defined as penetration of the fission product barrier (i.e., the fuel rod clad).

** In any case, the fraction of fuel rods damaged must be limited so as to meet the dose guideline of 10 CFR ~~100~~. 50.67

4.3 NUCLEAR DESIGN

4.3.1 DESIGN BASES

This section describes the design bases and functional requirements used in the nuclear design of the fuel and reactivity control system and relates these design bases to the General Design Criteria (GDC) presented in 10 CFR 50, Appendix A. Where applicable, supplemental criteria such as the "Final Acceptance Criteria for Emergency Core Cooling Systems" are addressed. But, before discussing the nuclear design bases, it is appropriate to briefly review the four major categories ascribed to conditions of plant operation.

The full spectrum of plant conditions is divided into four categories, in accordance with the anticipated frequency of occurrence and risk to the public:

a. Condition I - Normal Operation
b. Condition II - Incidents of Moderate Frequency
c. Condition III - Infrequent Faults
d. Condition IV - Limiting Faults

In general, the Condition I occurrences are accommodated with margin between any plant parameter and the value of that parameter which would require either automatic or manual protective action. Condition II incidents are accommodated with, at most, a shutdown of the reactor with the plant capable of returning to operation after corrective action. Fuel damage (fuel damage as used here is defined as penetration of the fission product barrier, i.e., the fuel rod clad) is not expected during Condition I and Condition II events. It is not possible, however, to preclude a very small number of rod failures. These are within the capability of the chemical and volume control system (CVCS) and are consistent with the plant design basis.

Condition III incidents do not cause more than a small fraction of the fuel elements in the reactor to be damaged, although sufficient fuel element damage might occur to preclude immediate resumption of operation. The release of radioactive material due to Condition III incidents is not sufficient to interrupt or restrict public use of those areas beyond the exclusion radius. Furthermore, a Condition III incident does not by itself generate a Condition IV fault or result in a consequential loss of function of the reactor coolant or reactor containment barriers.

Condition IV occurrences are faults that are not expected to occur but are defined as limiting faults which must be designed against. Condition IV faults do not cause a release of radioactive material that results in exceeding the limits of 10 CFR ~~100~~. 50.67

The core design power distribution limits related to fuel integrity are met for Condition I occurrences through conservative design and maintained by the action of the control system. The requirements for Condition II occurrences are met by providing an adequate protection system which monitors reactor parameters. The control and protection systems are described in Chapter 7.0, and the consequences of Condition II, III, and IV occurrences are given in Chapter 15.0.

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TABLE OF CONTENTS (Continued)

CHAPTER 6.0

ENGINEERED SAFETY FEATURES

Engineered safety features (ESF) are those safety-related systems and components designed to directly mitigate the consequences of a design basis accident by:

a. Protecting the fuel cladding

b. Ensuring the containment integrity

c. Limiting fission product releases to the environment within the guideline values of 10 CFR, Part ~~100~~ 50.67

The limiting design basis accidents which are discussed and analyzed in Chapter 15.0 and Section 6.3 are:

a. Loss-of-coolant accident (LOCA)

b. Main steam line break (MSLB)

c. Steam generator tube rupture

d. Fuel handling accident

(Items a and b are also discussed in Section 6.2)

The engineered safety features consist of the following systems:

a. Containment (Section 6.2.1)

b. Containment heat removal (Section 6.2.2)

c. Containment isolation (Sections 6.2.4 and 6.2.6)

d. Containment combustible gas control (Section 6.2.5)

e. Emergency core cooling (Section 6.3)

f. Fission product removal and control systems (Section 6.5)

g. Emergency HVAC and filtration (Section 9.4)

h. Control room habitability (Section 6.4)

i. Auxiliary feedwater (Section 10.4.9)

The containment is provided to contain radioactivity following a LOCA.

6.2 CONTAINMENT SYSTEMS

The containment systems include the containment, the containment heat removal systems, the containment isolation system, and the containment combustible gas control system.

The design basis accident (DBA) is defined as the most severe of a spectrum of hypothetical loss-of-coolant accidents (LOCA). The ability of the containment systems to mitigate the consequences of a DBA depends upon the high reliability of these systems. This section provides the design criteria and evaluations to demonstrate that these systems function within the specified limits throughout the unit operating lifetime.

6.2.1 CONTAINMENT FUNCTIONAL DESIGN

A physical description of the containment and the design criteria relating to construction techniques, static loads, and seismic loads is provided in Section 3.8. This section pertains to those aspects of containment design, testing, and evaluation that relate to the accident mitigation function.

6.2.1.1 Containment Structure

6.2.1.1.1 Design Bases

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the DBA without exceeding the design leakage rate, as required by 10 CFR 50, Appendix A, General Design Criterion 50, and that, in conjunction with the other containment systems and the other engineered safety features, the release of radioactive material subsequent to a DBA does not result in doses in excess of the guideline values specified in 10 CFR ~~100~~ 50.67. The radiological consequences of the DBA are presented in Section 15.6.

a. Assumed Accident Conditions

For the purpose of determining the design pressure requirements for the containment structure and the containment internal structures, the following simultaneous occurrences were assumed:

1. The postulated reactor coolant system pipe rupture, as listed in Table 6.2.1-1, was assumed to be concurrent with the loss of offsite power and the worst single active failure. No two pipe breaks were assumed to occur simultaneously or consecutively.

f. Bases for Containment Depressurization Rate

To meet the containment safety design basis of limiting the release of radioactive material subsequent to a DBA so that the doses are within the guideline values specified in 10 CFR 100, the containment pressure is reduced to less than 50 percent of the peak calculated pressure within 24 hours after an accident. Chapter 15.0 contains the assumptions used in the analysis of the offsite radiological consequences of the accident.

50.67

g. Bases for Minimum Containment Pressure Used in ECCS Performance Studies

The minimum containment pressure transient used in the analysis of the emergency core cooling system's capability is based on the conservative overestimated heat removal capability and pressure reduction capability of the containment structures and the containment systems and on the conservative reactor coolant system thermal analysis provided in Section 15.6. The determination and evaluation of the minimum containment pressure transient are provided in Section 6.2.1.5.

6.2.1.1.2 Design Features

The principal containment and containment subcompartment design parameters are provided in Table 6.2.1-2. General arrangement drawings for the reactor containment are provided in Figures 1.2-9 through 1.2-18. Simplified arrangement drawings illustrating the nodalization model used for the containment subcompartment analyses are provided in Figures 6.2.1-27 through 6.2.1-33, 6.2.1-43 through 6.2.1-55, and 6.2.1-76.

a. Missile and Pipe Whip Protection

Missile shield considerations are described in Section 3.5. The structural design of the containment and the containment subcompartments is discussed in Section 3.8. The designed structural strength considers the effects of pipe whip and jet forces, as discussed in Section 3.6.

b. Codes and Standards

The codes, standards, and guides applied in the design of the containment structure and the containment internal structures are identified in Section 3.8.

SAFETY DESIGN BASIS EIGHT - The hydrogen purge subsystem serves as a backup to the hydrogen recombiners and is capable of venting and purging the containment atmosphere in order to maintain the hydrogen concentration below 4.0 volume percent following a LOCA. With the purge system operating, the doses at the exclusion area boundary and the low population zone outer boundary does not exceed the guideline values of 10 CFR ~~100~~ 50.67. Except for the containment penetration and associated isolation valves, the purge subsystem is not redundant or seismic Category I, as allowed by Regulatory Guide 1.7.

SAFETY DESIGN BASIS NINE - The containment design and hydrogen mixing provisions ensure adequate mixing of the containment atmosphere in order to eliminate stagnant pockets and prevent stratification of the hydrogen-air mixture.

SAFETY DESIGN BASIS TEN - The hydrogen monitoring subsystem is designed to inform the operator of the hydrogen concentration inside the containment and to provide periodic samples of the post-LOCA containment atmosphere to be analyzed for hydrogen and/or oxygen and other substances, if required.

SAFETY DESIGN BASIS ELEVEN - The HCS is designed with provisions for periodic inspection and testing of all safety-related components (GDC-42 and 43).

6.2.5.1.2 Power Generation Design Bases

POWER GENERATION DESIGN BASIS - The hydrogen mixing subsystem provides continual mixing of the containment air during normal plant operation. The containment penetrations in the hydrogen monitoring subsystem are closed during normal plant operation. The remainder of the HCS performs no function during normal plant operations.

6.2.5.2 System Design

6.2.5.2.1 General Description

The total system for control of combustible hydrogen concentrations in the containment following a LOCA, shown schematically in Figures 6.2.5-1 and 9.4-6, consists of a hydrogen monitoring subsystem that provides containment atmosphere samples, hydrogen mixing provisions which assure a nearly uniform hydrogen concentration in the containment atmosphere, electric (thermal) hydrogen recombiners which provide the primary means of reducing containment hydrogen concentrations, and a hydrogen purge subsystem which is used as a backup system to the recombiners.

9. Bordelon, F. M., Massie, H. W., Jr., Zordon, T. A., "Westinghouse Emergency Core Cooling System Evaluation Model Summary", WCAP-8339, June 1974.

10. Topical Report AAF-TR-7101, "Design and Testing of Fan Cooler-Filter Systems for Nuclear Applications"; February 20, 1972; American Air Filter Co., Inc.; Louisville, KY.

11. Topical Report OCF-1, "Nuclear Containment Insulation System," August 1977, Owens-Corning Fiberglas Corporation, Lenexa, KS.

12. WAPD-PT 24, "Fission Product Decay Energy" (December 1961).

13. ~~TID 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," J. J. DiNunno, F. D. Anderson, R. E. Baker, R. L. Waterfield, March 23, 1962, Division of Licensing and Regulation, USAEC, Washington, D. C.~~

Replace with "Deleted"

14. Wilson, J. F., "Electrical Hydrogen Recombiner for Water Reactor Containments," WCAP-7709-L (Proprietary), July 1971, and WCAP-7820 (Non-Proprietary) December 1971.

15. Wilson, J. F., "Electric Hydrogen Recombiner for PWR Containments - Final Development Report," WCAP-7709-L, Supplement 1 (Proprietary), April 1972, and WCAP-7820, Supplement 1 (Non-Proprietary), May 1972.

16. Wilson, J. F., "Electric Hydrogen Recombiner for PWR Containments - Equipment Qualification Report," WCAP-7709-L, Supplement 2 (Proprietary), September 1973, and WCAP-7820, Supplement 2 (Non-Proprietary), October 1973.

17. Wilson, J. F., "Electric Hydrogen Recombiner for PWR Containments - Long-Term Tests," WCAP-7709-L, Supplement 3 (Proprietary), January 1974, and WCAP-7820, Supplement 3 (Non-Proprietary), February 1974.

18. Wilson, J. F., "Electric Hydrogen Recombiner for PWR Containments," WCAP-7709-L, Supplement 4 (Proprietary), April 1974, and WCAP-7820, Supplement 4 (Non-Proprietary), May 1974.

19. Wilson, J. F., "Electric Hydrogen Recombiner Special Tests," WCAP-7709-L, Supplement 5 (Proprietary) and WCAP-7820, Supplement 5 (Non-Proprietary), December 1975.

20. Wilson, J. F., "Electric Hydrogen Recombiner IEEE 323-1974 Qualification," WCAP-7709-L, Supplement 6 (Proprietary) and WCAP-7820, Supplement 6 (Non-Proprietary), October 1976.

Steam System Pipe Failure

The steam release arising from a rupture of a main steam pipe would result in energy removal from the RCS, causing a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity. There is an increased possibility that the core will become critical and return to power.

The core is ultimately shut down by the boric acid injection delivered by the safety injection system. Capability for injection of the boric acid solution is maintained, assuming any single failure in the safety injection system.

For cases where offsite power is assumed to be available, the sequencing of events in the safety injection system is the following. After the generation of the SIS (appropriate delays for instrumentation, logic, and signal transport included), the appropriate valves begin to operate and the centrifugal charging pumps start. In 12 seconds, the valves are assumed to be in their final position, and the pumps are assumed to be at full speed. This delay, described above, is included in the calculations.

In cases where offsite power is not available, an additional 12-second delay is assumed to start the diesels and to load the necessary safety injection equipment onto them.

The analysis has shown that even assuming a stuck RCCA with or without offsite power, and assuming a single failure in the engineered safeguards, the core remains in place and intact. Radiation doses will not exceed 10 CFR ~~100~~ 50.67 guidelines.

DECREASE IN HEAT REMOVED BY THE SECONDARY SYSTEM

Feedwater System Pipe Break

A major feedwater line rupture is defined as a break in a feedwater line large enough to prevent the addition of sufficient feedwater to the steam generators to maintain shell side fluid inventory in the steam generators. If the break is postulated in a feedwater line between the check valve and the steam generator, fluid from the steam generator may also be discharged through the break. Further, a break in this location could preclude the subsequent addition of auxiliary feedwater to the affected steam generator. (A break upstream of the feedwater line check valve would affect the NSSS only as a loss of feedwater. This case is covered by the evaluation in Sections 15.2.6 and 15.2.7).

6.5.1.3 Safety Evaluation

Safety evaluations are numbered to correspond to the safety design bases given in Section 6.5.1.1.1.

SAFETY EVALUATION ONE - Table 6.5-1 lists the ESF filtration systems' design parameters used to determine the radiological consequences for the postulated accidents analyzed in Chapter 15.0. The results of these analyses demonstrate that the emergency exhaust system reduces and controls fission products released from the fuel building following a fuel handling accident or released from the auxiliary building following a LOCA, such that the offsite radiation exposures are within the guidelines of 10 CFR ~~100~~. The safety evaluations which demonstrate the design and construction 50.67 e ESF filtration systems are provided in Sections 9.4.2 and 9.4.3.

50.67

SAFETY EVALUATION TWO - The results of the analyses described in Chapter 15.0 demonstrate that the control building HVAC systems reduce and control fission product release to the control room following a LOCA, such that the offsite radiation exposures are within the guidelines of 10 CFR ~~100~~. The safety evaluations which demonstrate the design and construction of these control building HVAC systems are provided in Sections 9.4.1 and 6.4.

6.5.1.4 Tests and Inspections

Tests and inspections for ESF filter systems are described in Section 9.4.

6.5.1.5 Instrumentation Requirements

Instrumentation and controls are provided to facilitate automatic operation and remote control of the system and to provide continuous indication of system parameters. Further descriptions are provided in Section 9.4.

6.5.1.6 Materials

The materials used for ESF filtration systems were chosen considering the environmental conditions and are commensurate with acceptable construction practices. Further information is provided in Section 9.4.

6.5.2 CONTAINMENT SPRAY SYSTEM

The containment spray system (CSS) is an ESF, the functions of which are to reduce pressure and temperature in the containment atmosphere following a postulated LOCA and to remove radioactive

fission products from the containment atmosphere. These functions are performed by spraying water into the containment atmosphere through a large number of nozzles on spray headers located in the containment dome. Reduction of pressure and temperature in the containment with the CSS is discussed in Section 6.2.2.

Radioiodine in its various forms is the fission product of primary concern in the evaluation of a LOCA. It is absorbed by the containment spray from the containment atmosphere. To enhance this iodine absorption capacity of the spray, the spray solution is adjusted to an alkaline pH which promotes iodine hydrolysis, in which iodine is converted to nonvolatile forms.

The physical characteristics of the CSS are discussed in Section 6.2.2. Discussed herein are the spray additive portion of the system and the containment spray system's fission product removal capability following a LOCA.

6.5.2.1 Design Bases

6.5.2.1.1 Safety Design Bases

SAFETY DESIGN BASIS ONE - The CSS is designed to provide a spray solution while the spray additive portion of the system is in operation in the pH range of 9.0 to 11.0 and a final containment recirculation sump solution with a pH of at least 8.5.

SAFETY DESIGN BASIS TWO - The CSS is capable of reducing the iodine and particulate fission product inventories in the containment atmosphere such that the offsite radiation exposures resulting from a design basis LOCA are within the plant siting dose guidelines of 10 CFR ~~100~~ 50.67

Additional safety design bases are included in Section 6.2.2, in which the capability of the spray system to remove heat from the containment atmosphere is discussed.

6.5.2.1.2 Power Generation Design Basis

The CSS has no power generation design basis.

6.5.2.2 System Design

6.5.2.2.1 General Description

The containment spray additive portion of the CSS provides for eduction of 30 weight percent (nominal) sodium hydroxide into the spray injection water. This yields a spray mixture with a pH of from 9.0 to 11.0 during the initial period of operation, when radioiodine is being removed from the containment atmosphere.

The design minimum spray water pH of 9.0 coupled with the dependent parameters identified in Safety Evaluation Two below, assure the minimum elemental iodine removal coefficient of 10.0 per hour during the spray injection phase. The design minimum sump pH of 8.5 assures long-term iodine retention in the recirculated spray liquid.

The maximum pH of the spray solution in the CSS during the spray injection phase is 11.0, based on the maximum allowable eductor sodium hydroxide flow rate and minimum boric acid concentration in the RWST.

The system is designed to provide a spray solution in the CSS during the spray recirculation phase with a maximum spray pH of less than 11.0 based on a sump pH of at least 8.5 (due to prior addition of NaOH), design spray recirculation flow rate, as noted in Table 6.2.2-2, and maximum spray additive flow rate greater than 46 gpm. To preclude closure of the valve between the spray additive tank and the spray additive eductors before sufficient NaOH has been added to meet the sump pH criterion, an interlock is provided on the motor-operated valves from the spray additive tank to prohibit closure of the valves before the prescribed amount of NaOH has been added to the sump. The total volume of sodium hydroxide added to the containment following a LOCA results in a minimum pH of 8.5 in the sump, and the rate of addition maintains the spray solution pH in the CSS between 9.0 and 11.0 for all single failures within the system. Single failure analysis for the spray additive subsystem is given in Table 6.5-4. The sump pH, as a function of time, is provided in Figure 6.5-5.

SAFETY EVALUATION TWO - The spray ~~iodine~~ removal analysis is based on the assumptions that:

a. Only one out of two spray pumps is operating

b. The ECCS is operating at its maximum capacity

which bounds both injection and recirculation spray design flow rates

The spray system is assumed to spray approximately 85 percent of the total containment net free volume. This volume consists of those areas directly sprayed plus those volumes which have good communication with the directly sprayed volumes. The remaining 15 percent of the containment free volume has restricted communication with the sprayed volumes and is assumed to be unsprayed. A description of the unsprayed volumes is presented in Table 6.5-2.

The containment spray additive subsystem is used to maintain the spray solution at a minimum pH of 9.0 during NaOH injection to ensure efficient and rapid removal of the iodine from the containment atmosphere.

The performance of the spray system was evaluated at the containment post-LOCA calculated saturation temperature corresponding to the calculated peak pressures provided in Table 6.2.1-8 and the containment design pressure provided in Table 6.2.1-2. The spray ~~design~~ flow rate of ~~2,995~~ 2900 gpm per train was used in the calculations provided in Appendix 6.5A.

Based on Regulatory Guide ~~1.4~~ 1.183, three species of airborne iodine are postulated to exist in the containment atmosphere following a LOCA. These are elemental, particulate, and organic species.

It has been assumed in these evaluations of spray removal effectiveness that organic iodine forms are not removed by the sodium hydroxide spray. A limited credit for the removal of airborne particulates ~~containing iodine~~ has been taken, assuming that the spray removal rate is ~~0.45~~ 5.0 hr^{-1} until a DF of ~~100~~ 50 is attained. Credit for removal of elemental iodine is based on a spray removal rate of 10 hr^{-1} until a ~~df of 100 is~~ DF of 200 attained. These assumptions underestimate the actual amounts of iodine removed and thus result in calculated accident doses higher than could realistically be expected.

Utilizing the dose analysis input parameters indicated above and in Table 6.5-2, the dose analysis of Chapter 15.0 demonstrates that offsite radiation exposures resulting from a design basis LOCA are within the plant siting dose guidelines of 10 CFR ~~100~~ 50.67.

Appendix 6.5A provides the model used to calculate the ~~iodine~~ removal coefficients provided in Table 6.5-2.

6.5.2.4 Tests and Inspections

All active components in the spray additive subsystem are tested both by performance tests in the manufacturer's shop and by inplace testing after installation.

and the spray removal rate is 0.5 hr^{-1} after a DF of 50 is attained.

The corrosion of materials in the NSSS and the containment building, resulting from the spray solution used for iodine absorption, has been tested by the Reactor Division at ORNL (Ref. 2). The spray solutions provided in Table 6.5-5 result in negligible corrosion, based on these studies.

Sodium hydroxide does not undergo radiolytic decomposition in the post-LOCA environment. Sodium has a low neutron absorption cross section and will not undergo significant activation.

With respect to the potential for pyrolytic decomposition, NaOH is stable to at least its melting point temperature of 604°F. It may convert to sodium oxide (Na_2O) upon removal of the water.

6.5.3 FISSION PRODUCT CONTROL SYSTEMS

6.5.3.1 Primary Containment

The containment consists of a prestressed post-tensioned, reinforced concrete structure with cylindrical walls, hemispherical dome, and base slab lined with welded quarter-inch carbon steel liner plate, which forms a continuous, leaktight membrane. Details of the containment structural design are discussed in Section 3.8. Layout drawings of the containment structure and the related items are given in the general arrangement drawings of Section 1.2.

The containment walls, liner plate, penetrations, and isolation valves function to limit the release of radioactive materials, subsequent to postulated accidents, such that the resulting offsite doses are less than the guideline values of 10 CFR ~~100~~ 50.67. Containment parameters affecting fission product release accident analyses are given in Appendix 15A.

Long-term containment pressure response to the design basis accident is shown in Figure 6.2.1-1. Relative to this time period, the CSS is operated to reduce iodine concentrations and containment atmospheric temperature and pressure commencing with system initiation, at approximately 60 seconds, as shown in Table 6.2.2-3 and ending when containment pressure has returned to normal. For the purpose of post-LOCA dose calculations discussed in Chapter 15.0, two dose models have been assumed, the 0-2 hour case and the 0-30 day case, as shown in Appendix 15A.

The containment minipurge system may be operated for personnel access to the containment when the reactor is at power, as discussed in Section 9.4.6.

TABLE 6.5-2 (Sheet 1)

INPUT PARAMETERS AND RESULTS OF
SPRAY ~~IODINE~~ REMOVAL ANALYSIS

Ultimate core power rating	3,565 MWt
Total containment free volume, maximum	2.7 ~~2.50~~ x 10^6 ft^3 **
Unsprayed containment free volume	<15.0 percent
~~Area coverage at the operating deck~~	
~~design~~	~~>90 percent~~
~~Calculated~~	~~>93 percent~~
~~Mixing rate between sprayed and unsprayed volumes~~	~~85,000 cfm ***~~
~~Dose model~~	~~One region~~
Minimum vertical distance to operating deck from lowest spray header	118 feet ~~— 2 in.~~
Net spray flow rate per train~~, injection phase~~	~~3,131 gpm~~ 2900 gpm *****
~~Design NaOH flow rate per eductor~~	~~44.0 ±2 gpm~~
Number of spray pumps operating	1
~~Spray solution pH~~	~~9.0 to 11.0~~
Elemental iodine absorption coefficient, s, used in accident dose calculations	10 hr^{-1} *
~~Expected s~~	~~25.7 hr-1 **~~
Particulate ~~iodine~~ absorption coefficient, p, used in accident ~~c~~alculations dose	~~0.45 hr^{-1} *~~ 5.0 hr^{-1} initially, changing abruptly to 0.5 hr^{-1} ***
~~Calculated p~~	~~0.73 hr^{-1} **~~
Spray drop ~~size, design~~ diameter	~~See Figure 6.5-2~~ 0.116 cm ****
Spray drop fall time	8.764 sec

TABLE 6.5-2 (Sheet 2)

~~Schmidt number~~	~~11.58~~
~~Gas diffusivity~~	~~0.064~~
~~Partition coefficient~~	~~5,000~~

* Used DFs of up to ~~100~~ 200.

~~** As calculated from Appendix 6.5A.~~

~~*** Adequate mixing of the containment atmosphere following a LOCA is ensured by effects of the initial blowdown, containment sprays, natural convection and forced air ventilation provided by the containment coolers without reliance on the hydrogen mixing fans. Refer to Section 6.2.5 for additional information.~~

** The maximum volume is conservative for the calculation of containment spray removal coefficients in Appendix 6.5A since the equations used include the containment volume as a divisor.

*** Changed when DF of 50 is reached.

****Note that this is larger (more conservative) than the results obtained by tests, as shown in Sheet 3.

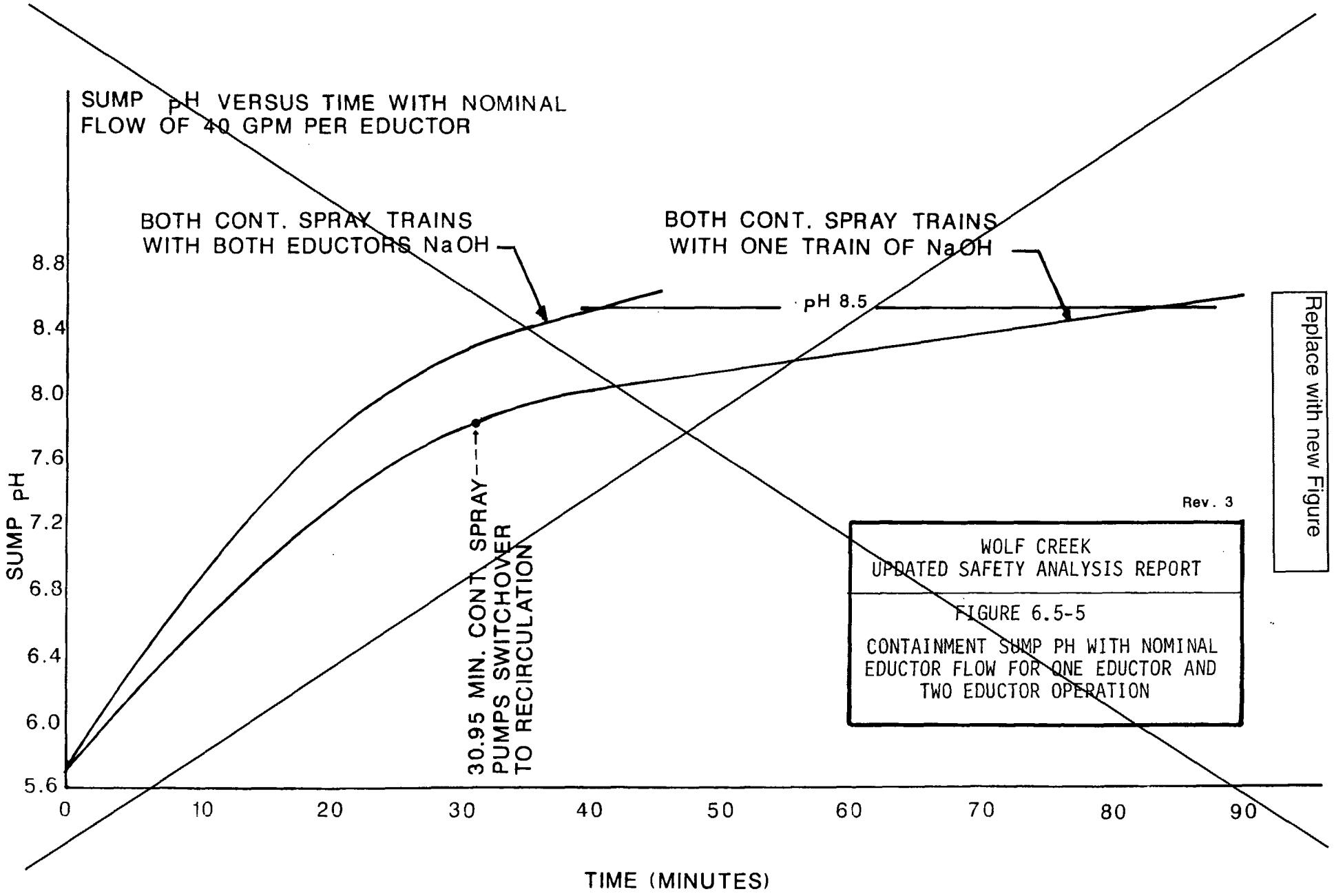
***** Bounds both injection and recirculation.

TABLE 6.5-2 (Sheet 4)

UNSPRAYED CONTAINMENT FREE VOLUME

Unsprayed Region	Volume (ft^3)
Pressurizer enclosure and overhang	26,511
Region below the four RC pump hatches	44,245
Pressurizer safety valve enclosure	14,392
Region below the four containment coolers and two filter adsorber units	63,852
Pressurizer spray valve enclosure	8,920
Region under CRDM PLENUM/SEISMIC SUPPORT PLATE	3,189
Elevator machine room and elevator shaft	16,596
Region under concrete flooring used for structural strength and shielding	182,821
Total unsprayed free volume	360,526
Percentage of free volume unsprayed	~14.4% *

* Based on a free volume of 2.5×10^6 ft^3.



Rev. 3

WOLF CREEK
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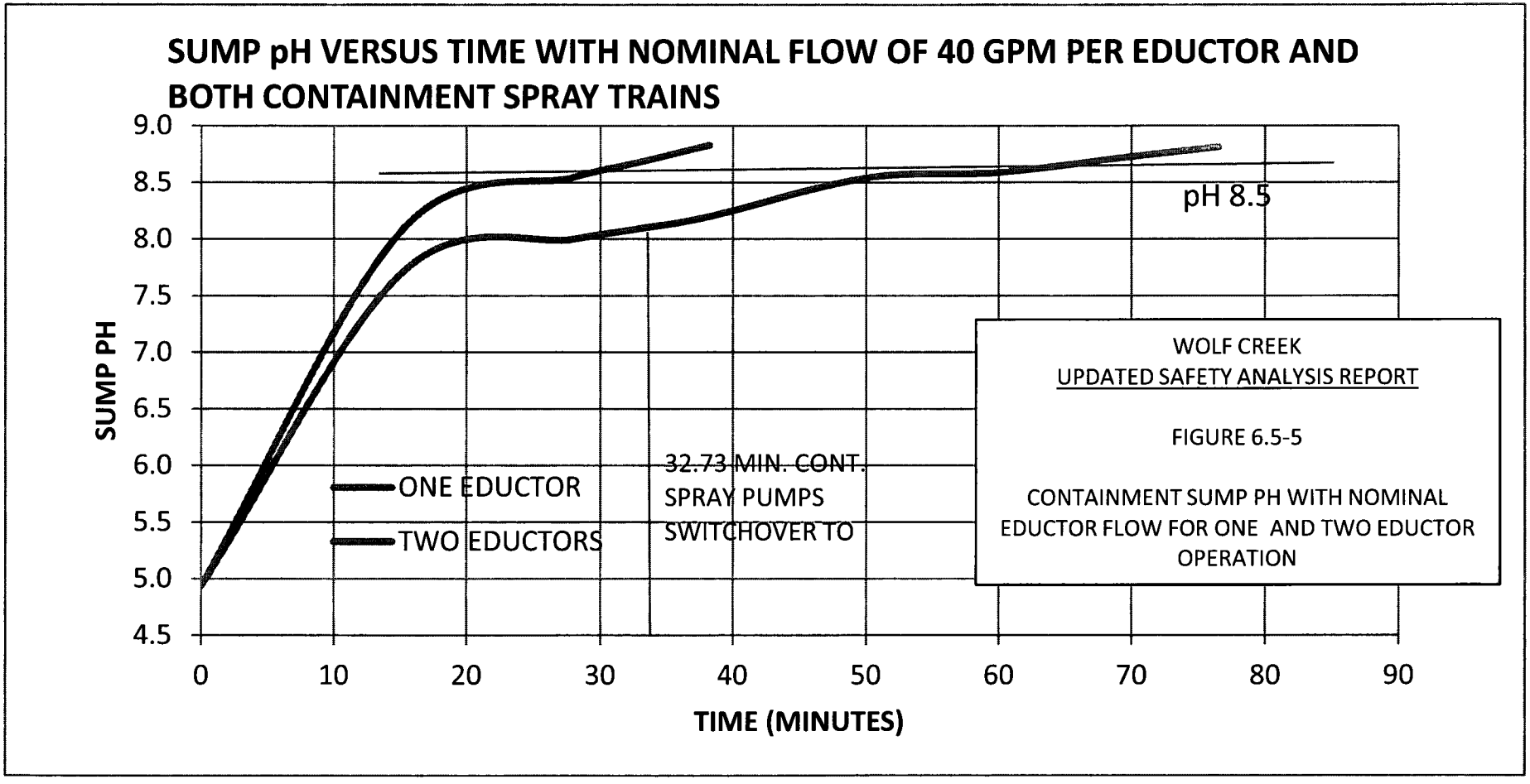
FIGURE 6.5-5

CONTAINMENT SUMP PH WITH NOMINAL EDUCTOR FLOW FOR ONE EDUCTOR AND TWO EDUCTOR OPERATION

Replace with new Figure


SUMP pH VERSUS TIME WITH NOMINAL FLOW OF 40 GPM PER EDUCTOR AND BOTH CONTAINMENT SPRAY TRAINS
SUMP PH
9.0
8.5
8.0
7.5
7.0
6.5
6.0
5.5
5.0
4.5
0
10
20
30
40
50
60
70
80
90
TIME (MINUTES)
ONE EDUCTOR
TWO EDUCTORS
32.73 MIN. CONT. SPRAY PUMPS SWITCHOVER TO
pH 8.5
WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT
FIGURE 6.5-5
CONTAINMENT SUMP PH WITH NOMINAL EDUCTOR FLOW FOR ONE AND TWO EDUCTOR OPERATION

APPENDIX 6.5A

~~IODINE~~ REMOVAL MODELS FOR THE CONTAINMENT SPRAY SYSTEM

6.5A.1 PARTICULATE ~~IODINE~~ MODEL

The spray washout model for aerosol particles is represented in equation form as follows:

$$l_P = \frac{3hEF}{2dV} \qquad (6.5A\text{-}1)$$

by SRP 6.5.2 (Ref. 1)

Where:

l_P = spray removal constant for particles

h = fall height

/d

E = ~~total~~ collection efficiency for ~~a single~~ drops of diameter d

F = spray flow rate

~~d = mean drop diameter~~

V = ~~volume of gas space~~ sprayed volume

/d

10 m^{-1} until a DF of 50 is reached and is 1 m^{-1} after a DF of 50 is reached

~~The capture of particles by falling drops results from Brownian diffusion, diffusiophoresis, interception, and impaction. Early in the injection phase, particles are removed mainly by impaction. Following injection, when the larger particles have already been removed, the removal rate is controlled by diffusiophoresis, which is the collection of particulates by steam condensing on the spray drops.~~ The ~~single~~ drop collection efficiency, E, is taken as ~~0.0015, the minimum value observed in experimental tests~~ (Ref. 1). ~~The minimum collection efficiency, 0.0015, was only attained after the major fraction of airborne particles was removed. For early time periods, the removal rates were much higher than the minimum values ultimately reached.~~

The spray removal constant (l_P) for particulate ~~iodine~~ has been calculated to be ~~0.73 hr^{-1}~~, based on equation 6.5A-1.

~~A limited and conservative credit for spray removal of airborne particulates containing iodine has been taken, assuming the spray removal constant is 0.45 hr^{-1} for the 0 to 2-hour period following the postulated LOCA (see Table 6.5-2).~~

~~Particle spray removal constants considerably larger and of longer duration than those conservatively chosen above have been reported from the Battelle Northwest Containment Systems Experiment (Ref. 2) and by the Oak Ridge National Laboratories Nuclear Safety Pilot Plant (Ref. 4).~~

5.0 hr^{-1} until a DF of 50 is reached and 0.5 hr^{-1} after a DF of 50 is reached

6.5A.2 ELEMENTAL IODINE MODEL

The spray system, by virtue of the large surface area provided between the droplets and the containment atmosphere, will afford an excellent means of absorbing elemental radioactive iodine released as a consequence of a LOCA. Sodium hydroxide is added to the spray fluid to increase the solubility of iodine in the spray to the point where the rate of absorption is largely dependent on the concentration of radioiodine in the air surrounding the drops.

Insert N

~~The basic model of the containment atmosphere and spray system is given by Parsley (Ref. 4). The containment atmosphere is viewed as a "black box" having a sprayed volume, V, and containing iodine at some uniform concentration Cg. Liquid enters at a flow of F volumes per unit time, containing iodine at a concentration of CL1, and leaves at the same flow, at concentration CL2. A material balance for the containment vessel as a function of time is given by:~~

~~$-VdCg = F(CL2 - CL1)dt$ (6.5A-2)~~

~~Where:~~

~~CL1 = the iodine concentration in the liquid entering the dispersed phase, gm/cm^3~~

~~CL2 = the iodine concentration in the liquid leaving the dispersed phase, gm/cm^3~~

~~V = sprayed volume of containment, cm^3~~

~~Cg = the iodine concentration in the containment atmosphere, gm/cm^3~~

~~F = the spray flow rate, cm^3/sec~~

~~t = spray time, sec~~

~~A drop absorption efficiency, E, which may be described as the fraction of saturation, is defined as:~~

~~$E = (CL2 - CL1)/(CL^* - CL1)$ (6.5A-3)~~

~~In addition, the equilibrium distribution of iodine between the vapor and liquid phases is given by:~~

~~$H = CL^*/Cg$ (6.5A-4)~~

Insert N

The removal model for elemental iodine is represented in equation form by SRP 6.5.2 (Ref. 1) as follows:

$$l_s = \frac{6(K_g)(T)(F)}{(V)(D)}$$

Where:

l_s = spray removal constant for elemental iodine

K_g = gas-phase mass-transfer coefficient

T = time of fall of the spray drops

F = spray flow rate

V = sprayed volume

D = diameter of the spray drops

The minimum observed gas-phase mass-transfer coefficient, K_g, is 3 m/min (Ref. 2).

~~Where:~~

~~H = The iodine partition coefficient (gm/liter of liquids)/(gm/liter of gas)~~

~~CL* = The equilibrium concentration in the liquid, gm/cm3~~

~~Substitution of equation 6.5A-4 into equation 6.5A-3 yields~~

~~E = (CL2 - CL1)/(HCg - CL1)~~ ~~(6.5A-5)~~

~~Solving equation 6.5A-5 for (CL2 - CL1) and inserting the result into equation 6.5A-2 gives~~

~~-(V)dCg = EF(HCg - CL1)dt~~ ~~(6.5A-6)~~

~~During the injection phase, CL1 = 0, so that~~

~~-(V)dCg = (EFHCg)dt~~ ~~(6.5A-7)~~

~~Equation 6.5A-7 can be integrated to solve for Cg. The concentration of iodine in the containment atmosphere during injection as a function of time is given by:~~

~~Cg = Cg_o exp [-EHFt/V]~~ ~~(6.5A-8)~~

~~Where:~~

~~Cg_o = The initial iodine concentration in the containment atmosphere, gm/cm^3~~

~~Equation 6.5A-8 is applicable up to the time the spray solution is recirculated and is based on the following assumptions:~~

~~a. Cg is uniform throughout the containment~~

~~b. There are no iodine sources after the initial release~~

~~c. The concentration of iodine in the spray solution entering the containment is zero~~

~~From equation 6.5A-8, the spray removal constant, l, is given by~~

$$l = \frac{EHF}{V}$$ ~~(6.5A-9)~~

~~The above equation for is independent of the models on which the numerical evaluation of the drop absorption efficiency, E, and the iodine partition coefficient, H, may be based.~~

~~Absorption efficiency for elemental iodine may be calculated from the time-dependent diffusion equation for a rigid sphere with the gas film mass transfer resistance as a boundary condition. This mass transfer model was suggested by L. F. Parsley (Ref. 4), who gives the solution to the diffusion equation, with the above given boundary condition, as:~~

$$\cancel{E = 1 - \sum_{n=1}^{\infty} \frac{6\,Sh^2 \exp\left(-a_n^2 Q_f\right)}{\left[a_n^2 + (Sh)\,(Sh-1)\right] a_n^2}} \qquad \cancel{(6.5A\text{-}10)}$$

~~Where:~~

~~Sh = the dimensionless group = kg a/HDL~~

~~a = the drop radius, cm~~

~~k_g = the gas film mass transfer coefficient, cm/sec~~

~~DL = the liquid diffusivity, cm2/sec~~

~~Qf = the dimensionless drop residence time~~

~~a_n = the eigenvalues of the solution~~

~~It should be noted that this solution, which applies to the rigid drop model, is based on the assumption that molecular diffusion is the only mechanism by which iodine is transported from the surface to the interior of the drop. Since a high degree of mixing is expected in the drops, particularly in the presence of sizable temperature and concentration gradients, it is apparent that this stagnant drop model presents a conservative approach to the calculation of iodine absorption by the drops.~~

~~The gas film mass transfer coefficient required for the above calculation is computed by the equation of Ranz and Marshall (Ref. 5).~~

$$\cancel{k_g = \frac{D_g}{d}\left(2 \times 0.6\ Re^{0.5}\ Sc^{0.33}\right)} \qquad \cancel{(6.5A\text{-}11)}$$

~~Where:~~

~~d = drop diameter, cm~~

~~D_g = diffusion coefficient in vapor, cm2/sec~~

~~Re = Reynold's number~~

~~Sc = Schmidt number~~

~~A more conservative numerical value of E is obtained from equation 6.5A-12 given below, which is quoted by Postma and Pasedag (Ref. 6):~~

$$E = 1 = \exp\left[\left[-\frac{6\ k_g t_e}{d\left(\left(\frac{H + k_g}{k_L}\right)\right)}\right]\right] \qquad (6.5A\text{-}12)$$

~~Where:~~

~~E = drop absorption efficiency~~

~~k_L = liquid phase mass transfer coefficient, cm/sec~~

~~t_e = drop exposure time, sec~~

~~d = drop diameter, cm~~

~~H = equilibrium partition coefficient~~

~~Equation 6.5A-12 is based on a model in which it is assumed that the drop consists of an outer stagnant film and a well-mixed interior. Though this model is basically nonconservative compared with the stagnant drop model represented by equation 6.5A-10, conservatism is introduced into equation 6.5A-12 when the following expression is used for kL:~~

$$kL = \frac{2p^2\ D_L}{3d} \qquad (6.5A\text{-}13)$$

~~Where:~~

~~D^L = liquid diffusivity of iodine in water, cm^2/sec~~

~~d = drop diameter, cm~~

~~Equation 6.5A-13 results from a truncated approximation (Ref. 6) to the rigid drop diffusion equation due to Griffith (Ref. 7). Griffith's approximation is conservative in that it predicts lower absorption than would be predicted without such approximation for stagnant drop absorption.~~

~~The numerical value of E obtained from equation 6.5A-12 is more conservative than the one obtained from equation 6.5A-10, as shown by Postma and Pasedag (Ref. 6) by comparing them with the numerical value of E based upon another model. The reference model chosen by Postma and Pasedag (Ref. 6) for comparison is the completely well mixed model in which the solution in the entire drop, including the interior as well as the gas-liquid interface, is in equilibrium with the iodine concentration in the gas phase outside the drop. The expression in this reference model is:~~

$$\cancel{E = 1 - \exp\left(-\frac{6\, k_g t_e}{dh}\right)} \qquad \cancel{(6.5A\text{-}14)}$$

~~The absorption efficiency is a function of the drop size, the gas phase mass transfer coefficient, diffusion in the liquid phase, the partition coefficient, and the drop fall time.~~

~~Eggleton's equation (Ref. 8) for the equilibrium elemental iodine decontamination factors, DF, is given by:~~

$$\cancel{DF = 1 + H(VL)/(VG)} \qquad \cancel{(6\text{-}5A\text{-}15)}$$

~~Where:~~

~~VG = Gaseous volume of the containment~~

~~VL = Liquid volume of the containment, which may be used for calculation of the partition coefficient, H, for a given value of the DF. However, equation 6.5A-15 was not used in the present analysis; instead, a numerical value of 5,000 for H, the minimum found from CSE tests (Refs. 9 and 10) for sodium hydroxide spray, was used in the evaluation of λ.~~

~~Since the spray does not consist of a uniform droplet size, a spectrum of drop sizes and their corresponding volume percentage (for the specific nozzle design) were used to determine the individual spray removal constant for each droplet size. The total spray removal constant is equal to the sum of the individual spray removal constants, i.e.:~~

$$\cancel{l = \sum_{i=1}^{n} li = \sum_{i=1}^{n} \sum_{J=1}^{m} iJ} \qquad \cancel{(6.5A\text{-}16)}$$

~~Since the fall time, t_e, is dependent on distance from the spray header to the operating deck, and each spray header consists of ring headers ( ) located at various levels, i was calculated for each spray ring header ( ), utilizing the appropriate drop distance for each header.~~

~~Therefore,~~

$$\cancel{l_{i_J} = \frac{E_{iJ}\ H\ F_{iJ}}{V}} \qquad \cancel{(6.5A\text{-}17)}$$

~~Where:~~

$\cancel{E_{i_J}}$ = ~~collection efficiency for a single drop of micron increment i for ring header~~

$\cancel{F_{i_J}}$ = ~~spray flow rate for micron increment i for header J~~

~~and,~~

$$\cancel{F_{i_J} = (F_i/\text{nozzle}) \cdot (N_J)} \qquad \cancel{(6\ 5A\text{-}18)}$$

~~Where:~~

$$\cancel{F_i/\text{nozzle} = \frac{15.2\ (N_i) \div (V_i)}{\sum_{i=1}^{n} N_i\ V_i}}$$

$\cancel{N_J}$ = ~~number of nozzles on ring header~~

$\cancel{N_i}$ = ~~number frequency for micron increment i (Figure 6.5-2)~~

$\cancel{V_i}$ = ~~volume of a drop in micron increment i~~

~~As the spray solution enters the high-temperature containment atmosphere, steam will condense on the spray drops. The amount of condensation is easily calculated by a mass balance of the drop:~~

$$\cancel{mh + m_c h_g = m'h_f}$$

~~where:~~

~~m and m' = the mass of the drop before and after condensation, lbs~~

~~m_c = the mass of condensate, lbs~~

~~h = the initial enthalpy of the drop, Btu/lb~~

~~h_g and h_f = The saturation enthalpy of water vapor and liquid, Btu/lb~~

~~The increase in each drop diameter in the distribution, therefore, is given by:~~

$$\left(\frac{d'}{d}\right)^3 = \left(\frac{v}{v_f}\right) - \left(\frac{h_g - h}{h_{fg}}\right)$$

~~Where:~~

~~v_f = the specific volume of liquid at saturation, ft^3/lb~~

~~v = the specific volume of the drop before condensation, ft^3/lb~~

~~h_{fg} = the latent heat of evaporation, Btu/lb~~

~~h_g = the enthalpy of steam at saturation, Btu/lb~~

~~d and d' = the drop diameter before and after condensation, cm~~

~~Postma and Pasedag (Ref. 6) conclude that condensation will tend to increase the iodine washout rate due to the increased volume of the spray. Their effect has been conservatively ignored.~~

~~The drop exposure time calculated is based on the assumption that the drops were sprayed in such a manner that the initial downward velocity of the drops at the spray ring header elevation was zero. The drops fall under the effect of gravity from the spray ring header to the operating deck. The minimum height is given in Table 6.5-2. As the drop size increases, the average residence time decreases from about 20 to 5 seconds. Incorporating the above parameters into equation 6.5A-16 with the sprayed containment volume, V, and assuming a single spray header flow rate, the value of the spray removal coefficient calculated is presented in Table 6.5-2.~~

The resulting elemental iodine spray removal constant is greater than 10 hr^{-1}. A ~~Only this~~ conservative removal constant of 10 hr^{-1} is assumed and used in the design basis LOCA evaluations presented in Section 15.4.

