STP
Nuclear Operating Company

South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

March 22, 2007
NOC-AE-07002127
10CFR50.67
10CFR50.90

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
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Rockville, MD 20852-2738

South Texas Project
Units 1 and 2
Docket Nos. STN 50-498, STN 50-499
Request for License Amendment Related to Application of the Alternate Source Term

References:

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

2. U. S. Nuclear Regulatory Commission Standard Review Plan 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, July 2000

In accordance with 10 CFR 50.67, "Accident Source Term," and 10 CFR 50.90, "Application for amendment of license or construction permit," STP Nuclear Operating Company (STPNOC) requests an amendment to Appendix A, Technical Specifications (TS), of Facility Operating Licenses NPF-76 and NPF-80 for South Texas Project (STP) Units 1 and 2. The proposed changes are requested to support application of an alternative source term (AST) methodology, with the exception that Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," will continue to be used as the radiation dose basis for equipment qualification. The proposed AST methodology is consistent with the guidance in References 1 and 2, including alternate methods for complying with the specified portions of the NRC's regulations as allowed by the guidance in Reference 1. Documentation of conformance to Reference 1 and the allowed alternate methods are presented in Tables A through G in Attachment 6 of this submittal.

Application of the AST methodology is being used to resolve a non-conforming condition at STP where testing resulted in control room unfiltered in-leakage at a value greater than the current accident dose analysis assumption. The application of the AST methodology also allows for cost beneficial changes resulting from the relaxation of certain Technical Specification requirements.

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Implementation of an AST methodology changes the regulatory assumptions regarding the analytical treatment of the design basis accidents (DBAs). Approval of this proposed change will provide a source term for the STP facility that will result in a more accurate assessment of the DBA radiological doses.

By a separate action, STPNOC plans to submit a revision to TS 3/4.7.7, "Control Room Makeup and Cleanup Filtration System." This revision will be based on TS Task Force (TSTF) Traveler 448, Revision 3 that includes a surveillance program for measuring control room inleakage. The inleakage acceptance criterion in the surveillance is the value assumed in the design basis accident analyses. Implementation of the separate TS based on TSTF 448 will be predicated on approval and implementation of this AST amendment because the AST amendment establishes the revised inleakage assumption in the design basis accident analyses that is required for implementation of the TSTF 448 amendment.

STPNOC identified the following condition related to this amendment request. Westinghouse Electric Company Nuclear Safety Advisory Letter NSAL-06-15, dated December 13, 2006, advised operators of Westinghouse plants that the single failure scenario for the steam generator tube rupture (SGTR) analysis that licensees used in their accident analysis may not be limiting. STPNOC has evaluated the applicability of this NSAL against the accident analysis assumptions. The current single failure assumption for the STP SGTR analysis is not limiting so that STPNOC is operating under compensatory measures to meet regulatory dose limits. STPNOC plans to resolve this condition at the earliest opportunity so that the assumptions, including the limiting single failure, for the SGTR accident analysis performed for this amendment request are consistent with the plant response to this event. Section 1.0 of Attachment 1 to this letter provides additional detail. The action to resolve this condition is provided as licensing commitment in Attachment 5 to this letter.

This request is subdivided as follows:

- Attachment 1 provides the Licensee's Evaluation for this change including a description of proposed changes, technical analysis, and regulatory analysis.
- Attachment 2 provides the markup of Technical Specification (TS) pages.Attachment 3 provides a markup of the associated TS Bases pages (for information only)
- Attachment 4 provides planned changes to the Technical Requirements Manual (for information only)
- Attachment 5 provides the List of Commitments resulting from the proposed changes.
- Attachment 6 contains "Regulatory Guide 1.183 Conformance Tables" providing detailed verification that the AST methodology conforms to the guidance in Regulatory Guide 1.183.
- Attachment 7 addresses information discussed in NRC Regulatory Issue Summary (RIS) 2006-04. A table provides a description of how each issue discussed in the RIS is addressed in the STPNOC application.

- Attachment 8 provides a markup of the affected pages to the STP Updated Final Safety Analysis Report (UFSAR) (for information only)

The Plant Operations Review Committee has approved the proposed change. STPNOC has notified the State of Texas in accordance with 10CFR50.91(b). A No Significant Hazards Consideration Determination is provided in Attachment 1.

STPNOC requests approval of the proposed amendment by March 30, 2008. Once approved, the amendment shall be implemented within 120 days due to the significant implementation scope of the subject changes. This implementation period will provide adequate time for the affected station documents to be revised using the appropriate change control mechanisms.

The NRC previously approved implementation of the AST methodology at a number of other nuclear power stations. This change is consistent with the guidance of NRC Regulatory Guide 1.183 and Standard Review Plan 15.0.1.

If there are any questions regarding this proposed amendment, please contact Mr. Ken Taplett at (361) 972-8416 or me at (361) 972-7867.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 22 MAR 2007

David W. Rencurrel
Vice President,
Engineering and Strategic Projects

Attachments:

1. Licensee's Evaluation
2. Markup of Technical Specification pages
3. Markup of Technical Specification Bases pages (information only)
4. Planned Changes to the Technical Requirements Manual (information only)
5. List of Commitments
6. Regulatory Guide 1.183 Conformance Tables
7. NRC Regulatory Issue Summary 2006-04 Table
8. Markup of UFSAR pages (information only) **(Not for Public Disclosure)**

cc:

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Attachment 1

Licensee's Evaluation

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LICENSEE'S EVALUATION

1.0 DESCRIPTION

In accordance with 10 CFR 50.67, "Accident Source Term," and 10 CFR 50.90, "Application for an amendment of license or construction permit," STP Nuclear Operating Company (STPNOC) requests an amendment to Appendix A, Technical Specifications (TS), of Facility Operating Licenses NPF-76 and NPF-80 for South Texas Project (STP) Units 1 and 2. The proposed changes are requested to support application of an alternative source term (AST) methodology, with the exception that Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," (Reference 1) will continue to be used as the radiation dose basis for equipment qualification. The proposed AST methodology is consistent with the guidance in Standard Review Plan 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," (Reference 2) and Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," (Reference 3) except where alternate methods for complying with the specified portions of the NRC's regulations have been used as allowed by RG 1.183. Documentation of conformance to RG 1.183 and the allowed alternate methods are presented in Tables A through G in Attachment 6 of this submittal.

NRC Generic Letter (GL) 2003-01, 'Control Room Habitability," required licensees, in part, to confirm that the most limiting unfiltered in-leakage into the control room envelope (CRE) is no more than the value assumed in the current design basis radiological analyses. The current analyses at STPNOC assume no unfiltered in-leakage other than 10 cfm for ingress and egress into the CRE. The American Society for Testing and Materials (ASTM) E741 tracer gas test was performed in Unit 1 in March 2004 and in Unit 2 in March 2007. The test results in both units were greater than the unfiltered in-leakage assumed in the current licensing basis accident analyses. Therefore, STPNOC opted for full-scope implementation of the AST methodology to address the test results and attain additional cost benefits described below.

In support of a full-scope implementation of the AST methodology, STPNOC, supported by Polestar Applied Technology, Inc., performed radiological consequence analyses for the following DBAs that result in control room (CR) and offsite exposure as specified in Reference 3.

- Loss of Coolant Accident (LOCA)
- Fuel Handling Accident (FHA)
- Main Steam Line Break (MSLB)
- Steam Generator Tube Rupture (SGTR)
- Control Rod Ejection Accident (CREA)
- Locked Rotor Accident (LRA)

Proposed changes to the current licensing basis for the South Texas Project (STP) justified by the AST analyses include the following items.

- The use of updated meteorological data to calculate onsite and offsite atmospheric dispersion
- Relies on less filtration
  - No credit taken for Fuel Handling Building Exhaust Air Ventilation filtration
  - No credit taken for Control Room Ventilation makeup filtration
  - No credit taken for either Control Room Ventilation makeup or recirculation cleanup filtration for the Fuel Handling Accident
- Containment isolation capability is no longer required to mitigate a FHA.
- Analysis of only a single limiting FHA rather than one analysis for an FHA inside containment and a second analysis for an FHA in the fuel handling building (FHB)
- Revised control room unfiltered in-leakage assumption

The proposed changes related to the applicability requirements during movement of irradiated fuel use insights from Technical Specification Task Force Traveler TSTF-51, Revision 2 (Reference 5). TSTF-51, Revision 2, was approved by the NRC on July 31, 2003. TSTF-51 changed the TS operability requirements for certain engineered safety features such that they are not required to be operable after sufficient radioactive decay has occurred to ensure that offsite doses remain within limits. STPNOC's change differs from TSTF-51 in that the definition of "recently irradiated fuel" is not used. Instead, the STPNOC change is based upon a specification in the STP Technical Requirements Manual that precludes the movement of irradiated fuel until a "42 hour decay time" has occurred following the achievement of subcriticality. This is consistent with the AST accident analysis assumption. Defining "recently irradiated fuel" has no practical application for the STPNOC proposed change. The "decay time" definition is consistent with the accident analyses assumption so that STPNOC would not be in a condition where recently irradiated fuel could be moved. The accident analyses were performed with a decay time such that the impacted engineered safety features are not required at any time that fuel is being moved.

Approval of this change will provide a source term for STP that will result in a more accurate assessment of the DBA radiological doses. The improved dose assessment allows relaxation of some current licensing basis requirements as described in Section 2.0. Upon implementation of AST, containment closure capability will no longer be required to mitigate an FHA. This proposed change provides flexibility when performing refueling activities by allowing movement of equipment through the containment boundary in support of outage activities while meeting accident radiological acceptance criteria. Outages can be optimized to achieve an overall risk reduction while also reducing outage time and cost. Outage resources can be directed to other activities, which ultimately should result in improvements in plant maintenance, operations and overall safety.

The revised radiological dose to the control room operator allows for a revised air unfiltered in-leakage assumption that provides a conservative margin over that determined by air in-leakage testing. In the Spring of 2004, STPNOC tested the Unit 1 control room envelope for unfiltered in-leakage using the tracer gas method. The highest measured unfiltered in-leakage was 9.4 cfm compared to the current licensing basis assumption of 0 cfm. Although the resultant dose increase was more than minimal as defined by 10 CFR 50.59 regulatory guidance, the result was within the regulatory limits. The assumption of 100 cfm unfiltered in-leakage used in the revised analyses was based on the Unit 1 results. The Unit 2 Control Room Envelope was recently tested for unfiltered in-leakage using the tracer gas method. The highest unfiltered in-leakage was 64 cfm. Regulatory limits are met in Unit 2 with compensatory measures. STPNOC is currently operating under a non-conforming licensing basis condition. The conditions in Units 1 and 2 are documented in the corrective action program. A revision to the source term and the assumed unfiltered in-leakage will assure that the revised dose analyses proposed by this amendment request are met.

Westinghouse Electric Company Nuclear Safety Advisory Letter NSAL-06-15, dated December 13, 2006, advised operators of Westinghouse plants that the single failure scenario for the steam generator tube rupture (SGTR) analysis may not be limiting. The methodology included evaluations of various single failures for a reference plant. Recent industry operating experience identified a condition where a failed-open main steam line isolation valve (MSIV) on the steamline from the ruptured steam generator (SG) may result in a steam flow that is higher than that previously assumed in the accident analysis and thus higher offsite doses rates.

The STP current SGTR analysis and the SGTR analysis presented in the safety evaluation for this licensing amendment request assumes a failed open SG power operated relief valve as the limiting single failure as far as assumed total steam release. An evaluation of NSAL-06-15 has resulted in a revised conclusion that the failed open MSIV results in the greater steam release at STP. This is because the steam valves to the moisture-separator reheater fail open on a loss of instrument air resulting from a loss of offsite power. The steam valves to the moisture-separator reheater fail closed for the reference plant thus significantly reducing the steam release from a failed open MSIV.

STP is currently operating under an administrative limit for reactor coolant system dose equivalent iodine that is lower than the Technical Specification limit. STP plans to correct this non conforming condition at the earliest opportunity. The most likely path to resolution will be a plant modification. Therefore, the assumptions, including the limiting single failure regarding total steam release, for the SGTR accident analysis performed for this amendment request will be consistent with the plant response to this event after the modification is completed. Until the modification is completed, STP will continue to maintain an administrative limit for reactor coolant system dose equivalent iodine so that the radiological dose limits for the SGTR analysis remain bounding. A commitment to maintain these controls until the plant is modified is described in Attachment 5. This condition is documented under Condition Report 07-2887.

STPNOC requests approval of the proposed amendment by March 30, 2008. Once approved, the amendment shall be implemented within 120 days due to the significant implementation scope of the subject changes. This implementation period will provide adequate time for the affected station documents to be revised using the appropriate change control mechanisms.

2.0 PROPOSED CHANGE

The proposed changes related to the applicability requirements during movement of irradiated fuel assemblies use insights from Technical Specification Task Force Traveler (TSTF)-51, "Revise Containment Requirements During Handling of Irradiated Fuel and Core Alterations," Revision 2. The NRC approved TSTF-51 on July 31, 2003. TSTF-51 changes the TS operability requirements for engineered safety features such that they are not required to be operable after sufficient radioactive decay has occurred to ensure that offsite doses remain within limits.

STPNOC's change differs from TSTF-51 in that the definition of "recently irradiated fuel" is not used. Instead, the STPNOC change is based upon a specification in the STP Technical Requirements Manual (TRM) that precludes the movement of irradiated fuel until a "42 hour decay time" has occurred following the achievement of sub-criticality. This is consistent with the AST accident analysis assumption. The STP TS changes are based upon the TRM decay time requirement. The decay time specification was relocated from TS to the TRM with approval of STPNOC licensing amendments 145 and 133 to Unit 1 and Unit 2 respectively (Reference 7). In the safety evaluation for this TS change, the NRC staff found that the "decay time requirement" specification does not need to be in the TS because it is not needed to ensure the decay time limit is met. This is because certain operational steps, such as containment entry, pressure vessel head removal, and cavity flood-up must be completed before fuel movement in the vessel is possible following critical operation. These preliminary activities require more than 42 hours to complete. The NRC staff found that these physical limitations are adequate to ensure compliance with the 42-hour limit (relocated to the TRM). Thus including the decay time limit in TSs is not needed to ensure this limit is met. Using insight from TSTF-51, the proposed change also deletes CORE ALTERATIONS from applicability requirements for some Limiting Conditions for Operation.