6.5A.3 REFERENCES

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~~9. Postma, A. K., Coleman, L. F., and Hilliard, R. K., "Iodine Removal from Containment Atmospheres by Boric Acid Spray," BNP-100, Battelle-Northwest, Richland, Washington, 1970.~~

~~10. Coleman, L. F., "Iodine Gas-Liquid Partition," Nuclear Safety Quarterly Report, February, March, April 1970, BNWL-1315-2, Battelle-Northwest, Richland, Washington, p. 2.12-2.19, 1970.~~

1. NUREG-0800, Standard Review Plan 6.5.2, Revision 4, "Containment Spray as a Fission Product Cleanup System," March 2007.

2. BNL-Technical Report A-3788, "Fission Product Removal Effectinveness of Chemical Additives in PWR Containment Sprays," August 1986.

CHAPTER 7.0

INSTRUMENTATION AND CONTROLS

7.1 INTRODUCTION

This section describes the various plant instrumentation and control systems and the functional performance requirements, design bases, system descriptions, design evaluations, and tests and inspections for each. The information provided in this chapter emphasizes those instruments and associated equipment which constitute the protection system, as defined in IEEE Standard 279-1971, "IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating Stations."

The instrumentation and control systems provide automatic protection and exercise proper control against unsafe and improper reactor operation during steady state and transient power operations (Conditions I, II and III) and to provide initiating signals to mitigate the consequences of emergency and faulted conditions (Condition III and IV). ANS conditions are discussed in Chapter 15.0.

Applicable criteria and codes are listed in Table 7.1-2.

7.1.1 IDENTIFICATION OF SAFETY-RELATED SYSTEMS

Safety-related instrumentation and control systems and their supporting systems are those systems required to ensure:

a. The integrity of the reactor coolant pressure boundary.

b. The capability to shut down the reactor and maintain it in a safe shutdown condition following any design basis accident.

c. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR ~~100~~. 50.67

The definitions provided below are used to classify the instrumentation systems into the categories defined for Chapter 7.0 by Regulatory Guide 1.70.

A listing of these systems, by categories, that are comparable to those of nuclear power plants of similar design is given in Table 7.1-1. Table 7.1-1 also identifies the systems that are different with references to discussions of those differences.

WOLF CREEK

TABLE 7A-3, DATA SHEET 13.1

I. REGULATORY GUIDE 1.97 TABLE 2 RECOMMENDATIONS

VARIABLE IDENT. NO.	VARIABLE	RANGE	CATEGORY	PURPOSE
E.6.1	Primary Coolant	Grab Sample	3[5,18]	Release assessment, verification analysis
E.6.1.1	Gross Activity	10 µCi/ml to 10 Ci/ml		
E.6.1.2	Gamma Spectrum	(Isotopic Analysis)		
E.6.1.3	Boron Content	0 to 6,000 ppm		
E.6.1.4	Chloride Content[2]	0 to 20 ppm		
E.6.1.5	Dissolved Hydrogen or Total Gas [19]			
E.6.1.6	Dissolved Oxygen[19]	0 to 20 ppm		
E.6.1.7	pH	1 to 13		
B.1.3	RCS Soluble Boron Concentration	0 - 6,000 ppm	3	Verification
C.1.3	Analysis of Primary Coolant (Gamma Spectrum)	10 µCi/gm to 10 Ci/gm or ~~TID-14844~~ source term in coolant volume	3[5]	Detail analysis, accomplishment of mitigation, verification, long-term surveillance
E.6.3	Containment Air	Grab Sample		Release assessment, verification analysis
E.6.3.2	Oxygen Content	0 to 30 percent		Release assessment, verification analysis
E.6.3.3	Gamma Spectrum	(Isotopic Analysis)		Release assessment, verification analysis

Regulatory Guide 1.183

SAFETY DESIGN BASIS ONE - The FHS is protected from the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, and external missiles (GDC-2).

SAFETY DESIGN BASIS TWO - The FHS is designed to remain intact after an SSE or following the postulated hazards of fire, internal missiles, or pipe breaks (GDC-3 and 4).

SAFETY DESIGN BASIS THREE - The FHS components are capable of being tested during plant operation. Provisions are made to allow for inservice inspection and testing of components at appropriate times.

SAFETY DESIGN BASIS FOUR - The FHS is designed and fabricated to codes consistent with the seismic category assigned by Regulatory Guide 1.29 and industry standard specifications.

SAFETY DESIGN BASIS FIVE - The containment isolation provisions for the system are selected, tested, and located in accordance with the requirements of GDC-54 and 10 CFR 50, Appendix J, Type B testing.

SAFETY DESIGN BASIS SIX - The FHS is designed and arranged so that there are no loads which, if dropped, could result in damage, leading to the release of radioactivity in excess of 10 CFR ~~100~~ 50.67 guidelines, or impair the capability to safely shut down the plant.

This meets the requirements of Regulatory Guide 1.13 and excludes the system from the requirements of Regulatory Guide 1.104.

9.1.4.2 System Description

9.1.4.2.1 General Description

The fuel handling system consists of the equipment needed to refuel the reactor core. Basically, this equipment is composed of cranes, handling equipment, and a fuel transfer system.

The associated fuel handling structures are divided into seven areas. In general, these areas are:

a. The refueling pool

b. The fuel transfer canal

SAFETY EVALUATION THREE - The FHS is initially tested with the program given in Chapter 14.0. Periodic inservice functional testing is done in accordance with Section 9.1.4.4. The fuel transfer tube is inspected in accordance with the technical requirements of ASME Section XI.

SAFETY EVALUATION FOUR - Section 3.2 delineates the seismic category applicable to the safety-related portions of this system. Table 9.1-7 shows that the components meet the design and fabrication codes given in Section 3.2.

SAFETY EVALUATION FIVE - Sections 6.2.4 and 6.2.6 provide the safety evaluation for the system containment isolation arrangement and testability.

SAFETY EVALUATION SIX - In the event of a fuel handling accident in the fuel building, the radiological consequences analyzed in Chapter 15.0 demonstrate that the 10 CFR Part ~~100~~ 50.67 guideline values are not exceeded. The circumstances resulting in a handling accident are limited to the following conditions.

a. Fuel drop from a lifting device

b. Improper operation of the transfer equipment and cranes

c. Drop of the fuel shipping cask

d. Drop of the RV head

The fuel handling equipment is designed to prevent a fuel assembly drop by providing special gripping devices which are locked in a manner which will not allow the release of the fuel assembly during transfer. The special features are described in Section 9.1.4.2.2.

Improper operation of the fuel transfer system is prevented by the location of special limit switches and interlocks which will not allow the movement of fuel assemblies unless they are properly oriented, horizontal, thus avoiding a fuel handling accident. Further description of these devices is given in Section 9.1.4.2.2.

Limit switches and interlocks located on the fuel handling cranes, in conjunction with administrative limits, prevent any improper operations which may result in a fuel handling accident. The limiting devices on the refueling machine and spent fuel pool bridge crane do not allow fuel to be moved unless it is in the proper orientation and handled correctly in the gripping tool of the crane.

TABLE 9.1-3

DESIGN COMPARISON TO REGULATORY POSITIONS OF REGULATORY GUIDE 1.13 REVISION 1, DATED DECEMBER 1975, TITLED "SPENT FUEL STORAGE FACILITY DESIGN BASIS"

Regulatory Guide 1.13 Position	WCGS
1. The spent fuel storage facility (including its structures and equipment, except as noted in Paragraph 6 below) should be designed to Category I seismic requirements.	1. Complies as described in Section 9.1.2.1.1.
2. The facility should be designed (a) to keep tornadic winds and missiles generated by these winds from the fuel storage pool and (b) to keep missiles generated by tornadic winds from contacting fuel within the pool.	2. Complies as described in Section 3.5, and 3.8.
3. Interlocks should be provided to prevent cranes from passing over stored fuel (or near stored fuel in a manner such that if a crane failed the load could tip over on stored fuel) when fuel handling is not in progress. During fuel handling operations, the interlocks may be bypassed and administrative control used to prevent the crane from carrying loads that are not necessary for fuel handling over the stored fuel or other prohibited areas. The facility should be designed to minimize the need for bypassing such interlocks.	3. Complies as described in Section 9.1.4.
4. A controlled leakage building should enclose the fuel pool. The building should be equipped with an appropriate ventilation and filtration system to limit the potential release of radioactive iodine and other radioactive materials. The building need not be designed to withstand extremely high winds, but leakage should be suitably controlled during refueling operations. The design of the ventilation and filtration system should be based on the assumption that the cladding of all of the fuel rods in one fuel bundle might be breached. The inventory of radioactive materials available for leakage from the building should be based on the assumptions given in Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors" (Safety Guide 25).	4. Complies as described in Section 9.4.2 and 15.7.4. Note that Regulatory Guide 1.25 has been replaced by following Regulatory Guide 1.183.

SAFETY EVALUATION FOUR - The fuel storage pool cooling pump room coolers, the emergency exhaust system, and the fuel building HVAC boundary penetration isolation provisions are initially tested with the program given in Chapter 14.0. Periodic inservice functional testing is done in accordance with Section 9.4.2.4.

Section 6.6 provides the ASME Boiler and Pressure Vessel Code, Section XI requirements that are appropriate for the fuel storage pool cooling pump room coolers.

SAFETY EVALUATION FIVE - Section 3.2 delineates the quality group classification and seismic category applicable to the safety-related portion of this system and supporting systems. All the power supplies and control functions necessary for safe function of the fuel storage pool cooling pump room coolers, emergency exhaust system, and the fuel building HVAC boundary penetration isolation provisions are Class IE, as described in Chapters 7.0 and 8.0.

SAFETY EVALUATION SIX - Section 9.4.2.2.3 describes the provisions made to assure the isolation of the auxiliary building following a DBA.

SAFETY EVALUATION SEVEN - The emergency exhaust system maintains a negative pressure of no less than 1/4 in. w.g. in the fuel building to prevent unprocessed exfiltration following a fuel handling accident which releases radioactivity. The emergency exhaust system is monitored for radioactivity both upstream and downstream of the filter adsorber unit prior to release to the site. The filter adsorber unit limits the radiological consequences of a fuel handling accident to less than 10 CFR ~~100~~ 50.67 limits.

SAFETY EVALUATION EIGHT - Room coolers are installed in each fuel storage pool cooling pump room and are designed to limit pump room ambient temperature to assure operability of the fuel storage pool cooling pump.

9.4.2.4 Tests and Inspections

Preoperational testing is described in Chapter 14.0.

Filters and adsorbers for the emergency exhaust system are tested in the manufacturer's shop, after initial installation and subsequent to each filter or adsorber change. After installation, interim tests and inspections are performed after every 720 hours of operation and once per 18 months in accordance with the requirements of Regulatory Guide 1.52 and the Technical Specifications, to detect any deterioration of components that may develop under service or standby conditions.

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CHAPTER 11.0

RADIOACTIVE WASTE MANAGEMENT

11.1 SOURCE TERMS

This section presents the design bases for determining the source terms for radioactive releases from the plant, for shielding within the plant, and for accident analysis performed in Chapter 15.0. The source terms used for releases, shielding, and accident analyses are based on 0.12, 0.25, and 1.0 percent fuel defects, respectively.

Actual release data is contained in Annual Radioactive Effluent Release Reports filed with the NRC in accordance with Offsite Dose Calculation Manual (ODCM) requirements. ~~Data supporting Chapter 15 accident analyses is not considered historical and is maintained current.~~

11.1.1 RADIOACTIVE CONCENTRATIONS AND RELEASES

Reactor coolant and secondary coolant specific activities for an assumed 0.12-percent fuel defects and an assumed 100 pounds per day primary-to-secondary leakage are listed in Table 11.1-1. The basis for calculating these sources is Regulatory Guide 1.112. Compliance with Regulatory Guide 1.112 is discussed in Table 11.1-3. Appendix 11.1A provides a description of the input used.

The decontamination factors applied are based on Regulatory Guide 1.112. A description of liquid leakage rates, process paths, and associated component activity levels is contained in Section 11.2 and Appendix 11.1A. A description of gaseous leakage rates, process paths, and associated activity levels is contained in Appendix 11.1A and Sections 11.3 and 9.4. In-plant airborne activity concentrations and other data regarding the ventilation systems are provided in Sections 12.3 and 12.4.

11.1.2 SHIELDING

Reactor coolant and secondary coolant source terms used for shielding are based on 0.25-percent fuel defects. The source terms and the parameters used to calculate the source terms are given in Table 11.1-4 and Appendix 11.1A, respectively. Table 11.1-6 provides the isotopic composition of the contained sources for radioactive waste management systems and for large, potentially radioactive outside storage tanks.

11.1.3 ACCIDENT ANALYSIS SOURCE TERMS

~~Except for a LOCA and a fuel handling accident, the specific activity used for accident analysis releases is based on operating with 1-percent fuel defects which results in a RCS activity limit more limiting than the Technical Specification 3.4.16 limit of 500 uci/gm DOSE EQUIVALENT XE-133. Table 11.1-5 provides the isotopic~~

~~composition of the reactor coolant based on 1 percent fuel defects. Table 11.1-5 also identifies those isotopes excluded from the Technical Specification definition of DOSE EQUIVALENT XE-133 based on low concentration, short half life, or small dose conversion factors. Table 11.1-6 provides the inventory of the contained sources for radioactive waste management systems and for large, potentially radioactive outside storage tanks.~~

~~Sources for the LOCA are based on TID 14844.~~

~~Sources for the fuel handling accident are based on Regulatory Guide 1.25.~~

Chapter 15.0 provides a complete discussion and a listing of the source terms for each accident analyzed.

Solid radwaste is shipped from the site in Department of Transportation-approved containers by Department of Transportation-approved carriers. Containers with any significant surface dose rate are moved remotely from the shielded storage areas to the transporting vehicle.

Radiation measurements made at the time of shipment of any radioactive waste material ensure that all shipments leave the site well within prescribed limits. Similarly, external contamination measurements are made to detect any potential release of radioactive material from the container prior to shipment.

Mixed waste will be stored in liquid and solid form in the MWSF. The total Curie content of the MWSF will be restricted accordingly to maintain doses to the maximally exposed individual during an extreme environmental event (e.g. fire, tornado, etc.) below the applicable limits in 10 CFR 20 and 10 CFR ~~100~~. 50.67

11.4.4 TESTS AND INSPECTIONS

The SRS is in intermittent use throughout normal reactor operation. Periodic visual inspection and preventive maintenance are conducted using normal industry practice. Refer to Chapter 14.0 for information on preoperational and startup testing.

11.4.5 INSTRUMENTATION APPLICATION

Two control panels are provided for the equipment in the SRS which contains or processes potentially radioactive fluids or slurries. One control panel is located in the radwaste building control room and contains the instrumentation for the equipment which interfaces the influent systems (i.e., evaporator bottoms tank - primary, evaporator bottoms tank - secondary, spent resin storage tank - primary, and spent resin storage tank - secondary) and for the equipment used for process control (i.e., acid addition tank, acid addition metering pump, caustic addition tank, and caustic addition metering pump).

The second control panel (radwaste crane control panel) is located in a separate room in close proximity to the solid radwaste processing area. The control panel contains all instrumentation, including television monitors, required for remote operations. Pertinent instruments and controls for the transferring of the wastes from the tanks containing the wastes are duplicated on this panel so that the solid radwaste system operator can transfer the waste from these tanks to the waste disposal station.

Table 11.1-5

Primary Coolant Activity Concentrations (1)

Nuclide	RCS Activity* (µCi/gram)	Nuclide	RCS Activity* (µCi/gram)
Br-83	9.86E-02	Sr-89	4.04E-03
Br-84	4.88E-02	SR-90	2.59E-04
Br-85	5.75E-03	Y-90	7.34E-05
I-127 (grams)	1.24E-10	Y-91m	3.01E-03
I-129	7.17E-08	Sr-91	5.60E-03
I-130	4.65E-02	Y-91	5.64E-04
I-132	3.39E+00	Sr-92	1.31E-03
1-134	7.30E-01	Y-92	1.13E-03
Kr-83m	4.62E-01	Y-93	3.82E-04
Kr-85m	1.83E+00	Zr-95	7.02E-04
Kr-85	1.00E+01	Nb-95	7.03E-04
Kr-87	1.19E+00	Mo-99	8.94E-01
Kr-88	3.29E+00	Tc-99m	8.22E-01
Kr-89	9.30E-02	Ru-103	7.43E-04
I-131	3.28E+00	Rh-103m	7.44E-04
Xe-131m	3.74E+00	Ru-106	3.25E-04
Xe-133m	5.54E+00	Ag-110m	3.34E-03
I-133	5.04E+00	Te-125m	6.03E-04
Xe-133	3.08E+02	Te-127m	4.49E-03
Xe-135m	6.15E-01	Te-127	1.52E-02
I-135	2.85E+00	Te-129m	1.46E-02
Xe-135	8.12E+00	Te-129	1.63E-02
Xe-137	2.18E-01	Te-131m	4.18E-02
Xe-138	7.59E-01	Te-131	1.63E-02
Rb-86	4.33E-02	Te-132	3.43E-01
Rb-88	4.08E+00	Te-134	3.51E-02
Rb-89	1.89E-01	Ba-140	4.55E-03
Cs-134	4.82E+00	La-140	1.53E-03
Cs-136	4.35E+00	Ce-141	6.94E-04
Cs-137	2.68E+00	Ce-143	5.46E-04
Cs-138	1.16E+00	Pr-143	6.46E-04
		Ce-144	5.37E-04

*** - Results include a "fuel management multiplier of 1.04**

(1) Refer to Table 11.1A-1 for assumptions

TABLE 11.1-5

Reactor Coolant and Secondary Coolant
Specific Activities - 1% Fuel Defects[1]

Class 1	Reactor Coolant µCi/gm	Secondary Coolant µCi/gm
Kr-83m*	5.54E-01	1.92E-05
Kr-85m	2.26E+00	7.10E-05
Kr-85*	9.41E+00	2.96E-04
Kr-87	1.47E+00	4.62E-05
Kr-88	4.26E+00	1.34E-04
Kr-89*	1.21E-01	3.71E-06
Xe-131m*	3.41E+00	1.07E-04
Xe-133	2.90E+02	9.12E-03
Xe-133m	5.37E+00	1.73E-04
Xe-135m	6.04E-01	8.62E-05
Xe-135	9.82E+00	3.20E-04
Xe-137*	2.24E-01	6.90E-06
Xe-138	8.15E-01	2.55E-05
Total noble gas	3.29E+02	1.04E-02
Class 2		
Br-83	1.09E-01	1.98E-04
Br-84	5.82E-02	4.38E-05
Br-85	6.86E-03	6.00E-07
I-130	3.57E-02	9.62E-05
I-131	2.80E+00	8.46E-03
I-132	3.14E+00	5.97E-03
I-133	4.93E+00	1.39E-02
I-134	7.52E-01	8.27E-04
I-135	2.88E+00	7.05E-03
Total halogens	1.47E+01	3.66E-02

*Excluded from Technical Specification definition of DOSE EQUIVALENT XE-133.

TABLE 11.1-5 (Sheet 2)

Reactor Coolant and Secondary Coolant
Specific Activities - 1% Fuel Defects[1]

Class 3	Reactor Coolant µCi/gm	Secondary Coolant µCi/gm
Rb-86	2.85E-02	1.56E-04
Rb-88	5.36E+00	2.72E-03
Rb-89	2.46E-01	1.08E-04
Cs-134	2.35E+00	1.30E-02
Cs-136	2.82E+00	1.54E-02
Cs-137	1.94E+00	1.07E-02
Cs-138	1.25E+00	1.07E-03
Total Cs, Rb	1.40E+01	4.32E-02
Class 4		
N-16	1.31E+02	3.12E-10
Water activation product		
Class 5		
H-3	3.50E+00	2.19E+00
Tritium		
Class 6		
Cr-51	1.90E-03	5.83E-06
Mn-54	3.10E-04	9.53E-07
Fe-55	1.60E-03	4.92E-06
Fe-59	1.00E-03	3.07E-06
Co-58	1.60E-02	4.92E-05
Co-60	2.00E-03	6.15E-06
Sr-89	5.11E-03	2.84E-05
Sr-90	1.90E-04	1.05E-06
Sr-91	6.73E-03	2.84E-05
Sr-92	1.48E-03	3.93E-06
Y-90	5.19E-05	1.82E-07
Y-91m	3.95E-03	1.49E-05
Y-91	5.71E-04	1.79E-06

TABLE 11.1-5 (Sheet 3)

Reactor Coolant and Secondary Coolant
Specific Activities - 1% Fuel Defects[1]

Class 6	Reactor Coolant µCi/gm	Secondary Coolant µCi/gm
Y-93	4.37E-04	1.16E-06
Zr-95	6.52E-04	2.00E-06
Nb-95	6.53E-04	2.01E-06
Mo-99	8.19E-01	2.46E-03
Tc-99m	7.54E-01	2.28E-03
Ru-103	5.35E-04	1.64E-06
Ru-106	1.65E-04	5.07E-07
Rh-103m	5.31E-04	1.64E-06
Rh-106	1.65E-04	2.57E-09
Ag-110m	1.31E-03	4.03E-06
Te-125m	5.92E-04	1.82E-06
Te-127m	2.95E-03	9.07E-06
Te-127	1.30E-02	3.54E-05
Te-129m	1.03E-02	3.16E-05
Te-129	1.36E-02	2.91E-05
Te-131m	2.55E-02	7.44E-05
Te-131	1.43E-02	1.96E-05
Te-132	2.99E-01	9.01E-04
Te-134	3.70E-02	3.42E-05
Ba-137m	1.83E+00	1.00E-02
Ba-140	4.14E-03	1.27E-05
La-140	1.35E-03	4.48E-06
Ce-141	6.34E-04	1.95E-06
Ce-143	5.53E-04	1.62E-06
Ce-144	4.69E-04	1.44E-06
Pr-143	6.13E-04	1.88E-06
Pr-144	4.69E-04	1.44E-06
Total other isotopes	3.88E+00	1.61E-02 Note (3)

~~(1) Refer to Table 11.1A 1 for assumptions.~~
~~(2) For the secondary side, the noble gas activities are for the~~ steam phase; all other activities are for the steam generator water activities.
(3) Lower blowdown rates result in higher secondary system activities. A 60-gpm blowdown will result in a total of 4.65E-1 µCi/gm (excluding noble gases, N-16, and tritium) in the steam generator. A maximum blowdown rate was used in this table.

APPENDIX 11.4A

INTERIM ON-SITE STORAGE FACILITY

11.4A.1 Introduction

In order to permit plant operation in the event that a permanent disposal site is unavailable, it is necessary to store waste temporarily on-site. This temporary storage is provided by the Interim On-Site Storage (IOS) Facility. The existing waste bale drum structure, which is South of the Radwaste Building, will be used as the IOS facility.

Temporary storage, on a concrete slab or within a building addition located West of the IOS facility and South of the Radwaste Building, provides temporary indoor/outdoor storage of large waste material which becomes activated during reactor operation. Each stored item will be unique, therefore procedures for storing items outdoors will be determined on a case by case basis.

11.4A.2 Design Objectives

The design of the IOS facility provides storage for solid waste produced at WCGS based on five years of processed waste (i.e. resins and sludges, including filter cartridges) and, due to storage capacity limitations, three and one half years of Dry Active Waste (DAW) generated as a result of normal operation of WCGS. The values contained in Table 11.4A-4, "Estimated Capacity and Radwaste Container Distribution for the IOS Facility", serve as the basis for the design storage capacity.

11.4A.3 Description of Containers

Containers used for packaging of radioactive material, and stored in the IOS, shall meet the applicable DOT requirements for quantity and form or the current burial site regulations for disposal (HIC) when placed in storage. Typical containers expected to be stored in the IOS facility are detailed in Table 11.4A-4. All containers are designed to reduce the occurrence of uncontrolled releases of radioactive materials due to handling, transportation, and storage. All containers are designed with materials compatible with the stored waste to prevent significant container corrosion.

11.4A.4 Description of Stored Wastes

Solidified radwaste, or waste meeting the no free standing water criteria of Branch Technical Position ETSB 11-3 (i.e. dewatered), shall be stored in the IOS facility. These wastes satisfy all applicable transportation and disposal requirements.

Wet radioactive waste, defined as any waste which does not meet receiving burial site free liquid requirements may be temporarily stored in the IOS facility.

11.4A.4.1 Dry Active Waste (DAW)

This includes contaminated trash (paper, cloth, plastic, etc.) super compacted into drums, typically by an off-site vendor. The exposure rate from these containers is low (2 mrem/hr to about 100 mrem/hr with a majority less than 10 mrem/hr).

11.4A.4.2 Solidified/Dewatered Wastes

Resin, filter cartridges and filter sludges will be transferred into HICs, and dewatered to less than 1% free standing water. Tables 11.1-6 (Sheet 1) to 11.1-6 (Sheet ~~2~~4) and 11.4-4 provide normal activity concentrations in the input streams.

radiation dose through any wall. The shielding thickness was designed such that the aggregate postulated radiation level from all contributing sources, at this point, would be attenuated to meet the criteria of the adjoining radiation zone specifications.

Where shielded entryways to compartments containing high radiation sources are anticipated, labyrinths or mazes are used. The mazes are constructed so that the scattered dose rate plus the transmitted dose rate through the shield wall from all contributing sources are consistent with the radiation zone specified for each plant area.

12.3.3 VENTILATION

The plant heating, ventilating, and air-conditioning (HVAC) systems are designed to provide a suitable environment for personnel and equipment during normal operation and anticipated operational occurrences. Parts of the plant HVAC systems perform safety-related functions.

12.3.3.1 Design Objectives

The plant HVAC systems for normal plant operation and anticipated operational occurrences are designed to meet the requirements of 10 CFR 20, "Standards for Protection Against Radiation," and 10 CFR 50, "Licensing of Production and Utilization Facilities."

12.3.3.2 Design Criteria

Design criteria for the plant HVAC systems include the following:

a. During normal operation and anticipated operational occurrences, the average and maximum airborne radioactivity levels to which plant personnel are exposed in the restricted areas of the plant are ALARA and within the limits specified in 10 CFR 20.

b. During normal operation and anticipated operational occurrences, the dose from concentrations of airborne radioactive material in unrestricted areas beyond the site boundary are ALARA and within the limits specified in 10 CFR 20 and 10 CFR 50.

c. The plant siting dose guidelines of 10 CFR ~~100~~ 50.67 are satisfied, following those hypothetical accidents described in Chapter 15.

d. No filter bank is more than three filter units high, where each filter unit is 2 feet by 2 feet. The access to the level or platform at which the filter is serviced is by stairs or elevators.

e. The clear space for doors is a minimum of 3 feet by 7 feet.

f. The filters are designed with replaceable 2 feet by 2 feet units that are clamped in place against compression seals. The filter housing is designed, tested, and proven to be airtight with bulkhead type doors that are closed against compression seals.

12.3.4 AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING INSTRUMENTATION

12.3.4.1 Area Radiation Monitoring

The area radiation monitoring system (ARMS) is provided to supplement the personnel and area radiation survey provisions of the plant health physics program described in Section 12.5 and to ensure compliance with the personnel radiation protection guidelines of 10 CFR 20, 10 CFR 50, 10 CFR 70, and Regulatory Guides 8.2, 8.8, and 8.12.

12.3.4.1.1 Design Bases

The principal objectives and criteria of the ARMS are provided below.

12.3.4.1.1.1 Safety Design Bases

The area radiation monitoring system has no function related to the safe shutdown of the plant or the capability to mitigate the consequences of accidents that could result in offsite exposures comparable to the guideline exposure of 10 CFR ~~100~~ and, therefore, has no safety design bases. See Appendix 7A for a 50.67 sion of Regulatory Guide 1.97.

12.3.4.1.1.2 Power Generation Design Bases

POWER GENERATION DESIGN BASIS ONE - The ARMS functions continuously to immediately alert plant personnel entering or working in nonradiation or low-radiation areas of increasing or abnormally high radiation levels which, if unnoticed, could possibly result in inadvertent overexposures.

TABLE OF CONTENTS (CONTINUED)

15A.3.1

App. 15B Regulatory Guide 1.183, Reivison 0 Conformance Tables 15B-1

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TABLE OF CONTENTS (CONTINUED)

LIST OF TABLES

CHAPTER 15 - LIST OF FIGURES

*Refer to Section 1.6 and Table 1.6-3. Controlled drawings were removed from the USAR at Revision 17 and are considered incorporated by reference.

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CHAPTER 15 - LIST OF FIGURES

*Refer to Section 1.6 and Table 1.6-3. Controlled drawings were removed from the USAR at Revision 17 and are considered incorporated by reference.

k. Partial loss of forced reactor coolant flow (Section 15.3.1).

l. Uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low power startup condition (Section 15.4.1).

m. Uncontrolled rod cluster control assembly bank withdrawal at power (Section 15.4.2).

n. Rod cluster control assembly misalignment (dropped RCCA, dropped RCCA bank, or statically misaligned RCCA) (Section 15.4.3).

o. Startup of an inactive reactor coolant pump at an incorrect temperature (Section 15.4.4).

p. Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant (Section 15.4.6).

q. Inadvertent operation of the emergency core cooling system during power operation (Section 15.5.1).

r. Chemical and volume control system malfunction that increases reactor coolant inventory (Section 15.5.2).

s. Inadvertent opening of a pressurizer safety or relief valve (Section 15.6.1).

t. Break in instrument line or other lines from reactor coolant pressure boundary that penetrate the containment (Section 15.6.2).

15.0.1.3 Condition III - Infrequent Faults

Condition III events are faults which may occur very infrequently during the life of the plant. They may result in 50.67 ilure of only a small fraction of the fuel rods. The release of radioact will not be sufficient to interrupt or restrict public use of those areas beyond the exclusion area boundary, per the guidelines of 10 CFR ~~100~~. A Condition III event will not, by itself, generate a Condition IV event or result in a consequential loss of function of the reactor coolant system or containment barriers. The following faults are included in this category:

a. Steam system piping failure (minor) (Section 15.1.5).

b. Complete loss of forced reactor coolant flow (Section 15.3.2).

c. Rod cluster control assembly misalignment (single rod cluster control assembly withdrawal at full power) (Section 15.4.3).

d. Inadvertent loading and operation of a fuel assembly in an improper position (Section 15.4.7).

e. Loss-of-coolant accidents resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (small break) (Section 15.6.5).

f. Radioactive gas waste system leak or failure (Section 15.7.1).

g. Radioactive liquid waste system leak or failure (Section 15.7.2).

h. Postulated radioactive releases due to liquid tank failures (Section 15.7.3).

i. Spent fuel cask drop accidents (Section 15.7.5).

15.0.1.4 Condition IV - Limiting Faults

Condition IV events are faults which are not expected to take place, but are postulated because their consequences would include the potential of the release of significant amounts of radioactive material. They are the ~~most~~ drastic which must be designed against and represent limiting design 50.67 Condition IV events are not to cause a fission product release to the environment resulting in doses in excess of guideline values of 10 CFR ~~100~~. A single Condition IV event is not to cause a consequential loss of required functions of systems needed to cope with the fault, including those of the emergency core cooling system and the containment. The following faults have been classified in this category:

a. Steam system piping failure (major) (Section 15.1.5).

b. Feedwater system pipe break (Section 15.2.8).

c. Reactor coolant pump shaft seizure (locked rotor) (Section 15.3.3).

d. Reactor coolant pump shaft break (Section 15.3.4).

15.0.8 MAJOR PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR MITIGATION OF TRANSIENT AND ACCIDENT CONDIITONS

The plant is designed to afford protection against the possible effects of natural phenomena, postulated environmental conditions, and dynamic effects of the postulated accidents. In addition, the design incorporates features which minimize the probability and effects of fires and explosions. Chapter 17.0 discusses the quality assurance program which has been implemented to assure that the NSSS will satisfactorily perform its assigned safety functions. The incorporation of these features in the plant, coupled with the reliability of the design, ensures that the normally operating systems and equipment listed in Table 15.0-6 are available for mitigation of the events discussed in Chapter 15.0. In determining which systems are necessary to mitigate the effects of these postulated events, the classification system of ANSI-N18.2-1973 is utilized. The design of "systems important to safety" (including protection systems) is consistent with IEEE Standard 379-1972 and Regulatory Guide 1.53, in the application of the single failure criterion.

In the analysis of the Chapter 15.0 events, control system action is considered only if that action results in more severe accident results. No credit is taken for control system operation if that operation mitigates the results of an accident. ~~For some~~ accidents, the analysis is performed both with and without cont operation to determine the worst case. The pressurizer heaters are to be energized during any of the Chapter 15.0 events.

plus 2% uncertainty

15.0.9 FISSION PRODUCT INVENTORIES

are calculated by the industry standard ORIGEN-S code (Ref 1).

15.0.9.1 Activities in the Core

The calculation of the core iodine fission product inventory is ~~consistent with the inventories given in TID-14844 (Ref. 1) and is~~ based on a core power level of 3565 MWt. The fission product inventories ~~for other isotopes that are important from a health hazards point of view are consistent with the data from APED-5398 (Ref. 2).~~ These inventories are given in Table 15A-3. The isotopes included in Table 15A-3 (Appendix 15A) are ~~the isotopes controlling from considerations of inhalation dose (iodines) and from direct dose due to immersion (noble gases).~~

the dose-significant isotopes from the radionuclide groups identified in Regulatory Guide 1.183 that should be considered in design basis analyses.

~~The isotopic data of APED-5398, utilizing the f uranium-235 as the sole fission ntory resulting from the fissioni d. The results of this review indicated that inclusion of all fission source data would result in small (less than 10 percent) change in the isotopic inventories.~~

15.0.9.2 Activities in the Fuel Pellet Clad Gap

The fuel-clad gap activities are determined, using the model given in Regulatory Guide ~~1.77. Thus,~~ the amount of activity accumulated in the fuel clad gap is assumed to be ~~10 percent of the core activity for all isotopes except for Kr-85. For Kr-85 it is assumed to be 30 percent of the core activity. The gap activities are given in Table 15A-3.~~

8 percent for I-131, 10 percent for Kr-85, 12 percent for alkali metals, and 5 percent for other noble gases and iodines, with certain exceptions. For the rod ejection accident, it is assumed to be 10% for iodines and noble gases and 12% for alkali metals. For LOCA, it is assumed to be 5% for these three groups of radionuclides.

1.183, except for fuel handling accident, which used the more conservative gap fractions listed in Regulatory Guide 1.25 (see Section 15.7.4). In accordance with Regulatory Guide 1.183,

As soon as primary temperature begins to decrease, the operator can use the steam dump system or the steam generator atmospheric relief valves to begin a controlled cooldown. In addition, if the U-tubes of the intact steam generators are covered with water as indicated by post accident monitoring system (PAMS) steam generator water level instrumentation (see Chapter 7.0), the operator can modulate the high-head charging pumps, so that the primary pressure decreases while ensuring that voiding does not occur within the RCS. The primary pressure-temperature relationship can be monitored by the operator via the PAMS widerange RCS pressure and temperature instruments.

Using the above-mentioned PAMS indications, the operator can maintain the plant in a hot shutdown condition for an extended period of time, or can proceed to a cold shutdown condition as desired.

The safety-related indicators for steamline pressure and pressurizer water level noted above are further discussed in Section 7.5.

Tables 15.0-8 and 15.0-9 list the short term operator actions required to bring the plant to a stable condition for the LOCA and steam generator tube rupture (SGTR). Further information (including alarms which alert the operator) on operator action for these two accidents are given in Section 6.3.2.8 for the LOCA and Section 15.6.3 for the SGTR.

Process information available to the operator in the control room following either of these accidents (LOCA or SGTR) is given in Section 7.5.

Instrumentation and controls provided to allow the operator to complete required manual actions are classified as Class IE. Electrical components are also classified as Class IE. Mechanical components are classified as Safety Class 1, 2, or 3.

Safety systems required for accident mitigation are designed to function after the occurrence of the worst postulated single failure. There are no adverse impacts as a result of these actions.

15.0.14 REFERENCES

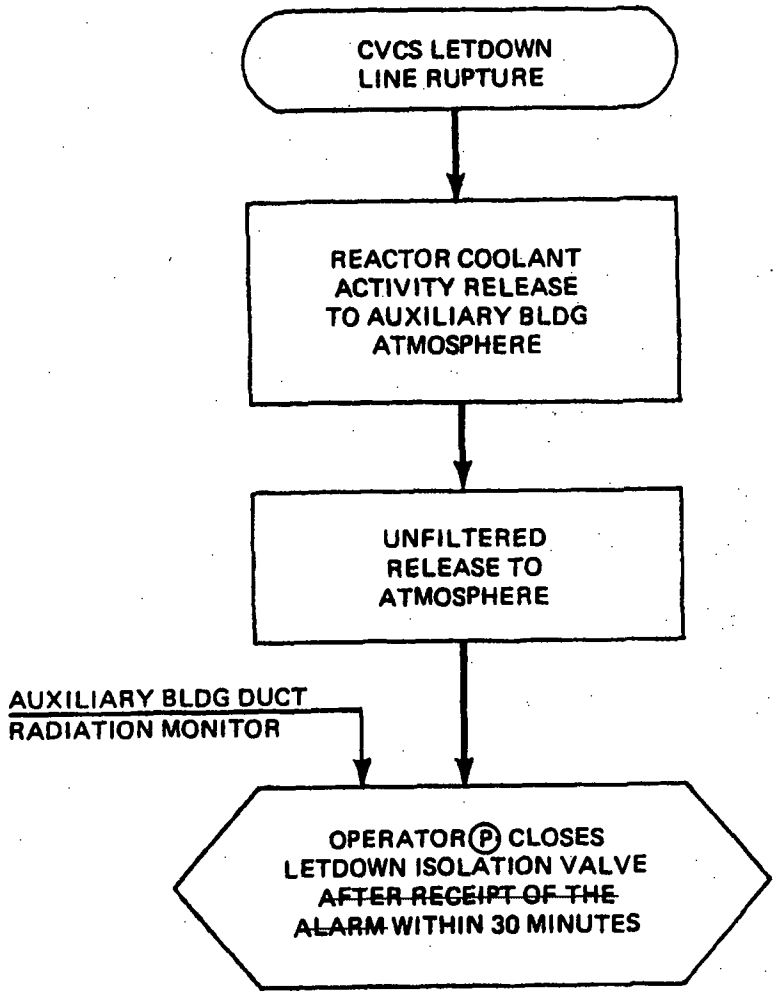
1. ~~DiNunno, J. J., et al., "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, March 1962.~~

2. ~~Meek, M. E. and Rider, B. F., "Summary of Fission Product Yields for U-235, U-238, Pu-239, and Pu-241 at Thermal Fission Spectrum and 14 Mev Neutron Energies," APED-5398, March 1968~~.

Replace with "Deleted"

3. Bordelon, F. M., et al., "SATAN-VI Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant," WCAP-8302 (Proprietary) and WCAP-8306 (Non-Proprietary), June 1974.

1. Gauld, I. C., Hermann, O. W., Westfall, R. M., "ORIGEN-S: Scale System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms," Oak Ridge National Laboratory, ORNL/TM-2005/39, Vol. II, Book 1, Sect. F7, February 2005.



WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.0-26

CVCS LETDOWN LINE RUPTURE

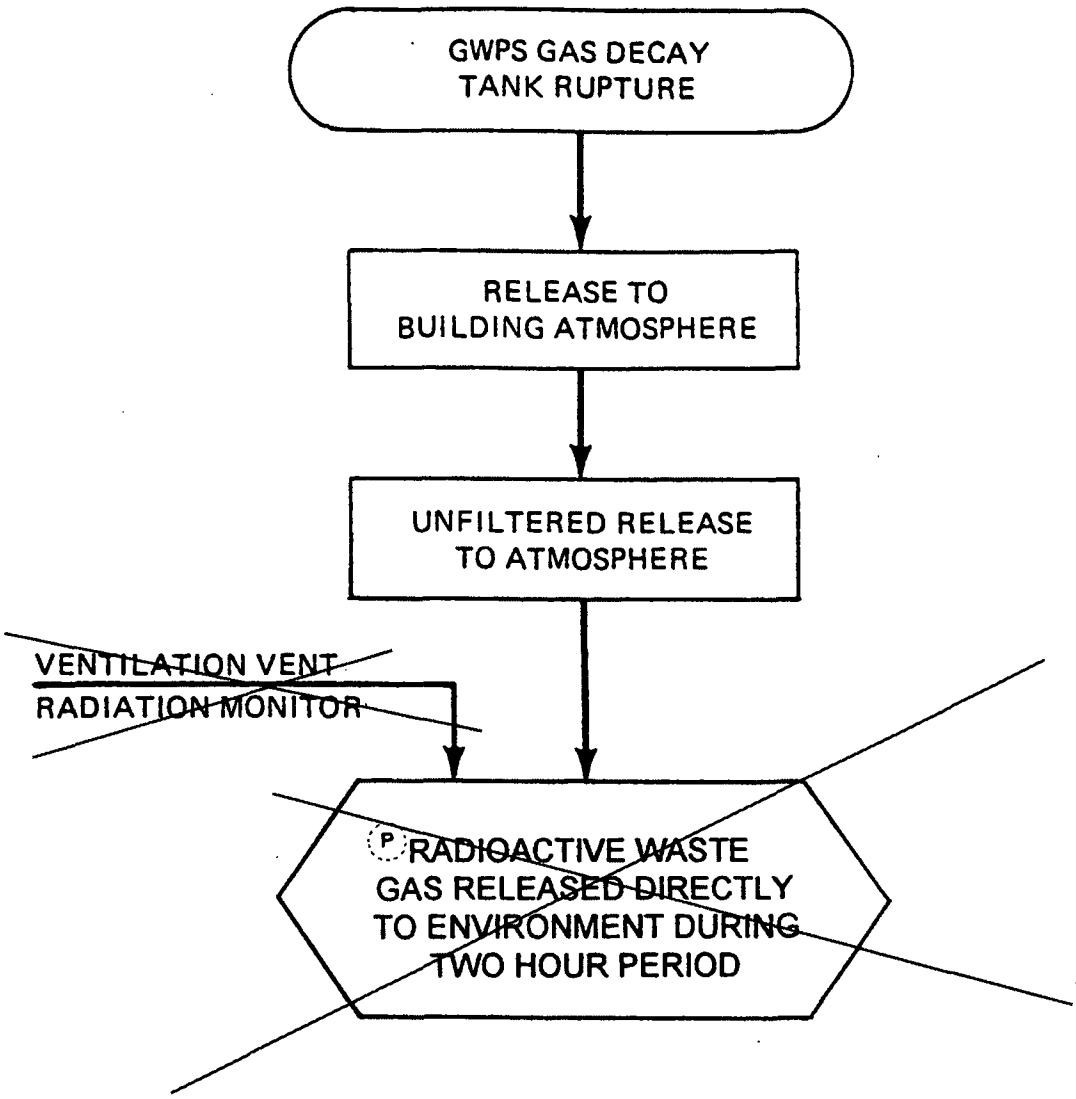


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FIGURE 15.0-27

GWPS GAS DECAY TANK RUPTURE

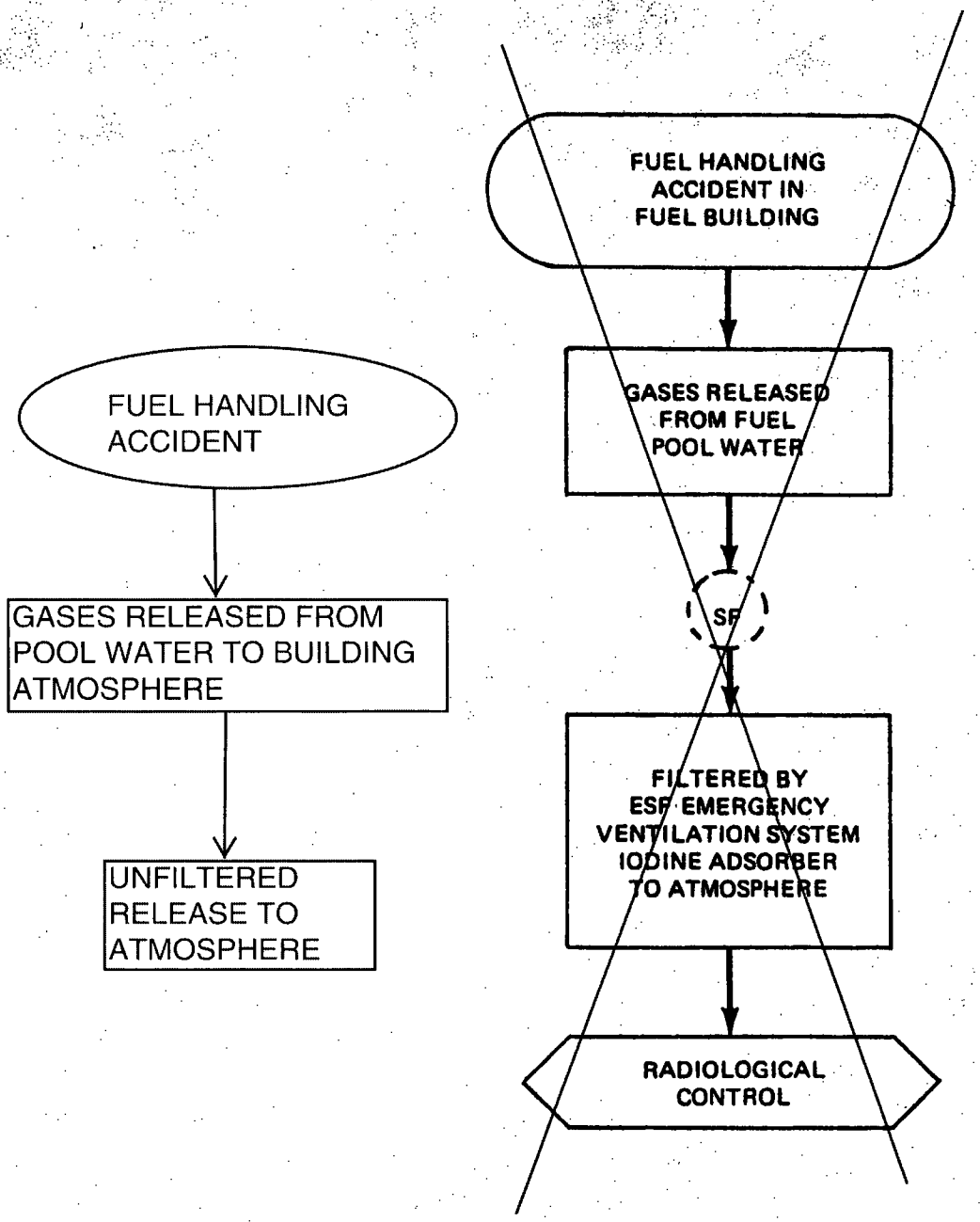

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FIGURE 15.0-29

FUEL HANDLING ACCIDENT ~~IN FUEL BUILDING~~

Delete

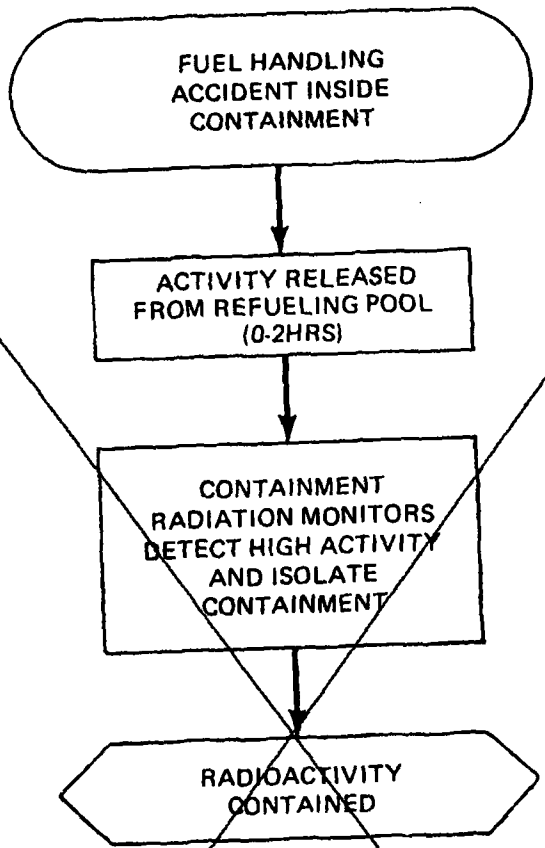

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FIGURE 15.0-30

FUEL HANDLING ACCIDENT INSIDE CONTAINMENT

The analysis of a main steam line rupture is performed to demonstrate that the following criteria are satisfied:

Assuming a stuck RCCA with or without offsite power, and assuming a single failure in the engineered safety features system, the core cooling capability is maintained. Radiation doses do not exceed the guidelines of 10 CFR ~~100~~. 50.67

Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable, the following analysis, in fact, shows that the DNB design basis is not exceeded for any rupture, assuming the most reactive control rod assembly stuck in its fully withdrawn position. The DNBR design basis is discussed in Section 4.4.