Using insights from TSTF-51, STPNOC is committing to the applicable provisions of Nuclear Utilities Management and Resources Council (NUMARC) 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 3, (Reference 6) as described in TSTF-51 and documented in Attachment 5 to this submittal. Additional discussion regarding these provisions is described in Section 4.4.8 of this evaluation.

Proposed changes to the TS resulting from this submittal are summarized below.

<u>Section 1.0, Definitions</u>

The dose conversion factors used to calculate the dose from DOSE EQUIVALENT I-131 concentration are revised to those listed in Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion," 1988; (Table 2.1, Exposure-to-Dose Conversion Factors for Inhalation) (Reference 20) instead of Table E-7 of NRC Regulatory Guide 1.109, Revision 1, October 1977 (Reference 32). "Thyroid" is dropped from the definition of dose to reflect that the dose is now the total effective dose equivalent based on Alterative Source Term methodology.

Table 3.3-3, Engineered Safety Features Actuation System Instrumentation

- Modes 5 and 6 for Functional Unit 3.b.4), "Containment Ventilation Isolation RCB Purge Radioactivity – High," are deleted as APPLICABLE MODES because automatic isolation is no longer required during core alterations or movement of irradiated fuel within containment to meet the AST design basis accident analysis. ACTION 18 is modified appropriately.

- Modes 5 and 6 for Functional Unit 10, "Control Room Ventilation," are deleted as APPLICABLE MODES because the accident mitigation capabilities of this system are no longer credited in AST design basis accident analysis for activities performed during these MODES.

- The ACTION 28 requirement for Functional Unit 10.d, "Control Room Intake Air Radioactivity – High," is modified to delete suspension of core alterations, movement of irradiated fuel, and crane operation with loads over the spent fuel pool because the accident mitigation capabilities of this system are no longer credited in AST design basis accident analysis during these activities.

- The requirement for an operable Functional Unit 11, Fuel Handling Building (FHB) Heating, Ventilation and Air Conditioning (HVAC), actuation instrumentation is deleted since the accident mitigation capabilities of the FHB HVAC system are no longer credited in AST design basis accident analysis.

- An administrative change is made to remove a Note from ACTION 20 because the provisions of the Note have expired.

Table 3.3-4, Engineered Safety Features Actuation System Instrumentation Trip Setpoints

The trip setpoints and allowable values for Functional Unit 11.a., "FHB HVAC," are deleted since the accident mitigation capabilities of this system are no longer credited in AST design basis accident analyses.

Table 4.3-2, Engineered Safety Features Actuation System Instrumentation Surveillance Requirements

- Modes 5 and 6 for Functional Unit 3.b.4), "Containment Ventilation Isolation RCB Purge Radioactivity – High," are deleted as APPLICABLE MODES because automatic isolation is no longer required during core alterations or movement of irradiated fuel within containment to meet the AST design basis accident analysis.

- Modes 5 and 6 for Functional Unit 10, "Control Room Ventilation," are deleted as APPLICABLE MODES because the accident mitigation capabilities of this system are no longer credited in AST design basis accident analysis for activities performed during these MODES.

- The requirement for performing surveillances for Functional Unit 11, Fuel Handling Building (FHB) Heating, Ventilation and Air Conditioning (HVAC), actuation instrumentation is deleted because the accident mitigation capabilities of the FHB HVAC system are no longer credited in AST design basis accident analysis.

TS 3/4.7.7, Control Room Makeup and Cleanup Filtration System

- The APPLICABILITY for Modes 5 and 6 is deleted. Requirements to suspend all operations during core alterations, movement of irradiated fuel, and crane operation with loads over the spent fuel pool are deleted because the accident mitigation capabilities of this system are no longer credited in AST design basis accident analysis during these activities. Requirements to suspend operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or required boron concentration are deleted. Adequate SHUTDOWN MARGIN is controlled by TS 3/4.1.1, "Shutdown Margin Boration Control" and adequate boron concentration is controlled by TS 3/4.9.1, "Boration Concentration." The safety analysis concludes that administrative controls and operator response time are adequate measures to preclude a loss of required SHUTDOWN MARGIN or required boron concentration. In addition, requirements to suspend operations involving positive reactivity additions are not found in Standard Technical Specifications for control room ventilation systems. Thus there are no radiological consequences.

- ACTION C for MODES 1, 2, 3,and 4 is modified to delete the requirements to suspend all operations involving movement of spent fuel and crane operations with loads over the spent fuel pool because the accident mitigation capabilities of this system are no longer credited in AST design basis accident analysis.

- An administrative change is made to remove a footnote from surveillance requirement 4.7.7.e 3) because the provisions of the footnote have expired.

TS 3/4.7.8, Fuel Handling Building (FHB) Exhaust Air System

This specification is deleted because the accident mitigation capabilities of the FHB Exhaust Air HVAC system are no longer credited in AST design basis accident analysis.

TS 3.8.1.2, A.C. Sources Shutdown

The actions to suspend movement of irradiated fuel or crane operation with loads over the spent fuel pool if the Limiting Condition for Operation is not met are deleted because the fuel handling accident analysis no longer credits the mitigation systems that are dependent upon this source of electrical power.

TS 3.8.1.3, A.C. Sources Shutdown

The actions to suspend movement of irradiated fuel or crane operation with loads over the spent fuel pool if the Limiting Condition for Operation is not met are deleted because the fuel handling accident analysis no longer credits the mitigation systems that are dependent upon these sources of electrical power.

TS 3.8.2.2, D.C. Sources Shutdown

The action to suspend movement of irradiated fuel if the Limiting Condition for Operation is not met are deleted because the fuel handling accident analysis no longer credits the mitigation systems that are dependent upon these sources of electrical power.

TS 3.8.2.3, Onsite Power Distribution Shutdown

The action to suspend movement of irradiated fuel if the Limiting Condition for Operation is not met are deleted because the fuel handling accident analysis no longer credits the mitigation systems that are dependent upon these sources of electrical power.

TS 3/4.9.4, Containment Building Penetrations during Refueling Operations

This specification is deleted because containment isolation is no longer credited in AST design basis accident analysis during core alterations or movement of irradiated fuel.

TS 3/4.9.9, Containment Ventilation Isolation during Refueling Operations

This specification is deleted because containment isolation is no longer credited in AST design basis accident analysis during core alterations or movement of irradiated fuel.

TS 3/4.9.12, Fuel Handling Building Exhaust Air System during Refueling Operations

This specification is deleted because the accident mitigation capabilities of the FHB HVAC system are no longer credited in AST design basis accident analysis.

In summary, this proposal (1) removes the Fuel Handling Building Exhaust Air System LCO for all MODES, (2) removes the Control Room Makeup and Cleanup Filtration System LCO during MODES 5 and 6, and (3) removes the Containment Building Penetrations LCO including the Containment Ventilation Isolation System LCO during Modes 5 and 6 including Refueling Operations from the Technical Specifications. The actions to suspend movement of irradiated fuel or crane operation with loads over the spent fuel pool if the Limiting Condition for Operation is not met are deleted from the Electrical Sources Technical Specifications while Shutdown because the fuel handling accident analysis no longer credits the mitigation systems that are dependent upon this source of electrical power. Finally, the definition for DOSE EQUIVALENT I-131 is revised.

The proposed changes do not impact Technical Specification requirements for systems needed to prevent or mitigate CORE ALTERATION events other than the FHA. The proposed changes also do not change the requirements of systems needed to mitigate potential vessel drain down events, systems needed for decay heat removal, or the requirements to maintain high water levels over irradiated fuel.

Corresponding changes to the TS Bases will be made following approval of the proposed amendment in accordance with the TS Bases Control Program and 10 CFR 50.59. The planned changes to the affected TS Bases pages are provided in Attachment 3 for information.

STPNOC plans to delete TS 3/4.7.8, 3/4.9.4 and 3/4.9.12 and add requirements to the Technical Requirements Manual (TRM) to facilitate restoration of Containment closure or the Fuel Handling Building Exhaust Air System, as applicable, and to provide a filtered and monitored release path as a defense-in-depth measure to mitigate the consequences of a postulated FHA. Further, STPNOC plans to insert a new TRM requirement to facilitate the restoration of one train of Control Room Makeup and Cleanup Filtration System as a defense-in-depth measure to mitigate the consequences of a postulated FHA. Attachment 4 provides the planned TRM pages for information. See Section 4.4.8 for further discussion.

The planned TRM specification for the FHB Exhaust Air System only requires one train to be OPERABLE or capable of being restored to an OPERABLE status to meet the Limiting Condition for Operation (LCO) whenever irradiated fuel is in the spent fuel pool. Movement of fuel within the spent fuel pool or crane operation with loads over the spent fuel pool will be required to be suspended if at least one FHB Exhaust Air Train can not be restored to OPERABLE status within the time required by the LCO. Surveillance Requirements will remain the same as TSs with the exception that the surveillance to verify that the system automatically starts upon initiation of a high radiation or safety injection test signal will not be included as a surveillance. This change does not propose to downgrade the safety classification of this system.

Amendments 125/113 (Reference 8) to Units 1 and 2, respectively, provide an allowed outage time of up to 12 hours to restore at least one train of control room makeup and cleanup filtration system or one train of fuel handling building exhaust air filtration to an operable status when multiple trains of either system are inoperable in MODES 1, 2, 3 or 4. As a compensatory measure to ensure that applicable regulatory limits continue to be met, STPNOC committed to not intentionally enter the action for multiple trains of the Control Room Makeup and Cleanup Filtration System and the Fuel Handling Building Exhaust Air System simultaneously in MODES 1, 2, 3 or 4. Although this proposed amendment request will relocate TS 3.7.8 to the TRM, the compensatory measure described for Amendments 125/113 will remain in place with procedural requirements revised as appropriate. TS Bases page B 3/4 7-5 of Attachment 3 reflects this change.

The planned TRM specification for Containment Building Penetrations requires containment building penetrations to be closed or capable of being closed within two hours following a fuel handling accident. Surveillance Requirements will remain the same as those previously in the TS with the exception that the surveillance to verify that the containment purge and exhaust isolation valves automatically close upon initiation of an isolation test signal will not be included.

The planned TRM specification for the Control Room Makeup and Cleanup Filtration System requires only one system to be OPERABLE or capable of being restored to an OPERABLE status to meet the Limiting Condition for Operation (LCO) whenever irradiated fuel is in the spent fuel pool or during the movement of irradiated fuel, which includes refueling operations in the reactor containment building. Movement of irradiated fuel or crane operation with loads over the spent fuel pool will be required to be suspended if at least one Control Room Makeup and Cleanup Filtration train can not be restored within the time required by the LCO. Surveillance Requirements will remain the same as TSs with the exception that the surveillance to verify that the system automatically starts upon initiation of a high radiation or safety injection test signal will not be included as a surveillance. This amendment change request does not propose to downgrade the safety classification of this system.

The two hours to restore the FHB Exhaust Air System and the Control Room Makeup and Cleanup Filtration System to OPERABLE status and to close containment closures in the event of a fuel handling accident is reasonable because these systems are not required to mitigate the accident. These systems are not credited in the accident analyses. Dose limits are within requirements assuming an instantaneous release from the FHA. These additional administrative actions are taken to further filter and monitor the release as a defense-in-depth measure.

3.0 BACKGROUND

3.1 Systems Affected by the Proposed Change

The following systems are affected by this proposed amendment:

1. The **Containment Ventilation Isolation System** closes the containment isolation valves in the Normal Containment Purge System and the Supplementary Containment Purge System. This action isolates the containment atmosphere from the environment to minimize releases of radioactivity in the event of an accident. These systems are described in Sections 9.4.5.2.6 and 9.4.5.2.7 of the Updated Final Safety Analyses Report (UFSAR).

2. The **Control Room Makeup and Cleanup Filtration System** provides a protected environment from which the operators can control the unit following an uncontrolled release of radioactivity by maintaining the control room envelope at a positive pressure. Outside air is filtered and mixed with the air being re-circulated within the control room. Pressurization of the control room minimizes the infiltration of unfiltered air from the surrounding areas of the building. The Control Room Makeup and Cleanup Filtration System satisfies the design requirement of limiting dose to the control room operators following the design basis accident in accordance with General Design Criterion 19 of 10 CFR 50, Appendix A. This system consists of three 50-percent-capacity redundant

trains. The system isolates normal supply ventilation and initiates filtered makeup and cleanup ventilation of the control room envelope following receipt of an accident initiation signal. Each train of filtered makeup and cleanup ventilation consists of a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system. A second bank of HEPA filters, not credited in the accident analysis, follows the adsorber section to collect carbon fines and provides backup in case of failure of the main HEPA filter bank. This system is described in Sections 6.4, 6.5.1 and 9.4.1.1.1 of the UFSAR.

3. The **Fuel Handling Building Exhaust Air System** filters airborne radioactive particulates from the area of the fuel pool following a fuel handling accident or loss of coolant accident. This system consists of two independent and redundant filter trains. Each train consists of a heater, a prefilter, a high efficiency particulate air (HEPA) filter, and an activated charcoal adsorber section for removal of gaseous activity (principally iodines). Three 50-percent-capacity main and booster fans serve the redundant exhaust trains. Heaters, ductwork, valves or dampers, and instrumentation also form part of the system functioning to reduce the relative humidity of the airstream. A second bank of HEPA filters, not credited in the accident analysis, follows the adsorber section to collect carbon fines and provide backup in case the main HEPA filter bank fails. The system establishes a negative pressure in the Fuel Handling Building and initiates filtered exhaust ventilation from the building following receipt of a high radiation signal or a safety injection signal. This system is described in Sections 6.5.1 and 9.4.2 of the UFSAR.