A major steam line rupture is classified as an ANS Condition IV event. See Section 15.0.1 for a discussion of Condition IV events.

Effects of minor secondary system pipe breaks are bounded by the analysis presented in this section. Minor secondary system pipe breaks are classified as Condition III events, as described in Section 15.0.1.3.

The methodology for analyzing the reactor core response to excessive secondary steam releases is documented in WCAP-9226 (Reference 3). This WCAP, known as the "Steamline Break Topical," examined the effect of power level (including full power cases), break size, and plant variations for typical three-loop and four-loop Westinghouse-designed PWRs. This WCAP concludes that the largest double-ended steamline rupture at end-of-life, hot zero-power (Mode 2) conditions, with the most reactive RCCA in the fully withdrawn position, bounds all other power levels and other Modes for the post-trip phase of the transient.

Note that the conclusion that the hot-zero power case is the limiting case is based on certain specific protection system performance characteristics credited for "at power" steamline break analyses. Since the WCAP was first issued, plant modifications have been made which might not be bounded by the generic assumptions of the WCAP, such as increases in the Overpower ΔT response times, lead/lag time constant changes, reduced Low Steam Pressure setpoints, etc. As a result, the full power steamline break event, which credits the Overpower ΔT reactor trip function, could be initiated from the plant conditions that are outside the limits of applicability.

A Wolf Creek specific analysis for steamline breaks occurring while the reactor is at power has been performed to confirm the conclusions documented in WCAP-9226 are valid, even with assumed parameters being outside the generic limits established by Westinghouse. The analysis demonstrates the adequacy of the protection systems by showing that with the appropriate actions of these systems, the DNB design basis is satisfied and fuel centerline melting is precluded.

As such, a detailed analysis of this transient with the largest double-ended rupture at the hot-zero power conditions, is presented here.

During startup or shutdown evolutions, when the operator manually blocks the safety injection on low pressurizer pressure or low steamline pressure and steamline isolation on low steamline pressure when pressurizer pressure is less than P-11 setpoint (i.e., 1970 psig), the steamline pressure-negative rate-high signal is automatically enabled to provide steamline isolation. For inside containment breaks, steamline isolation may also be provided by the containment pressure High-2 signal and safety injection would be actuated by the

15.1.5.3 Radiological Consequences

15.1.5.3.1 Method Of Analysis

15.1.5.3.1.1 Physical Model

The radiological consequences of a MSLB inside the containment are less severe than the one outside the containment because the radioactivity released will be held up inside the containment, allowing decay and plateout of the radionuclides. To evaluate the radiological consequences due to a postulated MSLB (outside the containment), it is assumed that there is a complete severance of a main steam line outside the containment.

It is also assumed that there is a simultaneous loss of offsite power, resulting in reactor coolant pump coastdown. The safety injection system is actuated and the reactor trips.

The main steam line isolation valves, their bypass valves, and the steam line drain valves isolate the steam generators and the main steam lines upon a signal initiated by the engineered safety features actuation system under the conditions of high steam negative pressure rate or low steam line pressure. The main steam isolation valves are installed in the main steam lines from each steam generator downstream from the safety and atmospheric relief valves outside the containment. The break in the main steam line is assumed to occur outside of the containment. The affected steam generator (steam generator connected to a broken steam line) blows down completely. The steam is vented directly to the atmosphere.

Each of the steam generators incorporates integral flow restrictors, which are designed to limit the rate of steam blowdown from the steam generators following a rupture of the main steam line. This, in turn, reduces the cooling rate of the reactor coolant system to preclude departure from nucleate boiling (DNB).

In case of loss of offsite power, the remaining steam generators are available for dissipation of core decay heat by venting steam to the atmosphere via the atmospheric relief valves. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently so that the RHR system can be utilized to cool the reactor.

15.1.5.3.1.2 Assumptions and Conditions

The major assumptions and parameters assumed in the analysis are itemized in Tables 15.1-3 and 15A-1.

Tables 15B-1 and 15B-4 provide a comparison of the analysis to the guidelines of Regulatory Guide 1.183.

The assumptions used to determine the concentrations of radioactive isotopes within the secondary system for this accident are as follows:

a. The secondary system initial concentrations of radioactive isotopes are assumed ~~to be the dose~~ equivalent of 0.1 μCi/gm of ~~1~~-131. change to "I" instead of 1

b. A primary-to-secondary leakage rate of 1 gpm is assumed to exist and is assumed to be in the ~~affected~~ steam generator. faulted

c. The reactor coolant concentration of radioactive isotopes is determined by two methods, and both cases are analyzed. These are:

1. The initial reactor coolant concentrations of radioactive isotopes are assumed to be the dose equivalent of 1.0 μCi/gm of I-131 with an iodine spike that increases the rate of iodine release into the reactor coolant by a factor of 500. This increased rate of transfer from the fuel into the coolant is assumed for 8 hours.

2. An assumed reactor coolant concentration of radioactive isotopes with a dose equivalent of 60 μCi/gm of I-131 as a result of preaccident iodine spikes.

d. The reactor coolant concentrations of noble gas correspond to ~~1-percent failed fuel~~. 500 micro-Ci/gm DE Xe-133.

e. Partition factors used to determine the secondary system activities are given in Table 15.1-3. for iodines and alkali metals

The following specific assumptions and parameters are used to calculate the activity release:

a. Offsite power is lost, resulting in reactor coolant pump coastdown.

b. No condenser air removal system release and no normal operating steam generator blowdown is assumed to occur during the course of the accident.

c. ~~Eight~~ Twelve hours after the occurrence of the accident, the residual heat-removal system (RHRS) ~~starts~~ operation ~~to cool down the plant~~. matches decay heat and steam releases are terminated.

d. After the acc34nt, the primary-to-secondary leakage continues for ~~8~~ hours, at which time the reactor coolant system is ~~depressurized~~. cooled to 212°F such that there would be no flashing of the leaked fluid.

f. The reactor coolant concentrations of alkali metals correspond to 1-percent failed fuel as provided in Table 11.1-5.

g. The secondary system initial concentration of alkali metals is 10% of the reactor coolant concentrations.

e. The ~~affected~~ faulted steam generator (steam generator connected to the broken steam line) is allowed to blow down completely.

f. Steam release to the atmosphere and the associated activity release from the safety and atmospheric relief valves ~~and the broken steam line~~ is terminated ~~8~~ 12 hours after the accident on the intact steam generators when the RHRS is activated ~~to~~ complete cooldown.

g. The amount of noble gas activity released is equal to the amount present in the reactor coolant, which leaks to the secondary during the accident. The amount of iodine and alkali metal activity released is based on the activity present in the secondary system and the amount of leaked reactor coolant which is entrained in the steam that is discharged to the environment via the safety and atmospheric relief valves and the broken steam line. Partition factors used for the unaffected steam generators after the accident occurs are given in Table 15.1-3. An iodine partition factor of 1 is used for the ~~affected~~ faulted steam generator.

h. The activity released from the broken steam line and the safety and atmospheric relief valves during the ~~8-hour duration of the~~ accident is immediately vented to the atmosphere.

15.1.5.3.1.3 Mathematical Models Used in the Analysis

Mathematical models used in the analysis are described in the following sections:

a. The mathematical models used to analyze the activity released during the course of the accident are described in Appendix 15A.

b. The atmospheric dispersion factors used in the analysis were calculated based on the onsite meteorological measurement programs described in Section 2.3.

c. The ~~thyroid inhalation dose and total-body gamma immersion~~ TEDE doses to a receptor at the exclusion area boundary, ~~and~~ outer boundary of the low-population zone, and in the control room were analyzed, using the models described in Appendix 15A.

15.1.5.3.1.4 Identification of Leakage Pathways and Resultant Leakage Activity

For evaluating the radiological consequences due to a postulated MSLB, the activity released from the ~~affected~~ faulted steam generator (steam generator connected to the broken steam line) is released directly to the environment. The unaffected steam generators are assumed to continually discharge steam and entrained activity via the safety and atmospheric relief valves up to the time initiation of the RHRS can be accomplished.

Since the activity is released directly to the environment with no credit for plateout or retention, the results of the analysis are based on the most direct leakage pathway available. Therefore, the resultant radiological consequences represent the most conservative estimate of the potential integrated dose due to the postulated MSLB.

15.1.5.3.2 Identification of Uncertainties and Conservatisms in the Analysis

a. Reactor coolant ~~activities~~ iodine and noble gas are based on the Technical Specification ~~limit~~s ~~of 1.0 µCi/gm I-131 dose equivalent with extremely large iodine spike values persisting for the entire duration of the accident, resulting in equivalent concentrations many times greater~~ which are significantly higher than the reactor coolant activities ~~based on 0.12 percent failed fuel~~ associated with normal operating conditions.

b. A 1-gpm steam generator primary-to-secondary leakage is assumed, which is significantly greater than that anticipated during normal operation (150 gpd/SG). ~~Furthermore,~~ it was conservatively assumed that all 1 gpm leakage is to the faulted ~~affected~~ steam generator only and a total of 450 gpd is to the intact steam generators.

c. The meteorological conditions which may be present at the site during the course of the accident are uncertain. However, it is highly unlikely that the assumed meteorological conditions would be present during the course of the accident for any extended period of time. Therefore, the radiological consequences evaluated, based on the meteorological conditions assumed, are conservative.

d. Reactor coolant activities based on extreme iodine spiking effects are conservatively high.

15.1.5.3.3.1 Filter Loadings

The only ESF filtration system considered in the analysis which limits the consequences of the MSLB is the control room filtration system. Activity loadings on the control room charcoal filter are based on flow rate through the filter, the concentration of activity at the filter inlet, and the filter efficiency.

Activity in the control room filter as a function of time has been evaluated for the more limiting LOCA analysis, as discussed in Section 15.6.5.4.3.1. Since the control room filters are capable of accommodating the potential design basis LOCA fission product iodine loadings, more than adequate design margin is available with respect to postulated MSLB releases.

15.1.5.3.3.2 Dose to Receptor at the Exclusion Area Boundary, ~~and~~ Low-Population Zone Outer Boundary, and in the Control Room

The potential radiological consequences resulting from the occurrence of a postulated MSLB have been conservatively analyzed, using assumptions and models described. The ~~total-body gamma doses due to immersion from direct radiation and the thyroid dose due to inhalation~~ have been analyzed for the 0-2 hour dose TEDE doses at the exclusion area boundary, ~~and~~ for the duration of the accident (0 to ~~8~~ 34 hrs) at the low-population zone outer boundary. The results are listed in Table 15.1-4. The resultant doses are within the acceptance limits, a small fraction (10 percent) of exposure limits of 10CFR~~100, i.e., 2.5 rem and 30 rem respectively for the whole body and thyroid doses~~ for the case of concurrent iodine spike and the exposure limits of 10CFR~~100 i.e., 25 rem and 300 rem respectively for the whole body and thyroid doses~~ for the case of pre-accident iodine spike.

offsite

, and in the control room for 30 days.

50.67

15.1.5.4 Conclusions

The analysis has shown that the criteria stated earlier in Section 15.1.5.1 are satisfied.

Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable and not precluded by the criteria, the above analysis shows that the DNB design basis is met for any rupture, assuming the most reactive RCCA stuck in its fully withdrawn position.

15.1.5.5 Notes

(1) As discussed in Reference 4, the SI response time has an additional delay of 15 seconds. This makes the SI response time 27 and 39 seconds for the cases with and without offsite power respectively.

The resultant control room doses are within the acceptance criteria limits of 10CFR50.67.

TABLE 15.1-3

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

I.	Source Data:	
a.	Core power level, Mwt	~~3565~~ 3637 (includes uncertainty)
b.	Steam generator tube leakage, gpm	1 to the faulted steam generator and 450 gpd total to the remaining steam generators
c.	Reactor coolant iodine activity:	
1)	Case 1 — Dose equivalent of 1.0μCi/gm o assumed iodine spike that increases the rate of iodine release into the reactor coolant by a factor of 500	
2)	Case 2 — An assumed pre-accident iodine spike which has resulted in the dose equivalent of 60 μCi/gm of I-131	
d.	Reactor coolant noble gas activity:	
1)	Case 1	~~Based on 1-percent failed fuel as provided in~~ Dose equivalent of 500 microCi/gm of Xe-133
2)	Case 2	~~Based on 1-percent failed fuel as provided in~~ Dose equivalent of 500 microCi/gm of Xe-133
~~e.~~ f.	Secondary system initial iodine activity	Dose equivalent of 0.1 μCi/gm of I-131
~~f.~~ h.	Iodine partition factors	
1)	Faulted ~~d~~ steam generator	1.0
2)	Intact ~~ected~~ steam generators	0.01
j. ~~g.~~	Reactor coolant mass, lbs	~~4.94E+5~~
k. ~~h.~~	Steam generator mass	3.99E+5
1)	Faulted ~~d~~ steam generator, lbs	~~164,500~~ 165,000
2)	Each intact ~~cted~~ steam generator, lbs	~~95,500~~ 82,333
II.	Atmospheric Dispersion Factors	See Table 15A-2
III.	Activity Release Data:	
a.	Faulted ~~d~~ steam generator	
1)	Initial steam release, 0-~~30~~ 2 min, lbs	~~164,500~~ 165,000
2)	Reactor coolant release, 0-~~8~~ 34 hr, lbs	~~4,000~~ 17,018
b.	Intact ~~ected~~ steam generators	
1)	Steam release, 0-2 hr, lbs	~~404,452~~ 419,340
2)	Steam release, 2-~~8~~ 12 hr, lbs	~~945,973~~ 1,310,269
~~3)~~	~~Reactor coolant release, 0-8 hr, lbs~~	~~0~~

Insert A

g. Insert C

i. Insert B

Insert A

e. Reactor coolant alkali metal activity:

1) Case 1 Based on 1-percent failed fuel as provided in Table 11.1-5

2) Case 2 Based on 1-percent failed fuel as provided in Table 11.1-5

Insert B

Alkali metal partition factors

1)	Faulted steam generator	1.0
2)	Intact steam generators (based on full power moisture carryover of 0.25%)	0.0025

Insert C

Secondary system initial alkali metal activity	Based on the same ratio of primary to secondary system activity as iodine (i.e., 10% of primary side activity)

TABLE 15.1-3 (Sheet 2)

~~c. Activity released to the environment~~

~~1) Case 1~~

~~Isotope~~	~~0-2 hr (Ci)~~	~~0-8 hr (Ci)~~
~~I-131~~	~~2.45E+1~~	~~3.00E+2~~
~~I-132~~	~~5.81E+1~~	~~2.99E+2~~
~~I-133~~	~~5.01E+1~~	~~5.69E+2~~
~~I-134~~	~~1.66E+1~~	~~3.32E+1~~
~~I-135~~	~~3.73E+1~~	~~3.48E+2~~
~~Xe-131m~~	~~1.54~~	~~6.11~~
~~Xe-133m~~	~~2.40~~	~~9.23~~
~~Xe-133~~	~~1.31E+2~~	~~5.13E+2~~
~~Xe-135m~~	~~5.13E-2~~	~~5.15E-2~~
~~Xe-135~~	~~4.13~~	~~1.33E+1~~
~~Xe-137~~	~~4.68E-3~~	~~4.68E-3~~
~~Xe-138~~	~~6.29E-2~~	~~6.31E-2~~
~~Kr-83m~~	~~1.76E-1~~	~~3.14E-1~~
~~Kr-85m~~	~~8.81E-1~~	~~2.34~~
~~Kr-85~~	~~4.27~~	~~1.70E+1~~
~~Kr-87~~	~~4.06E-1~~	~~6.03E-1~~
~~Kr-88~~	~~1.53~~	~~3.39~~
~~Kr-89~~	~~2.09E-3~~	~~2.10E-3~~

~~2) Case 2~~

~~Isotope~~	~~0-2 hr (Ci)~~	~~0-8 hr (Ci)~~
~~I-131~~	~~2.54E+1~~	~~8.41E+1~~
~~I-132~~	~~2.30E+1~~	~~4.01E+1~~
~~I-133~~	~~4.37E+1~~	~~1.33E+2~~
~~I-134~~	~~4.17~~	~~4.86~~
~~I-135~~	~~2.43E+1~~	~~6.21E+1~~
~~Xe-131m~~	~~1.54~~	~~6.11~~
~~Xe-133m~~	~~2.40~~	~~9.23~~
~~Xe-133~~	~~1.31E+2~~	~~5.13E+2~~
~~Xe-135m~~	~~5.13E-2~~	~~5.15E-2~~
~~Xe-135~~	~~4.13~~	~~1.33E+1~~
~~Xe-137~~	~~4.68E-3~~	~~4.68E-3~~
~~Xe-138~~	~~6.29E-2~~	~~6.31E-2~~
~~Kr-83m~~	~~1.76E-1~~	~~3.14E-1~~
~~Kr-85m~~	~~8.81E-1~~	~~2.34~~
~~Kr-85~~	~~4.27~~	~~1.70E+1~~
~~Kr-87~~	~~4.06E-1~~	~~6.03E-1~~
~~Kr-88~~	~~1.53~~	~~3.39~~
~~Kr-89~~	~~2.09E-3~~	~~2.10E-3~~

TABLE 15.1-4

RADIOLOGICAL CONSEQUENCES OF A
MAIN STEAM LINE BREAK

	TEDE Dose (rem)
CASE 1, 1.0 μCi/gm I-131 equivalent w/I spike	
Exclusion area boundary (~~0-2~~ hr)	6.4E-1
~~Thyroid~~	~~2.76~~
7.8-9.8 ~~Whole body~~	~~9.98E-3~~
Low-population zone outer boundary (duration)	5.9E-1
~~Thyroid~~	~~4.33~~
~~Whole body~~	~~8.78E-3~~
Control Room (30 days)	1.1E+0
CASE 2, 60 μCi/gm I-131 equivalent	
Exclusion area boundary (0-2 hr)	2.2E-1
~~Thyroid~~	~~2.67~~
~~Whole body~~	~~5.34E-3~~
Low population zone outer boundary (duration)	1.3E-1
~~Thyroid~~	~~1.15~~
~~Whole body~~	~~1.69E-3~~
Control Room (30 days)	7.5E-1

b. The reactor coolant activity assumed for noble gas is the Technical Specification limit of 500 microCi/gm Xe-133 dose equivalent

c. The reactor coolant system activity assumed for alkali metals is based on 1% fuel defects, as provided in Table 11.1-5.

15.2.6.3 Radiological Conse

15.2.6.3.1 Method of Analys

15.2.6.3.1.1 Physical Model

The dose calculation for loss of ac power ~~is based on the sequence of events described in Table 15.2-1.~~ ~~It is~~ assumed that heat removal from the nuclear steam supply system is achieved by venting the steam for ~~8~~ 12 hours.

The reactor coolant is assumed to be contaminated by radioactive fission products introduced through fuel cladding defects. The secondary system is contaminated by the inleakage of reactor coolant through postulated steam generator tube leaks.

The radioactivity in the vented steam is dispersed in the atmosphere without any reduction due to plateout, fallout, filtering, etc.

15.2.6.3.1.2 Assumptions and Conditions

The major assumptions and parameters assumed in the analysis are found in Tables 15.2-2 and 15.A-1. The assumptions used to determine the activity released are as follows:

~~a. The reactor coolant activity assumed is the Technical Specification limit of 1.0 μCi/gm I-131 dose equivalent.~~

~~b.~~ d. The initial steam generator ~~activity~~ iodine assumed is the Technical Specification limit of 0.1 μCi/gm I-131 dose equivalent.

~~c.~~ f. A 1-gpm steam generator primary-to-secondary leakage is assumed for the duration of steam venting.

~~d.~~ g. For noble gases, the activity released is taken to be the activity introduced by reactor coolant inleakage without holdup in the steam system.

~~e.~~ h. The iodine activity present in the primary-to-secondary leakage is assumed to be homogeneously mixed with the iodine activity initially present in the steam generators. The iodine partition factor provided in Table 15.2-2 is utilized to determine the iodine activity released via steam venting from the steam generators.

e. The initial steam generator alkali metal activity is based on the same ratio of primary to secondary system activity as iodine (i.e., 10% of primary side activity)

j. ~~f.~~ The atmospheric dispersion factors are given in Table 15A-2.

a. An initial reactor coolant iodine activity equal to the dose equivalent of 1.0 microCi/gm of I-131 with an iodine spike that increases the escape rate from the fuel into the coolant by a factor of 500 immediately after the accident is assumed. This increased escape rate is assumed for 8 hours.

i. The alkali metal activity present in the primary-to-secondary leakage is assumed to be homogeneously mixed with the alkali metal activity initially present in the steam generators. The full power moisture carryover provided in Table 15.2-2 is utilized to determine the alkali metal activity released via steam venting from the steam generators.

15.2.6.3.1.3 Mathematical Models Used in the Analysis

Mathematical models used in the analysis are described in the following sections:

a. The mathematical models used to analyze the activity released during the course of the accident are described in Appendix 15A.

b. The atmospheric dispersion factors used in the analysis were calculated using the onsite meteorological measurement programs described in Section 2.3.

c. ~~The thyroid inhalation and total-body immersion~~ TEDE doses to a receptor at the exclusion area boundary, ~~or~~ outer boundary of the low population zone, and in the control room were analyzed using the models described in Appendix 15A.

15.2.6.3.1.4 Identification of Leakage Pathways and Resultant Leakage Activities

Normal activity paths from the secondary system, such as the condenser air removal system and steam generator blowdown, cease during station blackout. The steam is released to the atmosphere through the:

a. Atmospheric relief valves

b. Main steam safety valves

Since all these paths are taken as direct to the atmosphere without any form of decontamination, they are all radiologically equivalent and need not be distinguished.

15.2.6.3.2 Identification of Uncertainties in, and Conservative Aspects of, the Analysis

The principal uncertainties in the dose calculation arise from the uncertainties in the accident circumstances, particularly the extent of steam contamination, the weather at the time, and delay before preferred ac power is restored. Each of these uncertainties is handled by making very conservative or worst-case assumptions.

a. Reactor coolant activities are based on the Technical Specification limit, which ~~is~~ are significantly higher than the activities associated with normal operating conditions~~, based on 0.12-percent failed fuel~~.

iodine and noble gas

s

b. Reactor coolant activities based on extreme iodine spiking effects are conservatively high.

c. ~~b.~~ A 1-gpm steam generator primary-to-secondary leakage is assumed, which is significantly greater than that anticipated during normal operation.

d. ~~c.~~ The meteorological conditions which may be present at the site during the course of the accident are uncertain. However, it is highly unlikely that the assumed meteorological conditions would be present during the course of the accident for any extended period of time. Therefore, the evaluated radiological consequences, based on the meteorological conditions assumed, will be conservative.

15.2.6.3.3 Conclusions

15.2.6.3.3.1 Filter Loadings

No filter serves to limit the release of radioactivity in this accident. There is no significant activity buildup on any filters as a consequence of loss of ac power.

15.2.6.3.3.2 Doses to Receptor at Exclusion Area Boundary ~~and~~ , Low Population Zone Outer Boundary

The maximum doses to an individual who spends the ~~first 2 hours after los~~ duration of the accident (12 hours or longer) ~~ac power~~ at the exclusion area boundary~~, and the maximum doses for a long-term exposure (8 hours or longer)~~ ~~at~~ or the outer boundary of the low-population zone~~,~~ and in the control room for 30 days are given in Table 15.2-3. These doses are within ~~a small fraction of~~ the guideline values of 10 CFR ~~100~~ 20 for offsite locations and the 10 CFR 50.67 guideline value in the control room.

15.2.6.4 Conclusions

Results of the analysis show that, for the loss of non-emergency ac power to plant auxiliaries event, all safety criteria are met. Since the DNBR remains above the design limit, the core is not adversely affected.

Analysis of the natural circulation capability of the RCS demonstrates that sufficient long term heat removal capability exists following reactor coolant pump coastdown to prevent fuel or clad damage.

WOLF CREEK

TABLE 15.2-2

PARAMETERS USED IN EVALUATING RADIOLOGICAL CONSEQUENCES OF LOSS OF NONEMERGENCY AC POWER

I.	Source Data		
a. Core power level, MWt	~~a.~~ b.	Steam generator ~~type~~ tube leakage, gpm	1
	~~b.~~ c.	Reactor coolant initial iodine activity	1.0 µCi/gm of I-131 dose equivalent (with an assumed iodine spike that increases the rate of iodine release into the reactor coolant by a factor of 500. The increased rate is assumed for 8 hours.)
	d. ~~c.~~	Secondary system initial iodine activity	0.1 µCi/gm of I-131 dose equivalent
	e. ~~d.~~	Reactor coolant initial noble gas activity	500 microCi/gm of Xe-133 dose equivalent
	~~e.~~ g.	Iodine partition factor ~~in~~ for the steam generator s	0.01
	~~f.~~	~~Each steam generator water mass, lb~~	~~9.55E+4~~
II.	Atmospheric Dispersion Factors		See Table 15A-2
III.	Activity Release Data		
	a.	Total primary to secondary leakage 0-~~8~~ 12 hr, lb	~~4000~~ 6006
	b.	Steam release from all steam generators 0-2 hours, lb	~~5.49E+5~~ 419,846
		2-~~8~~ 12 hours, lb	~~1.03E+6~~ 1,352,918
	~~c.~~	~~Activity released to the environment~~	

move to next line

~~Isotope~~	~~0-2 hr (Ci)~~	~~0-8 hr (Ci)~~
~~I-131~~	~~1.83E-1~~	~~5.31E-1~~
~~I-132~~	~~1.54E-1~~	~~2.55E-1~~
~~I-133~~	~~3.12E-1~~	~~8.42E-1~~
~~I-134~~	~~2.47E-2~~	~~2.87E-2~~
~~I-135~~	~~1.71E-1~~	~~3.95E-1~~
~~Xe-131m~~	~~1.54~~	~~6.13~~
~~Xe-133m~~	~~2.41~~	~~9.26~~
~~Xe-133~~	~~1.31E+2~~	~~5.15E+2~~
~~Xe-135m~~	~~5.13E-2~~	~~5.15E-2~~
~~Xe-135~~	~~4.14~~	~~1.34E+1~~
~~Xe-137~~	~~4.68E-3~~	~~4.68E-3~~
~~Xe-138~~	~~6.30E-2~~	~~6.31E-2~~
~~Kr-83m~~	~~1.76E-1~~	~~3.15E-1~~
~~Kr-85m~~	~~8.82E-1~~	~~2.35~~
~~Kr-85~~	~~4.27~~	~~1.71E+1~~
~~Kr-87~~	~~4.06E-1~~	~~6.04E-1~~
~~Kr-88~~	~~1.53~~	~~3.40~~
~~Kr-89~~	~~2.09E-3~~	~~2.09E-3~~

WOLF CREEK

TABLE 15.2-3

RADIOLOGICAL CONSEQUENCES OF LOSS OF NON-EMERGENCY AC POWER

	Wolf Creek Dose (rem) TEDE
Exclusion area boundary ~~(0-2 hr)~~ (duration)	3.3E-2
~~Thyroid, rem~~	~~1.92E-2~~
~~Whole body, rem~~	~~3.84E-4~~
Low-population zone, outer boundary (duration)	3.6E-3
~~Thyroid, rem~~	~~7.24E-3~~
~~Whole body, rem~~	~~1.60E-4~~
Control Room (30 days)	1.1E-1

Also, the peak clad surface temperature is considerably less than 2,700 °F. It should be noted that the clad temperature was conservatively calculated, assuming that DNB occurs at the initiation of the transient.

The calculated sequence of events for the two cases analyzed is shown on Table 15.3-1. Figures 15.3-14 and 15.3-15 shows that the core flow rapidly reaches a new equilibrium value. With the reactor tripped, a stable plant condition will eventually be attained. Normal plant shutdown may then proceed.

15.3.3.3 Radiological Consequences

residual heat removal system can match decay heat and releases from the secondary system are terminated.

15.3.3.3.1 Method of Analysis

15.3.3.3.1.1 Physical Model

The instantaneous seizure of a reactor coolant pump rotor results in a reactor trip on a low coolant flow signal. With the coincident loss of offsite power, the condensers are not available, so the excess heat is removed from the secondary system by a steam dump through the steam generator safety and atmospheric relief valves. Steam generator tube leakage is assumed to continue until the ~~pressures in the reactor coolant and secondary systems are equalized.~~ The reactor coolant will contain the gap activities of the fraction of the fuel which undergoes DNB in addition to its assumed equilibrium activity.

15.3.3.3.1.2 Assumptions and Conditions

The major assumptions and parameters used in the analysis are itemized in Tables 15.3-3 and 15A-1 and summarized below.

Tables 15B-1 and 15B-6 provide a comparison of the analysis to the guidelines of Regulatory Guide 1.183.

The assumption used to determine the initial concentrations of isotopes in the reactor coolant and secondary coolant prior to the accident are as follows:

a. The reactor coolant iodine activity is based on the dose equivalent of 1.0 µCi/gm of I-131.

b. The noble gas activity in the reactor coolant is based on ~~1-percent failed fuel~~.

the dose equivalent of 500 microCi/gm of Xe-133.

c. The reactor coolant alkali metal activity is based on 1-percent failed fuel as provided in Table 11.1-5.

e. The secondary coolant alkali metal activity is based on the same ratio of primary to secondary system activity as iodine (i.e., 10% of primary side activity).

d. ~~e.~~ The secondary coolant ~~a~~iodine activity is based on the dose equivalent of 0.1 µCi/gm of I-131.

The following conditions are used to calculate the activity released.

a. 5 percent of fuel rod gap activity is additionally released to the reactor coolant, i.e., 5 percent of the rods experienced DNB

b. Offsite power is lost.

c. Following the incident, secondary steam is released to the environment for heat removal. The total quantity of steam released is given in Table 15.3-3.

d. Primary-to-secondary lea~~k~~12 continues after the accident for a period of ~~8~~ hours. At that time, ~~reactor coolant and secondary system pressures are equalized. Until the pressure equalizes,~~ the residual heat removal system has matched decay heat removal and releases from the secondary system are terminated. The leakage rate is assumed to be constant and equal to ~~the rate existing prior to the incident of~~ 1 gpm (500 lbs/hr).

e. Fission products released from the fuel-cladding gap of the damaged fuel rods are assumed to be instantaneously and homogeneously mixed with the reactor coolant.

f. The noble gas activity released is equal to the amount present in the reactor coolant which leaks into the secondary system after the accident.

g. The iodine activity present in the primary to secondary leakage is assumed to mix homogeneously with the iodine activity initially present in the steam generators.

h. A partition factor of 0.01 between the vapor and liquid phases for radioiodine in the steam generators is utilized to determine iodine releases to the environment via steam venting from the steam generators.

j. ~~i.~~ The activity released from the steam generators is immediately vented to the environment.

k. ~~j.~~ No credit is taken for radioactive decay or ground deposition during radioactivity transport to offsite location~~.~~s or the control room.

i. A full power moisture carryover of 0.25% is utilized to determine alkali metal releases to the environment via steam venting from the steam generators.

, 15A-1, and 15A-4, respectively

l. ~~k.~~ Short-term accident atmospheric dispersion factors corresponding to ground level releases, breathing rates, and dose conversion factors are given in Table 15A-2.

15.3.3.3.1.3 Mathematical Models Used in the Analysis s

Mathematical models used in the analysis are described in the following sections:

a. The mathematical models used to analyze the activity released during the course of the accident are described in Appendix 15A.

b. The atmospheric dispersion factors used in the analysis were calculated based on the onsite meteorological measurement programs described in Section 2.3 and are provided in Table 15A-2.

c. The ~~thyroid~~ TEDE ~~ation and total-body immersion~~ doses to a receptor at the exclusion area boundary, ~~or~~ outer boundary of the low-population zone ~~were~~ analyzed using the models described in Appendix 15A.

, and in the control room

15.3.3.3.1.4 Identification of Leakage Pathways and Resultant Leakage Activity

The leakage pathways are:

a. Direct steam dump to the atmosphere through the secondary system atmospheric relief and safety valves for the secondary steam

b. Primary-to-secondary steam generator tube leakage and subsequent steam dump to the atmosphere through the secondary system atmospheric relief and safety valves

~~Table 15.3-3 shows the total curies released.~~

15.3.3.3.2 Identification of Uncertainties and Conservative Elements in the Analysis

~~a. Reactor coolant and secondary coolant activities of 1.0 mCi/gm and 0.1 mCi/gm I-131 dose equivalent, respectively, are many times greater than assumed for normal operation conditions.~~

a. ~~b.~~ A 1-gpm steam generator primary-to-secondary leakage, which is significantly greater than that anticipated during normal operation, is assumed.

b. ~~c.~~ The coincident loss of offsite power with the occurrence of a reactor coolant pump locked rotor is a highly conservative assumption. In the event of the availability of offsite station power, the condenser steam dump valves will open, permitting steam dump to the condenser. Thus there is no direct release to the environment.

c. ~~d.~~ The meteorological conditions which may be present at the site during the course of the accident are uncertain. However, it is highly unlikely that the meteorological conditions assumed will be present during the course of the accident for any extended period of time. Therefore, the radiological consequences evaluated, based on the meteorological conditions assumed, are conservative.

15.3.3.3.3 Conclusions

15.3.3.3.3.1 Filter Loadings

The only ESF filtration system considered in the analysis which limits the consequences of the reactor coolant pump locked rotor accident is the control room filtration system. Activity loadings on the control room charcoal filter are based on the flow rate through the filter, the concentration of activity at the filter inlet, and the filter efficiency.

The activity in the control room filter as a function of time has been evaluated for the loss-of-coolant accident, Section 15.6.5. Since the control room filters are capable of accommodating the potential design-basis loss-of-coolant accident fission product iodine loadings, more than adequate design margin is available with respect to postulated reactor coolant pump locked rotor accident releases.

15.3.3.3.3.2 Doses to Receptor at the Exclusion Area Boundary, ~~and~~ Low-Population Zone Outer Boundary, and in the Control Room

The potential radiological consequences resulting from the occurrence of a postulated reactor coolant pump locked rotor have been conservatively analyzed, using assumptions and models described in previous sections.

The ~~total-body doses due to immersion from direct radiation and the thyroid dose due to inhalation~~ TEDE doses have been analyzed for the 0-2 hour dose at the exclusion area boundary, ~~and~~ for the duration of the accident (0 to ~~8~~ 12 hours) at the low-population zone outer boundary, and in the control room for 30 days. The results are listed in Table 15.3-4. The resultant doses are ~~well~~ within a small fraction of the guideline values of 10 CFR ~~100~~ 50.67 for offsite locations and the full 10 CFR 50.67 guideline value in the control room.

WOLF CREEK

TABLE 15.3-3

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

I.	Source Data	
	a. Power level, MWt	3637 (includes uncertainty) ~~3,565~~
	b. Steam generator tube leakage, gpm	1
	~~c. Reactor coolant iodine activity~~	~~Dose equivalent of 1.0 µCi/gm of I-131~~
	~~d. Reactor coolant noble gas activity~~	~~Based on 1-percent failed fuel, as provided in Table 11.1-5~~
	~~e. Secondary system activity~~	~~Dose equivalent of 0.1 µCi/gm of I-131~~
	~~f. Activity released to reactor coolant from failed fuel~~	
Insert D	~~1. Noble gas, percent of gap inventory~~	~~5~~
	~~2. Iodine, percent of gap inventory~~	~~5~~
	~~3. Gap inventory~~	~~Table 15A-3~~
	g. Iodine partition factor for steam generators	0.01
h. Insert E	~~h.~~ i. Reactor coolant mass, lbs	~~4.94E+5~~ 3.99E+5
j. Insert M	~~i. Steam generator mass, per generator, lbs~~	~~9.55E+4~~
II.	Atmospheric Dispersion Factors	see Table 15A-2
III.	Activity Release Data	
	~~a.~~ a. Total primary to secondary leakage 0-12 hr, lb	~~ge, 0-2 hrs coolant leakage,~~ ~~1,000~~ 6006
	~~2. Mass released from steam generators, lbs~~	~~5.49E+5~~
	~~b.~~ b. Steam release from all steam generators 0-2 hr, lb 2-12 hr, lb	~~e, 2-8 hrs coolant leakage,~~ ~~3,000~~ 419,846
		~~ased from steam s, lbs~~ ~~1.03E+6~~ 1,352,918

Insert D

c. Core Inventories	see Table 15A-3
d. Radial peaking factor	1.65
e. Extent of core damage	5 percent of the fuel rods experience cladding failure
f. Activity released to reactor coolant from fuel gap of failed fuel, percent of core inventory	
1. Kr-85	10
2. I-131	8
3. Other noble gases	5
4. Other iodines	5
5. Alkali metals	12

Insert E

Alkali metal partition factor for steam generators (based on full power moisture carryover of 0.25%)	0.0025

Insert M

Each steam generator water mass, lb	
0-2 hrs	82,500
>2 hrs (assumes water level reaches 0% narrow range span)	121,250

TABLE 15.3-3 (Sheet 2)

~~c. Activity released to the environment~~

~~Isotope~~	~~0-2 hr (Ci)~~	~~0-8 hr (Ci)~~
~~I-131~~	~~8.37~~	~~8.36E+1~~
~~I-132~~	~~6.86~~	~~2.63E+1~~
~~I-133~~	~~1.38E+1~~	~~1.23E+2~~
~~I-134~~	~~5.87~~	~~9.88~~
~~I-135~~	~~1.17E+1~~	~~8.15E+1~~
~~Xe-131m~~	~~1.17E+1~~	~~4.66E+1~~
~~Xe-133m~~	~~6.30E+1~~	~~2.42E+2~~
~~Xe-133~~	~~2.09E+3~~	~~8.24E+3~~
~~Xe-135m~~	~~6.99E+1~~	~~7.02E+1~~
~~Xe-135~~	~~4.45E+2~~	~~1.44E+3~~
~~Xe-137~~	~~8.02E+1~~	~~8.02E+1~~
~~Xe-138~~	~~2.81E+2~~	~~2.82E+2~~
~~Kr-83m~~	~~8.82E+1~~	~~1.58E+2~~
~~Kr-85m~~	~~2.33E+2~~	~~6.23E+2~~
~~Kr-85~~	~~3.52E+1~~	~~1.41E+2~~
~~Kr-87~~	~~3.19E+2~~	~~4.74E+2~~
~~Kr-88~~	~~5.85E+2~~	~~1.30E+3~~
~~Kr-89~~	~~3.43E+1~~	~~3.43E+1~~

TABLE 15.3-4

RADIOLOGICAL CONSEQUENCES OF A
LOCKED ROTOR ACCIDENT

	Wolf Creek Dose* (rem) TEDE
Exclusion Area Boundary (0-2 hr)	3.8E-1
~~Thyroid, rem~~	~~0.882~~
~~Whole body, rem~~	~~0.076~~
Low Population ZoneOuter Boundary (duration) **Insert Space**	3.0E-1
~~Thyroid, rem~~	~~1.130~~
~~Whole body, rem~~	~~0.021~~
Control Room (30 days)	4.7E+0

*Note that the doses reported here reflect the rod ejection doses from secondary releases shown in Table 15.4-4, as it was determined this release pathway is bounding.

severity of the present analysis does not exceed the "worst case" analysis, the accident for this plant will not result in an excessive pressure rise or further damage to the RCS.

Lattice Deformations

A large temperature gradient will exist in the region of the hot spot. Since the fuel rods are free to move in the vertical direction, differential expansion between separate rods cannot produce distortion. However, the temperature gradients across individual rods may produce a differential expansion tending to bow the midpoint of the rods toward the hotter side of the rod. Calculations have indicated that this bowing would result in a negative reactivity effect at the hot spot since Westinghouse cores are under-moderated, and bowing will tend to increase the under-moderation at the hot spot. In practice, no significant bowing is anticipated, since the structural rigidity of the core is more than sufficient to withstand the forces produced. Boiling in the hot spot region would produce a net flow of coolant away from that region. However, the heat from the fuel is released to the water relatively slowly, and it is considered inconceivable that cross flow will be sufficient to produce significant lattice forces. Even if massive and rapid boiling, sufficient to distort the lattice, is hypothetically postulated, the large void fraction in the hot spot region would produce a reduction in the total core moderator to fuel ratio, and a large reduction in this ratio at the hot spot. The net effect would therefore be a negative feedback. It can be concluded that no conceivable mechanism exists for a net positive feedback resulting from lattice deformation. In fact, a small negative feedback may result. The effect is conservatively ignored in the analysis.

15.4.8.3 Radiological Consequences

15.4.8.3.1 Method of Analysis

15.4.8.3.1.1 Physical Model

Prior to the accident, it is assumed that the plant has been operating with simultaneous fuel defects and steam generator tube leakage for a time sufficient to establish equilibrium levels of activity in the reactor coolant and secondary systems.

The RCCA ejection results in reactivity being inserted to the core which causes the local power to rise. In a conservative analysis, it is assumed that partial cladding failure and fuel melting occurs. The fuel pellet and gap activities are assumed to be immediately and uniformly released within the reactor coolant or containment, depending on which release path is being considered.

Two release paths to the environment exist which are analyzed separately and conservatively, as if all the activity is available for release from each path.

The activity released to the containment from the reactor coolant through the ruptured control rod mechanism pressure housing is assumed to be mixed instantaneously throughout the containment and is available for leakage to the atmosphere. The only removal processes considered in the containment are ~~iodine plateout,~~ radioactive decay~~,~~ and leakage from the containment.

The model for the activity available for release to the atmosphere from the relief valves assumes that the release consists of the ~~activity in the secondary system plus that~~ fraction of the activity leaking from the reactor coolant through the steam generator tubes. The leakage of reactor coolant to the secondary side of the steam generator continues until ~~the pressures in the reactor coolant and secondary systems equalize.~~

~~Primary and secondary pressures are equalized at 1100 seconds following the accident, thus terminating primary to secondary leakage in the steam generators. Refer to Figures 15.4-26 and 15.4-27.~~

~~Thereafter, no mass transfer from the reactor coolant system to the secondary system due to the steam generator tube leakage is assumed. Thus, in the case of coincident loss of offsite power, activity is released to the atmosphere from a steam dump through the relief valves.~~

15.4.8.3.1.2 Assumptions and Conditions

The major assumptions and parameters used in the analysis are itemized in Tables 15.4-3 and 15A-1 and summarized below. The assumptions are consistent with Regulatory Guide ~~1.77~~. 1.183

~~The assumption used to determine the initial concentrations of isotopes in the reactor coolant and secondary coolant prior to the accident are as follows:~~

~~a. The reactor coolant iodine activity is based on the dose equivalent of 1.0 μCi/gm of I-131.~~

~~b. The noble gas and iodine activity in the reactor coolant are based on 1-percent failed fuel.~~

~~c. The secondary coolant activity is based on the dose equivalent of 0.1 μCi/gm of I-131.~~

Tables 15B-1 and 15B-7 provide a comparison of the analysis to the guidelines of Regulatory Guide 1.183.

the residual heat removal system can match decay heat and steam releases from the steam generators are terminated. After this time, no more releases to the environment occur.

The following conditions are used to calculate the activity released and the offsite doses following a RCCA ejection accident.

a. 10 percent of the fuel rod gap activity~~, except for Kr-85 and I-131, which are 30 percent and 12 percent respectively,~~ is ~~additionally~~ released to the reactor coolant.

for iodine and noble gas and 12 percent of the fuel rod gap activity for alkali metal

b. 0.25 percent of the fuel is assumed to melt.

c. Following the incident until ~~primary and secondary side pressures equalize~~, secondary steam is released to the environment. The total quantity of steam released is given in Table 15.4.3.

RHR operation to take over decay heat removal at 12 hours

d. The 1-gpm primary-to-secondary ~~leak~~ leakage is to the unaffected steam generators.

e. All noble gas activity in the reactor coolant which is transported to the secondary system via the primary-to-secondary leakage is assumed to be immediately released to the environment.

f. Fission products released from the fuel-cladding gap of the damaged fuel rods are assumed to be instantaneously and homogeneously mixed with the reactor coolant water.

for the primary to secondary release pathway case and instantaneously and homogeneously mixed within containment for the containment leakage pathway case.

g. The iodine and alkali metal activity present in the primary to secondary leakage is assumed to mix homogeneously with the ~~iodine activity initially present~~ in the steam generators.

h. A partition factor of ~~0.1~~ 0.01 between the vapor and liquid phases for radioiodine in the steam generators is utilized to determine iodine releases to the environment via steam venting from the steam generators.

j. ~~i.~~ The activity released from the steam generators is immediately vented to the environment.

k. ~~j.~~ The containment is assumed to leak at 0.2 volume percent/day during the first 24 hours immediately following the accident and 0.1 volume percent/day thereafter.

l. ~~k.~~ No credit is taken for radioactive decay or ground deposition during radioactivity transport to offsite or control room location.

s

i. The full power moisture carryover of 0.25% is utilized to determine the alkali metal releases to the environment via steam venting from the steam generators.

m. ~~l.~~ Short-term accident atmospheric dispersion factors corresponding to ground level releases, breathing rates, and dose conversion factors are given in Table 15A-2, 15A-1, and 15A-4, respectively.

s

n. ~~m.~~ Offsite power is assumed lost.