4. The Engineered Safety Features (ESF) **AC and DC Electrical Power Systems** are designed with redundancy and independence of onsite power sources, distribution systems, and controls in order to provide a reliable supply of electrical power to the ESF electrical loads necessary to achieve safe plant shutdown, or to mitigate the consequences of postulated accidents. The ESF AC and DC Electrical Power Systems are described in Section 8.3 of the UFSAR.

5. The **Reactor Containment Building** 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 100.11. Additionally, the Reactor Containment Building provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

6. The **Fuel Handling Building** is designed as a controlled-leakage structure. The Fuel Handling Building in conjunction with the Fuel Handling Building Exhaust Air System creates an enclosure to direct radioactivity releases to the environment through a filter bank such that offsite radiation exposures are maintained well within the requirements of 10 CFR 100.11.

No changes to these systems are required to implement the proposed change. The change will allow the automatic start feature of systems no longer credited in the accident analyses for mitigation to be disabled through the STPNOC modification process. The modification process provides checks to assure that the modification does not invalidate assumptions made in the PRA or adversely impact the severe accident management program.

3.2 The AST Rule

On December 23, 1999, the NRC issued the Final Rule on "Use of Alternate Source Terms at Operating Reactors." The Final Rule, issued under 10 CFR 50.67, "Accident source term," allows holders of operating licenses issued prior to January 10, 1997, to voluntarily replace the traditional source term used in design basis accident analyses with alternative source terms. This action allows interested licensees to pursue cost beneficial licensing actions to reduce unnecessary regulatory burden without compromising the margin of safety of the facility.

The fission product release from the reactor core into containment is referred to as the "source term," and is characterized by the composition and magnitude of the radioactive material, the chemical and physical properties of the material, and the timing of the release from the reactor core. Since the publication of U.S. Atomic Energy Commission Technical Information Document, TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," which is the currently used design basis document for calculation of offsite dose for loss of coolant accidents, significant advances have been made in understanding the composition and magnitude, chemical form, and timing of fission product releases from severe nuclear power plant accidents. Many of these insights developed out of the major research efforts started by the NRC and the nuclear industry after the accident at Three Mile Island. NUREG-1465 (Reference 9) was published in 1995 with revised ASTs for use in the licensing of future Light Water Reactors (LWRs). The NRC, in 10 CFR 50.67, later allowed the use of the ASTs described in NUREG-1465 at operating plants. This NUREG represents the result of decades of research on fission product release and transport in Light-Water Reactors under accident conditions. One of the major insights summarized in NUREG-1465 involves the timing and duration of fission product releases.

The five release phases describing the progression of a severe accident in a LWR are listed in NUREG-1465 and are given below.

1. Coolant Activity Release
2. Gap Activity Release
3. Early In-Vessel Release
4. Ex-Vessel Release
5. Late In-Vessel Release

Phases 1, 2, and 3 are considered in current (i.e., pre-AST) DBA evaluations; however, they are all assumed to occur instantaneously. Phases 4 and 5 are related to severe accident evaluations. Under the AST methodology, only the coolant activity release (i.e., Phase 1) is assumed to occur instantaneously and end with the onset of the gap activity release (i.e., Phase 2). This approach represents a more realistic time sequence for activity release. The insights from NUREG-1465 were subsequently incorporated into RG 1.183.

3.3 STP Full Application of AST

In order to utilize this more realistic approach, this license amendment request proposes to implement a full-scope application of the AST methodology addressing the composition and magnitude of the radioactive material, its chemical and physical form, and the timing of its release. STPNOC has performed radiological consequence analyses of the below DBAs that result in the most significant offsite exposures. The AST analyses have been performed in accordance with the guidance in RG 1.183 and NRC Standard Review Plan 15.0.1.

- Loss of Coolant Accident (LOCA)
- Fuel Handling Accidents (FHA) in the Fuel Handling Building (FHB) and in Containment
- Main Steam Line Break (MSLB)
- Steam Generator Tube Rupture (SGTR)
- Control Rod Ejection Accident (CREA)
- Locked Rotor Accident (LRA)

Implementation of an AST methodology changes only the regulatory assumptions regarding the analytical treatment of the DBAs. Implementation of the AST methodology will provide a source term for STP that will result in a more accurate assessment of the DBA radiological doses.

3.4 Precedent

The NRC has previously approved implementation of the AST methodology at a number of nuclear power plants. The STPNOC analyses conform to NRC RG 1.183 as demonstrated in Attachment 6.

An aspect of the STPNOC analyses where specific precedence exists is the methodology and code for calculation of transient containment sump pH. Application of this computer code is found in the Vermont Yankee AST application (Reference 10). See Section 4.3.3.1.2 for further discussion.

The relocation of the Fuel Handling Building Air Exhaust System and the associated actuation instrumentation requirements and the relocation of the Containment Ventilation Isolation System and the associated actuation instrumentation requirements out of Technical Specifications based on using the alternate source term methodology were previously approved in the Surry Plant Amendment 230. (Reference 11) The deletion of the spent fuel pool ventilation filtration system from the Technical Specifications based on alternate source term methodology was previously approved in the Salem Plant Amendments 263 and 245. (Reference 12)

4.0 TECHNICAL ANALYSIS

The AST analyses for STP were performed following the guidance in Standard Review Plan 15.0.1 and Regulatory Guide 1.183 except where alternate methods for complying with the specified portions of the NRC's regulations were used as allowed by Regulatory Guide 1.183. Documentation of conformance to Regulatory Guide 1.183 and the allowed alternate methods are presented in Attachment 6 of this submittal.

The full-scope implementation consists of the following:

1. Identification of the core source term based on plant specific analysis of core fission product inventory.

2. Determination of the release fractions for the six Pressurized Water Reactor (PWR) Design Basis Accidents (DBAs) identified in Appendices A, B, E, F, G, and H of Regulatory Guide 1.183 that could potentially result in control room and offsite doses. These are the Loss Of Coolant Accident (LOCA), the Fuel Handling Accident (FHA), the Main Steam Line Break accident (MSLB), the Steam Generator Tube Rupture accident (SGTR), the Control Rod Ejection Accident (CREA), and the Locked Rotor Accident (LRA).

3. Calculation of new control room (CR), exclusion area boundary (EAB), and low population zone (LPZ) atmospheric dispersion factors (χ/Q) for the containment leakage, plant vent, and steam generator (SG) secondary side power-operated relief valve (PORV) release paths.

4. Calculation of offsite and control room personnel Total Effective Dose Equivalent (TEDE).

5. Evaluation of containment sump pH to ensure that the particulate iodine deposited into the containment water during the DBA LOCA does not re-evolve beyond the amount recognized in the DBA LOCA analysis.

6. Evaluation of other related design and licensing bases such as NUREG-0737 (Reference 13).

Implementation of AST includes changes to the methodology presently used at STP for dose consequence analysis. These include:

1. Use of updated meteorological data (five years from 2000 to 2004) to calculate onsite and offsite atmospheric dispersion.

2. No credit for any filtration other than for the recirculation filters for the control room (CR) and the recirculation filters for the Technical Support Center (TSC). No filtration whatsoever has been relied upon in the analysis of the Fuel Handling Accident (FHA).

3. Limits on DBA LOCA iodine removal from the containment atmosphere based on the containment sump pH going slightly below 7.0 after one day. (The lowest pH reached over 30 days post-accident is 6.8.)

4. Analysis of only a single limiting FHA rather than one analysis for an FHA inside containment and a second analysis for an FHA in the FHB.

The revised dose consequence analyses were prepared, reviewed, and are 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. The analyses were performed in a manner to ensure that dose analyses have not been "tuned" to a specific set of accident progression assumptions. Conservative parameters are used when calculating components in the dose analyses. For example, the determination of iodine re-evolution in the LOCA analysis assumes the maximum sump temperature, which occurs in the first minutes of the accident, in conjunction with the lowest pH, which occurs at the very end of the 30-day dose assessment period, to achieve the limiting results.

The accident analyses assumptions are not based upon risk insights.

The Technical Analysis will demonstrate that all post-accident radiological doses for DBAs required by Regulatory Guide 1.183 remain within regulatory limits.

Overview and Organization of Accident Descriptions

This section provides the information on the analyses performed in support of the change of the design basis source terms from the current TID-14844 source terms to the NUREG-1465 Alternate Source Term. The section begins with a presentation of the updated meteorological data that was used to determine revised atmospheric dispersion factors for both offsite locations and the control room (Section 4.1). These revised χ/Q values are used in all new analyses.

Presentations of the generic analytical models used in the revised analyses are then presented (Section 4.2). This includes the formulation for the offsite dose model and detailed HVAC models for both the control room and Technical Support Center (TSC). Development of the radiological source terms that are used as a basis for the revised analyses is discussed. This discussion includes physical nuclide parameters and dose conversion factors. The core nuclide inventory is presented, along with isotopic concentrations for the reactor coolant system at the Technical Specification normal maximum iodine concentration of 1 μCi/gm and at the Technical Specification iodine spike limit of 60 μCi/gm. Similarly, secondary system nuclide concentrations are developed for 1% failed fuel and the Technical Specification normal maximum iodine concentration of 0.1 μCi/gm. These discussions are followed by detailed descriptions of the following accidents:

Section 4.3 LOCA
Section 4.4 Fuel Handling Accident

Section 4.5	Main Steam Line Break
Section 4.6	Steam Generator Tube Rupture
Section 4.7	Control Rod Ejection
Section 4.8	Locked Rotor

The description of each accident follows the following format:

- An overview of the methodology of the analysis
- The analytical model(s) used to perform the analysis
- Development/discussion of the radiological source term
- Discussion of the radiological releases (usually steam flows)
- Assumptions and inputs
- A table of important parameters specific to that analysis
- Summary, including the dose results

The contents of each subsection may change depending on the needs for the specific accident under discussion.

A list of commonly used acronyms is presented in Table 4.0-1.

Table 4.0-1
Frequently Used Acronyms

AHU	Air handling unit
AST	The Alternative Source Term, as defined in NUREG-1465 and Regulatory Guide 1.183
CLB	The current licensing basis, including analyses, for the South Texas Project
CR	Control Room
CRE	Control Room (HVAC) Envelope
CREA	Control Rod Ejection Accident
CSS	Containment Spray System
DCF	Dose Conversion Factor
DEI	Dose Equivalent Iodine-131
EAB	Exclusion Area Boundary (site boundary)
ESF	Engineered Safety Feature
FHA	Fuel Handling Accident
FHB	Fuel Handling Building
IVC	Isolation Valve Cubicle (location of PORVs and MSIVs)
LPZ	Low Population Zone
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LRA	Locked Rotor Accident
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break
PORV	Power Operated Relief Valve
RCB	Reactor Containment Building
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
TSC	Technical Support Center
TSP	Trisodium phosphate

4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

4.1 Meteorology and Atmospheric Dispersion

The revised χ/Q values used for the AST application have been developed using more recent meteorological data than that used for the current licensing basis (CLB). These more recent data were obtained for the years 2000 to 2004 (five years worth of data). Polestar subcontracted ABS Consulting to perform the meteorological data analysis and the PAVAN and ARCON96 analyses.

The χ/Q values resulting at the Control Room intake are calculated using the NRC-sponsored computer code ARCON96 consistent with the procedures in Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," (Reference 14).

The χ/Q values resulting at the EAB and LPZ are calculated using the NRC-sponsored computer code PAVAN (Reference 15), consistent with the procedures in Regulatory Guide 1.145 (Reference 16).

Onsite Meteorological Monitoring Program

The meteorological measurement program at STP consists of a 60-meter primary tower and a free standing 10-meter tower which serves as a backup to the primary system. The two methods used to determine atmospheric stability are:

a. Delta Temperature (i.e., vertical temperature difference), which is the principal method, and
b. Sigma theta (i.e., standard deviation of the horizontal wind direction), as a backup method.

Data, gathered per Safety Guide 23, "Onsite Meteorological Programs" (Reference 17), are used to determine the meteorological conditions specific to the plant site. The meteorological program includes information on site specific instrumentation and calibration procedures.

The meteorological tower is equipped with instrumentation that conforms to the system accuracy recommendations in Safety Guide 23. The dew point instrument is less accurate than the Regulatory Guide requires. The meteorological instrumentation is placed on horizontal booms oriented into the generally prevailing wind direction at the site. Equipment signals are transmitted to an instrument building with controlled environmental conditions. The instrument building, at or near the base of the tower, houses the recording equipment, processor cards, and digital recording equipment, etc. This instrumentation is used to process and retransmit the data to the end-point users.

4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

Recorded meteorological data are used to create joint frequency tables by quarter and to provide input to dispersion estimates of airborne concentrations of gaseous effluents in support of offsite radiation dose assessments. Instrument calibrations and data consistency evaluations are performed according to Safety Guide 23 to ensure maximum data accuracy. Capability is maintained to evaluate atmospheric dispersion better than 90% of the time even though individual instruments may not record data with 90% reliability. Delta-temperature is the primary method used for determining atmospheric stability. When the delta-temperature instrument is inoperable, wind direction sigma theta data are used to maintain 90% availability. Likewise, when wind direction or speed is unavailable from the primary tower, data from the corresponding instruments on the backup tower are substituted.

Site Description
The minimum EAB and LPZ boundaries are located at 1430 m and 4800 m. Note that Plant North is also True North. A simplified diagram of the units and the release points and receptors is provided in Figure 4.1-13.

A description of the climate at STP is provided in Section 2.3.1.1 of the UFSAR (Revision 14):

"... The climate of the region is subtropical maritime and is characterized by short mild winters and long hot summers. In the vicinity of the site, the humidity is generally high and rainfall is abundant throughout the year.

"Summer type climate extends from about May through September, with the highest temperature occurring during July and August. The summer weather is normally dominated by tropical maritime air masses associated with the Bermuda High. Days are typically hot and humid, and convective showers and thunderstorms are relatively frequent.

"Winter type climate extends from December through February, with the coldest weather occurring in January. The Gulf of Mexico modifies outbreaks of polar air masses to such an extent that temperatures below 32°F occur on an average of less than four times per year ([UFSAR] Ref. 2.3-1).

"The fall type climate months are October and November, and the spring type climate months are March and April. Both transitional seasons are short and are characterized by mild, pleasant weather. The locations and a brief topographical description of the meteorological stations used to determine the general climate and local meteorology ([UFSAR] Section 2.3.2) in the STPEGS site region are presented in [UFSAR] Table 2.3-1.

"The STPEGS site is located approximately 8 miles north-northwest of Matagorda, Texas, just west of the Colorado River. The site elevation is approximately 25 ft above mean sea level (MSL). More detail on local topography and its influences on local meteorology is presented in Section [UFSAR] 2.3.2."

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A description of the topographic influences on its meteorology and diffusion estimates is provided in UFSAR Section 2.3.2.2.2 (Revision 14):

"The terrain in the region of the STPEGS site is generally flat. Elevations above MSL average approximately 25 ft. [UFSAR] Figure 2.3-10 is a topographic map of the site area within a 5-mile radius. [UFSAR] Figure 2.3-11 is a topographic map of the site area within a 50-mile radius.

"The major local effect on site meteorology is the presence of the Gulf of Mexico and the resultant sea and land breeze circulations. The sea breeze generally forms during the spring and summer when the Gulf of Mexico's water temperature is colder than the air temperature and results in an onshore wind-flow. During periods of light geostrophic winds, surface winds may develop which blow onshore (sea to land) during the day and offshore (land to sea) at night. The formation of the sea breeze is the result of the temperature variation between water and land. Turbulent mixing within the water effects a continuous downward transport of surface heat through large masses of water during spring and summer, thus lowering the surface water temperature (and also lowering the temperature of the overlying air), in contrast with the strong surface heating of the air over the shoreline region. This contrast is also intensified because the water has a higher thermal capacity than that of the soil. As a result of this situation, temperatures over land are greater than those over water during spring and summer; this difference diminishes toward sunset and may reverse during the night. As the warmed air over the land begins to rise, a horizontal density gradient is formed which causes the heavier, colder air over the water to flow underneath the warm air during these seasons. To ensure continuity of the circulation cell, there is a return motion of the warmer air from land to the Gulf at higher levels. Although formation of the sea breeze circulation is usually perpendicular to the shoreline, Coriolis forces become significant as the system matures. During the late afternoon, the sea breeze can be expected to have a major component parallel to the shore to the right of the onground trajectory. Land breezes are the converse of sea breezes and may develop when sea temperatures are warmer than the land, such as during the fall and early winter or during the night in the summer. However, land breezes are generally weaker and less frequent than sea breezes ([UFSAR] Ref. 2.3-35).

"Therefore, considering the basis and characteristics of sea breeze circulations, these local wind systems are a definite factor relevant to the STPEGS site. The sea breeze circulation in southeast Texas extends approximately 25 mile inland ([UFSAR] Ref. 2.3-1). A study was undertaken to determine instances of sea breeze penetration to the STPEGS site. Synoptic weather conditions and hourly station data for selected stations in the STPEGS site area, obtained from the NWS for the period July 21, 1973 to July 20,1977, along with historical data on sea surface temperatures of the Gulf of Mexico in the site regions, were used to identify periods of potential sea breeze penetration into the STPEGS site. The site data were then examined for these periods to confirm sea breeze occurrences. Temperature drops and relative humidity changes were noted for three selected NWS stations in the vicinity of the STPEGS site as well as for the site, and hydrographs were plotted and analyzed for the NWS stations to confirm the occurrence of sea breezes in the site area. Based on these analyses, 51 days (primarily in the spring and summer) were identified

4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

during which the sea breeze penetration might reach the STPEGS site area. Of these, there were 35 days (10 percent on an annual basis) on which sea breeze occurrence was confirmed at the site. There were four cases of sea breeze at the STPEGS site on 2 consecutive days and one case of 4 consecutive days.

"Considering both the low frequency of consecutive daily occurrences of sea breeze penetration to the site and the low overall frequency of occurrence of sea breeze penetration to the site, the impact of sea breeze upon dose estimates for the STPEGS is expected to be insignificant ([UFSAR] Ref. 2.3-53). Since the Gulf of Mexico is a relatively warm body of water, air flowing over the Gulf is heated from below (especially prevalent in the fall and winter when the Gulf water temperatures average 3°F higher than the temperature of the overlying air near the water). This heating from below tends to increase the instability of the overlying air, enhancing diffusion. During the night, onshore flows of air warmed over the Gulf have a tendency to inhibit inversion formation over the land, further increasing the dilution potential of the atmosphere ([UFSAR] Ref. 2.3-1)."

Meteorological Data
The CLB is based on meteorological data obtained from July 21, 1973, through September 30, 1977. This amendment uses meteorological data from the five-year period, (i.e., 2000 - 2004). This data set was used in the used in the PAVAN and ARCON96 analyses discussed in this amendment.

4.1.1 Analysis of the 2000-2004 Meteorology Data

The 2000-2004 hourly data was analyzed by ABS Consulting to ensure reasonableness and consistency. Only the lower level wind speed and direction from the 10m level and the 60-10m delta temperature were examined. The data were plotted and printed out for questionable periods. The questionable periods were examined more closely to verify the validity of the data. Periods where data from the primary tower were missing were replaced with data from the backup tower. Replacement of the primary tower data with the backup tower data was verified by an ABS meteorologist.

The data contains 6 hours of calm, 5 in 2001 (1 E stability from the N, 1 F stability from the N, 3 G stability from the SSE, WNW, and NW) and one in 2002 (G stability from the N). The decrease in the number of calms from the earlier data set is because of more sensitive instrumentation, so that a cutoff of 0.5 mph instead of 0.75 mph could be used.

When the final database was established, joint frequency distributions, wind roses and delta temperature plots were run to compare each year of data. The total wind frequency distribution for all stability classes is presented in Table 4.1-1. The joint frequency distributions by stability class are presented in Tables 4.1-2 through 4.1-8. A comparison of the distributions of stability classes with similar tables from the STP UFSAR[1] are shown in Table 4.1-9.

[1] UFSAR Revision 13, Tables 2.3-29 through -36.

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The results of these comparisons showed the five years of data were very consistent. There is a rather high percentage of unstable conditions over the five years, but it is consistent over the five year period. The stability class comparison in Table 4.1-9 shows about a five percent increase in A stability (from 7.6% to 12.4% of occurrences) in the recent data and about a five percent decrease in F/G stability (from 24.2% to 19.0% of occurrences). Section 2.3.2.1.3, Revision 14, of the STP UFSAR states that

"[UFSAR] Table 2.3-13 presents annual ΔT stability classifications for STPEGS data for the July 21, 1973 through September 30, 1977 period. These data indicate a predominance of neutral (D) to slightly stable (E) conditions. On an annual basis, stable conditions (E, F, and G) occur approximately 47 percent of the time while unstable conditions (A, B, and C) occur approximately 20 percent of the time."

From Table 4.1-9, for the average of the five years of data obtained from 2000 to 2004, stable conditions (E, F, and G) occurred approximately 46% of the time, similar to the 1973-1977 data. Unstable conditions (A, B, and C) occurred approximately 25% of the time in the 2000-2004 time period.

Table 4.1-10 provides a summary of the average wind speeds and peak wind directions during 2000-2004. A comparison of the wind speed distributions between the recent meteorological data and the CLB[2] data sets of data is shown on Figure 4.1-1. The 10^{th} and 20^{th} percentiles compare well (approximately 4.2 mph for the 10^{th} percentile for both data sets and 5.3 MPH for the 20^{th} percentile). Even though the 80^{th} percentile shows some difference (approximately 12.6 mph for the more recent data as compared to approximately 15.2 mph for the CLB), it is the low wind speeds that would be controlling for the χ/Q analysis.

The average wind speed for each year and the sector of the peak wind is presented in Table 4.1-10. Wind roses for each year are provided in Figures 4.1-2 through 4.1-7. The distribution of wind directions are presented in Table 4.1-11 and are comparable to the CLB.

The delta temperature plots (Figures 4.1-8 through -12) show that the average delta temperature reading for the five year period was -0.3°F.

The overall five year data base is a consistent set of data in stability classification, wind direction, and wind speed and represents a good data set for use in atmospheric diffusion calculations.

[2] From UFSAR, Rev 13, Table 2.3-29

4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

Table 4.1-1
Joint Frequency Distribution for 2000-2004
All Stability Classes

Wind Direction	Calm	Calm-3.5	3.6-7.5	7.6-12.5	12.6-18.5	18.6-24.5	24.6-32.5	> 32.5	Total	Percent	Average Speed
	Wind Speed (mph) @ 10m										
N	3	298	916	1082	788	101	3	0	3191	7.4	9.6
NNE	0	408	1300	1142	306	12	1	2	3171	7.4	7.5
NE	0	545	1502	1043	208	1	0	1	3300	7.7	6.9
ENE	0	457	1146	712	147	2	0	0	2464	5.7	6.7
E	0	403	1208	751	324	22	3	1	2712	6.3	7.5
ESE	0	377	1232	1073	608	54	3	1	3348	7.8	8.5
SE	0	248	1954	2177	1164	123	2	0	5668	13.2	9.4
SSE	1	103	1757	2760	1414	106	2	0	6143	14.3	10.0
S	0	65	1030	3034	1054	37	0	0	5220	12.2	10.1
SSW	0	35	675	872	139	3	0	0	1724	4.0	8.5
SW	0	36	282	416	52	2	0	0	788	1.8	8.3
WSW	0	37	131	133	26	1	0	0	328	0.8	7.6
W	0	71	275	66	20	0	0	0	432	1.0	5.9
WNW	1	197	395	93	29	1	0	0	716	1.7	5.4
NW	1	222	533	262	176	25	0	0	1219	2.8	7.5
NNW	0	245	785	771	520	115	9	0	2445	5.7	9.4
Total	6	3747	15121	16387	6975	605	23	5	42869	100.0	
Percent	0.0	8.8	35.3	38.2	16.3	1.4	0.1	0.0	100.0		

Average Speed for This Table	8.7	Number of Invalid Hours	978
Number of Calm Hours for this Table	6	Number of Valid Hours for this Table	42869
Number of Variable Direction Hours for this Table	0	Total Hours for this Period	43847

Table 4.1-2
Joint Frequency Distribution for 2000-2004
Stability Class: A Extremely Unstable

Wind Direction	Calm	Calm-3.5	3.6-7.5	7.6-12.5	12.6-18.5	18.6-24.5	24.6-32.5	> 32.5	Total	Percent	Average Speed
	Wind Speed (mph) @ 10m										
N	0	7	64	76	54	6	0	0	207	3.9	10.1
NNE	0	10	61	62	19	1	0	0	153	2.9	8.3
NE	0	8	50	72	12	0	0	0	142	2.7	8.3
ENE	0	3	34	59	12	1	0	0	109	2.0	8.9
E	0	3	26	34	30	0	0	0	93	1.7	10.0
ESE	0	8	29	90	116	7	0	0	250	4.7	12.0
SE	0	2	34	307	241	45	0	0	629	11.8	12.6
SSE	0	4	52	319	337	54	0	0	766	14.4	12.9
S	0	6	109	1039	544	25	0	0	1723	32.3	11.5
SSW	0	6	105	297	86	3	0	0	497	9.3	9.9
SW	0	6	53	101	24	2	0	0	186	3.5	9.1
WSW	0	4	15	33	3	0	0	0	55	1.0	8.2
W	0	5	30	11	1	0	0	0	47	0.9	6.6
WNW	0	3	52	28	7	0	0	0	90	1.7	7.4
NW	0	8	47	27	26	7	0	0	115	2.2	9.6
NNW	0	10	55	112	67	23	2	0	269	5.0	11.1
Total	0	93	816	2667	1579	174	2	0	5331	100.0	
Percent	0.0	1.7	15.3	50.0	29.6	3.3	0.0	0.0	100.0		

Average Speed for This Table	11.1	Number of Invalid Hours	978
Number of Calm Hours for this Table	0	Number of Valid Hours for this Table	5331
Number of Variable Direction Hours for this Table	0	Total Hours for this Period	43847

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Table 4.1-3
Joint Frequency Distribution for 2000-2004
Stability Class: B Moderately Unstable

Wind Direction	Calm	Calm-3.5	3.6-7.5	7.6-12.5	12.6-18.5	18.6-24.5	24.6-32.5	> 32.5	Total	Percent	Average Speed
	Wind Speed (mph) @ 10m										
N	0	0	27	63	48	4	0	0	142	5.7	11.0
NNE	0	2	40	70	28	0	0	0	140	5.6	9.4
NE	0	2	36	85	20	0	0	0	143	5.7	9.3
ENE	0	5	27	57	17	0	0	0	106	4.3	9.3
E	0	2	21	41	20	3	0	0	87	3.5	10.1
ESE	0	4	18	96	100	7	0	0	225	9.0	12.2
SE	0	0	27	203	132	23	0	0	385	15.4	12.2
SSE	0	1	33	186	119	12	0	0	351	14.1	11.8
S	0	1	75	244	87	1	0	0	408	16.4	10.3
SSW	0	0	55	52	11	0	0	0	118	4.7	8.3
SW	0	0	27	27	3	0	0	0	57	2.3	7.8
WSW	0	1	14	10	2	0	0	0	27	1.1	7.5
W	0	0	17	8	0	0	0	0	25	1.0	7.1
WNW	0	3	30	5	3	0	0	0	41	1.6	6.3
NW	0	1	26	27	21	5	0	0	80	3.2	10.4
NNW	0	2	32	57	55	12	1	0	159	6.4	11.5
Total	0	24	505	1231	666	67	1	0	2494	100.0	
Percent	0.0	1.0	20.2	49.4	26.7	2.7	0.0	0.0	100.0		