15.4.8.3.1.3 Mathematical Models Used in the Analysis

Mathematical models used in the analysis are described in the following sections:

a. The mathematical models used to analyze the activity released during the course of the accident are described in Appendix 15A.

b. The atmospheric dispersion factors used in the analysis were calculated based on the onsite meteorological measurement programs described in Section 2.3 and are provided in Table 15A-2.

c. The ~~thyroid inhalation and total-body immersion~~ TEDE doses to a receptor at the exclusion area boundary, ~~or~~ outer boundary of the low-population zone, and in the control room were analyzed, using the models described in Appendix 15A.

15.4.8.3.1.4 Identification of Leakage Pathways and Resultant Leakage Activity

The leakage pathways are:

a. Direct steam dump to the atmosphere through the secondary system relief valves for the secondary steam

b. Primary-to-secondary steam generator tube leakage and subsequent steam dump to the atmosphere through the secondary system relief valves

c. The resultant activity released to the containment is assumed available for leakage directly to the environment.

~~Table 15.4-3 shows the total curies released~~

15.4.8.3.2 Identification of Uncertainties and Conservative Elements in the Analysis

~~a. Reactor coolant and secondary coolant activities of 1 percent failed fuel and 0.1 μCi/gm I-131 dose equivalent, respectively, are many times greater than assumed for normal operation conditions.~~

a. ~~b.~~ A 1-gpm steam generator primary-to-secondary leakage, which is significantly greater than that anticipated during normal operation, is assumed.

b. ~~c.~~ The coincident loss of offsite power with the occurrence of a RCCA ejection accident is a highly conservative assumption. In the event of the availability of offsite station power, the condenser steam dump valves will open, permitting steam dump to the condenser. Thus there is no direct release via that path to the environment.

c. ~~d.~~ It is assumed that ~~50 percent~~ none of the iodines released to the containment atmosphere is adsorbed (i.e. plate out) onto the internal surfaces of the containment or adheres to internal components. However, it is ~~estimated~~ recognized that the removal of airborne iodines by various physical phenomena such as adsorption, adherence, and settling could reduce the resultant doses ~~by a factor of 3 to 10~~.

d. ~~e.~~ The activity released to the containment atmosphere is assumed to leak to the environment at the containment leakage rate of 0.2-volume percent/day for the first 24 hours and 0.1-volume percent/day thereafter. The initial containment leakage rate is based on the peak calculated internal containment pressure anticipated after a LOCA. The pressures associated with a RCCA ejection accident are considerably lower than that calculated for a LOCA. The pressure inside the containment also decreases considerably with time, with an expected decrease in leakage rates. Taking into account that the containment leak rate is a function of pressure, the resultant doses could be reduced by a factor of 5 to 10 (Ref. 10).

e. ~~f.~~ The meteorological conditions which may be present at the site during the course of the accident are uncertain. However, it is highly unlikely that the meteorological conditions assumed will be present during the course of the accident for any extended period of time. Therefore, the radiological consequences evaluated, based on the meteorological conditions assumed, are conservative.

15.4.8.3.3 Conclusions

15.4.8.3.3.1 Filter Loadings

The only ESF filtration system considered in the analysis which limits the consequences of the RCCA ejection accident is the control room filtration system. Activity loadings on the control

room charcoal filter are based on the flow rate through the filter, the concentration of activity at the filter inlet, and the filter efficiency.

The activity in the control room filter as a function of time has been evaluated for the loss-of-coolant accident, Section 15.6.5. Since the control room filters are capable of accommodating the potential design-basis loss-of-coolant accident fission product iodine loadings, more than adequate design margin is available with respect to postulated RCCA ejection accident releases.

15.4.8.3.3.2 Doses to Receptor at the Exclusion Area Boundary, ~~and~~ Low-Population Zone Outer Boundary, and in the Control Room

The potential radiological consequences resulting from the occurrence of a postulated RCCA ejection accident have been conservatively analyzed, using assumptions and models described in previous sections.

The ~~total-body doses due to immersion from direct radiation and the thyroid dose due to inhalation have~~ TEDE doses have been analyzed for the 0-2 hour dose at the exclusion area boundary, ~~and~~ for the duration of the accident at the low-population zone outer boundary, and for 30 days in the control room. The results are listed in Table 15.4-4. The resultant doses are well within the guideline values of 10 CFR ~~100~~ 50.67 for offsite locations and the full 10 CFR 50.67 guideline value in the control room.

15.4.8.4 Conclusions

Even on a conservative basis, the analyses indicate that clad limits are not exceeded. It is concluded that there sudden fuel dispersal into the coolant. Since the peak p exceed that which would cause stresses to exceed the faulted condition stress limits, it is concluded that there is no danger of further consequential damage to the RCS. The analyses have demonstrated that upper limit in fission product release as a result of a number of fuel rods entering DNB amounts to 10 percent.

~~The RCS integrated break flow to containment following a rod ejection accident is shown in Figure 15.4-28.~~

15.4.9 REFERENCES

1. Risher, D. H., Jr. and Barry, R. F., "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code," WCAP-7979-P-A(Proprietary) and WCAP-8028-A (Non-Proprietary), January, 1975.

2. Hargrove, H. G., "FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December 1989.

3. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-A, April 1984.

4. Barry, R. F. and Altomare, S., "The TURTLE 24.0 Diffusion Depletion Code," WCAP-7213-P-A (Proprietary) and WCAP-7758-A (Non-Proprietary), February 1975.

5. Barry, R. F., "LEOPARD - A Spectrum Dependent Non-Spatial Depletion Code for the IBM-7094," WCAP-3269-26, September 1963.

WOLF CREEK

TABLE 15.4-3

PARAMETERS USED IN EVALUATING THE
RCCA EJECTION ACCIDENT

I.	Source Data	
a.	Core power level, MWT	3637 (includes uncertainty) ~~3565~~
~~b.~~	~~Burnup, full power days~~	~~1000~~
b. ~~c.~~	Core inventories	Table 15A-3
c. ~~d.~~	Steam generator tube leakage, gpm	1
~~e.~~	~~Reactor coolant~~	~~Based on 1-percent failed fuel, provided in Table 11.1-5~~
~~f.~~	~~Secondary system activity~~	~~Based on 1-percent failed fuel, 4 times the values provided in Table 11.1-4~~
d.	Radial peaking factor	1.65
e. ~~g.~~	Extent of core damage	10 percent of fuel rods experience cladding failure; 0.25 percent of fuel experiences melting
f. ~~h.~~	Activity released to reactor coolant, percent of core activity	
	1. Cladding failure	
	(a) Noble gas gap activity	100
	(b) Iodine gap activity	~~100~~ 10
	(c) Alkali metal gap activity	12
	2. Fuel melting	
	(a) Noble gas ~~gap~~ fuel activity	100
	(b) Iodine fuel activity	50 100
	(c) Alkali metal fuel activity	
~~i.~~ g.	Iodine carryover factor for steam generators	0.1
h. Insert E		
i. ~~j.~~	Reactor coolant mass, lbs	~~4.94E+5~~ 3.99E+5
~~k.~~	~~Steam generator mass, lbs/steam generator~~	~~1.04E+5~~
j. Insert M		

WOLF CREEK
TABLE 15.4-3 (Sheet 2)

II. Atmospheric Dispersion Factors — Table 15A-2

III. Activity Release Data

a. Containment volume, ft^3 — 2.5E + 6

b. Containment leak rate, volume percent/day
   1. 0-24 hours — 0.20
   2. 1-30 days — 0.10

c. Percent of containment leakage that is unfiltered — 100

~~d. Plateout of iodine within containment, percent~~ — ~~50~~

e. Steam release from all steam generators
0-2 hr, lb — 419,846
2-12 hr, lb — 1,352,918

~~e.~~ d. Offsite power — Lost

~~f. Steam release from relief valves, lbs~~ — ~~48,600~~

f. ~~g.~~ Duration of release from relief valves, ~~sec~~ hr — ~~140~~ 12

~~h. Activity released to the environment via steam release and containment release~~

~~Isotope~~	~~Steam Generator Release (Ci) 0-2hr~~	~~Containment Release (Ci) 0-2hr~~	~~Containment Release (Ci) 0-30 days~~
~~I-131~~	~~5.369E+00~~	~~1.040E+02~~	~~7.275E+03~~
~~I-132~~	~~6.546E+00~~	~~9.655E+01~~	~~2.139E+02~~
~~I-133~~	~~9.393E+00~~	~~1.769E+02~~	~~2.142E+03~~
~~I-134~~	~~9.937E+00~~	~~1.012E+02~~	~~1.275E+02~~
~~I-135~~	~~8.703E+00~~	~~1.554E+02~~	~~8.184E+02~~
~~Xe-131m~~	~~5.287E-01~~	~~2.225E+00~~	~~2.028E+02~~
~~Xe-133m~~	~~3.038E+00~~	~~1.265E+01~~	~~3.124E+02~~
~~Xe-133~~	~~9.881E+01~~	~~4.146E+02~~	~~2.087E+04~~
~~Xe-135m~~	~~1.768E+01~~	~~1.469E+01~~	~~1.476E+01~~
~~Xe-135~~	~~2.325E+01~~	~~9.114E+01~~	~~5.940E+02~~
~~Xe-137~~	~~6.881E+01~~	~~1.640E+01~~	~~1.640E+01~~
~~Xe-138~~	~~7.650E+01~~	~~5.812E+01~~	~~5.829E+01~~
~~Kr-83m~~	~~6.080E+00~~	~~1.810E+01~~	~~3.400E+01~~
~~Kr-85m~~	~~1.316E+01~~	~~4.788E+01~~	~~1.774E+02~~
~~Kr-85~~	~~1.392E+00~~	~~5.874E+00~~	~~1.079E+03~~
~~Kr-87~~	~~2.521E+01~~	~~6.547E+01~~	~~9.863E+01~~
~~Kr-88~~	~~3.580E+01~~	~~1.201E+02~~	~~3.105E+02~~
~~Kr-89~~	~~3.454E+01~~	~~7.095E+00~~	~~7.095E+00~~

TABLE 15.4-4

RADIOLOGICAL CONSEQUENCES OF A
ROD-EJECTION ACCIDENT

	Wolf Creek Dose (rem) TEDE
Containment Leakage	
Exclusion Area Boundary (0-2 hr)	1.2E+0
~~Thyroid~~	~~1.17E+1~~
~~Whole body~~	~~5.85E-2~~
Low Population Zone Outer Boundary (duration)	2.0E+0
~~Thyroid~~	~~1.44E+1~~
~~Whole body~~	~~2.34E-2~~
Control Room (30 days)	1.7E+0
Secondary Releases	
Exclusion Area Boundary (0-2 hr)	3.8E-1
Low Population Zone Outer Boundary (duration)	3.0E-1
Control Room (30 days)	4.7E+0

WOLF CREEK

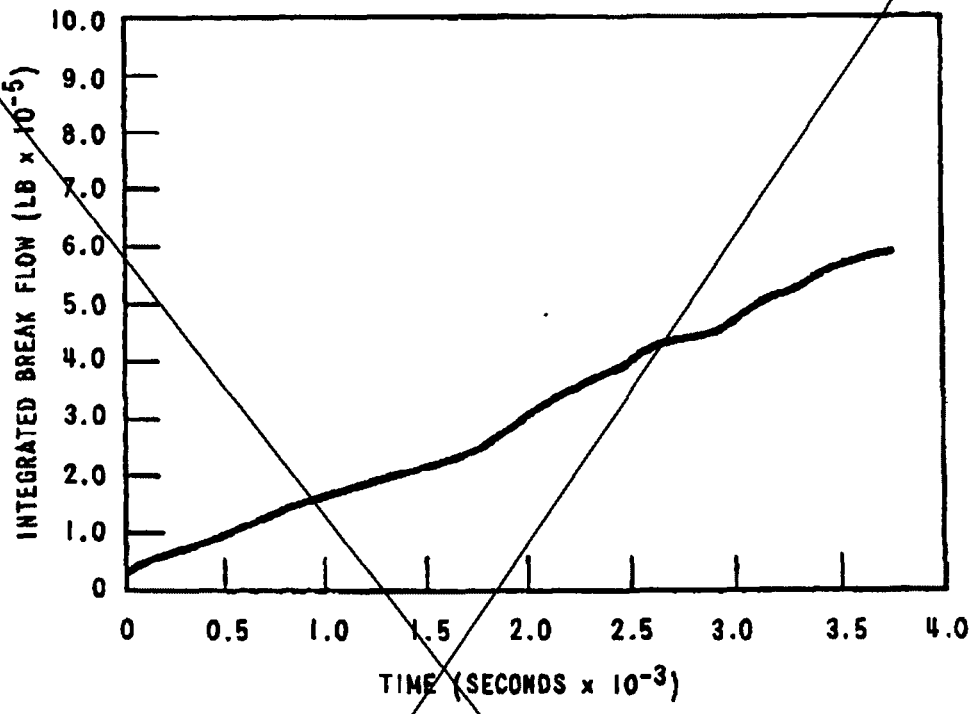

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FIGURE 15.4-28

REACTOR COOLANT SYSTEM
INTEGRATED BREAK FLOW FOLLOWING
A ROD EJECTION ACCIDENT

volume control system letdown line penetrating the containment. The grab sample lines are provided with normal closed isolation valves on both sides of the containment wall and are designed in accordance with the requirements of GDC-55.

The most severe pipe rupture with regard to radioactivity release during normal plant operation is rupture of the chemical and volume control system letdown line at a point outside of the containment. For such a break, the reactor coolant letdown flow would have passed sequentially from the cold leg and through the regenerative heat exchanger and letdown orifices. The letdown orifice reduces the letdown line pressure from 2,235 psig to less than 600 psig outside containment during normal plant operation when letdown flow is maintained at a nominal 120 gpm. Increase in flow will occur due to a rupture of the letdown line downstream of the orifices. It has been determined that the occurrence of a complete severance of the letdown line would result in a loss of reactor coolant at the rate of ~~222~~ gpm. 141

Since the reactor makeup water transfer pumps are designed to deliver 120 gpm to the boric acid blending tee, the capability of the reactor makeup system can not maintain VCT level. The imbalance of the VCT outflow and inflow would eventually result in water level dropping to VCT level Lo/Refueling Water Sequence setpoint (5%) and the suction of the charging pump would automatically be shifted from the VCT to the RWST. In addition, the calculated releases rate is beyond the capacity of a single charging pump, which is capable of delivering 150 gpm flow to the RCS under normal operating conditions, if the RCP seal leakoff and any identified leakage are accounted for. The control room operators would be alerted of the failure by a high charging flow alarm and/or continuous VCT makeup and a slowly decreasing VCT and pressurizer level. The high charging flow alarm procedure in conjunction with the plant off-normal procedure for high RCS leakage would require letdown isolation inside the containment which would terminate the coolant loss.

15.6.2.1 Radiological Consequen

15.6.2.1.1 Method of Analysis

15.6.2.1.1.1 Physical Model

a. An initial reactor coolant iodine activity equal to the dose equivalent of 1.0 microCi/gm of I-131 with an iodine spike that increases the escape rate from the fuel into the coolant by a factor of 500 immediately after the accident is assumed. This increased escape rate is assumed for 8 hours.

The volatile fractions of the spilled reactor coolant are assumed to be available for immediate release to the environment.

15.6.2.1.1.2 Assumptions and Conditions

The major assumptions and parameters used in the analysis are provided in Table 15.6-2 and summarized below:

~~a. The reactor coolant iodine activity is based on the dose equivalent of 1.0 µCi/gm of I-131.~~

b. The noble gas activity in the reactor coolant is based on ~~1-percent failed fuel.~~

500 microCi/gm of Xe-133 dose equivalent.

d. ~~c. A total of 111,600 pounds~~ of reactor coolant is spilled onto the floor of the auxiliary building~~. (Based on doubling the maximum flowrate of 222 gpm to account for backflow~~ over a thirty minute release, followed by a ten second valve closure period).

A flow rate of 141 gpm

e. ~~d.~~ All of the noble gases in the spilled reactor coolant are released to the environment.

c. The alkali metal activity in the reactor coolant is based on 1-percent failed fuel as provided in Table 11.1-5.

based on letdown line fluid conditions of 380°F and 600 psig and saturation at atmospheric pressure.

f. ~~e.~~ Eighteen ~~Ten~~ percent of the spill is assumed to flash. All of the iodine and alkali metal activity in the flashed fraction of the spill is assumed to be released.

g. ~~f.~~ No credit is taken for mixing and holdup of the releases within the auxiliary building, nor are the auxiliary building normal exhaust filters credited with reducing the release. That is, the release is modeled as being direct to the environment.

15.6.2.1.1.3 Mathematical Models Used in the Analysis

Mathematical models used in the analysis are described in the following sections:

a. The mathematical models used to analyze the activity released during the course of the accident are described in Appendix 15A.

b. The atmospheric dispersion factors used in the analysis were calculated based on the onsite meteorological measurement programs described in Section 2.3 and provided in Table 15A-2.

c. The ~~thyroid inhalation and total body immersion~~ TEDE doses to a receptor at the exclusion area boundary, ~~or~~ outer boundary of the low-population zone, and in the control room were analyzed, using the models described in Appendix 15A.

15.6.2.1.1.4 Identification of Leakage Pathways and Resultant Leakage Activity

The reactor coolant spilled in the auxiliary building will collect in the floor drain sumps. From there, it will be pumped to the radwaste treatment system. Therefore, the only release paths that present a radiological hazard involve the volatile fraction of spilled coolant.

Normally, gases released in the auxiliary building mix with the building atmosphere and are gradually exhausted through the filtered building ventilation system. The charcoal filters normally remove a very large fraction of the airborne iodine in the building atmosphere. However, the ventilation system is not designed to mitigate the consequences of an accident (e.g., it might not survive an earthquake more severe than the operating-basis earthquake), nor can the possibility of unplanned leakages from the auxiliary building be eliminated; hence, no credit is taken for these effects reducing the released activity.

The evaporated radionuclides are assumed to be available immediately to the outside atmosphere. ~~This activity is tabulated in Table 15.6-2.~~

15.6.2.1.2 Identification of Uncertainties and Conservatisms in the Analysis

The principal uncertainties in the calculation of doses following a letdown line rupture arise from the unknown extent of reactor coolant contamination by radionuclides, the quantity of coolant spilled, the fraction of the spilled activity that escapes the auxiliary building, and the environmental conditions at the time. Each of these uncertainties is treated by taking worst-case or extremely conservative assumptions.

The extent of coolant contamination assumed greatly exceeds the levels expected in practice. The rupture is postulated in a seismic Category I, ASME Section III, Class 2 piping system. It is assumed that the leak goes undetected for 30 minutes. It is expected that considerable holdup and filtration occurs in the auxiliary building, but no credit is assumed.

The purpose of all these conservatisms is to place an upper bound on doses.

15.6.2.1.3 Conclusions

15.6.2.1.3.1 Filter Loadings

and alkali metal

No filter is credited with the collection of radionuclides in this accident analysis. The buildup on these filters (auxiliary building and control building charcoal filters) that may be expected due to the adsorption of some of the iodine ~~is~~ very small compared with the design capacity of these filters.

15.6.2.1.3.2 Dose to Receptor at the Exclusion Area Boundary, ~~and~~ Low-Population Zone Outer Boundary, and in the Control Room

The radiological consequences resulting from the occurrence of a postulated letdown line rupture have been conservatively analyzed, using assumptions and models described in previous sections.

The ~~thyroid inhalation total-body immersion~~ TEDE doses have been analyzed for the 0-2 hour dose at the exclusion area boundary, ~~and~~ for the duration of the accident at the low-population zone outer boundary, and for 30 days in the control room. The results are listed in Table 15.6-3. The resultant doses are within a small fraction of the guideline values of 10 CFR ~~100~~ 50.67 for the offsite locations and the full 10 CFR 50.67 guideline value in the control room.

15.6.3 STEAM GENERATOR TUBE RUPTURE (SGTR)

The steam generator tube rupture (SGTR) examined is the complete severance of one single steam generator tube which results in the leakage of reactor coolant [illegible] generator. The consequences of SGTR [illegible]e operator to take the necessary actions [illegible] leakage. If the leakage continues for an [illegible]ry side of the steam generator may become filled and water may enter the steamline. As a result, the release of liquid through the secondary side safety/relief valves to the atmosphere may occur that could result in an increase in the radiological doses.

Two SGTR scenarios are evaluated in order to ensure that operators can respond to the accident in a timely manner so as to minimize the resulting offsite releases and prevent overfilling of the effected steam ~~line~~ generator. Those scenarios are described below:

and control room

tube rupture break flow and intact steam generator

activity initially in the primary and secondary systems

15.6.3.3 Radiological Consequences

15.6.3 3.1 Method of Analysis

, alkali metals,

The analysis of the radiological consequences of an SGTR considers the most severe release of secondary activity, as well as primary system activity leaked from the tube break. The inventory of iodine and noble gas fission product activity available for release to the environment depends on the primary to secondary coolant leakage rate, ~~the percentage of defective fuel in the core~~, and the mass of steam discharged to the environment.

1.183

The ~~re-~~analysis of the radiological consequences of a postulated SGTR uses the guidelines provided in Regulatory Guide ~~1.195~~ (Reference 7) for assumptions and methods to calculate the offsite dose consequences. ~~This involved changing three elements of the analysis of record methodology (Reference 1):~~

~~(1)~~ ~~The reanalysis uses thyroid dose conversion factors (DCFs) for inhalation of radionuclides based on the data provided in Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion". This is a departure from the more conservative DCF values listed in Regulatory Guide 1.109, which were used in the previous analysis of record,~~

~~(2)~~ ~~The effective dose conversion factors, provided in Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil", are used to calculate the whole body doses. This deviates from the previous analysis of record, which used the more conservative DCF values listed in Regulatory Guide 1.109, and~~

~~(3)~~ ~~The reanalysis uses a factor of 335 for the accident initiated iodine spike release rate, which is a departure from the more conservative factor of 500 modeled in the previous analysis of record.~~

~~These methodology changes reduce the magnitude of the accident source term and result in lower doses than would be obtained using the methodology previously presented in the SGTR with overfill analysis of record. However, the Regulatory Guide 1.195 source term methodology is recognized by the nuclear industry as having a better scientific basis and has been approved by the NRC for the design and licensing applications.~~

15.6.3.3.1.1 Physical Model

The evaluation of the radiological consequences of a postulated steam generator tube rupture (SGTR) utilizes the results of the RETRAN analyses to calculate releases of radioactive iodines and noble gases to the atmosphere to the time of RHR cut-in conditions.

, alkali metals,

ruptured

Concentrations of radioactivity in the RCS water and in the ~~faulted~~ and intact steam generators are calculated utilizing release rates from the fuel, calculated mass flows and conventionally used partitioning coefficients between the liquid and steam phases. These radioactivity concentrations and the calculated releases of mass to the atmosphere yield the released activity. Radiological consequences are calculated using atmospheric dispersion coefficients, breathing rates, and other aspects of conventional radiological consequence calculations.

15.6.3.3.1.2 Assumptions and Conditions

The major assumptions and parameters assumed in the analysis are itemized in Table 15.6-4 and 15A-1 and are summarized below.

Tables 15B-1 and 15B-5 provide a comparison of the analysis to the guidelines of Regulatory Guide 1.183.

d. The alkali metal activity in the reactor coolant is based on 1-percent failed fuel as provided in Table 11.1-5.

e. The initial secondary coolant alkali metal activity is based on the same ratio of primary to secondary system activity as iodine (i.e., 10% of primary side activity).

a. The assumed reactor coolant iodine activity is determined for the following two cases:

Case 1 - An initial reactor coolant iodine activity equal to the dose equivalent of 60.0 μCi/gm of I-131 due to a pre-accident iodine spike caused by RCS transients prior to the SGTR.

Case 2 - An initial reactor coolant iodine activity equal to the dose equivalent of 1.0 μCi/gm of I-131 with an iodine spike that increases the escape rate from the fuel into the coolant by a factor of 335 immediately after the accident. This increased escape rate is assumed for ~~the duration of the accident~~. 8 hours

b. The noble gas activity in the reactor coolant is based on ~~1-percent failed fuel as provided in Table 11.1-5.~~ the dose equivalent of 500 microCi/gm of Xe-133.

c. The initial secondary coolant iodine activity is based on the dose equivalent of 0.1 ~~m~~Ci/gm of I-131. micro

The following assumptions and parameters are used to calculate the activity released and the offsite ~~dose~~ and control room doses following an SGTR:

The total break flow to the ruptured steam generator is listed in Table 15.6-4.

a. The amount of discharge or reactor coolant in the secondary system as a function of time is as calculated by RETRAN analysis. ~~The analysis yields 195,371 pounds of reactor coolant transferred to the secondary side of the faulted steam generator.~~

b. It is assumed that all of the iodine and alkali metals ~~in the~~ fraction of reactor coolant that flashes to steam upon reaching the secondary side is released to the steam phase. No credit is taken for scrubbing.

c. A 1-gpm primary-to-secondary ~~leak~~ leakage ~~is~~ assumed to occur to the intact ~~unaffected~~ steam generators, through the accident sequence.

d. All noble gas activity in the reactor coolant that is transported to the secondary system via the tube rupture and the primary-to-secondary leakage is released to the atmosphere.

e. The iodine partition fraction between the liquid and steam in the steam generator is assumed to be 0.01.

g. ~~f.~~ The steam releases from the steam generators to the atmosphere are as calculated by RETRAN analysis and given in Table 15.6-4. ~~The total faulted feedwater flows to all steam generators are also listed in Table 15.6-4.~~

h. ~~g.~~ Radioactivity releases to the atmosphere are based on the concentrations of radioactivity in the steam phase times the calculated amounts of steam release.

i. ~~h.~~ No additional radioactivity releases occur after the initiation of RHR system cooling.

j. ~~i.~~ Radioactive decay prior to the release of activity is considered. No decay during transit or ground deposition is considered.

k. ~~j.~~ Short-term accident atmospheric dispersion factors, breathing rates, and dose conversion factors are provided in Tables 15A-2, 15A-1, and 15.6-4, respectively.

f. The full power moisture carryover of 0.25% is utilized to determine the alkali metal releases to the environment via steam venting from the steam generators.

15.6.3.3.1.3 Mathematical Models Used in the Analysis

Mathematical models used in the analysis are described in the following sections:

a. The mathematical models used to analyze the activity released during the course of the accident are described in ~~Reference 1 and 2~~. Appendix 15A

b. The atmospheric dispersion factors used in the analysis were calculated based on the onsite meteorological measurements program, as described in USAR Section 2.3, and are provided in Table 15A-2.

c. The ~~thyroid inhalation immersion~~ TEDE doses to a receptor at the exclusion area boundary, ~~and~~ outer boundary of the low-population zone ~~were~~ and in the control room analyzed, using the models described in Appendix 15A.

15.6.3.3.1.4 Identification of Leakage Pathways and Resultant Leakage Activity

The activity released from the ruptured steam generator, is released directly to the environment by the atmospheric relief valves. The intact steam generators discharge steam and entrained activity via the safety and atmospheric relief valves until the time that initiation of the RHR system can be accomplished. ~~In addition, the steam release via the exhaust stack of the TDAFW pumps is also considered in the SGTR dose consequences shown in Table 15.6-4.~~ Since the activity is released directly to the environment with no credit for plateout or fall out, the results of the analysis are based on the most direct leakage pathway available. Therefore, the resulting radiological consequences represent a conservative estimate of the potential integrated dose to the postulated SGTR.

15.6.3.3.2 Identification of Uncertainties and Conservatisms in the Analysis

a. Reactor coolant activities based on extreme iodine spiking effects are conservatively high.

b. The assumed 1-gpm steam generator primary-to-secondary leakage is greater than that anticipated during normal operation.

c. Tube rupture of the steam generator is assumed to be a double-ended severance of a single steam generator tube. This is a conservative assumption, since the steam generator tubes are constructed of highly ductile materials. The more probable mode of tube failure is one or more minor leaks of undetermined origin. Activity in the secondary steam system is subject to continual surveillance, and the accumulation of activity from minor leaks that exceeds the limits established in the technical specifications would lead to reactor shutdown. Therefore, it is unlikely that the total amount of activity considered available for release in this analysis would ever be realized.

g. Reactor coolant iodine and noble gas activities are based on the Technical Specification limits which are significantly higher than the reactor coolant activities associated with normal operating conditions.

d. The coincident loss of offsite power with the occurrence of an SGTR is a conservative assumption. In the event of the availability of offsite power, the condenser dump valves will open, permitting steam dump to the condenser. This will reduce the amount of steam and entrained activity discharged directly to the environment from the ~~unaffected~~ intact steam generators.

e. The radiological consequences have been based on a worst-case scenario, i.e., ~~forced steam generator overfill with a stuck-open safety valve.~~ a stuck open ARV on the ruptured steam generator.

f. The meteorological conditions which may be present at the site during the course of the accident are uncertain. However, it is unlikely that meteorological conditions assumed will be present during the course of the accident for any extended period of time. Therefore, the radiological consequences evaluated, based on the meteorological conditions assumed, are conservative.

15.6.3.3.3 Conclusions

15.6.3.3.3.1 Filter Loadings

The only ESF filtration system considered in the analysis which limits the consequences of the steam generator tube rupture is the control room filtration system. Activity loadings on the control room charcoal filter are based on flow rate through the filter, concentration of activity at the filter inlet, and filter efficiency.

Activity in the control room filter as a function of time has been evaluated for the LOCA, Section 15.6.5. Since the control room filters are capable of accommodating the potential design-basis LOCA fission product iodine loadings, more than adequate design margin is available with respect to postulated SGTR accident releases. Insert G

15.6.3.3.~~3.2~~ Doses to Receptor at the Exclusion Area Boundary, Low-Population Zone Outer Boundary, and in the Control Room

~~Consistent with the current methodology for evaluating radiological consequences of a postulated SGTR, along with the changes mentioned in Section 15.6.3.3.1, the offsite (Exclusion Area Boundary and Low Population Zone) doses to the thyroid and whole body have been calculated, using the RADTRAD 3.03 computer code (Reference 8). The results of the dose calculations are presented in Table 15.6-5. As can be seen from this table, the calculated radiological consequences of a postulated steam generator tube failure accident do not exceed: (1) the exposure guidelines as set forth in 10 CFR Part 100, Section 11 for the accident with an assumed pre-accident iodine spike, and (2) a small fraction (i.e., 10%) of these exposure guidelines, for the accident with an equilibrium iodine concentration in combination with an assumed accident initiated iodine spike.~~

15.6.3.4 Conclusions

A steam generator tube rupture will cause no subsequent damage to the RCS or the reactor core. An orderly recovery from the accident can be completed, even assuming simultaneously loss of offsite power.

Insert G

The radiological consequences resulting from the occurrence of a postulated steam generator tube rupture have been conservatively analyzed, using assumptions and models described in previous sections.

The TEDE doses have been analyzed for the 0-2 hour dose at the exclusion area boundary, for the duration of the accident (0 to 12 hours) at the low-population zone outer boundary, and for 30 days in the control room. The results are listed in Table 15.6-5. The resultant offsite doses are within the guideline values of 10 CFR 50.67 for the case with an assumed pre-accident iodine spike and are within a small fraction of the guideline values (i.e., 10%) of 10 CFR 50.67 for the case with an accident-initiated iodine spike. The resultant control room doses are within the guideline value of 10 CFR 50.67.

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

Replace with "Deleted"

15.6.3.5 REFERENCES

1. ~~Letter SLNRC 86-1, Petrick, N. A., SNUPPS to Denton, H. R. (NRC), "Steam Generator Single-Tube Rupture Analysis for SNUPPS Plants - Callaway and Wolf Creek", dated January 8,1986.~~

2. ~~Letter WM 87-0145, Withers, B. D. WCNOC to US NRC, "Response to RAI Regarding the SGTR Overfill Case Analysis", dated May 15, 1987.~~

3. McFadden, J. H., et al, "RETRAN-02 - A Program for Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems", EPRI-NP-1850-CCM-A, October 1984.

4. ~~Letter from NRC to Withers, B. D., "Safety Evaluation Report for the Wolf Creek Generating Station Steam Generator Tube Rupture Analysis", TAC No. 57363, dated May 7, 1991.~~

5. Stewart, C. W., et. al., "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores," Battelle, Pacific Northwest Laboratories, EPRI NP-2511-CCM-A, August 1989.

6. Paulsen, M. P., et. al., "RETRAN-3D: A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," EPRI NP-7450(A). July 2001.

7. ~~Regulatory Guide 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," May 2003.~~

8. ~~NUREG/CR-6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," S. L. Humphreys et. al., November 1998.~~

15.6.4 SPECTRUM OF BWR STEAM SYSTEM PIPING FAILURES OUTSIDE OF CONTAINMENT

This section is not applicable to WCGS.

15.6.5 LOSS-OF-COOLANT ACCIDENTS RESULTING FROM A SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY

15.6.5.1 Identification of Causes and Frequency Classification

A loss-of-coolant accident (LOCA) is the result of a pipe rupture of the reactor coolant system (RCS) pressure boundary. For the analysis reported here, a small break is defined as a rupture of the RCS piping with a cross-sectional area less than 1.0ft^2, in which the normally operating charging system flow is not sufficient to sustain pressurizer level and pressure. A small break LOCA is classified as an ANS Condition III event (an infrequent fault), as defined in Section 15.0. A major break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0ft^2. This event is considered an ANS Condition IV event, a limiting fault, in that it is not expected to occur during the life of a plant but is postulated as a conservative design basis.

The Acceptance Criteria for the LOCA are described in 10 CFR 50.46 as follows:

A. The calculated peak fuel element clad temperature shall not exceed the requirement of 2200°F.

15.6.5.4 Radiological Consequences

15.6.5.4.1 Method of Analysis

The analysis of the radiological consequences of a postulated LOCA uses the recommended dose conversion factors (DCFs) as follows:

committed effective dose equivalent

(1) The analysis uses ~~thyroid~~ dose conversion factors (DCFs) for inhalation of radionuclides based on the data provided in Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion". ~~This is a departure from the more conservative DCF values listed in Regulatory Guide 1.109, which were used in the previous analysis of record,~~ (2) The effective dose conversion factors, provided in Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil", are used to calculate the ~~whole body~~ doses. ~~This deviates from the previous analysis of record, which used the more conservative DCF values listed in Regulatory Guide 1.109, and Using this DCFs results in lower does then would be obtained using the DCFs previously presented in the LOCA analysis of record.~~

dose equivalent

external exposure

15.6.5.4.1.1 Containment Leakage Contribution

PHYSICAL MODEL - Following a postulated double-ended rupture of a reactor coolant pipe with subsequent blowdown, the ECCS limits the clad temperature to well below the melting point and ensures that the reactor core remains intact and in a coolable geometry, minimizing the release of fission products to the containment. However, to demonstrate that the operation of a nuclear power plant does not represent any undue radiological hazard to the general public, a hypothetical accident involving a significant release of fission products to the containment is evaluated.

in phases

It is assumed that ~~100 percent of the noble gases and 50 percent of~~ the ~~iodine~~ equilibrium core saturation fission product inventory is ~~immediately~~ released to the containment atmosphere. ~~Of the iodine released to the containment, 50 percent is assumed to plateout onto the internal surfaces of the containment or adhere to internal components.~~ The ~~remaining iodine and the noble gas~~ activity ~~are~~ assumed to be immediately available for leakage from the containment.

Once the ~~gaseous~~ fission product activity is released to the containment atmosphere, it is subject to various mechanisms of removal which operate simultaneously to reduce the amount of activity in the containment. The removal mechanisms include radioactive decay, containment sprays, and containment leakage. For the noble gas fission products, the only removal processes considered in the containment are radioactive decay and containment leakage.

released from the core during each of the release phases is

sedimentation,

a. Radioactive Decay - Credit for radioactive decay for fission product concentrations located within the containment is assumed throughout the course of the accident. Once the activity is released to the environment, no credit for radioactive decay or deposition is taken.

b. Containment Sprays - The containment spray system is designed to absorb airborne iodine fission products within the containment atmosphere. To enhance the iodine-removal capability of the containment sprays, sodium hydroxide is added to the spray solution. The spray effectiveness for the removal of iodine is dependent on the iodine chemical form.

and retention

d. Sedimentation - Credit is taken for sedimentation of particulates in the regions of containment not impacted by sprays and in the sprayed region of containment when sprays are not on.

c. Containment Leakage - The containment leaks at a rate of 0.2 volume percent/day as incorporated as a Technical Specification requirement at peak calculated internal containment pressure for the first 24 hours and at 50 percent of this leak rate for the remaining duration of the accident. The containment leakage is assumed to be directly to the environment.

ASSUMPTIONS AND CONDITIONS - The major assumptions and parameters assumed in the analysis are ~~itemized~~ in Tables 15A-1 and 15.6-6.

In the evaluation of a LOCA, all the fission product release assumptions of Regulatory Guide ~~1.4~~ 1.183 have been followed. The following specific assumptions were used in the analysis. ~~Table 15.6-7 provides a comparison of the analysis to the requirements of Regulatory Guide 1.4.~~

a. The reactor core equilibrium ~~noble gas and iodine~~ inventories are based on long-term operation at the ultimate core power level of 3,565 MWt plus 2% uncertainty.

b. ~~One hundred percent of the core equilibrium radioactive noble gas inventory is immediately available for leakage from the containment.~~

The nuclide groups and their release fractions are presented in Table 15.6-6.

~~c. Twenty-five percent of the core equilibrium radioactive iodine inventory is immediately available for leakage from the containment.~~

c. ~~d.~~ Of the iodine fission product inventory released to the containment, ~~91~~ 4.85 percent is in the form of elemental iodine, ~~5~~ 95 percent is in the form of particulate iodine, and ~~4~~ 0.15 percent is in the form of organic iodine. of iodine and particulates

d. ~~e.~~ Credit for ~~iodine~~ removal by the containment spray system is taken, starting ~~at time zero~~ 2 minutes after event initiation and continuing until a decontamination factor of ~~100~~ 200 for the elemental ~~and particulate~~ species has been achieved.

The credit for the particulate species removal is continued for the duration of sprays but is reduced by a factor of 10 after a decontamination factor of 50 is achieved.

e. ~~f.~~ The following ~~iodine~~ removal constants for the containment spray system are assumed in the analysis:

Elemental iodine	10.0 hr^{-1}
Organic iodine	0.0 hr^{-1}
Particulate s ~~iodine~~	5.0 ~~0.45~~ hr^{-1} prior to DF of 50

0.5 hr^{-1} after DF of 50

f. ~~g.~~ The following parameters were used in the two-region spray model:

Fraction of containment sprayed - 0.85
Fraction of containment unsprayed - 0.15
Mixing rate (cfm) between sprayed and unsprayed regions - ~~85,000~~ 69,400

Time (min) to achieve full fan cooler flow - 2

g. The removal constant in containment for sedimentation of particulates assumed in the analysis is 0.1 hr^{-1}.

Section 6.2

Section 6.5 contains a detailed analysis of the sprayed and unsprayed volumes and includes an explanation of the mixing rate between the sprayed and unsprayed regions.

h. The containment is assumed to leak at 0.2 volume percent/day during the first 24 hours immediately following the accident and 0.1 volume percent/day thereafter.

i. The containment leakage is assumed to be direct unfiltered to the environment.

~~j. The ESF filters are 90 percent efficient in the removal of all species of iodine.~~

MATHEMATICAL MODELS USED IN THE ANALYSIS - Mathematical models used in the analysis are described in the following sections:

a. The mathematical models used to analyze the activity released during the course of the accident are described in Section 15A.2.

b. The atmospheric dispersion factors used in the analysis were calculated, based on the onsite meteorological measurements program described in Section 2.3 and are provided in Table 15A-2.

c. The ~~thyroid inhalation total-body immersion~~ TEDE doses to a receptor exposed at the exclusion area boundary and the outer boundary of the low population zone were analyzed, using the models described in Sections 15A.2.4 and 15A.2.5, respectively.

d. Buildup of activity in the control room and the integrated doses to the control room personnel are analyzed, based on models described in Section 15A.3.

IDENTIFICATION OF LEAKAGE PATHWAYS AND RESULTANT LEAKAGE ACTIVITY - For evaluating the radiological consequences of a postulated LOCA, the resultant activity released to the containment atmosphere is assumed to leak directly to the environment.

No credit is taken for ground deposition or radioactive decay during transit to the exclusion area boundary, ~~or~~ LPZ outer boundary, or control room

15.6.5.4.1.2 Radioactive Releases Due to Leakage from ECCS and Containment Spray Recirculation Lines

Subsequent to the injection phase of ESF system operation, the water in the containment recirculation sumps is recirculated by the residual heat removal, centrifugal charging and safety injection pumps, and the containment spray pumps. Due to the operation of the ECCS and the containment spray system, most of the radioiodine released from the core would be contained in the containment sump. It is conservatively assumed that a leakage rate of 2 gpm from the ECCS and containment spray recirculation lines exists for the duration of the LOCA. This leakage would occur inside the containment as well as inside the auxiliary building. For this analysis, all the leakage is assumed to occur inside the auxiliary building. Only trace quantities of radioiodine are expected to be airborne within the auxiliary building due to the temperature and pH level of the recirculated water. However, 10 percent of the radioiodine in the leaked water is assumed to become airborne. This airborne iodine is assumed to be released immediately to the environment from the unit vent, via the safety grade filters associated with the auxiliary building emergency exhaust system. No credit is taken for holdup or mixing in the auxiliary building; however, mixing and holdup in the containment sumps are included in the determination of radioactive material releases and radioiodine removal through radioactive decay for this leakage pathway.

Radiological Consequences of ECCS/CS Recirculation Line Leakage - The assumptions used to calculate the amount of radioiodine released to the environment are given in Table 15.6-6. The dose models are presented in Section 15.A. The offsite doses at the site boundary and LPZ and the doses to control room personnel from this pathway are given in Table 15.6-8.

15.6.5.4.1.3 Radioactive Releases Due to Operation of Containment Mini-Purge System

The containment mini-purge is designed to reduce the containment noble gas concentration. The containment mini-purge system will be operated during power operation if access to the containment is desired. The containment mini-purge isolation valves are automatically closed upon a containment purge isolation signal should a LOCA occur during containment purging when the reactor is at power. The radioactive release via containment mini-purge will exist until the containment isolation signal is received and the valves can be closed. Exhaust from the containment is processed through the containment purge exhaust system filter absorber train prior to discharge through the unit vent.

The maximum time for the purge valve closure is limited to ~~five~~ ten seconds to assure that the purge valves would be closed before the onset of fuel failures following a LOCA. Therefore, the source terms used in the radiological consequences calculation is based on the fission product activity in the primary coolant ~~with consideration of pre-existing iodine spike~~. The containment mini-pure system is assumed to be isolated within ~~5~~ 10 seconds following the initiation of the accident. The release rate from the containment mini-purge system is assumed at 4680 cfm. ~~Filter efficiency of 90% for the~~ removal of ~~all species of~~ iodine and particulates is ~~assumed~~ not. ~~Credit~~ for ~~iodine~~ removal by the containment spray system ~~is not assumed~~ in the analysis, nor is credit for activity release via containment mini-purge system.

~~From the safety analysis perspective, it is acceptable to use either the shutdown purge or mini-purge during refueling operations. This conclusion is based on an assumption used in the fuel handling accident (FHA) involving the radioactive material relief rate. To comply with Reg. Guide 1.25, all of the gap activity in the damaged rods is assumed as a result of a FHA, to be released and escape to the environment over a two-hour time period. The analysis does not assume pathway, only that all the radioactivity is released~~

~~from containment within a two hour period in some fashion. Thus the operation of a particular purge system, mini or shutdown is not of importance in the analysis. This assumption translates into a very large air exhaust rate. Therefore, the difference between the large volume containment shutdown purge (20,000 cfm) and mini-purge (4,000 cfm) would have no impact on the calculated dose consequences.~~

15.6.5.4.1.4 Radioactive Releases Due to Leakage from the Containment Sump to the RWST

The leakage pathway is from the containment recirculation sump through ECCS boundary valves back to the RWST, which is vented to the atmosphere. It is assumed that the activity released to the holdup system (in this case, the containment recirculation sump) instantaneously diffuses to uniformly occupy the sump volume. Removal mechanisms from the sump include decay and release (i.e., leakage) to the RWST.

It is assumed that 10% of the radioiodine leaked to the RWST becomes airborne, mixes with the RWST air volume, and is released to the environment. Credit is taken for radioactive decay in the RWST. The leakage rate from containment sump to the ~~RWST~~ liquid is assumed at 3.8 gpm for the duration of the accident. The RWST air volume when leakage starts is assumed to be ~~400,000~~ 354,000 gallons. Other assumptions used to calculate the amount of radioiodine released to the environment are the same as in the calculation for ECCS recirculation leakage inside the auxiliary building.

The release from the RWST is assumed to match the volumetric flow into RWST liquid (i.e., 3.8 gpm).

15.6.5.4.2 Identification of Uncertainties and Conservatisms in the Analysis

The uncertainties and conservatisms in the assumptions used to evaluate t radiological consequences of a LOCA result principally from assumptions m involving the amount of the gaseous fission products available for releas the environment and the meteorology present at the site during the course the accident. The most significant of these assumptions are:

a. The ECCS is designed to prevent fuel cladding damage that would allow the release of the fission products contained in the fuel to the reactor coolant. Severe degradation of the ECCS (i.e., to the unlikely extent of simultaneous failure of redundant components) would be necessary in order for the release of fission products to occur of the magnitude assumed in the analysis.

~~b. The release of fission products to the containment is assumed to occur instantaneously.~~

b. ~~c.~~ It is assumed that ~~50 percent~~ none of the iodines released to the containment atmosphere is plated-out onto the internal surfaces of the containment or adheres to

internal components; however, it is ~~estimated~~ recognized that the removal of airborne iodines by various physical phenomena such as adsorption, adherence, and settling could reduce the resultant doses ~~by a factor of 3 to 10 (Ref. 24)~~.

c. ~~d.~~ The activity released to the containment atmosphere is assumed to leak to the environment at the containment leakage rate of 0.2-volume percent/day for the first 24 hours and 0.1-volume percent/day thereafter. The initial containment leakage rate is based on the peak calculated internal containment pressure anticipated after a LOCA. The pressure within the containment actually decreases with time. Taking into account that the containment leak rate is a function of pressure, the resultant doses could be reduced by a factor of 5 to 10 (Ref. ~~24~~ 21).

d. ~~e.~~ The meteorological conditions assumed to be present at the site during the course of the accident are based on X/Q values, which are ~~expected to be exceeded 5 percent of the time~~ the larger of the 5 percent overall site values and the 0.5 percent maximum sector values. This condition results in the poorest values of atmospheric dispersion calculated for the exclusion area boundary and the LPZ outer boundary. Furthermore, no credit has been taken for the transit time required for activity to travel from the point of release to the exclusion area boundary and LPZ outer boundary. Hence, the radiological consequences evaluated under these conditions are conservative.

15.6.5.4.3 Conclusions

15.6.5.4.3.1 Filter Loadings

No recirculating or single-pass filters are used for fission product cleanup and control within the containment following a postulated LOCA. The only ESF filtration systems expected to be operating under post-LOCA conditions are the control room HVAC system and the auxiliary building emergency exhaust filtration system.

Activity loadings on the control room charcoal adsorbers are based on the flowrate through the adsorber, the concentration of activity at the adsorber inlet, and the adsorber efficiency. Based on the ~~radioactive iodine~~ release assumptions previously described~~, the assumption that 25 percent of the core inventory of isotopes I-127 and I-129 is available for release from the containment atmosphere~~ and the assumption that the charcoal adsorber is ~~100~~ 90 percent efficient, the calculated filter loadings ~~are in accordance with Regulatory Guide 1.52, which limits the maximum loading to 2.5 mg of iodine per gram of activated charcoal.~~ dose to control room operators from is 0.0489 rem TEDE. for iodine and particulates

15.6.5.4.3.2 Doses to a Receptor at the Exclusion Area Boundary and Low Population Zone Outer Boundary

The potential radiological consequences resulting from the occurrence of the postulated LOCA have been conservatively analyzed, using assumptions and models described in previous sections.

The ~~total-body dose due to immersion and the thyroid dose due to inhalation~~ TEDE doses have been analyzed for the ~~0-~~2 hour dose at the exclusion area boundary and for the duration of the accident at the LPZ outer boundary. The results are listed in Table 15.6-~~5~~ 8. The resultant doses are within the guideline values of 10 CFR ~~100~~ 50.67. limiting

15.6.5.4.3.3 Doses to Control Room Personnel

Radiation doses to control room personnel following a postulated LOCA are based on the ventilation, cavity dilution, and dose model discussed in Section 15A.3.

Control room personnel are subject to a ~~total body dose due to immersion and a thyroid dose due to inhalation~~ TEDE dose. ~~These doses have~~ This dose has been analyzed, and ~~are~~ is provided in Table 15.6-8. The resultant ~~doses are~~ is within the limit~~s~~ established by ~~GDC-19~~ 10 CFR 50.67.

15.6.6 A NUMBER OF BWR TRANSIENTS

This section is not applicable to WCGS.

15.6.7 REFERENCES

1. Deleted

2. U.S. Nuclear Regulatory Commission 1975, "Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400, NUREG-75/014.

3. Bordelon, F. M.; Massie, H. W.; and Zordan, T. A., "Westinghouse ECCS Evaluation Model - Summary," WCAP-8339, July 1974.

4. Bordelon, F. M. et al., "SATAN-VI Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant," WCAP-8302 (Proprietary) and WCAP-8306 (Non-Proprietary), June 1974.

5. Kelly, R. D. et al., "Calculation Model for Core Reflooding After a Loss-of-Coolant Accident (WREFLOOD Code)," WCAP-8170 (Proprietary) and WCAP-8171 (Non-Proprietary), June 1974.

6. Young, M. Y. et al., "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," WCAP-10266-P-A Rev. 2 (Proprietary), March 1987.

with assumed iodine spike that increases the rate of iodine release into the reactor coolant by a factor of 500. This increased rate is assumed for 8 hours.

TABLE 15.6-2

PARAMETERS USED IN EVALUATING
THE RADIOLOGICAL CONSEQUENCE OF
THE CVCS LETDOWN LINE RUPTURE OUTSIDE OF CONTAINMENT

I.	Source Data	
	a. Core power level, MWt	3637 (includes uncertainty) ~~3,565~~
	b. Reactor coolant iodine activity	Dose equivalent of 1.0 µCi/gm of I-131
	c. Reactor coolant noble gas activity	~~Based on 1-percent failed fuel. See Table 11.1-5.~~
	d. Insert F	Dose equivalent of 500 microCi/gm of Xe-133
II.	Atmospheric Dispersion Factors	See Table 15A-2.
III.	Activity Release	
	a. Break flow rate, gpm	~~444~~ 141
	b. Duration, secs	1810
	c. Fraction of iodine and alkali metal activity in the spill that is airborne	~~0.10~~ 0.18
	~~d. Activity released to the environment~~	

~~Isotope~~	~~0-2 hr (Ci)~~
~~I-131~~	~~3.70~~
~~I-132~~	~~3.89~~
~~I-133~~	~~6.47~~
~~I-134~~	~~8.47E-01~~
~~I-135~~	~~3.72~~
~~Xe-131m~~	~~1.73E+02~~
~~Xe-133m~~	~~2.71E+02~~
~~Xe-133~~	~~1.47E+04~~
~~Xe-135m~~	~~1.92E+01~~
~~Xe-135~~	~~4.89E+02~~
~~Xe-137~~	~~3.39~~
~~Xe-138~~	~~2.52E+01~~
~~Kr-83m~~	~~2.59E+01~~
~~Kr-85m~~	~~1.11E+02~~
~~Kr-85~~	~~4.76E+02~~
~~Kr-87~~	~~6.65E+01~~
~~Kr-88~~	~~2.05E+02~~
~~Kr-89~~	~~1.58~~

Insert F

Reactor coolant initial alkali metal activity	Based on 1-percent failed fuel as provided in Table 11.1-5

TABLE 15.6-3

RADIOLOGICAL CONSEQUENCES OF A
CVCS LETDOWN LINE BREAK OUTSIDE
OF CONTAINMENT

	Doses (rem) TEDE
Exclusion Area Boundary (0-2 hr)	
~~Thyroid,~~	~~3.91e-1~~
~~Whole body~~	~~4.43e-2~~ 2.3E-1
Low Population Zone Outer Boundary (duration)	
~~Thyroid,~~	~~5.22e-2~~
~~Whole body~~	~~5.91e-3~~ 7.3E-2
Control Room (30 days)	8.3E-1

Table 15.6-4

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A STEAM GENERATOR TUBE RUPTURE ~~WITH FORCED OVERFILL~~

I.	Source Data	
a.	Core power level, MWt	363~~6.3~~ 7 (1)
b.	Steam generator tube leakage, gpm	1 (2)
c.	Reactor coolant iodine activity:	
	1. Case 1	Pre-accident iodine spike per ~~Standard Review Plan 15.6.3~~. A pre-accident reactor transient has occurred which raised the primary coolant I-131 dose equivalent concentration to the maximum value permitted by Technical Specifications (60 µCi/gm).
	2. Case 2	Accident initiated iodine spike per ~~Standard Review Plan 15.6.3~~. An iodine spike occurs concurrently with reactor trip where the release rate from the fuel rods increases to a value 335 time greater than the release rate corresponding to the I-131 ~~equivalent~~ dose concentration at the equilibrium value (1 µCi/gm) stated in the Technical Specifications.
e. ~~d.~~	Reactor coolant noble gas activity	~~Based on 1% failed fuel as provided in USAR Table 11.1-5.~~
f. ~~e.~~	Secondary system initial ~~activity~~ iodine	Dose equivalent of 0.1 µCi/gm of I-131
h. ~~f.~~	Reactor coolant mass, lbs	~~4.964E+5~~ 3.99E+5
i. ~~g.~~	Steam generator mass (each), lbs	
	1. ~~Water~~ Ruptured	~~9.884E+4~~ 70,000
	2. ~~Steam~~ Intact	~~8.269E+3~~ 65,000
j. ~~h.~~	Offsite power	Lost at reactor trip

Regulatory Guide 1.183

Dose equivalent of 500 microCi/gm of Xe-133

d. Insert F

g. Insert C

Table 15.6-4 (Sheet 2)

~~i.~~	~~Primary-to-secondary leakage duration~~	~~Break flow continues throughout the transient, due to primary-secondary pressure inequalities, although most of the flow is terminated beyond the end of the second depressurization at approximately 6200 seconds~~ [3]~~.~~
II.	Atmospheric Dispersion Factor	The atmospheric dispersion factors used are the same as those listed in USAR Table 15A-2.

III. Activity Release Data

a. Ruptured steam generator

1.	Reactor coolant discharged to steam generator, lbs	~~195,371~~ 307,665
2.	Flashed reactor coolant, lbs	~~Discussion below~~ 20,015

~~The fraction of the primary-to-secondary leakage that flashes to steam is defined as, X(t), where~~

$$\cancel{X(t) = \frac{h_{avg}(t) - h_f(t)}{h_g(t) - h_f(t)}}$$

~~and where~~ $\cancel{h_g(t) =}$ ~~saturated vapor specific enthalpy corresponding to the faulted steam generator water temperature,~~

$\cancel{h_f(t) =}$ ~~saturated liquid specific enthalpy corresponding to the faulted steam generator water temperature.~~

~~The value~~ $\cancel{h_{avg}(t)}$ ~~is the average specific enthalpy and is defined as:~~

$$\cancel{h_{avg} = \frac{[G_{HL}(t)*h_{HL}(t)] + [G_{CL}(t)*h_{CL}(t)]}{G_{HL}(t) + G_{CL}(t)}}$$

~~Where~~ $\cancel{h_{HL}(t) =}$ ~~specific enthalpy of the fluid from the hot leg,~~

$\cancel{h_{CL}(t) =}$ ~~specific enthalpy of the fluid from the cold leg,~~

Table 15.6-4 (sheet 3)

~~$G_{HL}(t)$ = leakage flow from the hot leg to the faulted steam generator,~~

~~$G_{CL}(t)$ = leakage flow from the cold leg to faulted steam generator.~~

and alkali metal

3. Iodine partition factor for flashed fraction of reactor coolant — 1.0

4. Steam release to atmosphere, lbs

0 - break flow termination

~~0-2 hours~~ — ~~140,406~~[4] 365,420

break flow termination - RHR cut-in conditions

~~2 hour-RHR cut-in conditions~~ — ~~120,960~~[4] 2530

Iodine

5. ~~Iodine~~ C arryover factor s for the non-flashed fraction of reactor coolant that mixes with the initial iodine activity in the steam generator — 0.01

Alkali Metal

0.0025

b. Intact steam generators

1. Primary-to-secondary leakage, lbs — 6012 ~~4004~~[2]

2. Flashed reactor coolant, percent — 0

~~3. Feedwater flow rate, lbs~~

~~0-2 hours~~ — ~~1,020,231~~

~~2 hours-RHR cut-in conditions~~ — ~~280,800~~

3. ~~4.~~ Steam release to atmosphere, lbs

0-break flow termination — ~~309,069~~ 955,791

break flow termination -RHR cut-in conditions — ~~299,520~~ 1,645,930

5. Alkali Metal carryover factor

4. ~~5.~~ Iodine carryover factor — 0.01 0.0025

~~6. Transient end time~~ — ~~The WCNOC analysis to force SGTR steam generator overfilling was carried out until RHR cut-in conditions were achieved at 28,800 seconds.~~

Table 15.6-4 (sheet 4)

Condensor

c. ~~Steam release from the turbine-driven AFW pump~~ ~~19,334~~

1. Steam release to atmosphere, lbs — 750,750
2. Iodine and alkali metal partition factor — 0.01

IV Dose Conversion Factors

~~Nuclide~~	See Table 15A-4	~~Whole Body Rem-m^3/Ci-sec~~	~~Thyroid Rem/Ci~~
~~I-131~~		~~6.73E-02~~	~~1.08E+06~~
~~I-132~~		~~4.14E-01~~	~~6.44E+03~~
~~I-133~~		~~1.09E-01~~	~~1.80E+05~~
~~I-134~~		~~4.81E-01~~	~~1.07E+03~~
~~I-135~~		~~2.95E-01~~	~~3.13E+04~~
~~Kr-83m~~		~~5.55E-06~~	~~NA~~
~~Kr-85m~~		~~2.77E-02~~	~~NA~~
~~Kr-85~~		~~4.40E-04~~	~~NA~~
~~Kr-87~~		~~1.52E-01~~	~~NA~~
~~Kr-88~~		~~3.77E-01~~	~~NA~~
~~Kr-89~~		~~3.23E-01~~	~~NA~~
~~Xe-131m~~		~~1.44E-03~~	~~NA~~
~~Xe-133m~~		~~5.07E-03~~	~~NA~~
~~Xe-133~~		~~5.77E-03~~	~~NA~~
~~Xe-135m~~		~~7.55E-02~~	~~NA~~
~~Xe-135~~		~~4.40E-02~~	~~NA~~
~~Xe-137~~		~~3.03E-02~~	~~NA~~
~~Xe-138~~		~~2.13E-01~~	~~NA~~

Table 15.6-4 (sheet 5)

Notes:

1. This value is 102% of the guaranteed core thermal power output. This power level was chosen to maximize steam generator ~~overfill~~, and thus the radiological consequences ~~resulting from the overfill~~ were calculated from the results of this model.

2. A constant 1 gpm primary-to-secondary leakage with an assumed density of 62.4 lbm/ft^3, is assumed to occur to the ~~unaffected~~ intact steam generators~~, for the duration of the event~~ until the RHR cut-in conditions were achieved at 12 hours.

3. ~~Steam generator overfill was forced by delaying the termination of safety injection to provide enough liquid in the faulted steam generator steam line to force safety valve opening and liquid relief. The extended SI flow time caused the primary pressure to remain high, thus the primary/secondary pressure remained unequal and allowed for break flow to continue for an extended period of time.~~

4. ~~The SGTR with Forced Overfill analysis assumed that at liquid relief through the faulted steam generator steam line safety valve, the valve would fail open with an effective flow area of 5%, with consequent continued secondary blowdown until RHR cut-in conditions were reached.~~

TABLE 15.6-5

RADIOLOGICAL CONSEQUENCES OF A
STEAM GENERATOR TUBE RUPTURE

Case 1	Dose (rem) TEDE
Exclusion Area Boundary (0-2 hr)	1.1E+0
~~Thyroid~~	~~51.769~~
~~Whole body~~	~~0.226~~
Low Population Zone (duration)	3.5E-1
~~Thyroid~~	~~7.192~~
~~Whole body~~	~~0.034~~
Control Room (30 days)	1.1E+0
Case 2	
Exclusion Area Boundary (0-2 hr)	8.6E-1
~~Thyroid~~	~~22.797~~
~~Whole body~~	~~0.132~~
Low Population Zone (duration)	2.9E-1
~~Thyroid~~	~~4.825~~
~~Whole body~~	~~0.025~~
Control Room (30 days)	4.5E-1

Case 1 - Pre-accident iodine spike per ~~SRP 15.6.3~~ RG 1.183

Case 2 - Concurrent iodine spike per ~~SRP 15.6.3~~ RG 1.183

TABLE 15.6-6

PARAMETERS USED IN EVALUATING
THE RADIOLOGICAL CONSEQUENCES OF A
LOSS-OF-COOLANT-ACCIDENT

I.	Source Data	
a.	Core power level, MWt	~~3,565~~ 3637 (inlcudes uncertainty)
b.	Burnup, full power days per cycle	~~510~~ 511
c.	~~Percent of core activity initially airborne in the containment~~	
~~1.~~	~~Noble gas~~ Insert H	~~100~~
~~2.~~	~~Iodine~~	~~25~~
d.	Percent of core activity in containment sump ~~@ 0.47 hours~~	
1.	Noble gases and Particulates	0
2.	Iodine	~~50~~ same release fraction and timing as Item c.2 above
e.	Core inventories	Table 15A-3
f.	Iodine distribution, percent Containment Leakage Release	
1.	Elemental	~~91~~ 97
2.	Organic	~~4~~ 0
3.	Particulate	~~5~~ 3
h. Insert O	g. Minimum sump pH 7.0	
II.	Atmospheric Dispersion Factors	See Table 15A-2
III.	Activity Release Data	
a.	Containment leak rate, volume percent/day	ECCS and mini-purge Release
1.	0-24 hours	0.20
2.	1-30 days	0.10
b.	Percent of containment leakage that is unfiltered	100
c.	Credit for containment sprays	
1.	Spray ~~iodine~~ removal constants (per hour)	
a.	Elemental (iodine)	10.0
b.	Organic (iodine)	0.0
c.	Particulates	~~0.45~~ 5.0 hr^{-1} prior to DF limit; 0.5 hr^{-1} thereafter

Insert H

c. Release fraction and timing of core activity in the containment

		Gap Release Phase Fraction (30 sec – 0.5 hr)	Early In-vessel Phase Fraction (0.5 hr – 1.8 hr)
1.	Noble gas	0.05	0.95
2.	Iodine	0.05	0.35
3.	Alkali metal	0.05	0.25
4.	Tellurium	0.00	0.05
5.	Barium and strontium	0.00	0.02
6.	Noble metal	0.00	0.0025
7.	Cerium	0.00	0.0005
8.	Lanthanide	0.00	0.0002

Insert O

Reactor Coolant Activity (mini-purge only)

1. Iodine	Dose equivalent of 1.0 μCi/gm of I-131
2. Noble gas	Dose equivalent of 500 μCi/gm of Xe-133
3. Alkali metal	Based on 1-percent failed fuel as provided in Table 11.1-5

TABLE 15.6-6 (Sheet 2)

2.	~~Maximum iodine~~ D econtamination factors for the containment atmosphere	
	a. Elemental	~~100~~ 200
	b. Organic	0 0
	c. Particulate s	~~100~~ 50
3.	Sprayed volume, percent	85
4.	Unsprayed volume, percent	15
5.	Sprayed-unsprayed mixing rate, CFM	~~85,000~~ 69,400
6.	Containment volume, ft^3	2.~~5~~E+6 7

Insert I

~~d. Activity released to containment~~

~~Isotope~~	~~Curies~~
~~I-131~~	~~2.37E+7~~
~~I-132~~	~~3.43E+7~~
~~I-133~~	~~4.88E+7~~
~~I-134~~	~~5.38E+7~~
~~I-135~~	~~4.58E+7~~
~~Xe-131m~~	~~1.01E+6~~
~~Xe-133m~~	~~6.06E+6~~
~~Xe-133~~	~~1.95E+8~~
~~Xe-135m~~	~~3.77E+7~~
~~Xe-135~~	~~4.70E+7~~
~~Xe-137~~	~~1.71E+8~~
~~Xe-138~~	~~1.64E+8~~
~~Kr-83m~~	~~1.24E+7~~
~~Kr-85m~~	~~2.67E+7~~
~~Kr-85~~	~~1.02E+6~~
~~Kr-87~~	~~5.16E+7~~
~~Kr-88~~	~~7.28E+7~~
~~Kr-89~~	~~8.94E+7~~

e. ECCS recirculation leakage

1.	Leak rate ~~(0.47~~ hours-30 day), gpm	2.0
	2. Leak rate to RWST (0 hours-30 day), gpm	3.8(1)

to auxiliary building

Insert I

d. Credit for Sedimentation of Particulates	0.1 hr^{-1} until DF of 1000 is reached

TABLE 15.6-6 (Sheet 3)

~~2.~~	~~Iodine inventory in sump @ 0.47 hour, curies~~	
	~~I-131~~	~~4.72E+7~~
	~~I-132~~	~~5.94E+7~~
	~~I-133~~	~~9.60E+7~~
	~~I-134~~	~~7.41E+7~~
	~~I-135~~	~~8.71E+7~~
3.	Sump volume, gal.	460,000
4.	Fraction iodine airborne	0.1
5.	ESF filter efficiency, %	90.0
		354,000

IV. Control room parameters — Tables 15A-1 and 15A-2

6. RWST Initial Gas Volume, gal

Note:
1. The release from the RWST is assumed to match the ECCS leak rate, i.e., the volumetric flow into and out of the tank are equal.

TABLE 15.6-7 Delete

DESIGN COMPARISON TO THE REGULATORY POSITIONS OF REGULATORY GUIDE 1.4 "ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A LOSS OF COOLANT ACCIDENT FOR PRESSURIZED WATER REACTORS," REVISION 2, JUNE 1974

Regulatory Guide 1.4 Position	Design
1. The assumptions related to the release of radioactive material from the fuel and containment are as follows:	
a. Twenty-five percent of the equilibrium radioactive iodine inventory developed from maximum full power operation of the core should be assumed to be immediately available for leakage from the primary reactor containment. Ninety-one percent of this 25 percent is to be assumed to be in the form of elemental iodine, 5 percent of this 25 percent in the form of particulate iodine, and 4 percent of this 25 percent in the form of organic iodides.	1a. Complies.
b. One hundred percent of equilibrium radioactive noble gas inventory developed from maximum full power operation of the core should be assumed to be immediately available for leakage from the reactor containment.	1b. Complies.
c. The effects of radiological decay during holdup in the containment or other buildings should be taken into account.	1c. Complies. Credit for radioactive decay is taken until the activity is assumed to be released.
d. The reduction in the amount of radioactive material available for leakage to the environment by containment sprays, recirculating filter systems, or other engineered safety features may be taken into account, but the amount of reduction in concentration of radioactive materials should be evaluated on an individual case basis.	1d. Complies. See Table 15.6-6 for reduction taken.

TABLE 15.6-7 (Sheet 2)

Regulatory Guide 1.4 Position	Design
e. The primary reactor containment should be assumed to leak at the leak rate incorporated or to be incorporated as a technical specification requirement at peak accident pressure for the first 24 hours, and at 50 percent of this leak rate for the remaining duration of the accident. Peak accident pressure is the maximum pressure defined in the technical specifications for containment leak testing.	1e. Complies.
2. Acceptable assumptions for atmospheric diffusion and dose conversion are:	
a. The 0-8 hour ground level release concentrations may be reduced by a factor ranging from one to a maximum of three (see Figure 1) for additional dispersion produced by the turbulent wake of the reactor building in calculating potential exposures. The volumetric building wake correction, as defined in section 3-3.5.2 of Meteorology and Atomic Energy 1968, should be used only in the 0-8 hour period; it is used with a shape factor of 1/2 and the minimum cross-sectional area of the reactor building only.	2a. Complies. Atmospheric dispersion factors were calculated based on the onsite meteorological measurement programs described in Section 2.3.
b. No correction should be made for depletion of the effluent plume of radioactive iodine due to deposition on the ground, or for the radiological decay of iodine in transit.	2b. Same as 2a above.
c. For the first 8 hours, the breathing rate of persons offsite should be assumed to be 3.47×10^{-4} cubic meters per second. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.75×10^{-4} cubic meters per second. After that until the end of the accident, the breathing rate should be assumed to be 1.75×10^{-4} cubic meters per second. After that until the end of the accident, the rate should be assumed to be 2.32×10^{-4} cubic meters	2c. Complies. See Table 15A-1

TABLE 15.6-7 (Sheet 3)

Regulatory Guide 1.4 Position	Design
per second. (These values were developed from the average daily breathing rate [2 x 10^7 cm3/day] assumed in the report of ICRP, Committee II-1959.)	
d. The iodine dose conversion factors are given in ICRP Publication 2, Report of Committee II, "Permissible Dose for Internal Radiation," 1959.	2d. The dose conversion factors provided in Regulatory Guide 1.109 are used. See Table 15A-4.
e. External whole body doses should be calculated using "Infinite Cloud" assumptions, i.e., the dimensions of the cloud are assumed to be large compared to the distance that the gamma rays and beta particles travel. "Such a cloud would be considered an infinite cloud for a receptor at the center because any additional [gamma and] beta emitting material beyond the cloud dimensions would not alter the flux of [gamma rays and] beta particles to the receptor" (Meteorology and Atomic Energy, Section 7.4.1.1-editorial additions made so that gamma and beta emitting material could be considered). Under these conditions the rate of energy absorption per unit volume is equal to the rate of energy released per unit volume. For an infinite uniform cloud containing curies of beta radioactivity per cubic meter the beta dose in air at the cloud center is: $\beta^{D\infty} = 0.457\ \bar{E}_\beta \chi$ The surface body dose rate from beta emitters in the infinite cloud can be approximated as being one-half this amount (i.e., $\beta^{D\infty} = 0.23\ \bar{E}_\beta \chi$) For gamma emitting material the dose rate in air at the cloud center is: $\gamma^{D\infty} = 0.507\bar{E}_\gamma x$ From a semi-infinite cloud, the gamma dose rate in air is: $\gamma^{D\infty} = 0.25\bar{E}_\gamma x$	2e. The dose factors given in Regulatory Guide 1.109, for noble gases; for iodine whole body dose factors with 5cm body tissue attenuation; and for beta-skin dose factors with credit for attenuation in the dead skin layer, are used. See Table 15A-4.

TABLE 15.6-7 (Sheet 4)

Regulatory Guide 1.4 Position	Design
Where $\beta^{D\infty}$ = beta dose rate from an infinite cloud (rad/sec) $\gamma^{D\infty}$ = gamma dose rate from an infinite cloud (rad/sec) $\bar{E}_\beta$ = average beta energy per disintegration (Mev/dis) $\bar{E}_\gamma$ = average gamma energy per disintegration (Mev/dis) χ = concentration of beta or gamma emitting isotope in the cloud (curie/m^3)	
f. The following specific assumptions are acceptable with respect to the radioactive cloud dose calculations:	2f.1 See response to 2e.
(1) The dose at any distance from the reactor should be calculated based on the maximum concentration in the plume at that distance taking into account specific meteorological, topographical, and other characteristics which may affect the maximum plume concentration. These site related characteristics must be evaluated on an individual case basis. In the case of beta radiation, the receptor is assumed to be exposed to an infinite cloud at the maximum ground level concentration at that distance from the reactor. In the case of gamma radiation, the receptor is assumed to be exposed to only one-half the cloud owing to the presence of the ground. The maximum cloud concentration always should be assumed to be at ground level.	
(2) The appropriate average beta and gamma energies emitted per disintegration, as given in the Table of Isotopes, Sixth Edition, by C. M. Lederer, J. M. Hollander, I. Perlman;	2f.2 See response to 2e.

TABLE 15.6-7 (Sheet 5)

Regulatory Guide 1.4 Position	Design
University of California, Berkeley; Lawrence Radiation Laboratory; should be used.	
g. The atmospheric diffusion model should be as follows:	
(1) The basic equation for atmospheric diffusion from a ground level point source is: $\chi/Q = \pi \frac{1}{u\sigma_y}$	2g.1 Short-term accident atmospheric dispersion factors were calculated based on onsite meteorological data measurement programs described in Section 2.3. These factors are for ground level releases and are based on Regulatory Guide 1.145 methodology and represent the worst of the 5 percent site meteorology and the 0.5 percent worst sector meteorology.
Where χ = the short term average centerline value of the ground level concentration (curie/meter^3) Q = amount of material released (curie/sec) u = windspeed (meter/sec) σ_y = the horizontal standard deviation of the plume (meters) [See Figure V-1, Page 48, Nuclear Safety, June 1961, Volume 2, Number 4, "Use of Routine Meteororological Observations for Estimating Atmospheric Dispersion," F.A. Gifford, Jr.] σ_z = the vertical standard deviation of the plume (meters) [See Figure V-2, Page 48, Nuclear Safety, June 1961, Volume 2, Number 4, "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," F.A. Gifford, Jr.]	
(2) For time periods of greater than 8 hours the plume should be assumed to meander and spread uniformly over a 22.5° sector. The resultant equation is: $\chi/Q = \frac{2.032}{\sigma_z ux}$	2g.2 See response to 2g.1 above.

TABLE 15.6-7 (Sheet 6)

Regulatory Guide 1.4 Position	Design
Where x = distance from point of release to the receptor; other variables are as given in g(1).	
(3) The atmospheric diffusion model[2] for ground level releases is based on the information in the following table.	2g.3 See response to 2g.1 above.

Time Following Accident	Atmospheric Conditions
0-8 hours	Pasquill Type F, windspeed 1 meter/sec, uniform direction
8-24 hours	Pasquill Type F, windspeed 1 meter/sec, variable direction within a 22.5° sector
1-4 days	(a) 40% Pasquill Type D, windspeed 3 meter/sec (b) 60% Pasquill Type F, windspeed 2 meter/sec (c) Wind direction variable within a 22.5° sector
4-30 days	(a) 33.3% Pasquill Type C, windspeed 3 meter/sec (b) 33.3% Pasquill Type D, windspeed 3 meter/sec (c) 33.3% Pasquill Type F, windspeed 2 meter/sec (d) Wind direction 33.3% frequency in a 22.5° sector

(4) Figures 2A and 2B give the ground level release atmospheric diffusion factors based on the parameters given in g(3).	2g.4 See response to 2g.1 above.

TABLE 15.6-8

Radiological Consequences of a
Loss-of-Coolant-Accident

	Total Reported Doses (rem)		Regulatory Limits (rem)	
		TEDE		TEDE
I. Exclusion Area Boundary (~~0-2~~ hr) 0.4-2.4				
~~Thyroid~~	~~61.95~~	5.1	~~300~~	25
~~Whole body~~	~~1.18~~		~~25~~	
II. Low Population Zone Outer Boundary (0-30 day)				
~~Thyroid~~	~~95.59~~	4.6	~~300~~	25
~~Whole body~~	~~0.41~~		~~25~~	
III. Control Room (0-30 day)				
~~Thyroid~~	~~17.99~~	4.7	~~30~~	5
~~Whole body~~	~~0.15~~		~~5~~	
~~Beta-skin~~	~~3.29~~		~~30~~	

15.7 RADIOACTIVE RELEASE FROM A SUBSYSTEM OR COMPONENT

This class of accident can be caused by any of the following events:

a. Radioactive gas waste system leak or failure - this is an ANS Condition III event.

b. Radioactive liquid waste system leak or failure - this is an ANS Condition III event.

c. Postulated radioactive release due to liquid tank failures - this is an ANS Condition IV event.

d. Fuel handling accident - this is an ANS Condition IV event.

e. Spent fuel cask drop accidents - this is an ANS Condition III event.

All of the accidents in this section have been analyzed. It has been determined that the most severe radiological consequences will result from the fuel handling accident analyzed in Section 15.7.4.

15.7.1 RADIOACTIVE WASTE GAS DECAY TANK FAILURE

15.7.1.1 Identification of Causes

This accident is an infrequent fault. Its consequences are considered in this section. The accident is defined as an unexpected and uncontrolled release of radioactive xenon, and krypton fission product gases stored in a waste gas decay tank as a consequence of a failure of a single gas tank or associated piping.

iodine,

15.7.1.2 Sequence of Events and System Operations

During a refueling shutdown, the radioactive gases are stripped from the primary coolant and are stored in the gas decay tanks. After the transfer has been completed, the tank is assumed to fail. This releases all of the contents of the tank to the radwaste building. Also, since the tanks are isolated from each other, the only radioactivity released is from the failed tank. ~~For conservatism, the tank is assumed to fail after 40 years, releasing the peak inventory expected in the tank.~~

For conservatism, the gas decay tank inventory for all radionuclides except Kr-85 is assumed to be the maximum activity for each nuclide during the degassing operation. Further, the Kr-85 activity inventory is assumed to be the total activity released to the primary coolant during the fuel cycle.

15.7.1.3 Core and System Performance

This accident occurs when the reactor is in the shutdown condition. There is no impact on the core or its system performance.

15.7.1.4 Barrier Performance

The only barrier between the released activity and the environment is the radwaste building. During the course of this accident, the radwaste building is assumed to remain intact. This means that the only method of release is through the radwaste building ventilation system.

15.7.1.5 Radiological Consequences

15.7.1.5.1 Method of Analysis

15.7.1.5.1.1 Physical Model

Radioactive waste gas decay tanks are used in the design to permit the decay of radioactive gases as a means of reducing or preventing the release of radioactive materials to the atmosphere. To evaluate the radiological consequences of the gaseous waste processing system, it is postulated that there is an accidental release of the contents of one of the waste gas decay tanks resulting from a rupture of the tank or from another cause, such as operator error or valve malfunction. The gaseous waste processing system is so designed that the tanks are isolated from each other during use, limiting the quantity of gas released in the event of an accident by preventing the flow of radioactive gas between the tanks.

The principal radioactive components of the waste gas decay tanks are the noble gases krypton and xenon~~, the particulate daughters of some of the krypton and xenon isotopes,~~ and trace quantities of halogens. The maximum amount of waste gases stored in any one tank occurs after a refueling shutdown, at which time the waste gas decay tanks store the radioactive gases stripped from the reactor coolant.

and iodine

is

The maximum content of a gas decay tank which is conservatively assumed for the purpose of computing the noble gas inventory available for release given in Table 15.7-3. Rupture of the waste gas decay tank is assumed to occur immediately upon completion of the waste gas transfer, releasing the entire contents of the tank to the radwaste building. For the purposes of evaluating the accident, it is assumed that all the activity is released directly to the environment during the 2-hour period immediately following the accident.

15.7.1.5.1.2 Assumptions and Conditions

The major assumptions and parameters assumed in the analysis are itemized in Table 15.7-3.

In the evaluation of the waste gas decay tank rupture, the ~~fission product accumulation and~~ release assumptions of Regulatory Guide 1.24 have been used. Table 15.7-1 provides a comparison of the assumptions used in the analysis to those of Regulatory Guide 1.24. The assumptions related to the release of radioactive gases from the postulated rupture of a waste gas decay tank are:

a. The reactor has been operating at full core power with 1 percent defective fuel, and a shutdown to cold condition has been conducted prior to the accident.

and gaseous iodine

b. All noble gas ~~activity~~ has been removed from the reactor coolant system ~~and~~ transferred to the gas decay tank that is assumed to fail.

via the CVCS volume control tank

c. The maximum content of the waste gas decay tank was conservatively assumed to calculate the isotopic activities given in Table 15.7-3 ~~for the accumulated radioactivity in the gaseous waste processing system after 40 years' operation and immediately following plant shutdown.~~ The source term determination ~~does take~~ into account degassing of the reactor coolant system at shutdown.

takes

d. The failure is assumed to occur immediately upon completion of the waste gas transfer, releasing the entire contents of the tank to the radwaste building.

, with no holdup

e. The dose is calculated as if the release were from the radwaste building at ground level during the 2-hour period immediately following the accident. ~~No credit~~ for radioactive decay is taken.

C

within the radwaste building

Once the activity is released to the environment, no credit for radioactive decay is taken.

f. The control building and control room ventilation systems are configured in the normal power generation lineup at the time of the accident and remain in this configuration for the duration o a postulated tank failure event. That is, the control room ventilation isolation signal will not be actuated and consequently, the control room emergency ventilation system is n assumed to operate following the initiation of the accident.

15.7.1.5.1.3 Mathematical Models Used in the Analysis

The mathematical models used in the analysis are described in the followin sections:

a. The mathematical models used to analyze the activity released during the course of the accident are described in Appendix 15A.

b. The atmospheric dispersion factors used in the analysis were calculated based on the onsite meteorological measurement programs described in Section 2.3.

c. The ~~thyroid inhalation and total-body immersion~~ TEDE doses to a receptor at the exclusion area boundary, ~~or~~ outer boundary of the low-population zone, and control room ~~were~~ analyzed, using the models described in Appendix 15A~~, Sections 15A.2.4 and 15A.2.5, respectively~~.

15.7.1.5.1.4 Identification of Leakage Pathways and Resultant Leakage Activity

For the purposes of evaluating the radiological consequences due to the postulated waste gas decay tank rupture, the resultant activity is conservatively assumed to be released directly to the environment during the 2-hour period immediately following the occurrence of the accident. This is a considerably higher release rate than that based on the actual building exhaust ventilation rate. Therefore, the results of the analysis are based on the most conservative pathway available.

15.7.1.5.2 Identification of Uncertainties and Conservatisms in the Analysis

The uncertainties and conservatisms in the assumptions used to evaluate the radiological consequences of a waste gas decay tank rupture result from assumptions made involving the release of the waste gas from the decay tank and the meteorology present at the site during the course of the accident.

a. The accumulated activity ~~in the gaseous waste processing system after 40 years' operation and immediately~~ following plant shutdown ~~with zero decay assumed to be~~ in the waste gas decay tank is based on 1 percent failed fuel, which is ~~eight times~~ greater than that assumed under normal operating conditions.

b. It is assumed that the waste gas decay tank fails immediately after the transfer of ~~the noble~~ radioactive gases from the reactor coolant to the waste gas decay tank is complete. These assumptions result in the greatest amount of ~~noble gas~~ activity available for release to the environment.

c. The ~~noble gas~~ activity contained in the ruptured waste gas decay tank was assumed to be released over a 2-hour period immediately following the accident. This is a conservative assumption. If the contents of the tank were assumed to mix uniformly with the volume of air within the radwaste building where the decay tanks are located, then, using the actual building exhaust ventilation rate, a considerable amount of holdup time would

be gained. ~~However, no credit for radioactive decay is taken.~~ This reduces the amount of ~~noble gas~~ activity released to the environment due to natural decay. Also no credit for iodine removal by the non-safety grade radwaste building HVAC charcoal adsorbers has been taken.

d. The meteorological conditions which may be present at the site during the course of the accident are uncertain. However, it is highly unlikely that meteorological conditions assumed will be present during the course of the accident for any extended period of time. Therefore, the radiological consequences evaluated, based on the meteorological conditions assumed, will be conservative.

15.7.1.5.3 Conclusions

15.7.1.5.3.1 Filter Loading

Since the accumulated iodine activity in the waste gas decay tanks is negligible, filter loading due to a waste gas decay tank rupture does not challenge ~~establish~~ the necessary design margin for the radwaste building exhaust or the control room intake filters. Hence, the respective filter loadings were not evaluated.

15.7.1.5.3.2 Dose to Receptor at the Exclusion Area Boundary, ~~and~~ the Low-Population Zone Outer Boundary, and in the Control Room

The radiological consequences resulting from the occurrence of a postulated waste gas decay tank rupture have been conservatively analyzed using the ~~RATRAD~~ RADTRAD code, based on assumptions and models described in previous sections.

The analysis results show that the radiological consequences of the postulated waste gas decay tank failure do not exceed ~~a small fraction (i.e., 10 percent) of the exposure limits set forth in 10 CFR 100 for offsite doses and well below the dose acceptance criteria specified in SRP Section 6.4, and GDC-19 for the control room~~, even with no credit taken for the CREVS and its actuation instrumentation. The results are listed in Table 15.7-4

the limit set in 10 CFR 20 for offsite doses. The resultant control room dose is less than the limit set forth in 10 CFR 50.67

15.7.2 RADIOACTIVE LIQUID WASTE SYSTEM LEAK OR FAILURE

15.7.2.1 Identification of Causes

This is an infrequent fault because, although it is unlikely to happen, the potential for release of significant amounts of radioactivity is present. The accident may be caused by an equipment malfunction or tank failure or rupture.

15.7.2.2 Sequence of Events and System Operation

The radioactive liquid tank is assumed to fail. This releases a maximum of 80 percent of the tank capacity to the equipment compartment.

15.7.2.3 Core and System Performance

This accident does not affect the core or the core system performance.

15.7.2.4 Barrier Performance

There are no barriers to the release of radioactivity from the radwaste building.

15.7.2.5 Radiological Consequences

15.7.2.5.1 Method of Analysis

a

15.7.2.5.1.1 Physical Model

The liquid radwaste tanks are used as a means of collecting waste to be: 1) processed through the liquid radwaste system, 2) pumped to the Solid Radwaste System, or 3) discharged from the plant. To evaluate the radiological consequences of the liquid waste processing system, it is postulated that there is an accidental release of the contents of one of the tanks.

~~Table 11.1-6 provides an inventory and the concentrations of stored radioactivity in all the liquid tanks~~. In the analyses, it is assumed that the liquid contents of ~~the~~ tank are released to the radwaste building and, subsequently, the airborne activity is released to the environment during the 2-hour period immediately following the tank failure.

Two tanks have been analyzed for this accident, and the radiological consequences for both tanks are provided. The boron recycle holdup tank was selected because it contained the maximum total inventory~~, and therefore, the highest whole body exposures~~. In addition a hypothetical tank containing the maximum possible amount of iodine was analyzed ~~in order to determine the thyroid exposures~~. Although the primary spent resin tank contains the highest inventory of airborne and soluble iodine, it is considered extremely unlikely that all the iodine activity will become airborne in case the tank fails. The assumptions, conditions, and mathematical models described in this section are identical for both tanks, except as stated.

15.7.2.5.1.2 Assumptions and Conditions

The major assumptions and parameters assumed in this analysis are listed below and in Tables 15.7-5 and 15A-1:

a. The isotopic ~~inventory of the ruptured tank is taken from Table 11.1-6, and is~~ based on 1-percent failed fuel. The isotopic inventory of the ruptured hypothetical tank ~~is also based on 1-percent failed fuel, but~~ was calculated assuming that all of the iodine in the streams entering the liquid radwaste system are concentrated in a hypothetical tank where the only means of depletion is radioactive decay. These input streams are shown on USAR Figure 11.1A-2 sheets 2, 3, and 4.

inventories of the ruptured tanks are given in Table 15.7-5 and are

b. The tank failure is assumed to occur when the contents of the tank are at a maximum.

c. The doses are calculated as if the release were from the radwaste building at ground level during the 2-hour period immediately following the accident. ~~No~~ Credit is taken for radioactive decay during holdup in the tank ~~or in transit to the site boundary~~.

Once the activity is released to the environment, no credit for radioactive decay is taken in the transit to the site boundary, outer boundary of the low population zone, or control room.

d. ~~For the boron recycle holdup tank~~ 100 percent of all noble gas ~~activity in the tank is released while 10 percent of the~~ iodine activity ~~is~~ released as airborne activity. 100 percent of all iodine activity is released from the hypothetical liquid waste tank.

and

in the tank

e. Credit for iodine removal by non-safety grade radwaste building HVAC charcoal adsorber is not taken.

15.7.2.5.1.3 Mathematical Models Used in the Analysis

a. The mathematical models used to analyze the activity released during the course of the accident are described in Appendix 15A.

b. The atmospheric dispersion factors used in the analysis were calculated, based on the onsite meteorological measurement programs described in Section 2.3, and are provided in Table 15A-2.

TEDE

c. The ~~thyroid inhalation dose and total-body immersion dose~~ to a receptor at the exclusion area boundary, ~~or~~ outer boundary of the low-population zone were analyzed, using the models described in Appendix 15A, ~~Sections 15A.2.4 and 15A.2.5, respectively~~.

s

, and in the control room

15.7.2.5.1.4 Identification of Leakage Pathways and Resultant Leakage Activity

For the purposes of evaluating the radiological consequences due to the postulated liquid radwaste tank rupture, the resultant activity is conservatively assumed to be released directly to the environment during the 2-hour period immediately following the occurrence of the accident. This is a considerably higher release rate than that based on the actual building exhaust ventilation rate. Therefore, the results of the analysis are based on the most conservative pathway available.

15.7.2.5.2 Identification of Uncertainties and Conservatisms in the Analysis

The uncertainties and conservatisms in the assumptions used to evaluate the radiological consequences of the liquid radwaste tank rupture result from assumptions made involving the release of the radioactivity from the tanks and the meteorology assumed for the site.

a. It was assumed that the liquid radwaste tank fails when the inventory in the tank is a maximum. This assumption results in the greatest amount of activity available for release to the environment.

b. The contents of the ruptured tank are assumed to be released over a 2-hour period immediately following the accident. If the contents of the tank were assumed to mix uniformly with the volume of air within the radwaste building where the tanks are located, then, using the actual building exhaust ventilation rate, a considerable amount of holdup time would be gained. This reduces the amount of activity released to the environment due to the natural decay. Also, no credit for iodine removal by the radwaste building HVAC charcoal adsorbers is taken.

c. The meteorological conditions which may be present at the site during the course of the accident are uncertain. However, it is highly unlikely that meteorological conditions assumed will be present during the course of the accident for any extended period of time.

d. A tank is assumed to have collected liquid waste based on operation at 100-percent power with 1 percent failed fuel for an extended period of time, which is ~~eight times~~ higher than under normal operating conditions.

15.7.2.5.3 Conclusions

15.7.2.5.3.1 Filter Loadings

The filter loading due to a liquid radwaste tank rupture does not ~~establish~~ challenge the necessary design margin for the control room intake filters. Thus, the filter loading was not evaluated.

15.7.2.5.3.2 Doses to Receptor at the Exclusion Area Boundary, ~~and~~ the Low-Population Zone Outer Boundary, and in the Control Room

The radiological consequences resulting from the occurrence of a postulated liquid radwaste tank rupture have been conservatively analyzed, using assumptions and models described in previous sections.

TEDE doses

, and for 30 days in the control room

The ~~total-body dose due to immersion and the thyroid dose due to inhalation~~ have been analyzed for the 0-2-hour dose at the exclusion area boundary, ~~and~~ for the duration of the accident at the low-population zone outer boundary. The results are listed in Table 15.7-6. ~~The resultant dose is a small fraction (≤ 10 percent) of the exposure limits set forth in 10 CFR 100.~~

15.7.3 POSTULATED RADIOACTIVE RELEASE DUE TO LIQUID TANK FAILURES

This analysis is presented in Section 2.4.13.3.

15.7.4 FUEL HANDLING ACCIDENTS

The postulated fuel handling accident has been analyzed for two cases: Case 1, a fuel handling accident outside the containment, and Case 2, a fuel handling accident inside the reactor building.

15.7.4.1 Identification of Causes and Accident Description

pool

The accident is defined as the dropping of a spent fuel assembly onto the fuel storage area floor or refueling pool floor, resulting in the rupture of the cladding of all the fuel rods in the assembly despite many administrative controls and physical limitations imposed on fuel handling operations. All refueling operations are conducted in accordance with prescribed procedures.

15.7.4.2 Sequence of Events and Systems Operations

The first step in fuel handling is the achievement of plant cold safe shutdown of the reactor. After a radiation survey of the containment, the disassembly of the reactor vessel is started. After disassembly is complete, the first fuel handling is started. It is estimated that the earliest time to first fuel transfer after shutdown is 76 hours.

The postulated fuel handling accident is assumed to occur during a core offload at least 76 hours after shutdown in either the reactor containment building, or in the fuel building subsequent to the transfer of a fuel assembly through the fuel storage pool transfer gate and prior to placement in a fuel storage pool storage rack designated location.

15.7.4.3 Core and System Performance

As fuel damage occurs outside the reactor vessel in either the reactor containment building or fuel building, a postulated fuel handling accident does not impair the safe operation of the reactor or its associated systems.