Average Speed for This Table	10.7	Number of Invalid Hours	978
Number of Calm Hours for this Table	0	Number of Valid Hours for this Table	2494
Number of Variable Direction Hours for this Table	0	Total Hours for this Period	43847

Table 4.1-4
Joint Frequency Distribution for 2000-2004
Stability Class: C Slightly Unstable

Wind Direction	Calm	Calm-3.5	3.6-7.5	7.6-12.5	12.6-18.5	18.6-24.5	24.6-32.5	> 32.5	Total	Percent	Average Speed
	Wind Speed (mph) @ 10m										
N	0	7	38	66	65	10	1	0	187	6.7	11.4
NNE	0	1	52	74	36	2	0	0	165	6.0	9.7
NE	0	6	60	112	31	0	0	1	210	7.6	9.3
ENE	0	5	39	72	21	0	0	0	137	4.9	9.3
E	0	5	43	53	28	4	3	0	136	4.9	10.1
ESE	0	0	40	94	92	13	0	0	239	8.6	11.9
SE	0	5	41	236	173	22	0	0	477	17.2	11.9
SSE	0	1	47	212	137	12	0	0	409	14.8	11.5
S	0	3	61	185	67	0	0	0	316	11.4	10.0
SSW	0	1	45	41	6	0	0	0	93	3.4	8.2
SW	0	1	33	28	6	0	0	0	68	2.5	7.7
WSW	0	0	10	10	1	0	0	0	21	0.8	8.1
W	0	3	16	2	3	0	0	0	24	0.9	6.6
WNW	0	5	42	5	3	0	0	0	55	2.0	6.0
NW	0	3	32	24	21	3	0	0	83	3.0	9.6
NNW	0	1	29	56	43	17	5	0	151	5.4	12.4
Total	0	47	628	1270	733	83	9	1	2771	100.0	
Percent	0.0	1.7	22.7	45.8	26.5	3.0	0.3	0.0			

Average Speed for This Table	10.6	Number of Invalid Hours	978
Number of Calm Hours for this Table	0	Number of Valid Hours for this Table	2771
Number of Variable Direction Hours for this Table	0	Total Hours for this Period	43847

4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

Table 4.1-5
Joint Frequency Distribution for 2000-2004
Stability Class: D Neutral

Wind Direction	Calm	Wind Speed (mph) @ 10m Calm-3.5	3.6-7.5	7.6-12.5	12.6-18.5	18.6-24.5	24.6-32.5	> 32.5	Total	Percent	Average Speed
N	0	36	224	583	538	76	2	0	1459	11.5	11.7
NNE	0	32	282	535	203	7	0	0	1059	8.4	9.5
NE	0	55	283	505	125	1	0	0	969	7.6	8.9
ENE	0	37	204	334	80	1	0	0	656	5.2	8.8
E	0	37	190	316	200	14	0	1	758	6.0	10.1
ESE	0	35	207	444	246	22	2	1	957	7.6	10.3
SE	0	18	304	830	532	32	2	0	1718	13.6	10.9
SSE	0	16	267	1006	632	25	2	0	1948	15.4	11.2
S	0	15	194	721	276	8	0	0	1214	9.6	10.4
SSW	0	8	111	155	18	0	0	0	292	2.3	8.4
SW	0	7	41	87	12	0	0	0	147	1.2	8.6
WSW	0	5	32	24	10	0	0	0	71	0.6	8.3
W	0	9	35	18	11	0	0	0	73	0.6	7.6
WNW	0	26	60	34	12	1	0	0	133	1.0	6.9
NW	0	32	117	90	89	9	0	0	337	2.7	9.4
NNW	0	29	170	316	306	55	1	0	877	6.9	11.4
Total	0	397	2721	5998	3290	251	9	2	12668	100.0	
Percent	0.0	3.1	21.5	47.3	26.0	2.0	0.1	0.0	100.0		

Average Speed for This Table	10.3	Number of Invalid Hours	978
Number of Calm Hours for this Table	0	Number of Valid Hours for this Table	12668
Number of Variable Direction Hours for this Table	0	Total Hours for this Period	43847

Table 4.1-6
Joint Frequency Distribution for 2000-2004
Stability Class: E Slightly Stable

Wind Direction	Calm	Wind Speed (mph) @ 10m Calm-3.5	3.6-7.5	7.6-12.5	12.6-18.5	18.6-24.5	24.6-32.5	> 32.5	Total	Percent	Average Speed
N	1	54	284	269	82	5	0	0	695	6.1	8.2
NNE	0	50	313	302	20	2	1	2	690	6.0	7.5
NE	0	55	397	218	20	0	0	0	690	6.0	6.9
ENE	0	58	351	175	17	0	0	0	601	5.2	6.6
E	0	52	415	277	46	1	0	0	791	6.9	7.4
ESE	0	51	485	340	53	5	1	0	935	8.1	7.5
SE	0	36	847	580	86	1	0	0	1550	13.5	7.6
SSE	0	22	873	989	188	3	0	0	2075	18.1	8.5
S	0	17	456	823	79	3	0	0	1378	12.0	8.6
SSW	0	7	314	320	18	0	0	0	659	5.7	7.8
SW	0	10	111	165	7	0	0	0	293	2.6	8.1
WSW	0	11	40	44	8	1	0	0	104	0.9	7.7
W	0	16	61	16	5	0	0	0	98	0.9	6.0
WNW	0	28	58	21	4	0	0	0	111	1.0	5.7
NW	0	38	121	68	19	1	0	0	247	2.2	6.9
NNW	0	52	243	209	49	8	0	0	561	4.9	8.0
Total	1	557	5369	4816	701	30	2	2	11478	100.0	
Percent	0.0	4.9	46.8	42.0	6.1	0.3	0.0	0.0	100.0		

Average Speed for This Table	7.8	Number of Invalid Hours	978
Number of Calm Hours for this Table	1	Number of Valid Hours for this Table	11478
Number of Variable Direction Hours for this Table	0	Total Hours for this Period	43847

4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

Table 4.1-7
Joint Frequency Distribution for 2000-2004
Stability Class: F Moderately Stable

Wind Direction	Calm	Calm-3.5	3.6-7.5	7.6-12.5	12.6-18.5	18.6-24.5	24.6-32.5	> 32.5	Total	Percent	Average Speed
	Wind Speed (mph) @ 10m										
N	1	66	161	19	1	0	0	0	248	5.6	5.1
NNE	0	99	239	75	0	0	0	0	413	9.4	5.4
NE	0	103	268	32	0	0	0	0	403	9.2	4.7
ENE	0	115	247	13	0	0	0	0	375	8.5	4.4
E	0	141	311	26	0	0	0	0	478	10.9	4.6
ESE	0	159	304	8	1	0	0	0	472	10.7	4.3
SE	0	114	555	20	0	0	0	0	689	15.7	4.7
SSE	0	35	413	42	1	0	0	0	491	11.2	5.7
S	0	11	124	22	1	0	0	0	158	3.6	6.0
SSW	0	8	42	7	0	0	0	0	57	1.3	5.9
SW	0	5	16	8	0	0	0	0	29	0.7	6.1
WSW	0	11	17	12	2	0	0	0	42	1.0	6.1
W	0	22	62	6	0	0	0	0	90	2.0	4.6
WNW	0	38	57	0	0	0	0	0	95	2.2	4.0
NW	0	47	92	20	0	0	0	0	159	3.6	4.9
NNW	0	48	138	17	0	0	0	0	203	4.6	5.0
Total	1	1022	3046	327	6	0	0	0	4402	100.0	
Percent	0.0	23.2	69.2	7.4	0.1	0.0	0.0	0.0	100.0		

Average Speed for This Table	4.9	Number of Invalid Hours	978
Number of Calm Hours for this Table	1	Number of Valid Hours for this Table	4402
Number of Variable Direction Hours for this Table	0	Total Hours for this Period	43847

Table 4.1-8
Joint Frequency Distribution for 2000-2004
Stability Class: G Extremely Stable

Wind Direction	Calm	Calm-3.5	3.6-7.5	7.6-12.5	12.6-18.5	18.6-24.5	24.6-32.5	> 32.5	Total	Percent	Average Speed
	Wind Speed (mph) @ 10m										
N	1	128	118	6	0	0	0	0	253	6.8	3.8
NNE	0	214	313	24	0	0	0	0	551	14.8	4.2
NE	0	316	408	19	0	0	0	0	743	19.9	4.1
ENE	0	234	244	2	0	0	0	0	480	12.9	3.7
E	0	163	202	4	0	0	0	0	369	9.9	3.9
ESE	0	120	149	1	0	0	0	0	270	7.2	3.8
SE	0	73	146	1	0	0	0	0	220	5.9	4.1
SSE	1	24	72	6	0	0	0	0	103	2.8	4.9
S	0	12	11	0	0	0	0	0	23	0.6	3.4
SSW	0	5	3	0	0	0	0	0	8	0.2	3.1
SW	0	7	1	0	0	0	0	0	8	0.2	2.1
WSW	0	5	3	0	0	0	0	0	8	0.2	3.6
W	0	16	54	5	0	0	0	0	75	2.0	4.7
WNW	1	94	96	0	0	0	0	0	191	5.1	3.6
NW	1	93	98	6	0	0	0	0	198	5.3	3.8
NNW	0	103	118	4	0	0	0	0	225	6.0	3.9
Total	4	1607	2036	78	0	0	0	0	3725	100.0	
Percent	0.1	43.1	54.7	2.1	0.0	0.0	0.0	0.0	100.0		

Average Speed for This Table	4.0	Number of Invalid Hours	978
Number of Calm Hours for this Table	4	Number of Valid Hours for this Table	3725
Number of Variable Direction Hours for this Table	0	Total Hours for this Period	43847

4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

Table 4.1-9
Stability Class Distribution
(%)

Stability Class	Updated Meteorological Data[3] 2000	2001	2002	2003	2004	Time Period Averages CLB[4] (1973-1977)	2000-2004	Difference
A	14.6 (15)	15.1 (15)	11.3 (13)	12.4 (14)	8.4 (9)	7.6	12.4	4.8
B	5.8 (6)	5.4 (5)	7.3 (7)	5.3 (5)	5.3 (5)	6.0	5.8	-0.2
C	6.3 (6)	6.6 (7)	6.4 (6)	6.5 (6)	6.5 (6)	6.9	6.5	-0.4
D	27.5 (27)	27.2 (27)	30.7 (31)	31.0 (31)	31.2 (31)	32.2	29.5	-2.7
E	31.0 (31)	23.2 (23)	24.9 (25)	25.4 (25)	29.5 (30)	23.1	26.8	3.7
F	8.7 (8)	11.5 (11)	10.3 (10)	10.2 (10)	10.7 (11)	14.1	10.3	-3.8
G	6.1 (6)	11.0 (11)	9.1 (9)	8.9 (9)	8.4 (9)	10.1	8.7	-1.4
% Data	97.2	95.7	99.7	99.9	96.3	-	97.8	-

() STP percentage

Table 4.1-10
Average Wind Speed and Peak Wind Direction: 2000-2004

Year	Average Wind Speed (mph)	Peak Wind Direction Sector
2000	9.4	SSE
2001	8.4	SE
2002	8.8	SSE
2003	8.1	S
2004	8.7	SSE
2000-2005	8.7	SSE

[3] Based on joint frequency data for 10m wind speed, wind direction, and delta T 60-1-m.
[4] From UFSAR, Rev 13, Table 2.3-13

4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

Table 4.1-11
Wind Direction Distribution

Wind Direction	Percent of CLB[5]	Percent of Recent Data
NNE	7.1	7.4
NE	7.3	7.7
ENE	5.4	5.8
E	5.7	6.3
ESE	6.4	7.8
SE	13.5	13.2
SSE	15.2	14.3
S	12.6	12.2
SSW	4.8	4.0
SW	2.3	1.8
WSW	1.1	0.8
W	1.3	1.0
WNW	1.3	1.7
NW	2.3	2.8
NNW	5.6	5.7
N	7.7	7.4
Calm	0.3	0.014

[5] From UFSAR, Rev 13, Table 2.3-29, [Wind Frequency Distribution for All Observations]

4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

Figure 4.1-1 Comparison of Wind Speed Distribution for STPEGS

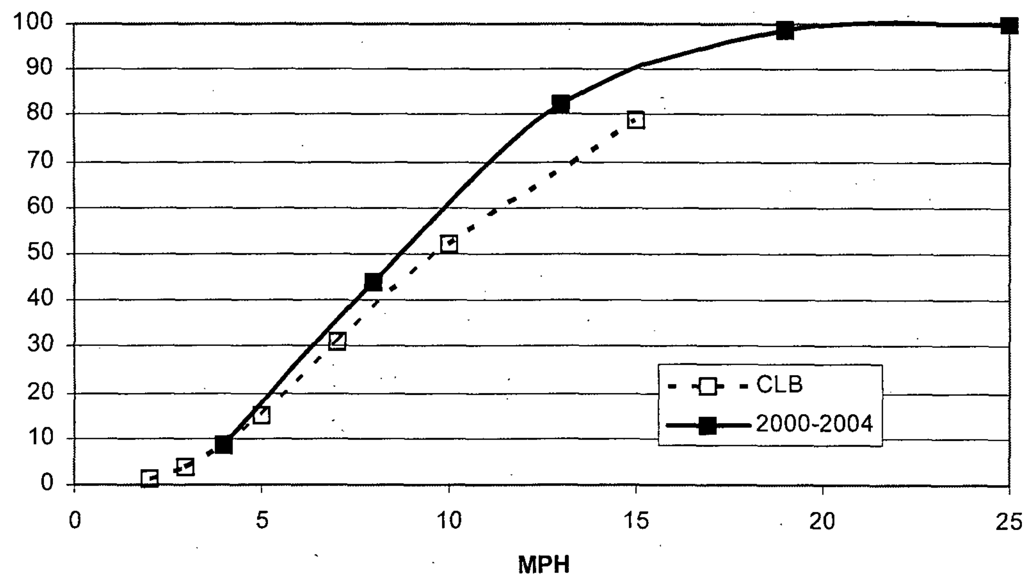
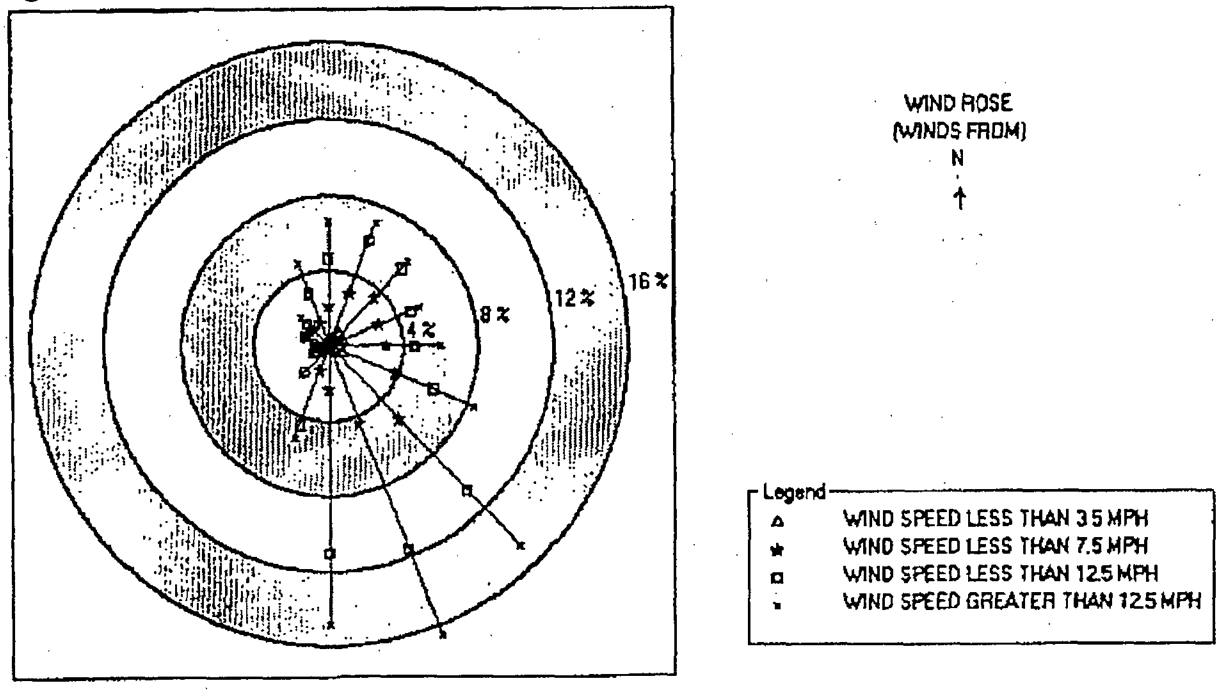


Figure 4.1-2 Wind Rose for 2000



4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

Figure 4.1-3 Wind Rose for 2001

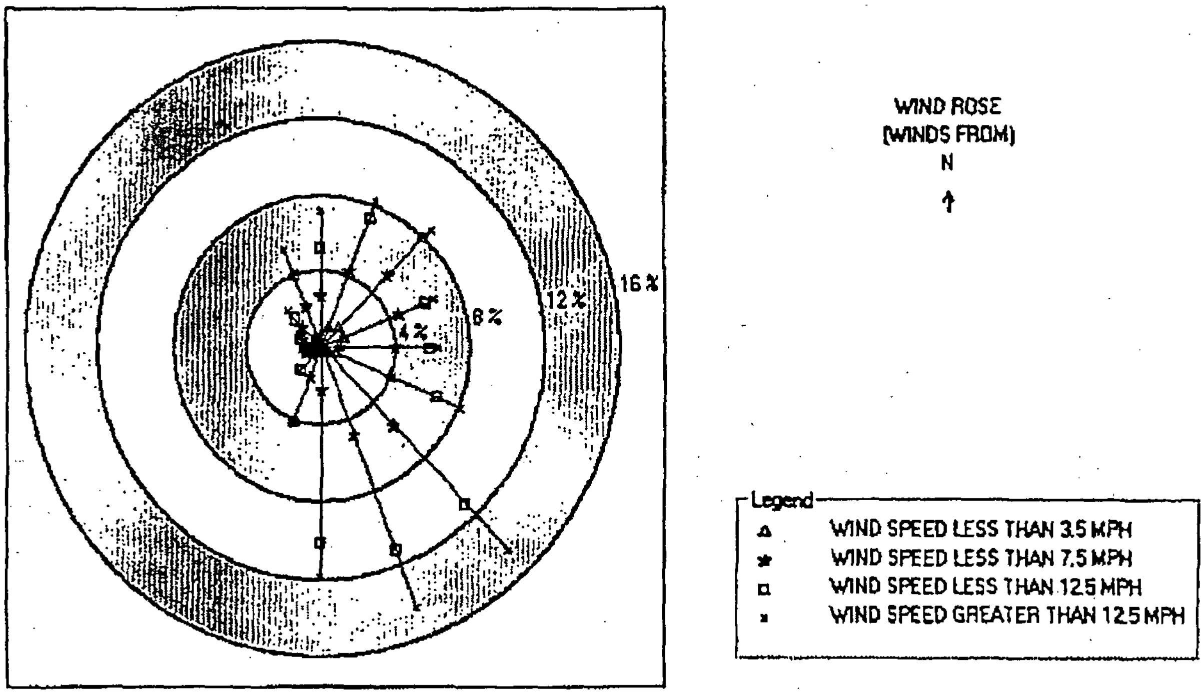
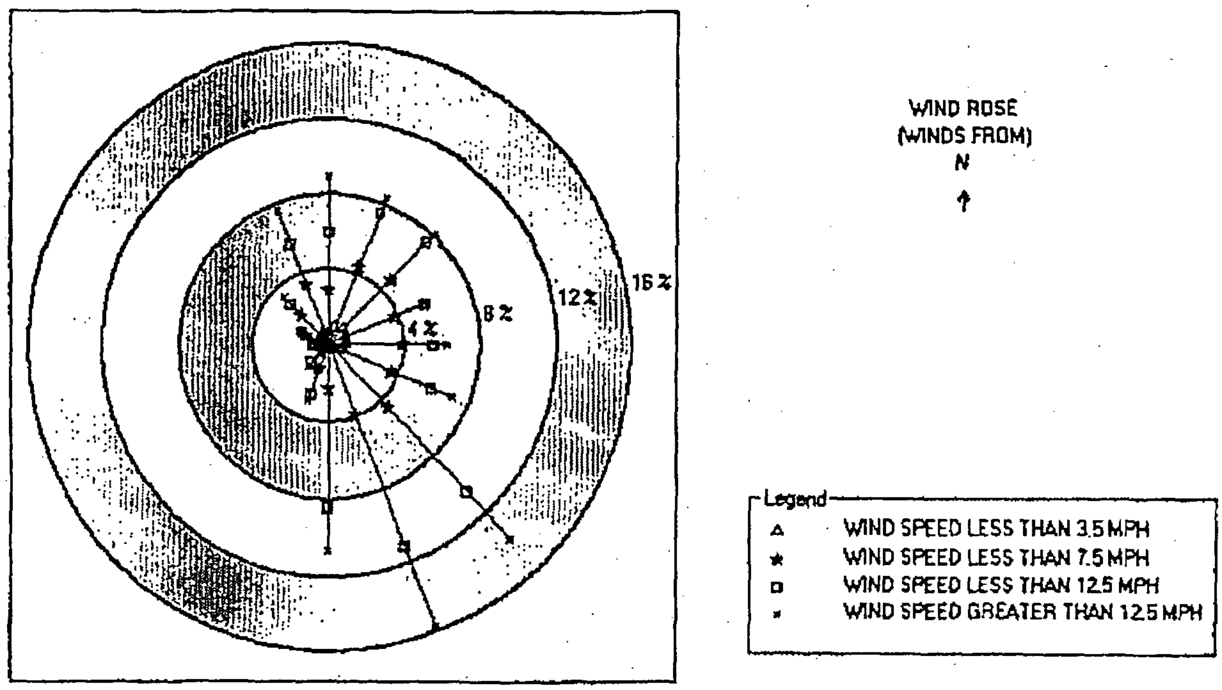


Figure 4.1-4 Wind Rose for 2002



4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

Figure 4.1-5 Wind Rose for 2003

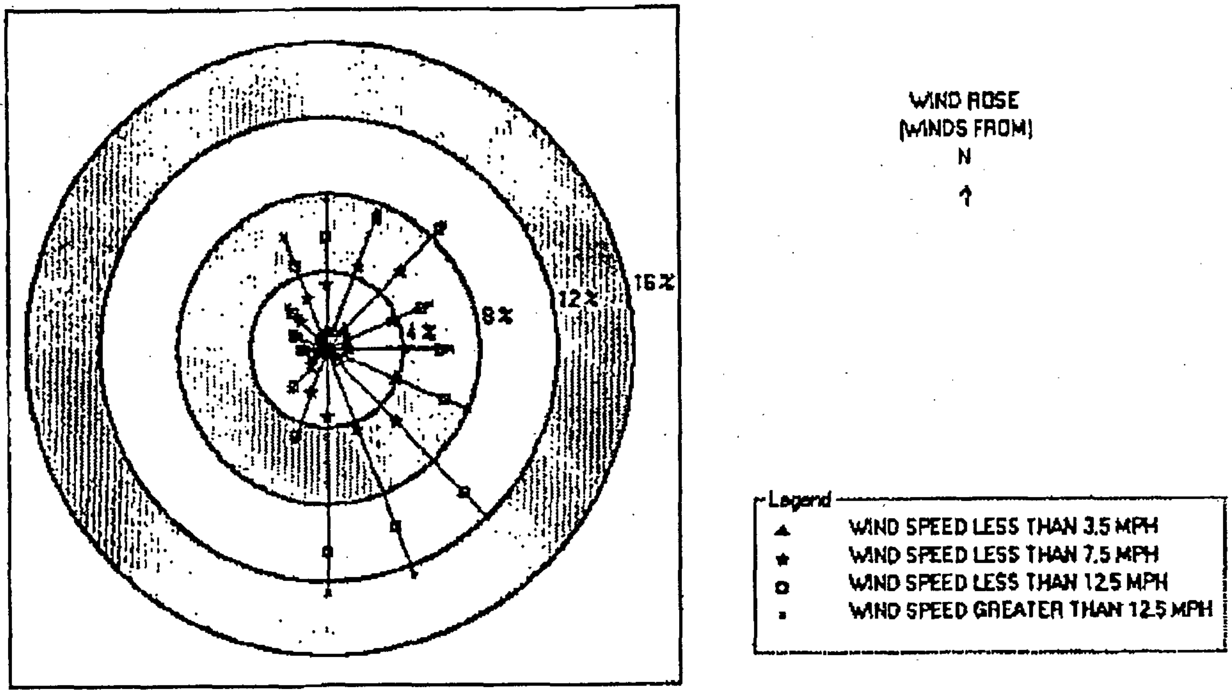
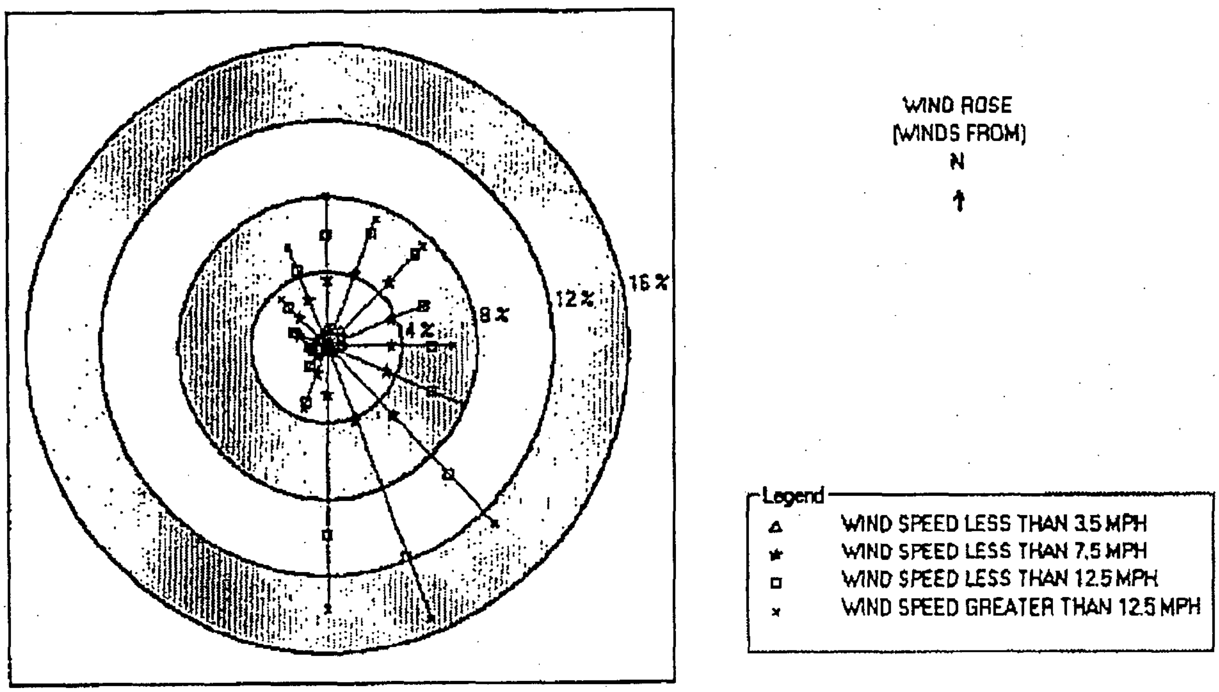
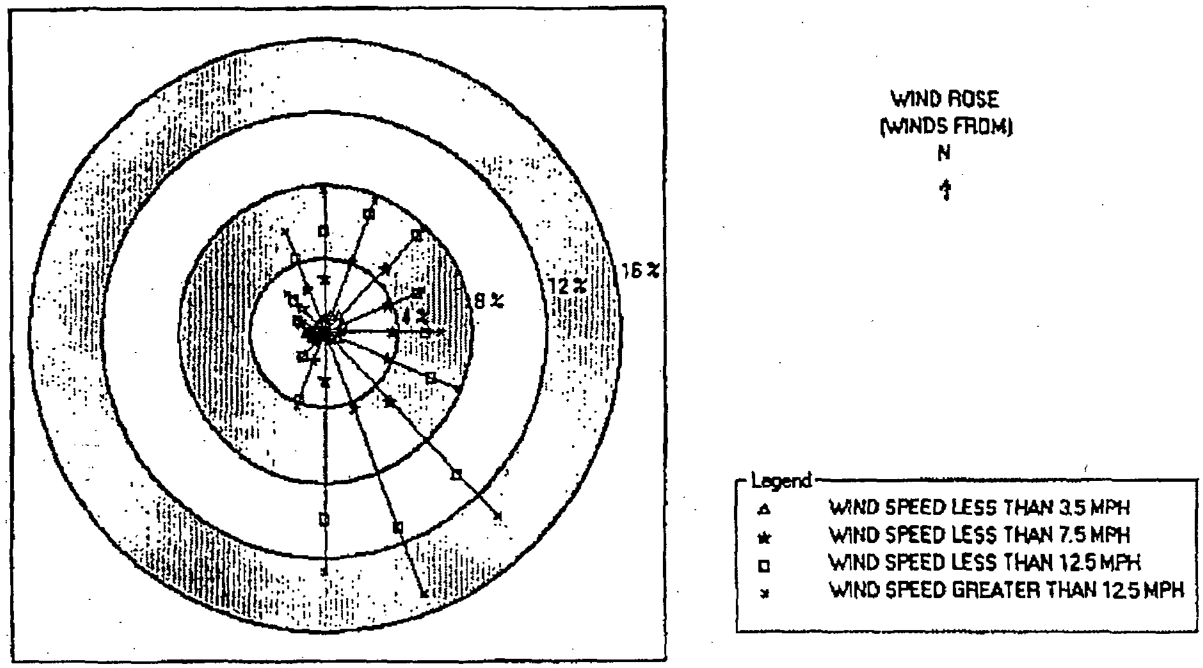