15.7.4.4 Barrier Performance

A barrier between the released activity and the environment is the reactor building and the fuel building. Since these buildings are designed seismic Category I, it is safe to assume that during the course of a fuel handling

The resultant offsite doses are less than the limit set in 10 CFR 20. The resultant control room dose is less than the limit set forth in 10 CFR 50.67.

accident their integrity is maintained. This means that the pathway for release of radioactivity for a postulated accident in the fuel building is initially via auxiliary/fuel building normal exhaust system. After it is isolated on a high radiation signal, the release pathway is via the ESF emergency filtration system. For a postulated accident in the reactor building, since the containment personnel airlock (PAL) doors and equipment hatch are allowed to be open during core alterations or movement of irradiated fuels, portion of the gaseous effluent escaping from the refueling water pool in the Reactor Containment Building could be released to the environment via the open personnel and equipment hatches. The fuel storage pool and the refueling pool provide minimum decontamination factors of ~~100~~ 200 for iodine.

15.7.4.5 Radiological Consequences

15.7.4.5.1 Method of Analysis

15.7.4.5.1.1 Physical Model

The possibility of a fuel-handling accident is remote because of the many administrative controls and physical limitations imposed on the fuel-handling operations (refer to Section 9.1.4). All refueling operations are conducted in accordance with prescribed procedures.

When transferring irradiated fuel from the core to the fuel storage pool for storage, the reactor cavity and refueling pool are filled with borated water at a boron concentration equal to that in the fuel storage pool, which ensures subcritical conditions in the core even if all rod cluster control (RCC) assemblies were withdrawn. After the reactor head and rod cluster control drive shafts are removed, fuel assemblies are lifted from the core, transferred vertically to the refueling pool, placed horizontally in a conveyor car and pulled through the transfer tube and canal, upended and transferred through the fuel storage pool transfer gate, then lowered into steel racks for storage in the fuel storage pool in a pattern which precludes any possibility of a criticality accident.

Fuel-handling manipulators and hoists are designed so that the fuel cannot be raised above a position that provides an adequate water shield depth for radiation protection of operating personnel.

The containment, fuel building, refueling cavity, refueling pool, and fuel storage pool are designed to seismic Category I requirements, which prevent the structures themselves from failing in the event of a safe shutdown earthquake. The spent fuel storage racks are also designed to prevent any credible external missile from reaching the stored irradiated fuel. The fuel-handling manipulators, cranes, trollies, bridges, and associated equipment above the water cavities through which the fuel assemblies move are

In addition to the area radiation monitor located on the bridge over the fuel storage pool, portable radiation monitors capable of emitting audible alarms are located in this area during fuel-handling operations. The doors in the fuel building are closed to maintain controlled leakage characteristics in the fuel storage pool region during operations involving irradiated fuel. Should a fuel assembly be dropped in the canal or in the pool and release radioactivity above a prescribed level, the radiation monitors sound an alarm,

If one of the redundant discharge vent radiation monitors indicates that the radioactivity in the vent discharge is greater than the prescribed levels, an alarm sounds and the auxiliary/fuel building normal exhaust is switched to the ESF Emergency Exhaust system to allow the spent fuel pool ventilation to exhaust through the ESF charcoal filters to remove most of the halogens prior to discharging to the atmosphere via the unit vent. The supply ventilation system servicing the fuel storage pool area is automatically shut down, thus ensuring controlled leakage to the atmosphere through charcoal adsorbers (refer to Section 9.4).

The probability of a fuel-handling accident is very low because of the safety features, administrative controls, and design characteristics of the facility, as previously mentioned.

analysis

15.7.4.5.1.2 Assumptions and Conditions

guidelines

The major assumptions and parameters assumed in the analysis are itemized in Tables 15.7-7 and 15A-1.

1.183

Tables 15B-1 and 15B-3

In the evaluation of the fuel-handling accident, all the fission product release assumptions of Regulatory Guide ~~1.25~~ have been followed. ~~Table 15.7-2~~ provide~~s~~ a comparison of the ~~design~~ to the ~~requirements~~ of Regulatory Guide 1.183 ~~1.25~~. The following assumptions, related to the release of fission product gases from the damaged fuel assembly, were used in the analyses:

a. The dropped fuel assembly is assumed to be the assembly containing the peak fission product inventory. All the fuel rods contained in the dropped assembly are assumed to be damaged. In addition, ~~for the analyses for the accident in the reactor building~~ the dropped assembly is assumed to damage 20 percent of the rods of an additional assembly.

b. The assembly fission product inventories are based on a radial peaking factor of 1.65.

c. The accident occurs 76 hours after shutdown, which is the earliest time fuel-handling operations can begin. Radioactive decay of the fission product inventories was taken into account during this time period.

d. Only that fraction of the fission products which migrates from the fuel matrix to the gap and plenum regions during normal operation was assumed to be available for immediate release to the water following clad damage.

e. The gap activity released to the fuel pool from the damaged fuel rods consists of 10 percent of the total noble gases other than Kr-85, 30 percent of the Kr-85, and 10 percent of the total radioactive iodine other than I-131, 12 percent of the I-131, contained in the fuel rods at the time of the accident.

The gap fractions are obtained for high burnup fuel from Regulatory Guide 1.25 as modified by NUREG/CR-5009 to support the conservative assumption that 100 percent of the rods do not meet the burnup and kW/ft limits set forth in Footnote 11 of Regulatory Guide 1.183.

f. The pool decontamination factor is 1.0 for noble gases.

g. The effective pool decontamination factor is ~~100~~ 200 for iodine assuming at least 23 feet of water above the top of the damaged fuel assemblies is maintained in the pool. 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.

h. The iodine above the fuel pool is assumed to be composed of ~~75~~ 70 percent inorganic and ~~25~~ 30 percent organic species.

i. The activity which escapes from the pool is assumed to be available for release to the environment ~~in~~ over a time period of 2 hours.

j. No credit for decay or depletion during transit to the site boundary, ~~and~~ outer boundary of the low-population zone ~~is assumed~~, or in the control room.

k. No credit is taken for mixing or holdup in the fuel building atmosphere. ~~The filter efficiency for the ESF emergency filtration system is assumed to be 82.5 percent which is based on the assumption of the failure of the humidity control system.~~

l. ~~The fuel building is switched from the auxiliary/fuel building normal exhaust system to the ESF emergency exhaust system within one minute from the time the activity reaches the exhaust duct.~~ The activity released ~~before completion of the switchover~~ is assumed to be discharged directly to the environment with no credit for filtration or dilution.

m. For the inside the reactor building case, the containment personnel airlock doors and equipment hatch are assumed to be open at the time of the accident. For added conservatism, the gaseous effluent escaping from the refueling water pool in the Reactor Containment Building is assumed to be released immediately to the environment through the open personnel and equipment hatch and the adjacent Auxiliary Building without mixing in the surrounding atmosphere. The activity releases continue until the containment personnel airlock doors and equipment hatch are closed (assumed to be accomplished within two hours). The Auxiliary Building atmosphere is normally exhausted through filter absorbers designed to remove iodine. However, no credit is taken for iodine removal by the atmosphere filtration system filters. It is also assumed that no containment coolers or hydrogen mixing fans are operating and ~~99.99~~% of the activity escaping from the pool to the containment building is released to the environment over a two-hour period following the accident.

100

15.7.4.5.1.3 Mathematical Models Used in the Analysis

Mathematical models used in the analysis are described in the following sections:

a. The mathematical models used to analyze the activity released during the course of the accident are described in Appendix 15A~~, Section 15A.2~~.

b. The atmospheric dispersion factors are calculated, based on the onsite meteorological measurements programs described in Section 2.3 and are provided in Table 15A-2.

TEDE

c. The ~~thyroid inhalation and total-body immersion~~ doses to a receptor located at the exclusion area boundary, ~~and~~ outer boundary of the low population zone, and in the control room are described in Appendix 15A~~, Sections 15A.2.4 and 15A.2.5, respectively~~.

15.7.4.5.1.4 Identification of Leakage Pathways and Resultant Leakage Activity

For evaluating the radiological consequences due to the postulated fuel-handling accident in the fuel building and reactor building, the resultant activity is conservatively assumed to be released to the environment during the 0-2-hour period immediately following the occurrence of the accident. This is a considerably higher release rate than that based on the actual ventilation rate. Therefore, the results of the analysis are based on the most conservative pathway available.

15.7.4.5.2 Identification of Uncertainties and Conservatisms in Analysis

The uncertainties and conservatisms in the assumptions used to evaluate the radiological consequences of a fuel-handling accident result from assumptions made involving the amount of fission

product gases available for release to the environment and the meteorology present at the site during the course of the accident. The most significant of these assumptions are:

a. It is assumed in the analysis that all the fuel rods in the dropped assembly are damaged. This is a highly conservative assumption since, transferring fuel under strict fuel handling procedures, only under the worst possible circumstances could the dropping of a spent fuel assembly result in damage to all the fuel rods contained in the assembly.

b. The fission product gap inventory in a fuel assembly is dependent on the power rating of the assembly and the temperature of the fuel. It has been conservatively assumed that the core has been operating at 100 percent for the entire burnup period. ~~The gap activities are listed in Table 15A-3.~~

c. Iodine removal from the released fission product gas takes place as the gas rises to the pool surface through the body of liquid in the fuel storage and refueling water pools. The extent of iodine removal is determined by mass transfer from the gas phase to the surrounding liquid and is controlled by the bubble diameter and contact time of the bubble in the solution. The values used in the analysis result in a release of activity ~~approximately a factor of 5~~ greater than anticipated.

Radioactive material from the refueling water pool in the reactor containment building is assumed to be released directly to the environment through the open personnel and equipment hatch and the adjacent auxiliary building, without mixing in the surrounding atmosphere. Radioactive material is assumed to be released from the ~~auxiliary~~ building or from the fuel building ~~over~~ a two-hour time period.

reactor

The fuel building atmosphere filtration system has absorbers which normally filter any exhaust; however, credit for this system is not taken.

~~d. The ESF emergency filtration system charcoal filters are known to operate with at least a 99-percent efficiency. This means a further reduction in the iodine concentrations and thus a reduction in the thyroid doses at the exclusion area boundary and the outer boundary of the low-population zone for the fuel handling accident in the fuel building.~~

d. ~~e.~~ The containment purge exhaust system has charcoal adsorber units which filter any containment purge release. However, no credit has been taken for its capability (90-percent efficiency, minimum) since these units are not specifically designed to seismic Category I criteria. It is expected that for any event which would produce a catastrophic failure of the charcoal adsorber unit to the extent that its filtering capability would be negated

would also result in the purge exhaust fan becoming inoperable. Therefore, failure within the purge exhaust system would terminate any high volume release from the containment. In fact, the purge exhaust fan is considerably more likely to be inoperable following any postulated event than the failure of a passive charcoal adsorber unit. Thus, although no credit in the analysis has been given for the normal purge exhaust filters, any release prior to containment isolation would be filtered.

e. ~~f.~~ There is also conservatism in the time to first fuel transfer. Despite the fact that fuel could be transferred at 76 hours, it is probable that fuel handling will begin sometime later.

f. ~~g.~~ The meteorological conditions which may be present at the site during the course of the accident are uncertain. However, it is highly unlikely that meteorological conditions assumed will be present during the course of the accident for any extended period of time. Therefore, the radiological consequences evaluated, based on the meteorological conditions assumed, are conservative.

15.7.4.5.2.1 Filter Loadings

only

considered in the analysis which limits

~~The~~ ESF filtration systems ~~which function to limit~~ the consequences of a fuel-handling accident ~~in the fuel building are the ESF emergency filtration system and~~ the control room filtration system.

is

The activity loadings on the control room charcoal adsorbers as a function of time have been evaluated for the loss-of-coolant accident, Section 15.6.5. Since these filters are capable of accommodating the design basis LOCA fission product iodine loadings, more than adequate design margin is available with respect to postulated fuel-handling accident releases.

~~The activity loadings on the ESF filtration system charcoal adsorbers have been evaluated in accordance with Regulatory Guide 1.52, which limits the maximum loading to 2.5 mg of iodine per gram of activated charcoal.~~

15.7.4.5.2.2 Doses to Receptor at the Exclusion Area Boundary, ~~and~~ Low-Population Zone Outer Boundary, and in the Control Room

The potential radiological consequences resulting from the occurrence of a postulated fuel-handling accident occurring in the fuel building and in the reactor building have been conservatively

TEDE doses

analyzed, using assumptions and models described in previous sections. The ~~total-body dose due to immersion from direct radiation and the thyroid dose due to inhalation~~ have been analyzed for the 0-2-hour dose at the exclusion area boundary, ~~and~~ for the duration of the accident (0 to 2 hours) at the low-population zone outer boundary, and for 30 days in the control room. The results are listed in Table 15.7-8. The resultant doses are well within the guideline values of 10 CFR ~~100~~ 50.67.

15.7.5 SPENT FUEL CASK DROP ACCIDENTS

The design of the spent fuel cask handling equipment is such that no cask could be dropped more than the equivalent of 30 feet in the air. Therefore, no cask rupture will occur and thus no radioactivity will be released. Refer to Section 9.1.4 for a description of the spent fuel shipping procedures.

TABLE 15.7-1

DESIGN COMPARISON TO THE REGULATORY POSITIONS OF REGULATORY GUIDE 1.24 "ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL CONSEQUENCES OF A PRESSURIZED WATER REACTOR RADIOACTIVE GAS STORAGE TANK FAILURE" REVISION 0, DATED MARCH 23, 1972

Regulatory Guide 1.24 Position	Design
1. The assumptions related to the release of radioactive gases from the postulated failure of a gaseous waste storage tank are:	
a. The reactor has been operating at full power with one percent defective fuel and a shutdown to cold condition has been conducted near the end of an equilibrium core cycle. As soon as possible after shutdown, all noble gases have been removed from the primary cooling system and transferred to the gas decay tank that is assumed to fail.	1.a Complies.
b. The maximum content of the decay tank assumed to fail should be used for the purpose of computing the noble gas inventory in the tank. Radiological decay may be taken into account in the computation only for the minimum time period required to transfer the gases from the primary system to the decay tank.	1.b Complies.
c. The failure is assumed to occur immediately upon completion of the waste gas transfer, releasing the entire contents of the tank to the building. The assumption of the release of the noble gas inventory from only a single tank is based on the premise that all gas decay tanks will be isolated from each other whenever they are in use.	1.c Complies.

TABLE 15.7-1 (Sheet 2)

no changes

Regulatory Guide 1.24 Position	Design
d. All of the noble gases are assumed to leak out of the building at ground level over a 2-hour time period.	1.d Complies.
2. The atmospheric diffusion assumptions for ground level releases are:	2. Short-term atmospheric dispersion factors corresponding to a ground level release and accident conditions were calculated based on on-site meteorological measurement programs described in Section 2.3. The dispersion factors are in compliance with the methodology described in Regulatory Guide 1.145 and represent the worst of the 5 percent overall site meteorology and the 0.5 percent worst sector meteorology.
a. The basic equation for atmospheric diffusion from a ground level point source is: $\chi/Q = \frac{1}{\pi u \sigma_y \sigma^z}$ Where: χ = the short term average centerline value of the ground level concentration (curies/m^3) Q = amount of material released (curies/sec) u = windspeed (meters/sec) σ_y = the horizontal standard deviation of the plume (meters) [See Figure V-1, page 48, Nuclear Safety, June 1961, Volume 2, Number 4, "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," F. A. Gifford, Jr.]	

TABLE 15.7-1 (Sheet 3)

Regulatory Guide 1.24 Position	Design
σ_z = the vertical standard deviation of the plume (meters) [See Figure V-2, page 48, Nuclear Safety, June 1961, Volume 2, Number 4, "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," F. A. Gifford, Jr.]	
b. For ground level releases, atmospheric diffusion factors[1] used in evaluating the radiological consequences of the accident addressed in this guide are based on the following assumptions: (1) windspeed of 1 meter/sec; (2) uniform wind direction (3) Pasquill diffusion category F.	
c. Figure 1 is a plot of atmospheric diffusion factors (χ/Q) versus distance derived by use of the equation for a ground level release given in regulatory position 2.a. above under the meteorological conditions given in regulatory position 2.b. above.	
3. The following assumptions and equations may be used to obtain conservative approximations of external whole body dose from radioactive clouds:	3. ~~Dose factors given in Regulatory Guide 1.109 for noble gases and iodine thyroid dose factors; iodine whole body dose factors were calculated with 5 cm body tissued attenuation; see Table 15A-4.~~
a. External whole body doses are calculated using "Infinite Cloud" assumptions, i.e., the dimensions of the cloud are assumed to be large compared to the distances that the gamma rays and beta particles travel.	Does not comply. Analyses followed Regulatory Guide 1.183 for calculating total effective dose equivalent doses.

Delete

TABLE 15.7-2

DESIGN COMPARISON TO THE REGULATORY POSITIONS OF REGULATORY GUIDE 1.25 "ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT IN THE FUEL HANDLING AND STORAGE FACILITY FOR BOILING AND PRESSURIZED WATER REACTORS" REVISION 0, DATED MARCH 23, 1972

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
1. The assumptions[1] related to the release of radioactive material from the fuel and fuel storage facility as a result of a fuel handling accident are:		
a. The accident occurs at a time after shutdown identified in the technical specifications as the earliest time fuel handling operations may begin. Radioactive decay of the fission product inventory during the interval between shutdown and commencement of fuel handling operations is taken into consideration.	Complies, except the time after shutdown is identified in Section 9.1.4.2.3. Accident occurs 76 hours after shutdown.	Complies, except the time after shutdown is identified in Section 9.1.4.2.3. Accident occurs 76 hours after shutdown.
b. The maximum fuel rod pressurization[2] is 1200 psig.	Complies.	Complies.
c. The minimum water depth[2] between the top of the damaged fuel rods and the fuel pool surface is 23 feet.	Complies. Water depth is greater than 23 feet.	Complies. Water depth is greater than 23 feet.

TABLE 15.7-2 (Sheet 2)

	Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
d.	All of the gap activity in the damaged rods is released and consists of 10% of the total noble gases other than Kr-85, 30% of the Kr-85, and 10% of the total radioactive iodine in the rods at the time of the accident. For the purpose of sizing filters for the fuel handling accident addressed in this guide, 30% of the I-127 and I-129 inventory is assumed to be released from the damaged rods.	Complies.	Complies.
e.	The values assumed for individual fission product inventories are calculated assuming full power operation at the end of core life immediately preceding shutdown and such calculation should include an appropriate radial peaking factor. The minimum acceptable radial peaking factors are 1.5 for BWR's and 1.65 for PWR's.	Complies. A peaking factor of 1.65 is used.	Complies. A peaking factor of 1.65 is used.
f.	The iodine gap inventory is composed of inorganic species (99.75%) and organic species (.25%).	Complies.	Complies.
g.	The pool decontamination factors for the inorganic and organic species are 133 and 1, respectively, giving an overall effective decontamination factor of 100 (i.e., 99% of the total iodine	Complies.	Complies.

TABLE 15.7-2 (Sheet 3)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
released from the damaged rods is retained by the pool water). This difference in decontamination factors for inorganic and organic iodine species results in the iodine above the fuel pool being composed of 75% inorganic and 25% organic species.		
h. The retention of noble gases in the pool is negligible (i.e., decontamination factor of 1).	Complies. A decontamination factor of 1 is used.	Complies. A decontamination factor of 1 is used.
i. The radioactive material that escapes from the pool to the building is released from the building[3] over a 2-hour time period.	Complies. A 0-2 hour release from the pool to the building to the environment is assumed.	The containment shutdown purge lines are automatically isolated upon detection of high radioactivity in the containment. It is conservatively assumed that isolation does not occur until 25 seconds after the release. The containment minipurge lines are assumed to automatically isolate in less than 25 seconds after the release. Therefore, the greatest portion of the activity is contained in the reactor building following the event.
j. If it can be shown that the building atmosphere is exhausted through adsorbers designed to remove iodine, the removal efficiency is 90% for inorganic species and 70% for organic species.[4]	Not applicable; complies with Regulatory Guide 1.52 as described in Table 9.4-2.	No credit is taken for the normal purge filters.

TABLE 15.7-2 (Sheet 4)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
k. The effluent from the filter system passes directly to the emergency exhaust system without mixing5 in the surrounding building atmosphere and is then released (as an elevated plume for those facilities with stacks[6]).	Complies.	Complies.
2. The assumptions for atmospheric diffusion are:	Short-term atmospheric dispersion factors corresponding to ground level release and accident conditions were based on meteorological measurement programs described in Section 2.3. The dispersion factors are in compliance with the methodology described in Regulatory Guide 1.145 and represent the worst of the 5 percent overall site meteorology and the 0.5 percent worst sector meteorology.	
a. Ground Level Releases		
(1) The basic equation for atmospheric diffusion from a ground level point source is: $\chi/Q = \frac{1}{\pi u \sigma_Y \sigma_Z}$		
Where:		
χ = the short term average centerline value of the ground level concentration (curies/m^3)		
Q = amount of material released (curies/sec)		
u = windspeed (meters/sec)		
σ^Y = the horizontal standard deviation of the plume (meters) [See Figure V-1,		

TABLE 15.7-2 (Sheet 5)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
Page 48, Nuclear Safety, June 1961, Volume 2, Number 4, "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," F. A. Gifford, Jr.]		
σ_z = the vertical standard deviation of the plume (meters) [See Figure V-2, Page 48, Nuclear Safety, June 1961, Volume 2, Number 4, "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," F. A. Gifford, Jr.]		
(2) For ground level releases, atmospheric diffusion factors[7] used in evaluating the radiological consequences of the accident addressed in this guide are based on the following assumptions: (a) windspeed of 1 meter/sec; (b) uniform wind direction; (c) Pasquill diffusion category F.		
(3) Figure 1 is a plot of atmospheric diffusion factors (χ/Q) versus distance derived by use		

TABLE 15.7-2 (Sheet 6)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
of the equation for a ground level release given in regulatory position 2.a.(1) and under the meteorological conditions given in regulatory position 2.a.(2).		
(4) Atmospheric diffusion factors for ground level releases may be reduced by a factor ranging from one to a maximum of three (see Figure 2) for additional dispersion produced by the turbulent wake of the reactor building. The volumetric building wake correction as defined in Subdivision 3-3.5.2 of Meteorology and Atomic Energy-1968, is used with a shape factor of 1/2 and the minimum cross-sectional area of the reactor building only.		
b. Elevated Releases		
(1) The basic equation for atmospheric diffusion from an elevated release is: $\chi/Q = \frac{e^{-h^2/2\sigma_z^2}}{\pi u \sigma_y \sigma_z}$	Not applicable. Ground level releases were assumed.	Not applicable. Ground level releases were assumed.

Rev. 1

TABLE 15.7-2 (Sheet 7)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
Where:		
χ = the short term average centerline value of the ground level concentration (curies/m^3)		
Q = amount of material released (curies/sec)		
u = windspeed (meters/sec)		
σ_y = the horizontal standard deviation of the plume (meters) [See Figure V-1, Page 48, Nuclear Safety, June 1961, Volume 2, Number 4, "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," F. A. Gifford, Jr.]		
σ_z = the vertical standard deviation of the plume (meters) [See Figure V-2, Page 48, Nuclear Safety, June 1961, Volume 2, Number 4, "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," F. A. Gifford, Jr.]		
h = effective height of release (meters)		

TABLE 15.7-2 (Sheet 8)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
(2) For elevated releases, atmospheric diffusion factors[7] used in evaluating the radiological consequences of the accident addressed in this guide are based on the following assumptions: (a) windspeed of 1 meter/sec; (b) uniform wind direction; (c) envelope of Pasquill diffusion categories for various release heights; (d) a fumigation condition exists at the time of the accident.[8]		
(3) Figure 3 is a plot of atmospheric diffusion factors versus distance for an elevated release assuming no fumigation, and Figure 4 is for an elevated release with fumigation.		
(4) Elevated releases are considered to be at a height equal to no more than the actual stack height. Certain site conditions may exist, such as surrounding elevated topography or nearby structures, which will have the effect of reducing the effective stack height. The degree of stack height reduction will be evaluated on an individual case		

TABLE 15.7-2 (Sheet 9)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
3. The following assumptions and equations may be used to obtain conservative approximations of thyroid dose from the inhalation of radioiodine and external whole body dose from radioactive clouds:		
a. The assumptions relative to inhalation thyroid dose approximations are:	Complies. See Appendix 15A, Section 15A.2.4.	Complies. See Appendix 15A, Section 15A.2.4.
(1) The receptor is located at a point on or beyond the site boundary where the maximum ground level concentration is expected to occur.		
(2) No correction is made for depletion of the effluent plume of radioiodine due to deposition on the ground, or for the radiological decay or radioiodine in transit.		
(3) Inhalation thyroid doses may be approximated by use of the following equation: $D = \frac{F_q IFPBR(\chi/Q)}{(DF_p)(DF_f)}$		
Where:		
D = thyroid dose (rads)		

TABLE 15.7-2 (Sheet 10)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
F_g = fraction of fuel rod iodine inventory in fuel rod void space (0.1)		
I = core iodine inventory at time of accident (curies)		
F = fraction of core damaged so as to release void space iodine		
P = fuel peaking factor		
B = Breathing rate = 3.47 x 10^{-4} cubic meters per second (i.e., 10 cubic meters per 8 hour work day as recommended by the ICRP)		
DF_p = effective iodine decontamination factor for pool water		
DF_f = effective iodine decontamination factor for filters (if present)		
χ/Q = atmospheric diffusion factor at receptor location (sec/m^3)		

TABLE 15.7-2 (Sheet 11)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
R = adult thyroid dose conversion factor for the iodine isotope of interest (rads per curie). Dose conversion factors for Iodine 131-135 are listed in Table 1.[9] These values were derived from "standard man" parameters recommended in ICRP Publication 2.[10]		
TABLE 1 Adult Inhalation Thyroid Dose Conversion Factors Iodine Isotope — Conversion Factor (R) (Rads/curie inhaled) 131 — 1.48×10^6 132 — 5.35×10^4 133 — 4.0×10^5 134 — 2.5×10^4 135 — 1.24×10^5	Table 1; the thyroid dose conversion factors given in Regulatory Guide 1.109 are used.	Table 1; the thyroid dose conversion factors given in Regulatory Guide 1.109 are used.
b. The assumptions relative to external whole body dose approximations are:	Complies. See Appendix 15A, Section 15A.2.5.	Complies. See Appendix 15A, Section 15A.2.5.
(1) The receptor is located at a point on or beyond the site boundary where the maximum ground level concentration is expected to occur.		

TABLE 15.7-2 (Sheet 12)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
(2) External whole body doses are calculated using "Infinite Cloud" assumptions, i.e., the dimensions of the cloud are assumed to be large compared to the distances that the gamma rays and beta particles travel. The dose at any distance from the reactor is calculated based on the maximum ground level concentration at that distance. For an infinite uniform cloud containing curies of beta radioactivity per cubic meter, the beta dose rate in air at the cloud center is:[11] $\beta^{D'\infty} = 0.457\ \overline{E}_{\beta}\chi$ Where: $\beta^{D'\infty}$ = beta dose rate from an infinite cloud (rad/sec) $\overline{E}_{\beta}$ = average beta energy per disintegration (Mev/dis) χ = concentration of beta or gamma emitting isotope in the cloud (curie/m^3)	(2) The whole-body dose factors for gammas given in Regulatory Guide 1.109 are used; for iodines, the whole-body dose factors for gammas with credit for 5 cm body tissue attenuation are used. See Table 15A-4 for dose conversion factors.	

TABLE 15.7-2 (Sheet 13)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
Because of the limited range of beta particles in tissue, the surface body dose rate from beta emitters in the infinite cloud can be approximated as being one-half this amount or: $\beta^{D'\infty} = 0.23\ \overline{E}_{\beta}\chi$ For gamma emitting material the dose rate in tissue at the cloud center is: $\gamma^{D'\infty} = 0.507\ \overline{E}_{\gamma}\chi$ Where: $\gamma^{D'\infty}$ = gamma dose rate from an infinite cloud (rad/sec) $\overline{E}_{\gamma}$ = average gamma energy per disintegration (Mev/dis) However, because of the presence of the ground, the receptor is assumed to be exposed to only one-half of the cloud (semi-infinite) and the equation becomes: $\gamma^{D'} = 0.25\ \overline{E}_{\gamma}\chi$		

TABLE 15.7-2 (Sheet 14)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
Thus, the total beta or gamma dose to an individual located at the center of the cloud path may be approximated as: $\beta^{D\infty} = 0.23\ \bar{E}_\beta \Psi$ or $\gamma^{D\infty} = 0.23\ \bar{E}_\gamma \Psi$ Where Ψ is the concentration time integral for the cloud (curie sec/m^3).		
(3) The beta and gamma energies emitted per disintegration, as given in Table of Isotopes,[12] are averaged and used according to the methods described in ICRP Publication 2.		

Notes:

[1]The assumptions given are valid only for oxide fuels of the types currently in use and in cases where the following conditions are not exceeded:

a. Peak linear power density of 20.5 kW/ft for the highest power assembly discharged.

b. Maximum center-line operating fuel temperature less than 4500 F for this assembly.

TABLE 15.7-2 (Sheet 15)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
c. Average burnup for the peak assembly of 25,000 MWD/ton or less (this corresponds to a peak local burnup of about 45,000 MWD/ton).		

[2]For release pressures greater than 1200 psig and water depths less than 23 feet, the iodine decontamination factors will be less than those assumed in this guide and must be calculated on an individual case basis using assumptions comparable in conservatism to those of this guide.

[3]The effectiveness of features provided to reduce the amount of radioactive material available for release to the environment will be evaluated on an individual case basis.

[4]These efficiencies are based upon a 2-inch charcoal bed depth with 1/4 second residence time. Efficiencies may be different for other systems and must be calculated on an individual case basis.

[5]Credit for mixing will be allowed in some cases; the amount of credit will be evaluated on an individual case basis.

[6]Credit for an elevated release will be given only if the point of release is (a) more than two and one-half times the height of any structure close enough to affect the

TABLE 15.7-2 (Sheet 16)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
dispersion of the plume or (b) located far enough from any structure which could affect the dispersion of the plume. For those plants without stacks the atmospheric diffusion factors assuming ground level release given in regulatory position 2.b should be used.		
[7]These diffusion factors should be used until adequate site meteorological data are obtained. In some cases, available information on such site conditions as meteorology, topography and geographical location may dictate the use of more restrictive parameters to ensure a conservative estimate of potential offsite exposures.		
[8]For sites located more than 2 miles from large bodies of water such as oceans or one of the Greak Lakes, a fumigation condition is assumed to exist at the time of the accident and continue for 1/2 hour. For sites located less than 2 miles from large bodies of water a fumigation condition is assumed to exist at the time of the accident and continue for the duration of the release (2 hours).		

TABLE 15.7-2 (Sheet 17)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
[9]Dose conversion factors taken from "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, J. J. DiNunno, R. E. Baker, F. D. Anderson, and R. L. Waterfield (1962).		
[10]Recommendations of the International Commission on Radiological Protection, "Report of Committee II on Permissible Dose for Internal Radiation (1959)," ICRP Publication 2, (New York: Permagon Press, 1960).		
[11]Meteorology and Atomic Energy-1968, Chapter 7.		
[12]C. M. Lederer, J. M. Hollander, and I. Perlman, Table of Isotopes, Sixth Edition (New York: John Wiley and Sons, Inc. 1967).		

TABLE 15.7-3

PARAMETERS USED IN EVALUATING
THE RADIOLOGICAL CONSEQUENCES OF A
WASTE GAS DECAY TANK RUPTURE

I.	Source Data	
	a. Core power level, MWt	3637 (includes uncertainty) ~~3,565~~
	b. Failed fuel, percent	1
II.	Atmospheric Dispersion Factors	See Table 15A-2
III.	Activity Release Data	

Isotope	0-2 hr (Ci)	
I-131	~~4.02E-2~~	3.99E-2
I-132	~~0.00~~	2.30E-2
I-133	~~3.50E-2~~	4.28E-2
I-134	~~0.00~~	4.96E-3
I-135	~~1.39E-2~~	1.94E-2
Xe-131m	~~9.05E+2~~	1.07E+3
Xe-133m	~~1.14E+3~~	1.27E+3
Xe-133	~~7.09E+4~~	8.12E+4
Xe-135m	~~5.99E+1~~	5.97E+1
Xe-135	~~1.15E+3~~	1.02E+3
~~Xe-137~~	~~3.42E-1~~	
Xe-138	~~4.40~~	4.06E+0
~~Kr-83m~~	~~2.26E+1~~	
Kr-85m	~~1.85E+2~~	1.49E+2
Kr-85	~~4.75E+3~~	5.52E+3
Kr-87	~~3.64E+1~~	3.00E+1
Kr-88	~~2.25E+2~~	1.79E+2
~~Kr-89~~	~~1.76E-1~~	

TABLE 15.7-4

RADIOLOGICAL CONSEQUENCES OF A
WASTE GAS DECAY TANK RUPTURE

	Doses (rem) TEDE
Exclusion Area Boundary (0-2 hr)	9.0E-2
~~Thyroid~~	~~3.66E-03~~
~~Whole Body~~	~~1.31E-01~~
Low Population Zone Outer Boundary (duration)	2.9E-2
~~Thyroid~~	~~4.88E-04~~
~~Whole Body~~	~~1.74E-02~~
Control Room (30 days)	1.9E-2
~~Thyroid~~	~~1.45E-02~~
~~Whole Body~~	~~2.16E-02~~
~~Beta Skin~~	~~6.23E-01~~

TABLE 15.7-5

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LIQUID RADWASTE TANK FAILURE

I.	Source Data	
	a. Core power level, MWt	3637 (includes uncertainty) ~~3,565~~
	b. Failed fuel, percent	1
II.	Atmospheric Dispersion Factors	See Table 15A-2
III.	Activity Release Data	
	a. Noble gas activity, percent of tank contents	100
	b. Iodine gas activity, percent of tank contents	
	1. Boron recycle holdup tank	~~10~~ 100
	2. Hypothetical Liquid Waste tank	100
	~~c. Tank contents subject to release~~	
	~~1. Boron recycle holdup tank~~	~~Table 11.1-6 (sheet 13)~~
	~~2. Primary evaporator bottoms tank~~	~~Table 11.1-6 (sheet 16)~~
	~~d.~~ c. Activity released to the environment	
	1. Boron recycle holdup tank	

Isotope	0-2 hr (Ci)
I-130	6.36E-3
~~I-131~~	~~4.96E-1~~ 5.83E+0
I-132	~~7.90E-3~~ 8.52E-2
I-133	~~1.13E-1~~ 1.15E+0
I-134	~~7.25E-4~~ 7.04E-3
I-135	~~2.09E-2~~ 2.07E-1
Xe-131m	~~3.35E+2~~ 3.67E+2
Xe-133m	~~1.40E+2~~ 1.44E+2
Xe-133	~~1.68E+4~~ 1.79E+4
Xe-135m	~~1.06E-1~~ 1.09E-1
Xe-135	~~4.40E+1~~ 3.64E+1
~~Xe-137~~	~~6.97E-3~~
Xe-138	~~9.38E-2~~ 8.74E-2
~~Kr-83m~~	~~5.02E-1~~
Kr-85m	~~4.93~~ 4.00E+0
Kr-85	~~1.59E+3~~ 1.69E+3
Kr-87	~~9.06E-1~~ 7.34E-1
Kr-88	~~5.80~~ 4.49E+0
~~Kr-89~~	~~3.10E-3~~

TABLE 15.7-5 (Sheet 2)

2. Hypothetical Liquid Waste ~~Primary evaporator bottoms~~ tank

Isotope	0-2 hr (Ci)	
I-130		2.75E-3
~~I-131~~	~~1.92E+01~~	2.75E+1
I-132	~~9.939E-24~~	7.10E-3
I-133	~~7.464E-03~~	8.07E-1
I-134	~~3.192E-61~~	2.26E-4
I-135	~~7.190E-09~~	4.88E-2
Xe-131m	0.000E+00	
Xe-133m	0.000E+00	
Xe-133	0.000E+00	
Xe-135m	0.000E+00	
Xe-135	0.000E+00	
~~Xe-137~~	~~0.000E+00~~	
Xe-138	0.000E+00	
~~Kr-83m~~	~~0.000E+00~~	
Kr-85m	0.000E+00	
Kr-85	0.000E+00	
Kr-87	0.000E+00	
Kr-88	0.000E+00	
~~Kr-89~~	~~0.000E+00~~	

TABLE 15.7-6

RADIOLOGICAL CONSEQUENCES OF A
LIQUID RADWASTE TANK FAILURE

	Dose (rem)	TEDE
Boron Recycle Tank		
Exclusion Area Boundary (0-2 hr)		
~~Thyroid~~	~~4.01E-2~~	2.8E-2
~~Whole-body~~	~~2.48E-2~~	
Low Population Zone Outer Boundary (duration)		
~~Thyroid~~	~~5.35E-3~~	8.8E-3
~~Whole-body~~	~~3.31E-3~~	
Control Room (30 days)		5.5E-2
Hypothetical Liquid Waste Tank		
Exclusion Area Boundary (0-2 hr)		
~~Thyroid~~	~~1.49E+00~~	5.0E-2
Low Population Zone Outer Boundary (duration)		
~~Thyroid~~	~~2.00E-1~~	1.6E-2
Control Room (30 days)		2.4E-1

TABLE 15.7-7

PARAMETERS USED IN EVALUATING
THE RADIOLOGICAL CONSEQUENCES OF A
FUEL-HANDLING ACCIDENT *

	~~In Fuel Building~~	~~In Reactor Building~~
I. Source Data		
a. Core power level, MWt	~~3,565~~ 3637 (includes uncertainty)	~~3,565~~
b. Radial peaking factor	1.65	~~1.65~~
c. Decay time, hours	76	~~76~~
d. Number of fuel assemblies affected	~~1.0~~	1.2
e. Fraction of fission product gases contained in the gap region of the fuel assembly	Per R.G. 1.25 and NUREG/CR-5009	~~Per R.G. 1.25 and NUREG/CR-5009~~
II. Atmospheric Dispersion Factors	See Table 15A-2	~~See Table 15A-2~~
III. Activity Release Data		
a. Percent of affected fuel assemblies gap activity released	100	~~100~~
b. Pool decontamination factors		
1. Iodine	~~100~~ 200	~~100~~
2. Noble gas	1	~~1~~
~~c. Filter efficiency, percent~~	~~82.5*~~	~~0~~
c. ~~d.~~ Building mixing volumes assumed, percent of total volume	0	~~0~~
d. ~~e.~~ Activity release period, hrs	2	~~2~~

~~*NOTE: The postulated fuel handling accident in the Fueling Building was analyzed with a reduced filter efficiency, based upon the single failure assumption that one of the emergency Exhaust Filter-Adsorber units is operating with a failed heater or humidistat.~~

*Note: The parameter values are consistent to bound the release pathway from both the fuel building and reactor building.

TABLE 15.7-7 (Sheet 2)

h. Activity released to the environment

Isotope	~~Fuel Building~~ ~~Building~~ 0-2 hr (Ci)	~~Reactor~~ ~~0-2 hr (Ci)~~
~~I-131~~	~~1.84E+2~~	~~8.86E+2~~
~~I-133~~	~~3.28E+1~~	~~1.58E+2~~
~~Xe-131m~~	~~7.18E+2~~	~~8.61E+2~~
~~Xe-133m~~	~~1.90E+3~~	~~2.28E+3~~
~~Xe-133~~	~~1.10E+5~~	~~1.32E+5~~
~~Xe-135~~	~~1.22E+2~~	~~1.45E+2~~
~~Kr-85~~	~~2.62E+3~~	~~3.14E+3~~
I-130	1.44E-1	
I-131	4.89E+2	
I-132	3.91E+2	
I-133	8.81E+1	
I-135	3.39E-1	
Xe-131m	1.07E+3	
Xe-133m	3.48E+3	
Xe-133	1.62E+5	
Xe-135m	1.10E+1	
Xe-135	1.61E+3	
Kr-85m	2.18E-1	
Kr-85	3.38E+3	

TABLE 15.7-8

RADIOLOGICAL CONSEQUENCES OF A
FUEL HANDLING ACCIDENT

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	Dose (rem)	TEDE
~~In Fuel Building~~		
Exclusion Area Boundary (0-2 hr)		
~~Thyroid~~	~~1.48E+1~~	1.1E+0
~~Whole-body~~	~~1.72E-1~~	
Low Population Zone Outer Boundary (duration)		
~~Thyroid~~	~~1.97E+0~~	3.5E-1
~~Whole-body~~	~~2.30E-2~~	
~~In Reactor Building~~		
~~Exclusion Area Boundary (0-2 hr)~~		
~~Thyroid~~	~~7.09E+1~~	
~~Whole-body~~	~~2.04E-1~~	
~~Low Population Zone Outer Boundary (duration)~~		
~~Thyroid~~	~~9.46E+0~~	
~~Whole-body~~	~~2.72E-2~~	
Control Room (30 days) With emergency HVAC not in operation		4.0E+0

APPENDIX 15A

ACCIDENT ANALYSIS RADIOLOGICAL CONSEQUENCES EVALUATION MODELS AND PARAMETERS

15A.1 GENERAL ACCIDENT PARAMETERS

This section contains the parameters used in analyzing the radiological consequences of postulated accidents. Table 15A-1 contains the general parameters used in all the accident analyses. For parameters specific only to particular accidents, refer to that accident parameter section. The site specific, ground-level release, short-term dispersion factors (for accidents, ground-level releases are assumed) are based on Regulatory Guide 1.145 (Ref. 1) methodology and represent the worst of the 5 percent overall site meteorology and the 0.5 percent worst-sector meteorology and these are given in Table 15A-2 (see Section 2.3.4 for additional details on meteorology). The core ~~and gap~~ inventories are given in Table 15A-3. The ~~thyroid (via inhalation pathway), beta skin, and total-body (via submersion pathway)~~ dose factors based on References 2~~, 3a~~ and 3~~b~~ are given in Table 15A-4.

from CEDE (via inhalation) and EDE (via external exposure)

TEDE

15A.2 OFFSITE RADIOLOGICAL CONSEQUENCES CALCULATIONAL MODELS

and

This section presents the models ~~and equations~~ used ~~for~~ calculating the integrated activity released to the environment~~,~~ the accident flow paths~~, and the equations for dose calculations. Two major release models are considered: (1) a single holdup system with no internal cleanup and (2) a holdup system wherein a two-region spray model is used for internal cleanup.~~

15A.2.1 ACCIDENT RELEASE PATHWAYS

The release pathways for the major accidents are given in Figure 15A-1. The accidents and their pathways are as follows:

brought into

is calculated and

LOCA: ~~Immediately~~ F~~ollowing a postulated loss-of-coolant accident (LOCA), the release of radioactivity from the containment is to the environment with the containment spray and ESF systems ~~in full~~ operation. The release in this case is calculated ~~using Equation (8)~~ which takes into account a two-region spray model within the containment. The release of radioactivity to the environment due to assumed ESF system leakages in the auxiliary building will be via ESF filters ~~and is calculated using Equation (5)~~. ~~In addition,~~ T~~he release of radioactivity to the environment due to assumed ECCS boundary valves leakage through RWST is calculated ~~using Equation (11)~~.

In addition, the release of radioactivity to the environment due to assumed containment mini-purge is calculated.

The equations for dose calculations are consistent with those presented in References 4, 5, and 6 for the RADTRAD computer code.

over a two-hour period.

WGDTR: The activity release to the environment due to waste gas decay tank rupture (WGDTR) will be direct and unfiltered, with no holdup. The release pathway is A'-D. The total activity release in this case is therefore assumed to be the initial source activity itself.

FHA: The release to the environment due to a fuel handling accident (FHA) in the fuel building is via filters following the actuation of the emergency exhaust system. The release pathway is B-C-D. Since the release is calculated without any credit for holdup in the fuel building, the total release will be the unfiltered release ~~for the first minute plus the product of the initial activity and the filter nonremoval efficiency fraction (for noble gases, the nonremoval efficiency fraction is 1).~~ The release of radioactivity to the environment due to FHA inside the containment is direct and unfiltered, via the A'-D pathway, and occurs over a two-hour period (actually, the release is via the non-safety graded filters). ~~The release is calculated using Equation (8) based on a two-region spray model.~~

from the containment

(spray removal is not assumed)

CAE: Radioactivity release to the environment due to the control assembly ejection (CAE) accident is direct and unfiltered. The releases from the primary system are calculated ~~using equation 5 which~~ considers holdup in the ~~single-region~~ primary system ~~(the spray removal is not assumed)~~; the secondary (steam) releases via the relief valves are calculated with~~out any~~ holdup. The pathways for these releases are A-D and A'-D.

ing

(excluding noble gases)

MSLB, SGTR: Radioactivity releases to the environment due to main steam line break (MSLB) or steam generator tube rupture (SGTR) accidents are direct and unfiltered with no holdup via the A'-D pathway. The activity release calculations for these accidents are complex, involving spiking effects, time-dependent flashing fractions, and scrubbing of flashed activities; the release calculations are described in those sections that address these accidents.

~~15A.2.2 SINGLE-REGION RELEASE MODEL~~

~~It is assumed that any activity released to the holdup system instantaneously diffuses to uniformly occupy the system volume.~~

~~The following equations are used to calculate the integrated activity released from postulated accidents.~~

$$\cancel{A_1(0) = \text{initial source activity at time } t_0, \text{ Ci}}$$
$$\cancel{A_1(t) = \text{source activity at time t seconds, Ci}}$$
$$\cancel{A_1(t) = A_1(0)e^{-\lambda_1 t}} \qquad \cancel{(1)}$$

in the steam generators. Releases of noble gases are direct and unfiltered.

from the affected SG

The releases from the primary system into the intact steam generators are calculated considering holdup in the primary system (excluding noble gases); the secondary (steam) releases via the relief valves are calculated with holdup in the steam generators. Releases of noble gases are direct and unfiltered.

Locked Rotor: The releases from the primary system into the steam generator due to locked rotor are calculated considering holdup in the primary system (excluding noble gases); the secondary (steam) releases via the relief valves are calculated with holdup in the steam generators. Releases of noble gases are direct and unfiltered.