Figure 4.1-6 Wind Rose for 2004



4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

Figure 4.1-7 Wind Rose for 2000-2004



4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

Figure 4.1-8 Delta-T Frequency for 2000

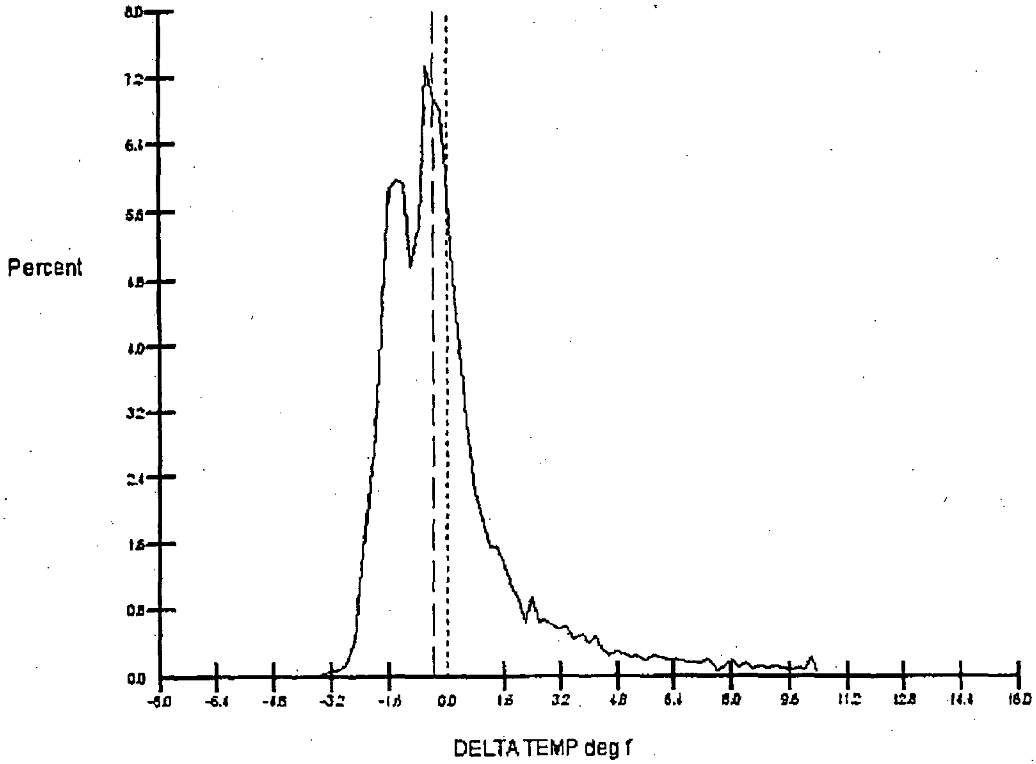
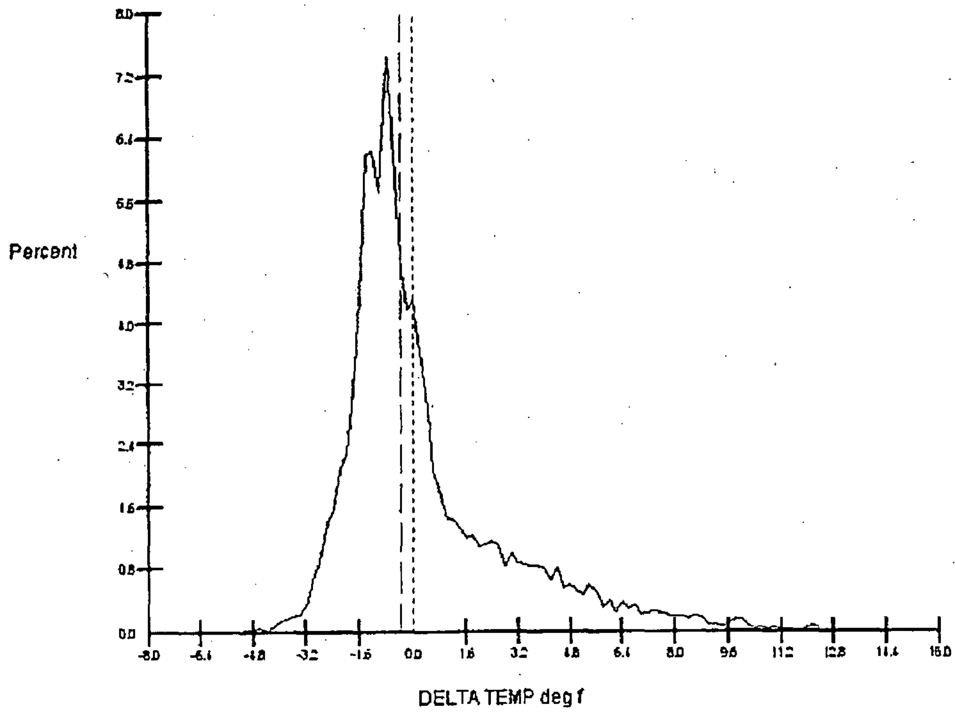


Figure 4.1-9 Delta-T Frequency for 2001



4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

Figure 4.1-10 Delta-T Frequency for 2002

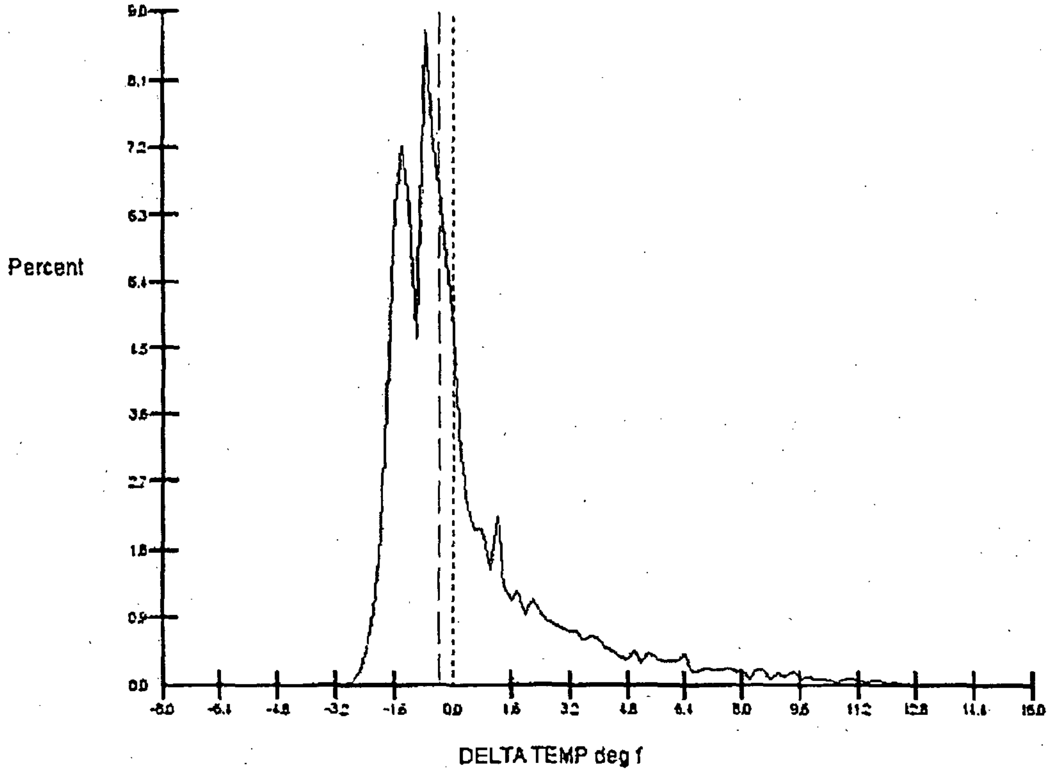
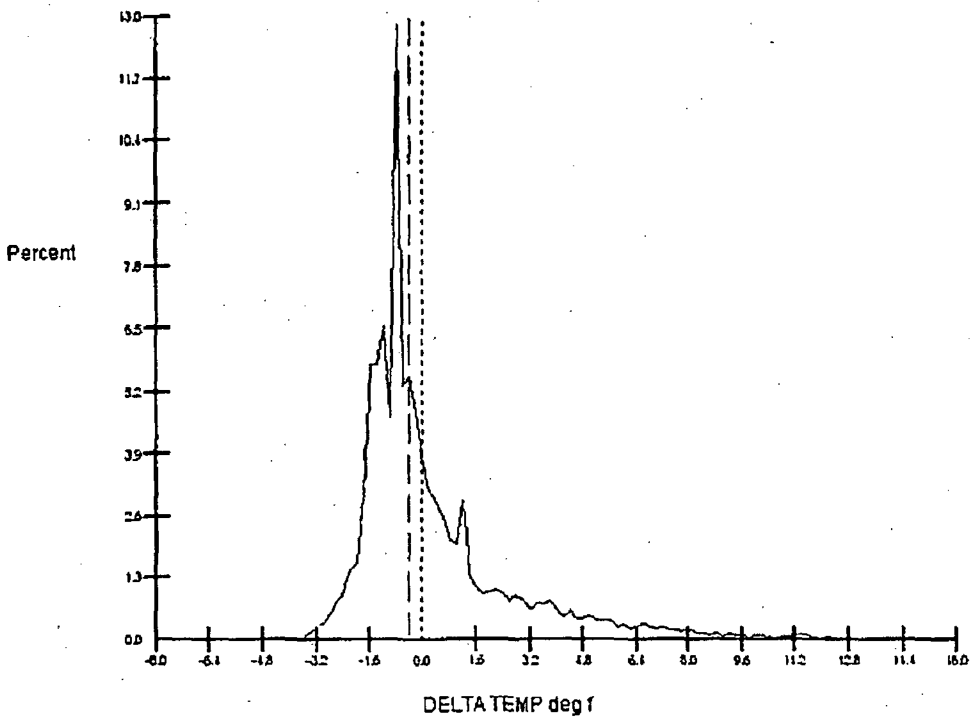
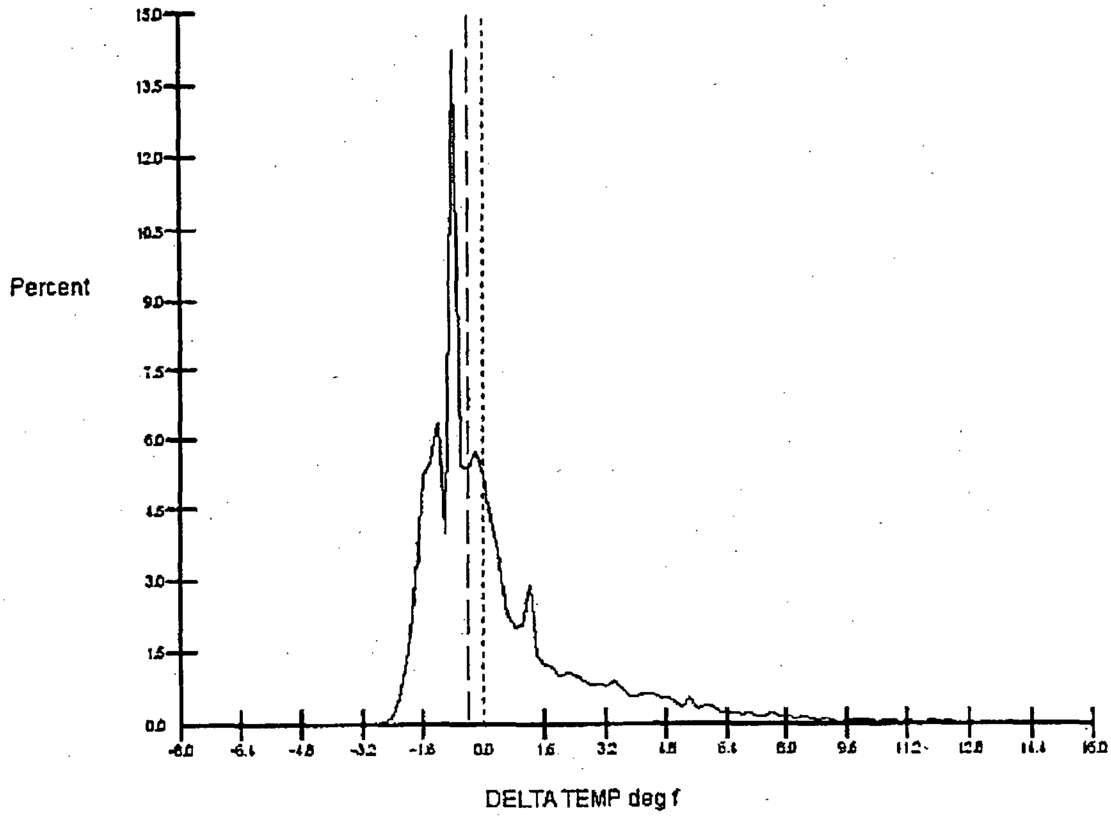