~~where~~ λ_1 = ~~total removal constant from primary holdup system, sec^{-1}~~

$$\lambda_1 = \lambda_d + \lambda_{1\ell} + \lambda_r \qquad (2)$$

~~where~~ λ_d = ~~decay removal constant, sec^{-1}~~

$\lambda_{1\ell}$ = ~~primary holdup leak or release rate, sec^{-1}~~

λ_r = ~~internal removal constant (i.e., sprays, plateout, etc.), sec^{-1}~~

~~Thus, the direct release rate to the atmosphere from the primary holdup system~~

$$R_u(t) = \lambda_{1\ell} A_1(t) \qquad (3)$$

$R_u(t)$ = ~~unfiltered release rate (Ci/sec)~~

~~The integrated activity release is the integral of the above equation.~~

$$IAR(t) = \int_0^t R_u(t) = \int_0^t \lambda_1 A_1(0) e^{-\lambda_1 t} \qquad (4)$$

~~This yields:~~

$$IAR(t) = (\lambda_{1\ell} A_1(0) / \lambda_1)(1 - e^{-\lambda_1 t}) \qquad (5)$$

~~15A.2.3a TWO-REGION SPRAY MODEL IN CONTAINMENT (LOCA)~~

~~A two-region spray model is used to calculate the integrated activity released to the environment. The model consists of a sprayed and unsprayed region in containment and a constant mixing rate between them.~~

~~As it is assumed that there are no sources after initial release of the fission products, the remaining processes are removal and transfer so that the multivolume containment is described by a system of coupled first-order differential equations of the form~~

$$\frac{dA_i}{dt} = -\sum_{j=1}^{K_i} -\lambda_{ij} A_i - \sum_{\ell=1}^{n-1} Q_{i\ell} \frac{A_i}{V_i} + \sum_{\ell=1}^{n-1} Q_{i\ell} \frac{A_i}{V_i} \qquad (6)$$

~~where~~

$\cancel{A_i}$ = ~~fission product activity in volume i, Ci~~

$\cancel{n}$ = ~~number of volumes considered in the model~~

$\cancel{Q_{il}}$ = ~~transfer rate from volume i to volume l, cc/sec~~

$\cancel{V_i}$ = ~~volume of the ith compartment, cc~~

$\cancel{l_{ij}}$ = ~~removal rate of the jth removal process in volume i, sec^{-1}~~

$\cancel{K_i}$ = ~~total number of removal processes in volume i~~

~~This system of equations is readily solved if the coefficients are known.~~

~~For a two-region model, the above system reduces to~~

$$\cancel{\frac{dA_1}{dt} = -\sum_{j=1}^{K_1} l_{1j}A_1 = Q_{12}\frac{A_1}{V_1} + Q_{21}\frac{A_2}{V_2}} \qquad \cancel{(6a)}$$

$$\cancel{\frac{dA_2}{dt} = -\sum_{j=1}^{K_1} l_{2j}A_2 = Q_{21}\frac{A_2}{V_2} + Q_{12}\frac{A_1}{V_1}} \qquad \cancel{(6b)}$$

~~Upon solving this coupled set of differential equations numerically, the release rate of activity is found from~~

$$\cancel{R(t) = \lambda_1 \ell\ A(t)} \qquad \cancel{(7)}$$

~~The integrated activity released from time~~ $\cancel{t_0}$ = $\cancel{t_1}$ ~~is shown in the following equation which is solved numerically.~~

$$\cancel{IAR = \int_{t_0}^{t_1} R(t)dt} \qquad \cancel{(8)}$$

~~15A.2.3b TWO-REGION RELEASE MODEL FOR LEAKAGE THROUGH RWST~~

~~It is assumed that the activity released to the holdup system (in this case, the containment recirculation sump) instantaneously diffuses to uniformly occupy the sump volume. Removal mechanisms from the sump include decay and release (i.e., leakage) to the RWST. The release rate from the RWST to the environment is given by~~

$$R_2(t) = f\ \lambda_{2\ell}\ A_2(t) \qquad (9)$$

~~where~~ $R_2(t)$ ~~= the unfiltered release rate from the RWST vent, Ci/sec~~

f ~~= assumed percent of radioiodine released to the RWST that becomes airborne~~

$\lambda_{2\ell}$ ~~= release rate constant for leakage from the RWST to the environment, based on an assumed leak rate from the sump that is uniformly mixed in the RWST volume, sec^{-1}~~

$A_2(t)$ ~~= RWST activity, Ci~~

~~The RWST activity can be calculated as~~

$A_2(t)$ ~~= Activity in RWST + Activity from sump - Activity released to environment~~

$$= A_2(t-\Delta t)e^{-\lambda_d \Delta t} + \lambda_{1\ell} A_1(t-\Delta t)\Delta t - R_2(t-\Delta t)\Delta t \qquad (10)$$

~~where~~ $\lambda_{1\ell}$ ~~= release rate constant for leakage from the uniformly mixed sump to the RWST, based on an assumed leak rate from the sump to the RWST, sec^{-1}~~

$A_1(t)$ ~~= containment sump activity, Ci~~

λ_d ~~= decay removal constant, sec^{-1}~~

~~The integrated release from the RWST is given by~~

$$IAR_2(t) = \int_0^t R_2(t)dt = f\ \lambda_{2\ell} \int_0^t A_2(t)dt \qquad (11)$$

~~and is calculated numerically by using Equation (10).~~

~~15A.2.4 OFFSITE THYROID DOSE CALCULATION MODEL~~

~~Offsite thyroid doses are calculated using the equation:~~

$$\cancel{D_{TH} = \sum_i DCF_{Thi} \sum_j (IAR)_{ij} \; (BR)_j \; (c/Q)_j \qquad (12)}$$

~~where~~

~~$(IAR)_{ij}$ = integrated activity of isotope i released* during the time interval j in Ci~~

~~and $(BR)_j$ = breathing rate during time interval j in meter^3/second~~

~~$(c/Q)_j$ = offsite atmospheric dispersion factor during time interval j in second/meter^3~~

~~$(DCF)^{Thi}$ = thyroid dose conversion factor via inhalation for isotope i in rem/Ci~~

~~DTh = thyroid dose via inhalation in rems~~

~~*No credit is taken for cloud depletion by ground deposition and radioactive decay during transport to the exclusion area boundary or the outer boundary of the low-population zone.~~

~~15A.2.5 OFFSITE TOTAL-BODY DOSE CALCULATIONAL MODEL~~

~~Assuming a semi-infinite cloud of gamma emitters, offsite total-body doses are calculated using the equation:~~

$$\cancel{D_{TB} = \sum_i DCF_{gi} \sum_j (IAR)_{ij} (c/Q)_j}$$

~~where~~

~~$(IAR)_{ij}$ = integrated activity of isotope i released* during the jth time interval in Ci~~

~~and $(c/Q)_j$ = offsite atmospheric dispersion factor during time interval j in second/meter^3~~

~~$(DCF)_{gi}$ = total-body gamma dose conversion factor for the ith isotope in rem-meter^3/Ci-sec~~

~~D_{TB} = total-body dose in rems~~

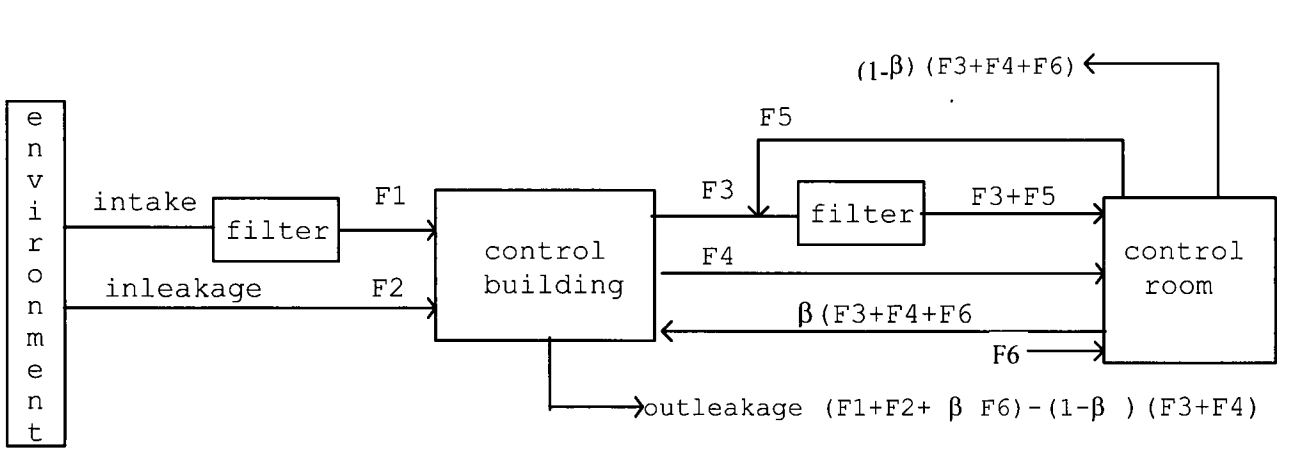
15A.3 CONTROL ROOM RADIOLOGICAL CONSEQUENCES CALCULATIONAL MODELS

~~Only radiation doses to a control room operator due to postulated LOCA are presented in this chapter since a study of the radiological consequences in the control room due to various postulated accidents indicate that the LOCA is the limiting case.~~

~~15A.3.1 INTEGRATED ACTIVITY IN CONTROL ROOM~~

Make-up air is brought into the control room via the control room filtration system which draws in air from the control building. Outside air is brought into the control building through safety grade filters via the control room pressurization fan. Some unfiltered air also may leak into the control building via an assumed inleakage rate. The activity concentrations at the control building intake for each time interval are found by multiplying the activity release to the environment by the appropriate c/Q for that time interval. The flow path model is shown below.

~~*No credit is taken for cloud depletion by ground deposition and radioactive decay during transport to the exclusion area boundary or the outer boundary of the low-population zone.~~



Once activity is brought into the control building, mixing within the control building is afforded by the control room pressurization fan. The control room filtration system fan takes air from the control building and the control room (recirculation) and discharges to the control room through the control room filtration safety grade filters. The radiological analysis input parameters are provided in Table 15A-1.

The control room ventilation isolation signal (CRVIS) starts both trains of the control room pressurization system and the control room filtration system. For the determination of dose to control room personnel, the worst single failure has been ascertained to be the failure of the filtration fan in one of the two filtration system trains.

Prior to operator action, a potential pathway would exist allowing air from the control building to enter the control room, bypassing the control room filtration filters. Operator action is required to ensure no bypass pathways then exist for unfiltered air to enter the control room.

Owning to this single failure of the control room filtration fan, the assumed failure of one of the two containment spray (CS) trains, and two of the four hydrogen mixing subsystem fans, inherent in the LOCA analysis parameters given in Table 15.6-6 should not be applied in this analysis. With both trains of CS and four hydrogen mixing fans operating, more volumetric coverage of the containment spray and more mixing between the new sprayed and unsprayed regions would be expected, thereby giving much greater ~~iodine~~ removal of iodine and particulates within the containment atmosphere. ~~However, the doses to control room personnel have been based on the LOCA analysis parameters given in Table 15.6-6.~~

~~The activity in the control building and control room is calculated by solving the following coupled set of first order differential equations.~~

$$\frac{dA_{CB}(t)}{dt} = \left[(1-\eta)F_1 + F_2\right]\chi/Q\left[R(t)\right] + \beta\lambda_{4\ell}A_{CR}(t) - \lambda_3 A_{CB}(t)$$

$$\frac{dA_{CR}(t)}{dt} = \left[(1-\eta)\lambda_{3f} + \lambda_{3u}\right]A_{CB}(t) - \lambda_4 A_{CR}(t) + F_6(\chi/Q)\left[R(t)\right]$$

~~where~~

$A_{CB}(t)$ = ~~activity in control building at time t, curies~~

$A_{CR}(t)$ = ~~activity in control room at time t, curies~~

η = ~~filter efficiency, fraction~~

F_1 = ~~filtered intake rate, meter^3/sec~~

F_2 = ~~unfiltered intake (inleakage), meter^3/sec~~

χ/Q = ~~atmospheric dispersion factor, sec/meter^3~~

$R(t)$ = ~~activity release rate in Ci/sec as given in Equation 3 of Section 15A.2.2 or Equations 7 and 9 of Sections 15A.2.3a and 15A2.3b~~

λ_3 = λ_d ~~+~~$\lambda_{3\ell}$ ~~+~~λ_{3f} ~~+~~λ_{3u} ~~, total removal rate from the control building, sec^{-1}~~

λ_d = ~~isotopic decay constant, sec^{-1}~~

$\lambda_{3\ell}$ = ~~outleakage to atmosphere from the control building (= (F1 + F2 + βF_6 = (1-β)(F3 +F4))/V_{CB} with V_{CB} being control building mixing volume in meter^3), sec^{-1}~~

λ_{3f} = ~~filtered flow from control building into control room (=F_3/V_{CB}, F_3 in meter^3/sec), sec^{-1}~~

λ_{3u} = ~~unfiltered flow from control building into control room (=F_4/V_{CB}, F_4 in meter^3/sec), sec^{-1}~~

λ_4 = $\lambda_d + \lambda_r + \lambda_{4\ell}$, total removal rate from the control room, sec^{-1}

λ_r = recirculation removal rate ($=\eta E_5/V_{CR}$ with E_5 being recirculation flow rate in $meter^3$/sec through filter with efficiency η and V_{CR} being control room volume in $meter^3$), sec^{-1}

$\lambda_{4\ell}$ = leakage to control building from the control room ($=[E_3 + E_4]/V_{CR}$), sec^{-1}

β = fraction of control room outleakage which returns to the control building mixing volume

E_6 = control room direct unfiltered intake, $meter^3$/sec

Upon solving this coupled set of differential equations, the integrated activity in the control room (IA_{CR}) is determined by the expression

$$IA_{CR(t)} = \int_0^t A_{CR}(t)\,dt$$

This $IA_{CR}(t)$ is used to calculate the doses to the operator in the control room. This activity is multiplied by an occupancy factor which accounts for the time fraction the operator is in the control room.

15A.3.2 CONTROL ROOM THYROID DOSE CALCULATIONAL MODEL

Control room thyroid doses via inhalation pathway are calculated using the following equation:

$$D_{Th-CR} = \frac{BR}{V_{CR}} \sum_i DCF_{Thi} \sum_j (IA_{CRij}) \times O_j$$

where

D_{Th-CR} = control room thyroid dose in rem

and

BR = breathing rate assumed to be always 3.47×10^{-4} $meter^3$/second

V_{CR} = volume of the control room in cubic meters

DCF_{Thi} = thyroid dose conversion factor for adult via inhalation in rem/Ci for isotope i

~~IA_{CRij} = integrated activity in control room in Ci-sec for isotope i during time interval j~~

~~O_j = control room occupancy fraction during time interval j~~

~~15A.3.3 CONTROL ROOM BETA-SKIN DOSE CALCULATIONAL MODEL~~

~~The beta-skin doses to a control room operator are calculated using the following equation:~~

$$\cancel{D_{b-CR} = \frac{1}{V_{CR}} \sum_i DCF_{bi} \sum_j (IA_{CRij}) \times O_j}$$

~~where D_{b-CR} and DCF_{bi} are the beta-skin doses in the control room in rem and the beta-skin dose conversion factor for isotope i in rem-meter^3/Ci-sec, respectively. The other symbols are explained in Section 15A.3.2.~~

~~15A.3.4 CONTROL ROOM TOTAL-BODY DOSE CALCULATION~~

~~Due to the finite structure of the control room, the total-body gamma doses to a control room operator will be substantially less than what they would be due to immersion in an infinite cloud of gamma emitters. The finite cloud gamma doses are calculated using Murphy's method (Ref. 4) which models the control room as a hemisphere. The following equation is used:~~

$$\cancel{D_{TB-CR} = \frac{1}{V_{CR}(GF)} \sum_i DCF_{gi} \sum_j (IA_{CRij}) \times O_j}$$

~~where~~

~~GF = dose reduction due to control room geometry factor~~

~~$GF = 1173/(V_1)^{0.338}$~~

~~V_1 = volume of the control room in cubic feet~~

~~D_{TB-CR} = total-body dose in the control room in rem,~~

~~and other quantities have been defined in subsections 15A.2.5 and 15A.3.2.~~

15A.3.1

~~15A.3.4.1~~ Model for Radiological Consequences Due to Radioactive Cloud External to the Control Room

This dose is calculated based on the semi-infinite cloud model which is modified using ~~the~~ protection factors ~~described in Section 7.5.4 of Reference 5~~ to account for the control room walls.

15A.4 REFERENCES

Rev. 1

1. USNRC Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," ~~August 1979~~. November 1982

~~2. USNRC Regulatory Guide 1.109, Rev. 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50 Appendix I," October 1977.~~

~~3a. Kocher, D.C., "Nuclear Decay Data for Radionuclides Occurring in Routine Releases from Nuclear Fuel Cycle Facilities," ORNL/NUREG/TM-102, August 1977.~~

~~3b. Berger, M.J., "Beta-Ray Dose in Tissue-Equivalent Material Immersed in a Radioactive Cloud," Health Physics, Vol. 26, pp. 1-12, January 1974.~~

~~4. Murphy, K.G. and Campe, K.M., "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19," Paper presented at the 13th AEC Air Cleaning Conference, August 1974.~~

~~5. "Meteorology and Atomic Energy 1968," D. H. Slade (ed.), USAEC Report, TID 24190, 1968.~~

2. Eckerman, K.F., et al., "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," Federal Guidance Report 11, EPA-520/1-88-020, Environmental Protection Agency, September 1988.

3. Eckerman, K.F. and Ryman, J.C., "External Exposure to Radionuclides in Air, Water, and Soil," Federal Guidance Report 12, EPA-402-R-93-081, Environmental Protection Agency, September 1993.

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

5. Supplement 1 to NUREG/CR-6604, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation," Bixler, N.E. and Erickson, C.M., June 1999.

6. Supplement 2 to NUREG/CR-6604, "RADTRAD: A Simplified Model for RADionuclide Transport And Removal And Dose Estimation," Arcieri, W.C., October 2002.

TABLE 15A-1

PARAMETERS USED IN ACCIDENT ANALYSIS

3637 (includes uncertainty)

I.	General	
	1. Core power level, Mwt	~~3565~~
	2. Full-power operation, days per cycle	~~510~~ 511
	3. Number of fuel assemblies in the core	193
	4. Maximum radial peaking factor	1.65
	5. Percentage of failed fuel	1.0
	6. Steam generator tube leak, lb/hr	500
II.	Sources	
	1. Core inventories, Ci	Table 15A-3
	~~2. Gap inventories, Ci~~	~~Table 15A-3~~
	2. ~~3.~~ Primary coolant specific activities, Ci/gm	Table 11.1-5
	4. ~~Primary coolant activity, technical specification limit for iodines - I-131 dose equivalent, µCi/gm~~	1.0
	5. Secondary coolant activity technical specification limit for iodines - I-131 dose equivalent, µCi/gm	0.1
III.	Activity Release Parameters	
	1. Free volume of containment, ft^3	2.5×10^6
	2. Containment leak rate	
	i. 0-24 hours, % per day	0.2
	ii. after 24 hrs, % per day	0.1
IV.	Control Room Dose Analysis (for LOCA)	
	1. Control building	
	i. Mixing volume, cf	239,000
	ii. Filtered intake, cfm	
	Prior to operator action (0-1.5 hours)	≥1350
	After operator action (1.5 hours-720 hours)	≥675
	iii. Unfiltered inleakage, cfm	≤~~300~~ 400
	iv. Filter efficiency (all forms of iodine), %	95

500

3. Primary coolant activity, technical specification limit for noble gas - Xe-133 dose equivalent, microCi/gm

TABLE 15A-1 (Sheet 2)

2. Control room	
i. Volume, cf	100,000
ii. Filtered flow from control building, cfm	≤550(*)
iii. Unfiltered flow from control building, cfm	
Prior to operator action	≤550(*)
(0-1.5 hours) After operator action (1.5 hours - 720 hours)	0
iv. Filtered recirculation, cfm	≥1250
v. Filter efficiency (all forms of iodine), %	95

(*)NOTE: Flows possible per train with two trains in operation. Each train is balanced for 400 cfm.

~~3. Total Flow Summary (Filtered plus unfiltered)~~	
~~i. 0-1.5 hours~~	
~~a. Control Room Pressurization~~	~~≥1650(**)~~
~~b. Control Room Filtration~~	~~≤1100~~
~~ii. 1.5 hours - 720 hours~~	
~~a. Control Room Pressurization~~	~~≥975(**)~~
~~b. Control Room Filtration~~	~~≤550~~

~~(**)NOTE: Includes 300 cfm of unfiltered Control Bldg inleakage.~~

~~4. Emergency Exhaust Filter Adsorber Unit Efficiency(all forms of iodine), %~~	~~90~~

V. Miscellaneous

1.Atmospheric dispersion factors, χ/Q sec/m^3		Table 15A-2
2.Dose conversion factors		
~~i. total body and beta skin, rem-meter^3/Ci-sec~~		Table 15A-4
~~ii. thyroid, rem/Ci~~		~~Table 15A-4~~
3.Breathing rates, meter^3/sec		
i. control room at all times	3.5	~~3.47~~ x 10^{-4}
ii. offsite		
0-8 hrs	3.5	~~3.47~~ x 10^{-4}
8-24 hrs	1.8	~~1.75~~ x 10^{-4}
24-720 hrs	2.3	~~2.32~~ x 10^{-4}
4.Control room occupancy fractions		
0-24 hrs		1.0
24-96 hrs		0.6
96-720 hrs		0.4

WOLF CREEK

TABLE 15A-2

LIMITING SHORT-TERM ATMOSPHERIC DISPERSION FACTORS
(χ/Qs) FOR ACCIDENT ANALYSIS
(sec/meter^3)

Location Type/ Time Interval (hrs)	χ/Q (Sec/Meters^3)	
Site boundary		
0-~~2~~ 720	~~1.5E-4~~	1.40E-4
Low-population zone		
0-2 ~~0-8~~	~~2.0E-5~~	4.50E-5
2-8 8-24	~~1.3E-5~~	2.39E-5
24-96	~~5.4E-6~~	1.29E-5
96-720	~~1.5E-6~~	5.49E-6
		1.61E-6
~~Control room (via containment leakage)~~		
~~0-8~~	~~5.3E-4~~	
~~8-24~~	~~3.6E-4~~	
~~24-96~~	~~6.6E-5~~	
~~96-720~~	~~0~~	
~~Control room (via unit vent exhaust)~~		
~~0-8~~	~~1.1E-4~~	
~~8-24~~	~~6.8E-5~~	
~~24-96~~	~~1.7E-5~~	
~~96-720~~	~~0~~	

Insert J

Insert J

Control room

Equipment Hatch (LOCA Containment Leakage, CAE Containment Leakage, FHA)

0 – 2	5.44E-4
2 – 8	4.35E-4
8 – 24	1.62E-4
24 – 96	1.22E-4
96 – 720	8.70E-5

Unit Vent Exhaust (LOCA Mini-Purge & ECCS Leakage, MSLB Faulted SG, Letdown Line Break)

0 – 2	6.12E-4
2 – 8	4.38E-4
8 – 24	1.79E-4
24 – 96	1.14E-4
96 – 720	8.94E-5

RWST Vent (LOCA RWST Backleakage, WGDTR, Liquid Waste Tank Failure)

0 – 2	6.80E-4
2 – 8	6.19E-4
8 – 24	2.27E-4
24 – 96	1.96E-4
96 – 720	1.53E-4

MSSV (MSLB Intact SGs, Loss of AC Power, Locked Rotor, CAE Primary-to-Secondary Leakage, SGTR)

0 – 2	1.04E-3
2 – 8	7.46E-4
8 – 24	3.03E-4
24 – 96	1.90E-4
96 – 720	1.39E-4

TABLE 15A-3

Core Inventory (Ci)

~~FUEL AND ROD GAP INVENTORIES - CORE (Ci)~~

Replace with Insert K

~~Core~~

~~Isotope~~	~~Fuel~~	~~Gap~~
~~I-131~~	~~9.46E+7~~	~~9.46E+6~~
~~I-132~~	~~1.37E+8~~	~~1.37E+7~~
~~I-133~~	~~1.95E+8~~	~~1.95E+7~~
~~I-134~~	~~2.15E+8~~	~~2.15E+7~~
~~I-135~~	~~1.83E+8~~	~~1.83E+7~~
~~Kr-83m~~	~~1.24E+7~~	~~1.24E+6~~
~~Kr-85m~~	~~2.67E+7~~	~~2.67E+6~~
~~Kr-85~~	~~1.02E+6~~	~~3.05E+5~~
~~Kr-87~~	~~5.16E+7~~	~~5.16E+6~~
~~Kr-88~~	~~7.28E+7~~	~~7.28E+6~~
~~Kr-89~~	~~8.94E+7~~	~~8.94E+6~~
~~Xe-131m~~	~~1.01E+6~~	~~1.01E+5~~
~~Xe-133m~~	~~6.06E+6~~	~~6.06E+5~~
~~Xe-133~~	~~1.95E+8~~	~~1.95E+7~~
~~Xe-135m~~	~~3.77E+7~~	~~3.77E+6~~
~~Xe-135~~	~~4.70E+7~~	~~4.70E+6~~
~~Xe-137~~	~~1.71E+8~~	~~1.71E+7~~
~~Xe-138~~	~~1.64E+8~~	~~1.64E+7~~

~~*Gap activity is assumed to be 10 percent of core activity for all isotopes except for Kr-85; for Kr-85 it is assumed to be 30 percent of the core activity. However, gap activity for I-131 is assumed to be 12% instead of 10% of the core activity for fuel handling accident, locked rotor accident and rod ejection accident analyses to account for extended burnup fuel.~~

Insert K

Isotope	Core Activity (Ci)	Isotope	Core Activity (Ci)
Kr-85m	2.69E+07	Sr-92	1.33E+08
Kr-85	1.10E+06	Ba-139	1.87E+08
Kr-87	5.30E+07	Ba-140	1.79E+08
Kr-88	7.12E+07	Ru-103	1.56E+08
Xe-131m	1.05E+06	Ru-105	1.08E+08
Xe-133m	6.06E+06	Ru-106	4.79E+07
Xe-133	2.01E+08	Rh-105	1.00E+08
Xe-135m	4.39E+07	Mo-99	1.91E+08
Xe-135	4.06E+07	Tc-99m	1.69E+08
Xe-138	1.80E+08	Ce-141	1.69E+08
I-130	1.98E+06	Ce-143	1.59E+08
I-131	1.01E+08	Ce-144	1.30E+08
I-132	1.49E+08	Pu-238	2.10E+05
I-133	2.10E+08	Pu-239	2.70E+04
I-134	2.36E+08	Pu-240	4.20E+04
I-135	2.00E+08	Pu-241	1.06E+07
Cs-134	1.65E+07	Np-239	1.94E+09
Cs-136	3.95E+06	Y-90	8.88E+06
Cs-137	1.11E+07	Y-91	1.30E+08
Cs-138	1.96E+08	Y-92	1.35E+08
Rb-86	1.86E+05	Y-93	1.52E+08
Te-127m	1.49E+06	Nb-95	1.76E+08
Te-127	9.09E+06	Zr-95	1.74E+08
Te-129m	5.04E+06	Zr-97	1.75E+08
Te-129	2.66E+07	La-140	1.85E+08
Te-131m	1.99E+07	La-142	1.64E+08
Te-132	1.45E+08	Nd-147	6.57E+07
Sb-127	9.23E+06	Pr-143	1.55E+08
Sb-129	2.85E+07	Am-241	9.94E+03
Sr-89	9.98E+07	Cm-242	2.82E+06
Sr-90	8.51E+06	Cm-244	2.11E+05
Sr-91	1.25E+08		

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TABLE 15A-4

DOSE CONVERSION FACTORS USED IN ACCIDENT ANALYSIS

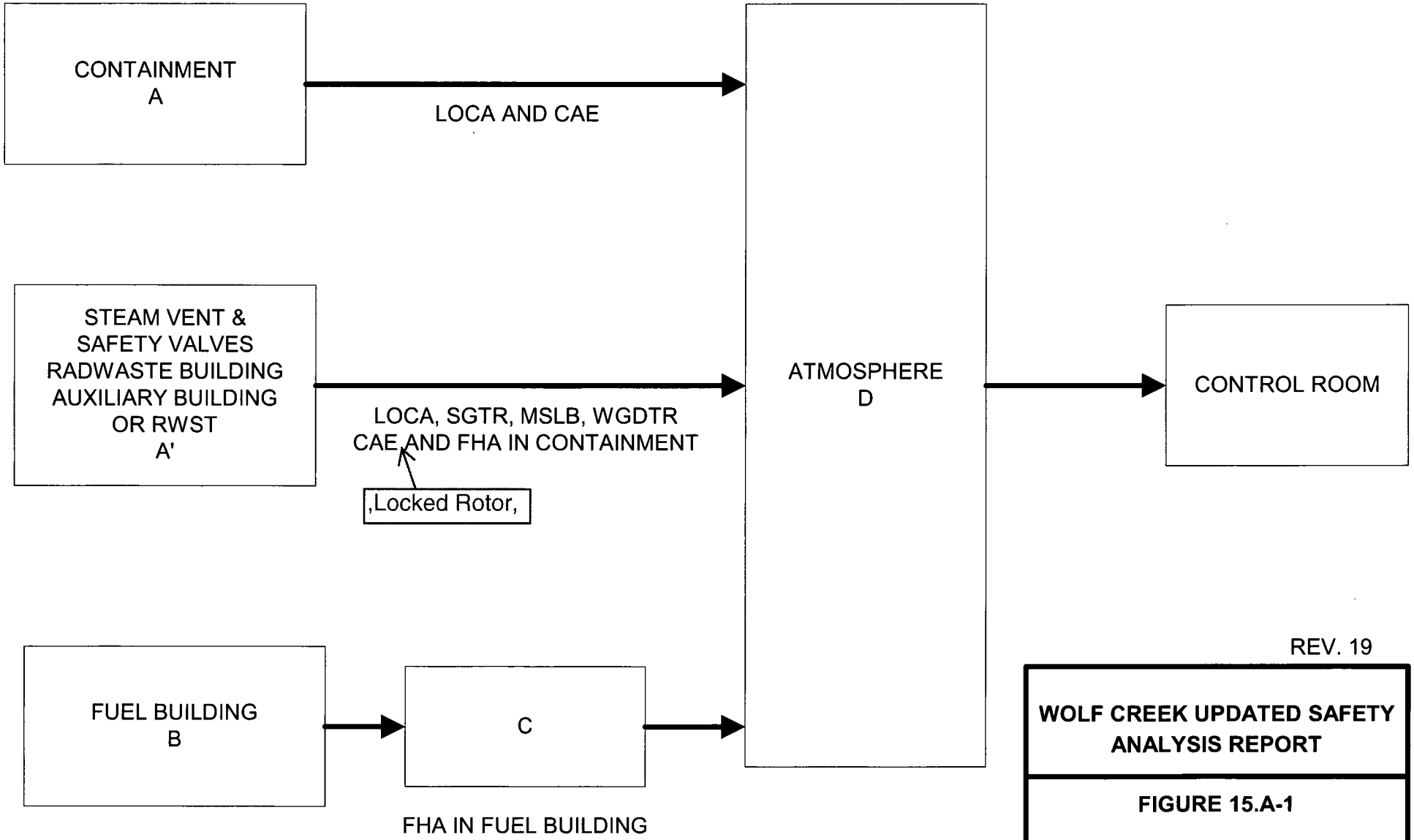
Replace with Insert L

~~Nuclide~~	~~Total Body Rem-meter^3 / Ci-sec~~	~~Beta Skin Rem-meter^3 / Ci-sec~~	~~Thyroid Rem/Ci~~
~~I-131~~	~~8.72E-2~~	~~3.17E-2~~	~~1.49E+6~~
~~I-132~~	~~5.13E-1~~	~~1.32E-1~~	~~1.43E+4~~
~~I-133~~	~~1.55E-1~~	~~7.35E-2~~	~~2.69E+5~~
~~I-134~~	~~5.32E-1~~	~~9.23E-2~~	~~3.73E+3~~
~~I-135~~	~~4.21E-1~~	~~1.29E-1~~	~~5.60E+4~~
~~Kr-83m~~	~~2.40E-6~~	~~0~~	~~NA~~
~~Kr-85m~~	~~3.71E-2~~	~~4.63E-2~~	~~NA~~
~~Kr-85~~	~~5.11E-4~~	~~4.25E-2~~	~~NA~~
~~Kr-87~~	~~1.88E-1~~	~~3.09E-1~~	~~NA~~
~~Kr-88~~	~~4.67E-1~~	~~7.52E-2~~	~~NA~~
~~Kr-89~~	~~5.27E-1~~	~~3.20E-1~~	~~NA~~
~~Xe-131m~~	~~2.91E-3~~	~~1.51E-2~~	~~NA~~
~~Xe-133m~~	~~7.97E-3~~	~~3.15E-2~~	~~NA~~
~~Xe-133~~	~~9.33E-3~~	~~9.70E-3~~	~~NA~~
~~Xe-135m~~	~~9.91E-2~~	~~2.25E-2~~	~~NA~~
~~Xe-135~~	~~5.75E-2~~	~~5.90E-2~~	~~NA~~
~~Xe-137~~	~~4.51E-2~~	~~3.87E-1~~	~~NA~~
~~Xe-138~~	~~2.80E-1~~	~~1.31E-1~~	~~NA~~

Insert L

Nuclide	CEDE (Sv/Bq)	EDE (Sv-m^3/Bq-sec)
Kr-85m	NA	7.48E-15
Kr-85	NA	1.19E-16
Kr-87	NA	4.12E-14
Kr-88	NA	1.02E-13
Xe-131m	NA	3.89E-16
Xe-133m	NA	1.37E-15
Xe-133	NA	1.56E-15
Xe-135m	NA	2.04E-14
Xe-135	NA	1.19E-14
Xe-138	NA	5.77E-14
I-130	7.14E-10	1.04E-13
I-131	8.89E-9	1.82E-14
I-132	1.03E-10	1.12E-13
I-133	1.58E-9	2.94E-14
I-134	3.55E-11	1.30E-13
I-135	3.32E-10	7.98E-14
Cs-134	1.25E-8	7.57E-14
Cs-136	1.98E-9	1.06E-13
Cs-137	8.63E-9	2.88E-14
Cs-138	2.74E-11	1.21E-13
Rb-86	1.79E-9	4.81E-15
Te-127m	5.81E-9	1.47E-16
Te-127	8.60E-11	2.42E-16
Te-129m	6.47E-9	1.55E-15
Te-129	2.42E-11	2.75E-15
Te-131m	1.73E-9	7.01E-14
Te-132	2.55E-9	1.03E-14
Sb-127	1.63E-9	3.33E-14
Sb-129	1.74E-10	7.14E-14
Sr-89	1.12E-8	7.73E-17
Sr-90	3.51E-7	7.53E-18
Sr-91	4.49E-10	3.45E-14
Sr-92	2.18E-10	6.79E-14
Ba-139	4.64E-11	2.17E-15
Ba-140	1.01E-9	8.58E-15

Nuclide	CEDE (Sv/Bq)	EDE (Sv-m^3/Bq-sec)
Ru-103	2.42E-9	2.25E-14
Ru-105	1.23E-10	3.81E-14
Ru-106	1.29E-7	0
Rh-105	2.58E-10	3.72E-15
Mo-99	1.07E-9	7.28E-15
Tc-99m	8.80E-12	5.89E-15
Ce-141	2.42E-9	3.43E-15
Ce-143	9.16E-10	1.29E-14
Ce-144	1.01E-7	8.53E-16
Pu-238	1.06E-4	4.88E-18
Pu-239	1.16E-4	4.24E-18
Pu-240	1.16E-4	4.75E-18
Pu-241	2.23E-6	7.25E-20
Np-239	6.78E-10	7.69E-15
Y-90	2.28E-9	1.90E-16
Y-91	1.32E-8	2.60E-16
Y-92	2.11E-10	1.30E-14
Y-93	5.82E-10	4.80E-15
Nb-95	1.57E-9	3.74E-14
Zr-95	6.39E-9	3.60E-14
Zr-97	1.17E-9	9.02E-15
La-140	1.31E-9	1.17E-13
La-142	6.84E-11	1.44E-13
Nd-147	1.85E-9	6.19E-15
Pr-143	2.19E-9	2.10E-17
Am-241	1.20E-4	8.18E-16
Cm-242	4.67E-6	5.69E-18
Cm-244	6.70E-5	4.91E-18



REV. 19

WOLF CREEK UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.A-1

RELEASE PATHWAYS

Insert Appendix 15B

Appendix 15B: Regulatory Guide 1.183, Revision 0 "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants" – Conformance Tables

NOTE: In Tables 15B-1 – 15B-7, the text shown in the "RG Position" columns is taken from Regulatory Guide 1.183. Therefore, references to footnotes, tables, and numbered references may be found in the regulatory guide.

Table 15B-1 Conformance with Regulatory Guide 1.183, Revision 0 Main Sections

Table 15B-2 Conformance with Regulatory Guide 1.183, Revision 0 Appendix A (Loss-of-Coolant-Accident)

Table 15B-3 Conformance with Regulatory Guide 1.183, Revision 0 Appendix B (Fuel Handling Accident)

Table 15B-4 Conformance with Regulatory Guide 1.183, Revision 0 Appendix E (PWR Main Steam Line Break)

Table 15B-5 Conformance with Regulatory Guide 1.183, Revision 0 Appendix F (PWR Steam Generator Tube Rupture Accident)

Table 15B-6 Conformance with Regulatory Guide 1.183, Revision 0 Appendix G (PWR Locked Rotor Accident)

Table 15B-7 Conformance with Regulatory Guide 1.183, Revision 0 Appendix H (PWR Rod Ejection Accident)

Table 15B-1 Conformance with Regulatory Guide 1.183, Revision 0 Main Sections

RG Section	RG Position	Analysis	Comments
3.1	The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty.[8] The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values.[9] The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 (Ref. 17) or ORIGEN-ARP (Ref. 18). Core inventory factors (Ci/MWt) provided in TID14844 and used in some analysis computer codes were derived for low burnup, low enrichment fuel and should not be used with higher burnup and higher enrichment fuels.	Conforms	The inventory of fission products in the reactor core and available for release to the containment was based on the maximum full power operation with a core thermal power of 3637 MWt (102% of 3565 MWt nominal power). Core design parameters (enrichment, burnup, and MTU loading) are based on the cycle 19 core design. Margin is added to the EOC core inventory, calculated with ORIGEN-S, to account for potential core design differences in future cycles. The magnitude of this margin is based on sensitivity studies that consider variations in enrichment and burnup.
3.1	For the DBA LOCA, all fuel assemblies in the core are assumed to be affected and the core average inventory should be used. For DBA events that do not involve the entire core, the fission product inventory of each of the damaged fuel rods is determined by dividing the total core inventory by the number of fuel rods in the core. To account for differences in power level across the core, radial peaking factors from the facility's core operating limits report (COLR) or technical specifications should be applied in determining the inventory of the damaged rods.	Conforms	For the DBA LOCA, all fuel assemblies were assumed to be affected and the core average inventory was used. A peaking factor of 1.65 was used for DBA events that do not involve the entire core (fuel handling accident, rod ejection, locked rotor), with fission product inventories for damages fuel rods determined by multiplying the total core inventory by the fraction of damaged rods.
3.1	No adjustment to the fission product inventory should be made for events postulated to occur during power operations at less than full rated power or those postulated to occur at the beginning of core life. For events postulated to occur while the facility is shutdown, e.g., a fuel handling accident, radioactive decay from the time of shutdown may be modeled.	Conforms	No adjustments for less than full power were made in any analysis. For the fuel handling accident, 76-hours of radioactive decay after shutdown was modeled.

Table 15B-1 Conformance with Regulatory Guide 1.183, Revision 0 Main Sections (cont.)

RG Section	RG Position	Analysis	Comments
3.2	The core inventory release fractions[10], by radionuclide groups, for the gap release and early in-vessel damage phases for DBA LOCAs are listed in Table 1 for BWRs and Table 2 for PWRs. These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1. **Table 2** **PWR Core Inventory Fraction Released Into Containment** **Group** / **Gap Release Phase** / **Early In-Vessel Phase** / **Total** Noble Gases 0.05 0.95 1.0 Halogens 0.05 0.35 0.4 Alkali Metals 0.05 0.25 0.3 Tellurium Metals 0.00 0.05 0.05 Ba, Sr 0.00 0.02 0.02 Noble Metals 0.00 0.0025 0.0025 Cerium Group 0.00 0.0005 0.0005 Lanthanides 0.00 0.0002 0.0002	Conforms	For the LOCA event, the core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage phases in Table 2 were utilized.
3.2	For non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in Table 3. The release fractions from Table 3 are used in conjunction with the fission product inventory calculated with the maximum core radial peaking factor. **Table 3[11]** **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	Conforms	For non-LOCA events, the fraction of the core inventory assumed to be in the gap by radionuclide group in Table 3 were utilized in conjunction with the maximum core radial peaking factor of 1.65. The control rod ejection accident was evaluated per Footnote 11 of RG 1.183 (the gap fractions are assumed to be 10% for iodines and noble gases). To account for possible damage to an assembly with high burnup and rod power and to address Footnote 11, the fuel handling accident used conservatively high gap fractions of 12% for I-131, 30% for Kr-85, and 10% for all other iodines and noble gases. These gap fractions were obtained from NUREG/CR-5009.

Table 15B-1 Conformance with Regulatory Guide 1.183, Revision 0 Main Sections (cont.)

RG Section	RG Position	Analysis	Comments
3.3	Table 4 tabulates the onset and duration of each sequential release phase for DBA LOCAs at PWRs and BWRs. The specified onset is the time following the initiation of the accident (i.e., time = 0). The early in-vessel phase immediately follows the gap release phase. The activity released from the core during each release phase should be modeled as increasing in a linear fashion over the duration of the phase.[12] For non-LOCA DBAs in which fuel damage is projected, the release from the fuel gap and the fuel pellet should be assumed to occur instantaneously with the onset of the projected damage. **Table 4** **LOCA Release Phases** **PWRs** / **BWRs** **Phase** / **Onset** / **Duration** / **Onset** / **Duration** Gap Release / 30 sec / 0.5 hr / 2 min / 0.5 hr Early In-Vessel / 0.5 hr / 1.3 hr / 0.5 hr / 1.5 hr	Conforms	The Table 4 PWR onset and durations for the DBA LOCA releases were utilized in the analysis. Note that the gap release was modeled beginning at 30 seconds and ending in the first half hour in order to model the early in-vessel release beginning at 0.5 hr.
3.3	For facilities licensed with leak-before-break methodology, the onset of the gap release phase may be assumed to be 10 minutes. A licensee may propose an alternative time for the onset of the gap release phase, based on facility-specific calculations using suitable analysis codes or on an accepted topical report shown to be applicable to the specific facility. In the absence of approved alternatives, the gap release phase onsets in Table 4 should be used.	Not Applicable	No additional delays in gap release were assumed for the DBA analyses.

Table 15B-1 Conformance with Regulatory Guide 1.183, Revision 0 Main Sections (cont.)

RG Section	RG Position	Analysis	Comments
3.4	Table 5 lists the elements in each radionuclide group that should be considered in design basis analyses. **Table 5** **Radionuclide Groups** **Group** — **Elements** Noble Gases — Xe, Kr Halogens — I, Br Alkali Metals — Cs, Rb Tellurium Group — Te, Sb, Se, Ba, Sr Noble Metals — Ru, Rh, Pd, Mo, Tc, Co Lanthanides — La, Zr, Nd, Eu, Nb, Pm, Pr Sm, Y, Cm, Am Cerium — Ce, Pu, Np	Conforms	The Table 5 elements in each radionuclide group were utilized in DBA analyses. Note that since RADTRAD is limited to modeling 63 nuclides, certain nuclides which were deemed to be insignificant from a dose perspective were not included.
3.5	Of the radioiodine released from the reactor coolant system (RCS) to the containment in a postulated accident, 95 percent of the iodine released should be assumed to be cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. This includes releases from the gap and the fuel pellets. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form. The same chemical form is assumed in releases from fuel pins in FHAs and from releases from the fuel pins through the RCS in DBAs other than FHAs or LOCAs. However, the transport of these iodine species following release from the fuel may affect these assumed fractions. The accident-specific appendices to this regulatory guide provide additional details.	Conforms	For releases from the reactor coolant system (RCS) to the containment, 95% of the iodine released was assumed to be cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. Fission products were assumed to be in particulate form with the exception of elemental and organic iodine and noble gases,

Table 15B-1 Conformance with Regulatory Guide 1.183, Revision 0 Main Sections (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
3.6	The amount of fuel damage caused by non-LOCA design basis events should be analyzed to determine, for the case resulting in the highest radioactivity release, the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and the fraction of fuel elements for which the fuel clad is breached. Although the NRC staff has traditionally relied upon the departure from nucleate boiling ratio (DNBR) as a fuel damage criterion, licensees may propose other methods to the NRC staff, such as those based upon enthalpy deposition, for estimating fuel damage for the purpose of establishing radioactivity releases.	Conforms	The amount of fuel damage caused by non-LOCA design basis events was analyzed. The conservatively calculated values were reflected in the rod ejection and locked rotor DBA analyses.
4.1.1	The dose calculations should determine the TEDE. TEDE is the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure. The calculation of these two components of the TEDE should consider all radionuclides, including progeny from the decay of parent radionuclides, that are significant with regard to dose consequences and the released radioactivity.[13]	Conforms	The dose calculations determine the TEDE and consider all radionuclides that are significant with regard to dose consequences. Progeny was not included in the dose calculations consistent with previously approved submittals, including: • Point Beach Units 1 & 2 – April 2011 (ADAMS Accession Number ML110240054) • Arkansas Nuclear One, Unit 2 – April 2011 (ADAMS Accession Number ML110980197)
4.1.2	The exposure-to-CEDE factors for inhalation of radioactive material should be derived from the data provided in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers" (Ref. 19). Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (Ref. 20), provides tables of conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the CEDE.	Conforms	CEDE Conversion factors for isotopes were taken from Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

Table 15B-1 Conformance with Regulatory Guide 1.183, Revision 0 Main Sections (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
4.1.3	For the first 8 hours, the breathing rate of persons offsite should be assumed to be 3.5 x 10^{-4} cubic meters per second. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.8 x 10-4 cubic meters per second. After that and until the end of the accident, the rate should be assumed to be 2.3 x 10^{-4} cubic meters per second.	Conforms	The breathing rates provided were utilized to calculate the offsite dose consequences. For determining a limiting 2-hour EAB dose, a constant breathing rate of 3.5 x 10^{-4} cubic meters per second was used.
4.1.4	The DDE should be calculated assuming submergence in semi-infinite cloud assumptions with appropriate credit for attenuation by body tissue. 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 may be used in lieu of DDE in determining the contribution of external dose to the TEDE. Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil" (Ref. 21), provides external EDE conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the EDE.	Conforms	EDE Conversion factors for isotopes were taken from Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil."
4.1.5	The TEDE should be determined for the most limiting person at the EAB. The maximum EAB TEDE for any two-hour period following the start of the radioactivity release should be determined and used in determining compliance with the dose criteria in 10 CFR 50.67.[14] The maximum two-hour TEDE should be determined by calculating the postulated dose for a series of small time increments and performing a "sliding" sum over the increments for successive two-hour periods. The maximum TEDE obtained is submitted. The time increments should appropriately reflect the progression of the accident to capture the peak dose interval between the start of the event and the end of radioactivity release (see also Table 6).	Conforms	The TEDE was determined for the most limiting person at the EAB. The maximum two-hour TEDE was determined by calculating the postulated dose for a series of small time increments and performing a "sliding" sum over the increments for successive two-hour periods. This was performed by the RADTRAD computer code with constant inputs for atmospheric dispersion factors and breathing rates.
4.1.6	TEDE should be determined for the most limiting receptor at the outer boundary of the low population zone (LPZ) and should be used in determining compliance with the dose criteria in 10 CFR 50.67.	Conforms	The TEDE was determined for the most limiting receptor at the outer boundary of the low population zone (LPZ).

Table 15B-1 Conformance with Regulatory Guide 1.183, Revision 0 Main Sections (cont.)

RG Section	RG Position	Analysis	Comments
4.1.7	No correction should be made for depletion of the effluent plume by deposition on the ground.	Conforms	No correction was made for the depletion of the effluent plume by deposition on the ground.
4.2.1	The TEDE analysis should consider all sources of radiation that will cause exposure to control room personnel. The applicable sources will vary from facility to facility, but typically will include: • Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility, • Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope, • Radiation shine from the external radioactive plume released from the facility, • Radiation shine from radioactive material in the reactor containment, • Radiation shine from radioactive material in systems and components inside or external to the control room envelope, e.g., radioactive material buildup in recirculation filters.	Conforms	The TEDE analysis considered all significant sources of radiation that would cause exposure to Control Room personnel. For WCGS, the limiting Control Room dose included: • Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility, • Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from the Control Building, • Radiation shine from the external radioactive plume released from the facility, • Radiation shine from radioactive material in the reactor containment, • Radiation shine from radioactive material in Control Room recirculation filters and radioactive material in the Control Building.
4.2.2	The radioactive material releases and radiation levels used in the control room dose analysis should be determined using the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values, unless these assumptions would result in nonconservative results for the control room.	Conforms	The radioactive material releases and radiation levels used in the Control Room dose analyses were determined using the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values.

Table 15B-1 Conformance with Regulatory Guide 1.183, Revision 0 Main Sections (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
4.2.3	The models used to transport radioactive material into and through the control room,[15] and the shielding models used to determine radiation dose rates from external sources, should be structured to provide suitably conservative estimates of the exposure to control room personnel.	Conforms	The models used to transport radioactive material into and through the Control Room, and the shielding models used to determine radiation dose rates from external sources, were developed to provide suitably conservative estimates of the exposure to Control Room personnel.
4.2.4	Credit for engineered safety features that mitigate airborne radioactive material within the control room may be assumed. Such features may include control room isolation or pressurization, or intake or recirculation filtration. Refer to Section 6.5.1, "ESF Atmospheric Cleanup System," of the SRP (Ref. 3) and Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Postaccident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" (Ref. 25), for guidance. The control room design is often optimized for the DBA LOCA and the protection afforded for other accident sequences may not be as advantageous. In most designs, control room isolation is actuated by engineered safeguards feature (ESF) signals or radiation monitors (RMs). In some cases, the ESF signal is effective only for selected accidents, placing reliance on the RMs for the remaining accidents. Several aspects of RMs can delay the control room isolation, including the delay for activity to build up to concentrations equivalent to the alarm setpoint and the effects of different radionuclide accident isotopic mixes on monitor response.	Conforms	Credit for engineered safety features that mitigate airborne radioactive material within the Control Room and Control Building were assumed as appropriate. Note that no credit for Control Room isolation was modeled for events that rely solely on radiation monitors.
4.2.5	Credit should generally not be taken for the use of personal protective equipment or prophylactic drugs. Deviations may be considered on a case-by-case basis.	Conforms	Credit was not taken for the use of personnel protective equipment or prophylactic drugs.

Table 15B-1 Conformance with Regulatory Guide 1.183, Revision 0 Main Sections (cont.)

RG Section	RG Position	Analysis	Comments
4.2.6	The dose receptor for these analyses is the hypothetical maximum exposed individual who is present in the control room for 100% of the time during the first 24 hours after the event, 60% of the time between 1 and 4 days, and 40% of the time from 4 days to 30 days.[16] For the duration of the event, the breathing rate of this individual should be assumed to be 3.5×10^{-4} cubic meters per second.	Conforms	The occupancy factors and breathing rate were utilized to determine the doses to the hypothetical maximum exposed individual who is present in the Control Room. Control Room χ/Q values were determined utilizing the ARCON96 computer code which does not incorporate occupancy factors. Occupancy factors were included in the RADTRAD computer code for the dose evaluations.
4.2.7	Control room doses should be calculated using dose conversion factors identified in Regulatory Position 4.1 above for use in offsite dose analyses. The DDE from photons may be corrected for the difference between finite cloud geometry in the control room and the semi-infinite cloud assumption used in calculating the dose conversion factors. The following expression may be used to correct the semi-infinite cloud dose, DDE_{∞}, to a finite cloud dose, DDE_{finite}, where the control room is modeled as a hemisphere that has a volume, V, in cubic feet, equivalent to that of the control room (Ref. 22). $DDE_{finite} = \frac{DDE_{\infty} V^{0.338}}{1173}$	Conforms	The DDE from photons was corrected for the difference between finite cloud geometry in the Control Room and the semi-infinite cloud assumption used in calculating the dose conversion factors by the given equation. This correction was performed by the RADTRAD computer code.
4.3	The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737 (Ref. 2). Design envelope source terms provided in NUREG-0737 should be updated for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of TEDE. Integrated radiation exposure of plant equipment should be determined using the guidance of Appendix I of this guide.	Conforms	Exception – The current TID-14844 accident source term will remain the licensing basis for equipment qualification and NUREG-0737 evaluations other than Control Room and Technical Support Center doses.

Table 15B-1 Conformance with Regulatory Guide 1.183, Revision 0 Main Sections (cont.)

RG Section	RG Position	Analysis	Comments
4.4	The radiological criteria for the EAB, the outer boundary of the LPZ, and for the control room are in 10 CFR 50.67. These criteria are stated for evaluating reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation, e.g., a large-break LOCA. The control room criterion applies to all accidents. For events with a higher probability of occurrence, postulated EAB and LPZ doses should not exceed the criteria tabulated in Table 6. The acceptance criteria for the various NUREG-0737 (Ref. 2) items generally reference General Design Criteria (GDC 19) from Appendix A to 10 CFR Part 50 or specify criteria derived from GDC-19. These criteria are generally specified in terms of whole body dose, or its equivalent to any body organ. For facilities applying for, or having received, approval for the use of AST, the applicable criteria should be updated for consistency with the TEDE criterion in 10 CFR 50.67(b)(2)(iii). **Table 6[17]** **Accident Dose Criteria** **Accident or Case** — **EAB and LPZ Dose Criteria** — **Analysis Release Duration** LOCA — 25 rem TEDE — 30 days for containment, ECCS, and MSIV (BWR) leakage BWR Main Steam Line Break — Instantaneous puff Fuel Damage or Pre-incident Spike — 25 rem TEDE Equilibrium Iodine Activity — 2.5 rem TEDE BWR Rod Drop Accident — 6.3 rem TEDE — 24 hours PWR Steam Generator Tube Rupture — Affected SG: time to isolate; Unaffected SG(s): until cold shutdown is established Fuel Damage or Pre-incident Spike — 25 rem TEDE Coincident Iodine Spike — 2.5 rem TEDE PWR Main Steam Line Break — Until cold shutdown is established Fuel Damage or Pre-incident Spike — 25 rem TEDE Coincident Iodine Spike — 2.5 rem TEDE PWR Locked Rotor Accident — 2.5 rem TEDE — Until cold shutdown is established PWR Rod Ejection Accident — 6.3 rem TEDE — 30 days for containment pathway; until cold shutdown is established for secondary pathway Fuel Handling Accident — 6.3 rem TEDE — 2 hours The column labeled "Analysis Release Duration' is a summary of the assumed radioactivity release durations identified in the individual appendices to this guide. Refer to these appendices for complete descriptions of the release pathways and durations.	Conforms	The DBAs were updated for consistency with the TEDE criterion in Table 6 for offsite doses and in 10 CFR 50.67(b)(2)(iii) for the Control Room and Technical Support Center doses.

Table 15B-1 Conformance with Regulatory Guide 1.183, Revision 0 Main Sections (cont.)

RG Section	RG Position	Analysis	Comments
5.1.1	The evaluations required by 10 CFR 50.67 are re-analyses of the design basis safety analyses and evaluations required by 10 CFR 50.34; they are considered to be a significant input to the evaluations required by 10 CFR 50.92 or 10 CFR 50.59. These analyses should be prepared, reviewed, and maintained in accordance with quality assurance programs that comply with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.	Conforms	The DBA analyses were prepared, reviewed, and maintained per 10 CFR 50 Appendix B and the guidance consistent with RG 1.183.
5.1.2	Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The single active component failure that results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences.	Conforms	Credit was taken for Engineered Safeguard Features with failure assumptions to maximize the calculated doses. Assumptions regarding the occurrence and timing of a loss of offsite power were also selected with the objective of maximizing the postulated radiological consequences.
5.1.3	The numeric values that are chosen as inputs to the analyses required by 10 CFR 50.67 should be selected with the objective of determining a conservative postulated dose. In some instances, a particular parameter may be conservative in one portion of an analysis but be nonconservative in another portion of the same analysis.	Conforms	The numeric values that were chosen as inputs to the analyses required by 10 CFR 50.67 were selected with the objective of determining a conservative postulated dose. For a range of values, the value that resulted in a conservative postulated dose was used.
5.1.4	Licensees should ensure that analysis assumptions and methods are compatible with the ASTs and the TEDE criteria.	Conforms	Licensee has ensured that analysis assumptions and methods are compatible with the AST and the TEDE criteria.

Table 15B-1 Conformance with Regulatory Guide 1.183, Revision 0 Main Sections (cont.)

RG Section	RG Position	Analysis	Comments
5.3	Atmospheric dispersion values (χ/Q) for the EAB, the LPZ, and the control room that were approved by the staff during initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified by this guide. Methodologies that have been used for determining χ/Q values are documented in Regulatory Guides 1.3 and 1.4, Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and the paper, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19" (Refs. 6, 7, 22, and 28). References 22 and 28 should be used if the FSAR χ/Q values are to be revised or if values are to be determined for new release points or receptor distances. Fumigation should be considered where applicable for the EAB and LPZ. For the EAB, the assumed fumigation period should be timed to be included in the worst 2-hour exposure period. The NRC computer code PAVAN (Ref. 29) implements Regulatory Guide 1.145 (Ref. 28) and its use is acceptable to the NRC staff. The methodology of the NRC computer code ARCON96[19] (Ref. 26) is generally acceptable to the NRC staff for use in determining control room χ/Q values. Meteorological data collected in accordance with the site-specific meteorological measurements program described in the facility FSAR should be used in generating accident χ/Q values. Additional guidance is provided in Regulatory Guide 1.23, "Onsite Meteorological Programs" (Ref. 30). All changes in χ/Q analysis methodology should be reviewed by the NRC staff.	Conform	The re-calculation of atmospheric dispersion factors was performed for the EAB and LPZ using the NRC computer code PAVAN according to the guidance of RG 1.145 and for the control room and TSC intakes with new release points using the NRC computer code ARCON96 according to the guidance of RG 1.194. The meteorological data used in the calculation were collected in accordance with WCGS site-specific measurements program and RG 1.23.

Table 15B-2 Conformance with Regulatory Guide 1.183, Revision 0 Appendix A (Loss-of-Coolant-Accident)			
RG Section	**RG Position**	**Analysis**	**Comments**
1	Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.	Conforms	Assumptions regarding core inventory and the release of radionuclides from the fuel were obtained from Regulatory Position 3 of RG 1.183.
2	If the sump or suppression pool pH is controlled at values of 7 or greater, the chemical form of radioiodine released to the containment should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine species, including those from iodine re-evolution, for sump or suppression pool pH values less than 7 will be evaluated on a case-by-case basis. Evaluations of pH should consider the effect of acids and bases created during the LOCA event, e.g., radiolysis products. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form.	Conforms	The sump pH is controlled at values of 7 or greater.
3.1	The radioactivity released from the fuel should be assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment in PWRs or the drywell in BWRs as it is released. This distribution should be adjusted if there are internal compartments that have limited ventilation exchange. The suppression pool free air volume may be included provided there is a mechanism to ensure mixing between the drywell to the wetwell. The release into the containment or drywell should be assumed to terminate at the end of the early in-vessel phase.	Conforms	The radioactivity released from the fuel was assumed to mix instantaneously and homogeneously throughout the region of containment not impacted by sprays as it was released. Recirculation fans provide a mechanism for mixing between the sprayed and unsprayed portions of containment.
3.2	Reduction in airborne radioactivity in the containment by natural deposition within the containment may be credited. Acceptable models for removal of iodine and aerosols are described in Chapter 6.5.2, "Containment Spray as a Fission Product Cleanup System," of the Standard Review Plan (SRP), NUREG-0800 (Ref. A-1) and in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments" (Ref. A-2). The latter model is incorporated into the analysis code RADTRAD (Ref. A-3). The prior practice of deterministically assuming that a 50% plateout of iodine is released from the fuel is no longer acceptable to the NRC staff as it is inconsistent with the characteristics of the revised source terms.	Conforms	No natural deposition was assumed for elemental or organic iodine. A sedimentation removal coefficient of 0.1 hr^{-1} for particulates was credited. This is consistent with previously approved submittals, including: • Point Beach Units 1 & 2 – April 2011 (ADAMS Accession Number ML110240054)

Table 15B-2 Conformance with Regulatory Guide 1.183, Revision 0 Appendix A (Loss-of-Coolant-Accident) (cont.)

RG Section	RG Position	Analysis	Comments
3.3	Reduction in airborne radioactivity in the containment by containment spray systems that have been designed and are maintained in accordance with Chapter 6.5.2 of the SRP (Ref. A-1) may be credited. Acceptable models for the removal of iodine and aerosols are described in Chapter 6.5.2 of the SRP and NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays"[1] (Ref. A-4). This simplified model is incorporated into the analysis code RADTRAD (Refs. A-1 to A-3). The evaluation of the containment sprays should address areas within the primary containment that are not covered by the spray drops. The mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building, provided that adequate flow exists between these regions, is assumed to be two turnovers of the unsprayed regions per hour, unless other rates are justified. The containment building atmosphere may be considered a single, well-mixed volume if the spray covers at least 90% of the volume and if adequate mixing of unsprayed compartments can be shown. The SRP sets forth a maximum decontamination factor (DF) for elemental iodine based on the maximum iodine activity in the primary containment atmosphere when the sprays actuate, divided by the activity of iodine remaining at some time after decontamination. The SRP also states that the particulate iodine removal rate should be reduced by a factor of 10 when a DF of 50 is reached. The reduction in the removal rate is not required if the removal rate is based on the calculated time-dependent airborne aerosol mass. There is no specified maximum DF for aerosol removal by sprays. The maximum activity to be used in determining the DF is defined as the iodine activity in the columns labeled "Total" in Tables 1 and 2 of this guide multiplied by 0.05 for elemental iodine and by 0.95 for particulate iodine (i.e., aerosol treated as particulate in SRP methodology).	Conforms	A spray system is available in containment. The spray removal model from Chapter 6.5.2 of NUREG-0800 was used in the analysis. The containment sprays cover 85% of containment. The mixing within containment is promoted by fan coolers which begin operation 2 minutes into the accident and are assumed on for the duration of the event. The elemental iodine removal coefficient was limited to 10 hr^{-1} and removal was terminated when a DF of 200 was reached. A particulate removal coefficient of 5 hr^{-1} was modeled until a DF of 50 was reached, at which time the coefficient was reduced to 0.5 hr^{-1} until sprays were terminated at 5 hours.

Table 15B-2 Conformance with Regulatory Guide 1.183, Revision 0 Appendix A (Loss-of-Coolant-Accident) (cont.)

RG Section	RG Position	Analysis	Comments
3.4	Reduction in airborne radioactivity in the containment by in-containment recirculation filter systems may be credited if these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. A-5 and A-6). The filter media loading caused by the increased aerosol release associated with the revised source term should be addressed.	Not Applicable	In-containment recirculation filters were not credited.
3.5	Reduction in airborne radioactivity in the containment by suppression pool scrubbing in BWRs should generally not be credited. However, the staff may consider such reduction on an individual case basis. The evaluation should consider the relative timing of the blowdown and the fission product release from the fuel, the force driving the release through the pool, and the potential for any bypass of the suppression pool (Ref. 7). Analyses should consider iodine re-evolution if the suppression pool liquid pH is not maintained greater than 7.	Not Applicable	Suppression pool scrubbing was not applicable to the WCGS design.
3.6	Reduction in airborne radioactivity in the containment by retention in ice condensers, or other engineering safety features not addressed above, should be evaluated on an individual case basis. See Section 6.5.4 of the SRP (Ref. A-1).	Not Applicable	No credit was taken for reduction of airborne radioactivity in the containment by retention in ice condensers or other engineering safety features not addressed above.

Table 15B-2 Conformance with Regulatory Guide 1.183, Revision 0 Appendix A (Loss-of-Coolant-Accident) (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
3.7	The primary containment (i.e., drywell for Mark I and II containment designs) should be assumed to leak at the peak pressure technical specification leak rate for the first 24 hours. For PWRs, the leak rate may be reduced after the first 24 hours to 50% of the technical specification leak rate. For BWRs, leakage may be reduced after the first 24 hours, if supported by plant configuration and analyses, to a value not less than 50% of the technical specification leak rate. Leakage from subatmospheric containments is assumed to terminate when the containment is brought to and maintained at a subatmospheric condition as defined by technical specifications. For BWRs with Mark III containments, the leakage from the drywell into the primary containment should be based on the steaming rate of the heated reactor core, with no credit for core debris relocation. This leakage should be assumed during the two-hour period between the initial blowdown and termination of the fuel radioactivity release (gap and early in-vessel release phases). After two hours, the radioactivity is assumed to be uniformly distributed throughout the drywell and the primary containment.	Conforms	The primary containment peak pressure leak rate was defined as 0.2% by weight of containment air. This leak rate was reduced to 0.1% after the first 24 hours of the accident.
3.8	If the primary containment is routinely purged during power operations, releases via the purge system prior to containment isolation should be analyzed and the resulting doses summed with the postulated doses from other release paths. The purge release evaluation should assume that 100% of the radionuclide inventory in the reactor coolant system liquid is released to the containment at the initiation of the LOCA. This inventory should be based on the technical specification reactor coolant system equilibrium activity. Iodine spikes need not be considered. If the purge system is not isolated before the onset of the gap release phase, the release fractions associated with the gap release and early in-vessel phases should be considered as applicable.	Conforms	Releases via the containment mini-purge system were calculated assuming 100% of the RCS activity (based on the Technical Specifications) was released to containment. The mini-purge release is isolated before the onset of the gap release phase.

Table 15B-2 Conformance with Regulatory Guide 1.183, Revision 0 Appendix A (Loss-of-Coolant-Accident) (cont.)

RG Section	RG Position	Analysis	Comments
4.1	Leakage from the primary containment should be considered to be collected, processed by engineered safety feature (ESF) filters, if any, and released to the environment via the secondary containment exhaust system during periods in which the secondary containment has a negative pressure as defined in technical specifications. Credit for an elevated release should be assumed only if the point of physical release is more than two and one-half times the height of any adjacent structure.	Not Applicable	A dual containment is not applicable to the WCGS design.
4.2	Leakage from the primary containment is assumed to be released directly to the environment as a ground-level release during any period in which the secondary containment does not have a negative pressure as defined in technical specifications.	Not Applicable	A dual containment is not applicable to the WCGS design.
4.3	The effect of high wind speeds on the ability of the secondary containment to maintain a negative pressure should be evaluated on an individual case basis. The wind speed to be assumed is the 1-hour average value that is exceeded only 5% of the total number of hours in the data set. Ambient temperatures used in these assessments should be the 1-hour average value that is exceeded only 5% or 95% of the total numbers of hours in the data set, whichever is conservative for the intended use (e.g., if high temperatures are limiting, use those exceeded only 5%).	Not Applicable	A dual containment is not applicable to the WCGS design.
4.4	Credit for dilution in the secondary containment may be allowed when adequate means to cause mixing can be demonstrated. Otherwise, the leakage from the primary containment should be assumed to be transported directly to exhaust systems without mixing. Credit for mixing, if found to be appropriate, should generally be limited to 50%. This evaluation should consider the magnitude of the containment leakage in relation to contiguous building volume or exhaust rate, the location of exhaust plenums relative to projected release locations, the recirculation ventilation systems, and internal walls and floors that impede stream flow between the release and the exhaust.	Not Applicable	A dual containment is not applicable to the WCGS design.

Table 15B-2 Conformance with Regulatory Guide 1.183, Revision 0 Appendix A (Loss-of-Coolant-Accident) (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
4.5	Primary containment leakage that bypasses the secondary containment should be evaluated at the bypass leak rate incorporated in the technical specifications. If the bypass leakage is through water, e.g., via a filled piping run that is maintained full, credit for retention of iodine and aerosols may be considered on a case-by-case basis. Similarly, deposition of aerosol radioactivity in gas-filled lines may be considered on a case-by-case basis.	Not Applicable	A dual containment is not applicable to the WCGS design.
4.6	Reduction in the amount of radioactive material released from the secondary containment because of ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).	Not Applicable	A dual containment is not applicable to the WCGS design.
5.1	With the exception of noble gases, all the fission products released from the fuel to the containment (as defined in Tables 1 and 2 of this guide) should be assumed to instantaneously and homogeneously mix in the primary containment sump water (in PWRs) or suppression pool (in BWRs) at the time of release from the core. In lieu of this deterministic approach, suitably conservative mechanistic models for the transport of airborne activity in containment to the sump water may be used. Note that many of the parameters that make spray and deposition models conservative with regard to containment airborne leakage are nonconservative with regard to the buildup of sump activity.	Conforms	With the exception of noble gases, all the fission products released from the fuel to the containment (as defined in Table 1 of RG 1.183) were assumed to instantaneously and homogenously mix in the primary containment sump.
5.2	The leakage should be taken as two times the sum of the simultaneous leakage from all components in the ESF recirculation systems above which the technical specifications, or licensee commitments to item III.D.1.1 of NUREG-0737 (Ref. A-8), would require declaring such systems inoperable. The leakage should be assumed to start at the earliest time the recirculation flow occurs in these systems and end at the latest time the releases from these systems are terminated. Consideration should also be given to design leakage through valves isolating ESF recirculation systems from tanks vented to atmosphere, e.g., emergency core cooling system (ECCS) pump miniflow return to the refueling water storage tank.	Conforms	The ESF leakage to the Plant Auxiliary Building was doubled in the analysis and was assumed to conservatively begin at the start of the event. The dose from ESF leakage to the refueling water storage tank was also included in the analysis.

Table 15B-2 Conformance with Regulatory Guide 1.183, Revision 0 Appendix A (Loss-of-Coolant-Accident) (cont.)

RG Section	RG Position	Analysis	Comments
5.3	With the exception of iodine, all radioactive materials in the recirculating liquid should be assumed to be retained in the liquid phase.	Conforms	Iodines were the only nuclides modeled to be released from the liquid.
5.4	If the temperature of the leakage exceeds 212°F, the fraction of total iodine in the liquid that becomes airborne should be assumed equal to the fraction of the leakage that flashes to vapor. This flash fraction, FF, should be determined using a constant enthalpy, h, process, based on the maximum time-dependent temperature of the sump water circulating outside the containment: $FF = \frac{h_{f_1} - h_{f_2}}{h_{fg}}$ Where: hf1 is the enthalpy of liquid at system design temperature and pressure; hf2 is the enthalpy of liquid at saturation conditions (14.7 psia, 212°F); and hfg is the heat of vaporization at 212°F.	Conforms	The calculated flashing fraction based on the maximum sump temperature was less than 0.10.
5.5	If the temperature of the leakage is less than 212°F or the calculated flash fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid, unless a smaller amount can be justified based on the actual sump pH history and area ventilation rates.	Conforms	An airborne fraction of 10% was assumed for the iodine in the liquid.
5.6	The radioiodine that is postulated to be available for release to the environment is assumed to be 97% elemental and 3% organic. Reduction in release activity by dilution or holdup within buildings, or by ESF ventilation filtration systems, may be credited where applicable. Filter systems used in these applications should be evaluated against the guidance of Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).	Conforms	The radioiodine that was postulated to be available for release to the environment was assumed to be 97% elemental and 3% organic. No reductions due to dilution or holdup were assumed. Credit was taken for the ESF ventilation filtration system in the Plant Auxiliary Building (PAB) exhaust.

Table 15B-2 Conformance with Regulatory Guide 1.183, Revision 0 Appendix A (Loss-of-Coolant-Accident) (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
7.0	The radiological consequences from post-LOCA primary containment purging as a combustible gas or pressure control measure should be analyzed. If the installed containment purging capabilities are maintained for purposes of severe accident management and are not credited in any design basis analysis, radiological consequences need not be evaluated. If the primary containment purging is required within 30 days of the LOCA, the results of this analysis should be combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. Reduction in the amount of radioactive material released via ESF filter systems may be taken into account provided that these systems meet the guidance in Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).	Not Applicable	The containment is not purged post-LOCA.

Table 15B-3 Conformance with Regulatory Guide 1.183, Revision 0 Appendix B (Fuel Handling Accident)			
RG Section	**RG Position**	**Analysis**	**Comments**
1	Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.	Conforms	Assumptions regarding core inventory and the release of radionuclides from the fuel were obtained from Regulatory Position 3 of RG 1.183.
1.1	The number of fuel rods damaged during the accident should be based on a conservative analysis that considers the most limiting case. This analysis should consider parameters such as the weight of the dropped heavy load or the weight of a dropped fuel assembly (plus any attached handling grapples), the height of the drop, and the compression, torsion, and shear stresses on the irradiated fuel rods. Damage to adjacent fuel assemblies, if applicable (e.g., events over the reactor vessel), should be considered.	Conforms	For the postulated fuel handling accident, one entire assembly and 20% of an adjacent assembly were assumed to be damaged as a result of this event.
1.2	The fission product release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. All the gap activity in the damaged rods is assumed to be instantaneously released. Radionuclides that should be considered include xenons, kryptons, halogens, cesiums, and rubidiums.	Conforms	The fission product release from the breached fuel was based on Regulatory Position 3.2 of RG 1.183 and the estimate of the number of fuel rods breached. All the gap activity in the damaged rods was assumed to be instantaneously released. Radionuclides that were considered include xenons, kryptons, halogens, cesiums, and rubidiums.
1.3	The chemical form of radioiodine released from the fuel to the spent fuel pool should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. The CsI released from the fuel is assumed to completely dissociate in the pool water. Because of the low pH of the pool water, the iodine re-evolves as elemental iodine. This is assumed to occur instantaneously. The NRC staff will consider, on a case-by-case basis, justifiable mechanistic treatment of the iodine release from the pool.	Conforms	All particulate iodine released to the spent fuel pool was assumed to dissociate and instantaneously re-evolve as elemental iodine.

Table 15B-3 Conformance with Regulatory Guide 1.183, Revision 0 Appendix B (Fuel Handling Accident) (cont.)

RG Section	RG Position	Analysis	Comments
2	If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 500 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85%) and organic iodine (0.15%) species results in the iodine above the water being composed of 57% elemental and 43% organic species. If the depth of water is not 23 feet, the decontamination factor will have to be determined on a case-by-case method (Ref. B-1).	Conforms	The minimum water depth over the reactor core when handling fuel and over the spent fuel in the fuel handling building is 23 feet, so the overall DF of 200 was applied.
3	The retention of noble gases in the water in the fuel pool or reactor cavity is negligible (i.e., decontamination factor of 1). Particulate radionuclides are assumed to be retained by the water in the fuel pool or reactor cavity (i.e., infinite decontamination factor).	Conforms	The analysis modeled a noble gas DF of 1 and an infinite DF for particulates.
4.1	The radioactive material that escapes from the fuel pool to the fuel building is assumed to be released to the environment over a 2-hour time period.	Conforms	The release of radioactive material was modeled as a linear release over a 2-hour period.
4.2	A reduction in the amount of radioactive material released from the fuel pool by engineered safety feature (ESF) filter systems may be taken into account provided these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. B-2, B-3). Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system1 should be determined and accounted for in the radioactivity release analyses.	Not Applicable	No engineered safety features were credited for the releases from the fuel building.

Table 15B-3 Conformance with Regulatory Guide 1.183, Revision 0 Appendix B (Fuel Handling Accident) (cont.)

RG Section	RG Position	Analysis	Comments
4.3	The radioactivity release from the fuel pool should be assumed to be drawn into the ESF filtration system without mixing or dilution in the fuel building. If mixing can be demonstrated, credit for mixing and dilution may be considered on a case-by-case basis. This evaluation should consider the magnitude of the building volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the pool, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the pool and the exhaust plenums.	Conforms	No credit was taken for mixing or dilution in the fuel building.
5.1	If the containment is isolated[2] during fuel handling operations, no radiological consequences need to be analyzed.	Not Applicable	No credit was taken for containment isolation in the analysis.
5.2	If the containment is open during fuel handling operations, but designed to automatically isolate in the event of a fuel handling accident, the release duration should be based on delays in radiation detection and completion of containment isolation. If it can be shown that containment isolation occurs before radioactivity is released to the environment,[1] no radiological consequences need to be analyzed.	Not Applicable	No credit was taken for containment isolation in the analysis.
5.3	If the containment is open during fuel handling operations (e.g., personnel air lock or equipment hatch is open),[3] the radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period.	Conforms	The containment was assumed to be open and the release of radioactive material was modeled as a linear release over a 2-hour period.
5.4	A reduction in the amount of radioactive material released from the containment by ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. B-2 and B-3). Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system should be determined and accounted for in the radioactivity release analyses.[1]	Not Applicable	No engineered safety features were credited for the releases from containment.

Table 15B-3 (cont.) Conformance with Regulatory Guide 1.183, Revision 0 Appendix B (Fuel Handling Accident)			
RG Section	**RG Position**	**Analysis**	**Comments**
5.5	Credit for dilution or mixing of the activity released from the reactor cavity by natural or forced convection inside the containment may be considered on a case-by-case basis. Such credit is generally limited to 50% of the containment free volume. This evaluation should consider the magnitude of the containment volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the reactor cavity, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the reactor cavity and the exhaust plenums.	Not Applicable	No credit was taken for mixing or dilution in containment.

Table 15B-4 Conformance with Regulatory Guide 1.183, Revision 0 Appendix E (PWR Main Steam Line Break)			
RG Section	**RG Position**	**Analysis**	**Comments**
1	Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this regulatory guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. The fuel damage estimate should assume that the highest worth control rod is stuck at its fully withdrawn position.	Not Applicable	No fuel damage was postulated to occur during the MSLB.
2	If no or minimal[2] fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by the technical specifications. Two cases of iodine spiking should be assumed.	Conforms	Since no fuel damage occurs, two cases of iodine spiking (pre-accident and accident-initiated) were modeled.
2.1	A reactor transient has occurred prior to the postulated main steam line break (MSLB) and has raised the primary coolant iodine concentration to the maximum value (typically 60 μCi/gm DE I-131) permitted by the technical specifications (i.e., a preaccident iodine spike case).	Conforms	The pre-accident iodine spike was modeled with a primary coolant iodine concentration of 60 μCi/gm DE I-131, consistent with the Technical Specification limit.
2.2	The primary system transient associated with the MSLB causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (typically 1.0 μCi/gm DE I-131) specified in technical specifications (i.e., concurrent iodine spike case). A concurrent iodine spike need not be considered if fuel damage is postulated. The assumed iodine spike duration should be 8 hours. Shorter spike durations may be considered on a case-by-case basis if it can be shown that the activity released by the 8-hour spike exceeds that available for release from the fuel gap of all fuel pins.	Conforms	The accident-initiated iodine spike was modeled with a spike factor of 500 and spike duration of 8 hours. The initial activity was based on 1.0 μCi/gm DE I-131, consistent with the Technical Specification limit.
3	The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant.	Conforms	The activity was modeled to be released instantaneously and homogenously throughout the primary coolant.

Table 15B-4 Conformance with Regulatory Guide 1.183, Revision 0 Appendix E (PWR Main Steam Line Break) (cont.)

RG Section	RG Position	Analysis	Comments
4	The chemical form of radioiodine released from the fuel should be 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 should be assumed to be 97% elemental and 3% organic. These fractions apply to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking.	Conforms	Iodine chemical fractions for releases to the environment (97% elemental and 3% organic) were modeled in the analysis.
5.1	For facilities that have not implemented alternative repair criteria (see Ref. E-1, DG-1074), the primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate limiting condition for operation specified in the technical specifications. For facilities with traditional generator specifications (both per generator and total of all generators), the leakage should be apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized.	Conforms	The accident-induced Technical Specification Bases leakage of 1 gpm was assumed to be entirely to the faulted steam generator and the normal operation Technical Specification primary-to-secondary leakage of 150 gpd/SG was assumed to be evenly apportioned between the intact steam generators.
5.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of the parameter being converted. The ARC leak rate correlations are generally based on the collection of cooled liquid. Surveillance tests and facility instrumentation used to show compliance with leak rate technical specifications are typically based on cooled liquid. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).	Conforms	A density of 1.0 gm/cc (62.4 lbm/ft^3) was used.
5.3	The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212°F). The release of radioactivity from unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms	The primary-to-secondary leakage was terminated at 34 hours when the reactor coolant system was cooled to 212°F. The release of radioactivity from the unaffected steam generators was terminated at 12 hours when the residual heat removal system was in service and removing all decay heat.
5.4	All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.	Conforms	No reduction or mitigation of noble gas activity in releases from the primary system was modeled.

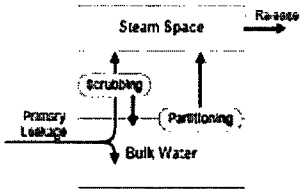
Table 15B-4 Conformance with Regulatory Guide 1.183, Revision 0 Appendix E (PWR Main Steam Line Break) (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
5.5	The transport model described in this section should be utilized for iodine and particulate releases from the steam generators. This model is shown in Figure E-1 and summarized below:  	Conforms	The transport model was utilized in the analysis.
5.5.1	A portion of the primary-to-secondary leakage will flash to vapor, based on the thermodynamic conditions in the reactor and secondary coolant. • During periods of steam generator dryout, all of the primary-to-secondary leakage is assumed to flash to vapor and be released to the environment with no mitigation. • With regard to the unaffected steam generators used for plant cooldown, the primary-to-secondary leakage can be assumed to mix with the secondary water without flashing during periods of total tube submergence.	Conforms	The faulted steam generator was assumed to blowdown to dryout conditions and any primary-to-secondary leakage was modeled as a release to the environment with no mitigation. Leakage to the intact steam generators was modeled to mix with the secondary water without flashing since the steam generator tubes are submerged.
5.5.2	The leakage that immediately flashes to vapor will rise through the bulk water of the steam generator and enter the steam space. Credit may be taken for scrubbing in the generator, using the models in NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident" (Ref. E-2), during periods of total submergence of the tubes.	Not Applicable	See Position 5.5.1 above.
5.5.3	The leakage that does not immediately flash is assumed to mix with the bulk water.	Not Applicable	See Position 5.5.1 above.

Table 15B-4 Conformance with Regulatory Guide 1.183, Revision 0 Appendix E (PWR Main Steam Line Break) (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
5.5.4	The radioactivity in the bulk water is assumed to become vapor at a rate that is the function of the steaming rate and the partition coefficient. A partition coefficient for iodine of 100 may be assumed. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators.	Conforms	A partition coefficient of 100 was modeled for iodine and the particulate radionuclides release was limited by the moisture carryover of 0.25%.
5.6	Operating experience and analyses have shown that for some steam generator designs, tube uncovery may occur for a short period following any reactor trip (Ref. E-3). The potential impact of tube uncovery on the transport model parameters (e.g., flash fraction, scrubbing credit) needs to be considered. The impact of emergency operating procedure restoration strategies on steam generator water levels should be evaluated.	Not Applicable	The issue of tube uncovery was addressed by the Westinghouse Owners Group (WOG) in WCAP-13247, "Report on the Methodology for the Resolution of the Steam Generator Tube Uncovery Issue," March 1992. The WOG program concluded that the effect of tube uncovery would be essentially negligible and the issue could be closed without any further investigation or generic restrictions. This position was accepted by the NRC in a letter dated March 10, 1993, from Robert C. Jones, Chief of the Reactor System Branch, to Lawrence A. Walsh, Chairman of the WOG. The letter states "... the Westinghouse analyses demonstrate that the effects of partial steam generator tube uncovery on the iodine release for SGTR and non-SGTR events is negligible. Therefore, we agree with your position on this matter and consider this issue to be resolved." Consistent with this position, the MSLB dose analysis did not model tube uncovery in the intact steam generators.

Table 15B-5 Conformance with Regulatory Guide 1.183, Revision 0 Appendix F (PWR Steam Generator Tube Rupture Accident)			
RG Section	**RG Position**	**Analysis**	**Comments**
1	Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are in Regulatory Position 3 of this guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.	Not Applicable	No fuel damage was postulated to occur during the SGTR.
2	If no or minimal[2] fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by technical specification. Two cases of iodine spiking should be assumed.	Conforms	Since no fuel damage occurs, two cases of iodine spiking (pre-accident and accident-initiated) were modeled.
2.1	A reactor transient has occurred prior to the postulated steam generator tube rupture (SGTR) and has raised the primary coolant iodine concentration to the maximum value (typically 60 μCi/gm DE I-131) permitted by the technical specifications (i.e., a preaccident iodine spike case).	Conforms	The pre-accident iodine spike was modeled with a primary coolant iodine concentration of 60 μCi/gm DE I-131, consistent with the Technical Specification limit.
2.2	The primary system transient associated with the SGTR causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 335 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (typically 1.0 μCi/gm DE I-131) specified in technical specifications (i.e., concurrent iodine spike case). A concurrent iodine spike need not be considered if fuel damage is postulated. The assumed iodine spike duration should be 8 hours. Shorter spike durations may be considered on a case-by-case basis if it can be shown that the activity released by the 8-hour spike exceeds that available for release from the fuel gap of all fuel pins.	Conforms	The accident-initiated iodine spike was modeled with a spike factor of 335 and spike duration of 8 hours. The initial activity was based on 1.0 μCi/gm DE I-131, consistent with the Technical Specification limit.
3	The activity released from the fuel, if any, should be assumed to be released instantaneously and homogeneously through the primary coolant.	Conforms	The activity was modeled to be released instantaneously and homogenously throughout the primary coolant.

Table 15B-5 Conformance with Regulatory Guide 1.183, Revision 0 Appendix F (PWR Steam Generator Tube Rupture Accident) (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
4	Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic.	Conforms	Iodine chemical fractions for releases to the environment (97% elemental and 3% organic) were modeled in the analysis.
5.1	The primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate limiting condition for operation specified in the technical specifications. The leakage should be apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized.	Conforms	The accident-induced Technical Specification Bases leakage of 1 gpm was assumed to be evenly apportioned between the intact steam generators. This leakage modeling was used since the leakage is negligible compared to the flow through the ruptured tube.
5.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests are typically based on cool liquid. Facility instrumentation used to determine leakage is typically located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).	Conforms	A density of 1.0 gm/cc (62.4 lbm/ft^3) was used.
5.3	The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212° F). The release of radioactivity from the unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms	The primary-to-secondary break flow was terminated when the reactor coolant system pressure equalized with the ruptured steam generator secondary side pressure. The release of radioactivity from the ruptured and intact steam generators was terminated at 12 hours when the residual heat removal system was in service and removing all decay heat.
5.4	The release of fission products from the secondary system should be evaluated with the assumption of a coincident loss of offsite power.	Conforms	A loss of offsite power was assumed coincident with reactor trip.
5.5	All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.	Conforms	No reduction or mitigation of noble gas activity in releases from the primary system was modeled.

Table 15B-5 Conformance with Regulatory Guide 1.183, Revision 0 Appendix F (PWR Steam Generator Tube Rupture Accident) (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
5.6	The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.	Conforms	The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E for iodine and particulates was considered as appropriate in the SGTR. In addition, flashing of break flow in the ruptured steam generator with a time dependent flashing fraction was considered and all activity in the flashed break flow was released to the environment with no mitigation, dilution, or credit for scrubbing.

Table 15B-6 Conformance with Regulatory Guide 1.183, Revision 0 Appendix G (PWR Locked Rotor Accident)			
RG Section	**RG Position**	**Analysis**	**Comments**
1	Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are in Regulatory Position 3 of this regulatory guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.	Conforms	Assumptions regarding core inventory and the release of radionuclides from the fuel were obtained from Regulatory Position 3 of RG 1.183. The analysis modeled the fraction of the core inventory assumed to be in the gap by radionuclide group consistent with Table 3 in Regulatory Position 3.2. The assumed number of fuel rods breached was 5% of the core and the radial peaking factor of 1.65 was applied.
2	If no fuel damage is postulated for the limiting event, a radiological analysis is not required as the consequences of this event are bounded by the consequences projected for the main steam line break outside containment.	Not Applicable	Fuel damage was postulated; therefore, an analysis was performed.
3	The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant.	Conforms	The activity was modeled to be released instantaneously and homogenously throughout the primary coolant.
4	The chemical form of radioiodine released from the fuel should be 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 should be assumed to be 97% elemental and 3% organic. These fractions apply to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking.	Conforms	Iodine chemical fractions for releases to the environment (97% elemental and 3% organic) were modeled in the analysis.
5.1	The primary-to-secondary leak rate in the steam generators should be assumed to be the leak-rate-limiting condition for operation specified in the technical specifications. The leakage should be apportioned between the steam generators in such a manner that the calculated dose is maximized.	Conforms	The accident-induced Technical Specification Bases leakage of 1 gpm was assumed to be evenly apportioned to the steam generators.

Table 15B-6 Conformance with Regulatory Guide 1.183, Revision 0 Appendix G (PWR Locked Rotor Accident) (cont.)

RG Section	RG Position	Analysis	Comments
5.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests are typically based on cool liquid. Facility instrumentation used to determine leakage is typically located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).	Conforms	A density of 1.0 gm/cc (62.4 lbm/ft^3) was used.
5.3	The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212° F). The release of radioactivity should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms	The primary-to-secondary leakage was terminated when the release of radioactivity from the steam generators was terminated. This occurred at 12 hours when the residual heat removal system was in service and removing all decay heat.
5.4	The release of fission products from the secondary system should be evaluated with the assumption of a coincident loss of offsite power.	Conforms	A loss of offsite power was assumed.
5.5	All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.	Conforms	No reduction or mitigation of noble gas activity in releases from the primary system was modeled.
5.6	The transport model described in assumptions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.	Conforms	The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E for iodine and particulates was considered as appropriate in the locked rotor event.

Table 15B-7 Conformance with Regulatory Guide 1.183, Revision 0 Appendix H (PWR Rod Ejection Accident)			
RG Section	**RG Position**	**Analysis**	**Comments**
1	Assumptions acceptable to the NRC staff regarding core inventory are in Regulatory Position 3 of this guide. For the rod ejection accident, the release from the breached fuel is based on the estimate of the number of fuel rods breached and the assumption that 10% of the core inventory of the noble gases and iodines is in the fuel gap. The release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and the assumption that 100% of the noble gases and 25% of the iodines contained in that fraction are available for release from containment. For the secondary system release pathway, 100% of the noble gases and 50% of the iodines in that fraction are released to the reactor coolant.	Conforms	Assumptions regarding core inventory and the release of radionuclides from the fuel were obtained from Regulatory Position 3 of RG 1.183. The analysis modeled gap release fractions of 10% for iodines and noble gases and 12% for alkali metals. The assumed number of fuel rods breached was 10% of the core and the radial peaking factor of 1.65 was applied. The analysis also modeled fuel melting release fractions of 100% of noble gases and 50% of iodines and alkali metals in the melted fuel rods. Note that the assumption of 50% iodine and alkali metal release from melted fuel was conservatively modeled for each pathway.
2	If no fuel damage is postulated for the limiting event, a radiological analysis is not required as the consequences of this event are bounded by the consequences projected for the loss-of-coolant accident (LOCA), main steam line break, and steam generator tube rupture.	Not Applicable	Fuel damage was postulated; therefore, an analysis was performed.
3	Two release cases are to be considered. In the first, 100% of the activity released from the fuel should be assumed to be released instantaneously and homogeneously through the containment atmosphere. In the second, 100% of the activity released from the fuel should be assumed to be completely dissolved in the primary coolant and available for release to the secondary system.	Conforms	Both cases were considered separately in the analysis.
4	The chemical form of radioiodine released to the containment atmosphere should be assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. If containment sprays do not actuate or are terminated prior to accumulating sump water, or if the containment sump pH is not controlled at values of 7 or greater, the iodine species should be evaluated on an individual case basis. Evaluations of pH should consider the effect of acids created during the rod ejection accident event, e.g., pyrolysis and radiolysis products. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form.	Conforms	The iodine chemical fractions for releases to containment were modeled as 95% CsI, 4.85% elemental, and 0.15% organic. All fission products, with the exception of elemental and organic iodine and noble gases, were assumed to be in particulate form. No removal processes were modeled in containment besides leakage and decay, so sump pH has no impact.

Table 15B-7 Conformance with Regulatory Guide 1.183, Revision 0 Appendix H (PWR Rod Ejection Accident) (cont.)

RG Section	RG Position	Analysis	Comments
5	Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic.	Conforms	Iodine chemical fractions for releases to the environment (97% elemental and 3% organic) were modeled in the analysis.
6	Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material in and from the containment are as follows.	Conforms	See below.
6.1	A reduction in the amount of radioactive material available for leakage from the containment that is due to natural deposition, containment sprays, recirculating filter systems, dual containments, or other engineered safety features may be taken into account. Refer to Appendix A to this guide for guidance on acceptable methods and assumptions for evaluating these mechanisms.	Not Applicable	No removal processes were modeled in containment besides leakage and decay.
6.2	The containment should be assumed to leak at the leak rate incorporated in the technical specifications at peak accident pressure for the first 24 hours, and at 50% of this leak rate for the remaining duration of the accident. Peak accident pressure is the maximum pressure defined in the technical specifications for containment leak testing. Leakage from subatmospheric containments is assumed to be terminated when the containment is brought to a subatmospheric condition as defined in technical specifications.	Conforms	The primary containment peak pressure leak rate was defined as 0.2% by weight of containment air. This leak rate was reduced to 0.1% after the first 24 hours of the accident.
7.1	A leak rate equivalent to the primary-to-secondary leak rate limiting condition for operation specified in the technical specifications should be assumed to exist until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms	The accident-induced Technical Specification Bases leakage of 1 gpm was assumed to be evenly apportioned to the steam generators. The primary-to-secondary leakage was terminated when the release of radioactivity from the steam generators was terminated. This occurred at 12 hours when the residual heat removal system was in service and removing all decay heat.

Table 15B-7 Conformance with Regulatory Guide 1.183, Revision 0 Appendix H (PWR Rod Ejection Accident) (cont.)			
RG Section	**RG Position**	**Analysis**	**Comments**
7.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests typically are based on cooled liquid. The facility's instrumentation used to determine leakage typically is located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).	Conforms	A density of 1.0 gm/cc (62.4 lbm/ft^3) was used.
7.3	All noble gas radionuclides released to the secondary system are assumed to be released to the environment without reduction or mitigation.	Conforms	No reduction or mitigation of noble gas activity in releases from the primary system was modeled.
7.4	The transport model described in assumptions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.	Conforms	The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E for iodine and particulates was considered as appropriate in the rod ejection event.