Figure 4.1-11 Delta-T Frequency for 2003



4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

Figure 4.1-12 Delta-T Frequency for 2004



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4.1.2 Determination of the EAB and LPZ χ/Qs

Source/Receptor Scenarios and Configurations
The minimum EAB and LPZ boundaries are located at 1430 m and 4800 m. Postulated releases do not qualify as elevated releases in accordance with Regulatory Guide 1.145; therefore, they were executed by PAVAN as "ground" type releases requiring an assumption of a 10 m release height. The minimum cross sectional area of the containment building used for the building wake calculation is 2734 m^2. A containment height of 61.9 meters was used for the building wake factor in the annual average calculation.

Meteorological Data (PAVAN)
Meteorological data from the five-year period, (i.e., 2000 - 2004), were used in the PAVAN analysis. Independent of the consistency analysis performed on the meteorology data described in Section 4.1.1, the hourly data was processed into joint wind-stability occurrence frequency distribution for input into PAVAN. The only differences from the processing and checks described in Section 4.1.1 were:

- the inclusion of the calms;
- the F stability observations that became G stability because of the different classification scheme; and,
- the addition of the 36 mph upper bound so that a representative wind speed could be used for the highest speed bin.

The data were independently processed to develop PAVAN input with the following accounting:

- Total number of observations: 43847
- Total number of observations found to be invalid: 978
- Total number of F Stability observations from met data evaluation: 4402 (including 1 calm)
- Total number of G Stability observations from met data evaluation: 3725 (including 4 calms)
- Total number of F Stability observations reclassified as G Stability for PAVAN: 157 (3.6% of F Stability observations from met data evaluation)
- Total number of F Stability observations for PAVAN: 4245
- Total number of G Stability observations for PAVAN: 3882 (including 5 calms)

Apparent Discrepancies in Stability Categories from Section 4.1.1 to the PAVAN Input
When comparing the distributions from the consistency checks in Section 4.1.1 to the PAVAN input, it appears that 153 observations (1 calm, 42 1-4 mph, 100 4-8 mph, and 10 8-13 mph observations) have been transferred from F Stability to G Stability. Safety Guide 23, Table 2, has the cutoffs for lapse rate, but does not clearly define what is to be done on the boundaries between the ranges. For instance, D is "-1.5 to -0.5" and E is "-0.5 to 1.5" °C/100m. F is defined as "1.5 to 4.0", and G is ">4.0". The ABS conversion program to convert the hourly data to PAVAN always puts cases on the boundary in the more stable category, making G ">=4.0". The

4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

temperatures supplied were °F per 50m, so 3.6 in raw data is exactly 4.0 °C/100m. In the consistency checks performed for Section 4.1.1, these cases are classified as F, while the conversion program for PAVAN classified them as G. This is the source of the discrepancy, and leads to more conservative PAVAN results.

Wind Speed Categories

Seven wind speed categories were defined according to Safety Guide 23 with the first category identified as "calm." The minimum wind speed (i.e., wind threshold) was set to 0.5 mph and "calm" wind speeds were distributed into the first wind speed group. The Safety Guide 23 wind speed categories and the categories used in this PAVAN analysis are presented in Table 4.1-12.

PAVAN requires an upper limit for the highest wind speed bin so as to have an average wind speed to use for computation. In the ABS conversion program, this is determined to be value for which the arithmetic mean wind speed in the bin matched the actual arithmetic mean wind speed of all of the hours in the bin.

NRC Regulatory Issues Summary 2006-4, *Experience With Implementation Of Alternative Source Terms* (Reference 43), states that:

> *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).*

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," provides suggestions for a set of wind speeds as follows:

> *"It has been found that ENVELOP produces the best results near the 0.5 percentile if the wind-speed data are classified into a large number of categories at the lower wind speeds, e.g., calm speed, 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/second (see Card Type 11). The important aspect of having a large number of lower wind speed categories is to generate more X/Q values at the lower values of the cumulative frequency since the 0.5% value is required."*

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The guidance given with the input description (Table 3.1, Card Type 10; Card type 11 is actually the downwind boundary distances) is as follows:

> *"Maximum wind speed in each wind-speed category, in either miles/hour or meters/second. (If in miles/hour, set UCOR greater than 100). So that calms can be properly apportioned a direction, it is preferable that the first wind speed category have a maximum wind speed less than 1.5 meters/second."*

ABS Consulting used the same seven wind speed groups for STP that are used in most of their work which are as follows: 0.5 mph, 3.5 mph, 7.5 mph, 12.5 mph, 18.5 mph, 24.5 mph, and 36.0 mph (i.e., 0.22 m/s, 1.56 m/s, 3.35 m/s, 5.59 m/s, 8.27 m/s, 10.95 m/s, and 16.09 m/s). These are appropriate for determining the offsite X/Q values using PAVAN.

Table 4.1-12
Defined Wind Speed Category Ranges For PAVAN Modeling

Category No.	Safety Guide 23 Speed Interval (mph)	PAVAN-Assumed Maximum Speed (mph)
1 (Calm)	0 to < 1	0.5[6]
2	1 to 3	3.5
3	4 to 7	7.5
4	8 to 12	12.5
5	13 to 18	18.5
6	19 to 24	24.5
7	> 24	36.0[7]

Joint Frequency data for the PAVAN input is presented in Tables 4.1-13 through 4.1-19. A summary is presented on Table 4.1-20.

Calculations

PAVAN output summaries are provided on Tables 4.1-21, -22, and -23. The χ/Q values for offsite locations were evaluated using the methods of Regulatory Guide 1.145. The offsite χ/Q values recalculated for AST are presented in Table 4.1-24 along with the CLB values for comparison. The 0-2 hr χ/Q is used for the worst 2 hour doses in the AST dose analyses. Note that in all cases, the revised χ/Q values are more conservative than the CLB values.

[6] Calms are distributed into the first wind speed group.
[7] Determined from the data in the last wind speed group such that the actual mean wind speed in that group matched the average of the upper and lower wind speed limits of that group.

4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

Table 4.1-13
PAVAN Input:
Joint Frequency Distribution for 2000-2004
Stability Class: A Extremely Unstable

Elevation: 10m

	Maximum Wind Speed (mph)							
Wind Direction	0.5	3.5	7.5	12.5	18.5	24.5	36.0	Total
N	7	64	76	54	6	0	0	207
NNE	10	61	62	19	1	0	0	153
NE	8	50	72	12	0	0	0	142
ENE	3	34	59	12	1	0	0	109
E	3	26	34	30	0	0	0	93
ESE	8	29	90	116	7	0	0	250
SE	2	34	307	241	45	0	0	629
SSE	4	52	319	337	54	0	0	766
S	6	109	1039	544	25	0	0	1723
SSW	6	105	297	86	3	0	0	497
SW	6	53	101	24	2	0	0	186
WSW	4	15	33	3	0	0	0	55
W	5	30	11	1	0	0	0	47
WNW	3	52	28	7	0	0	0	90
NW	8	47	27	26	7	0	0	115
NNW	10	55	112	67	23	2	0	269
Total	93	816	2667	1579	174	2	0	5331

4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

Table 4.1-14
PAVAN Input:
Joint Frequency Distribution for 2000-2004
Stability Class: B Moderately Unstable

Elevation: 10m

	Maximum Wind Speed (mph)							
Wind Direction	0.5	3.5	7.5	12.5	18.5	24.5	36.0	Total
N	0	27	63	48	4	0	0	142
NNE	2	40	70	28	0	0	0	140
NE	2	36	85	20	0	0	0	143
ENE	5	27	57	17	0	0	0	106
E	2	21	41	20	3	0	0	87
ESE	4	18	96	100	7	0	0	225
SE	0	27	203	132	23	0	0	385
SSE	1	33	186	119	12	0	0	351
S	1	75	244	87	1	0	0	408
SSW	0	55	52	11	0	0	0	118
SW	0	27	27	3	0	0	0	57
WSW	1	14	10	2	0	0	0	27
W	0	17	8	0	0	0	0	25
WNW	3	30	5	3	0	0	0	41
NW	1	26	27	21	5	0	0	80
NNW	2	32	57	55	12	1	0	159
Total	24	505	1231	666	67	1	0	2494

4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

Table 4.1-15
PAVAN Input:
Joint Frequency Distribution for 2000-2004
Stability Class: C Slightly Unstable

Elevation: 10m

	Maximum Wind Speed (mph)							
Wind Direction	0.5	3.5	7.5	12.5	18.5	24.5	36.0	Total
N	7	38	66	65	10	1	0	187
NNE	1	52	74	36	2	0	0	165
NE	6	60	112	31	0	1	0	210
ENE	5	39	72	21	0	0	0	137
E	5	43	53	28	4	3	0	136
ESE	0	40	94	92	13	0	0	239
SE	5	41	236	173	22	0	0	477
SSE	1	47	212	137	12	0	0	409
S	3	61	185	67	0	0	0	316
SSW	1	45	41	6	0	0	0	93
SW	1	33	28	6	0	0	0	68
WSW	0	10	10	1	0	0	0	21
W	3	16	2	3	0	0	0	24
WNW	5	42	5	3	0	0	0	55
NW	3	32	24	21	3	0	0	83
NNW	1	29	56	43	17	5	0	151
Total	47	628	1270	733	83	10	0	2771

4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

Table 4.1-16
PAVAN Input:
Joint Frequency Distribution for 2000-2004
Stability Class: D Neutral

Elevation: 10m

	Maximum Wind Speed (mph)							
Wind Direction	0.5	3.5	7.5	12.5	18.5	24.5	36.0	Total
N	36	224	583	538	76	2	1	1460
NNE	32	282	535	203	7	0	0	1059
NE	55	283	505	125	1	0	0	969
ENE	37	204	334	80	1	0	0	656
E	37	190	316	200	14	1	0	758
ESE	35	207	444	246	22	3	0	957
SE	18	304	830	532	32	2	0	1718
SSE	16	267	1006	632	25	2	0	1948
S	15	194	721	276	8	0	0	1214
SSW	8	111	155	18	0	0	0	292
SW	7	41	87	12	0	0	0	147
WSW	5	32	24	10	0	0	0	71
W	9	35	18	11	0	0	0	73
WNW	26	60	34	12	1	0	0	133
NW	32	117	90	89	9	0	0	337
NNW	29	170	316	306	55	1	0	877
Total	397	2721	5998	3290	251	11	1	12669

4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

Table 4.1-17
PAVAN Input:
Joint Frequency Distribution for 2000-2004
Stability Class: E Slightly Stable

Elevation: 10m

	Maximum Wind Speed (mph)							
Wind Direction	0.5	3.5	7.5	12.5	18.5	24.5	36.0	Total
N	54	284	269	82	5	0	0	695
NNE	50	313	302	20	2	3	0	690
NE	55	397	218	20	0	0	0	690
ENE	58	351	175	17	0	0	0	601
E	52	415	277	46	1	0	0	791
ESE	51	485	340	53	5	1	0	935
SE	36	847	580	86	1	0	0	1550
SSE	22	873	989	188	3	0	0	2075
S	17	456	823	79	3	0	0	1378
SSW	7	314	320	18	0	0	0	659
SW	10	111	165	7	0	0	0	293
WSW	11	40	44	8	1	0	0	104
W	16	61	16	5	0	0	0	98
WNW	28	58	21	4	0	0	0	111
NW	38	121	68	19	1	0	0	247
NNW	52	243	209	49	8	0	0	561
Total	557	5369	4816	701	30	4	0	11478

4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

Table 4.1-18
PAVAN Input:
Joint Frequency Distribution for 2000-2004
Stability Class: F Moderately Stable

Elevation: 10m

	Maximum Wind Speed (mph)							
Wind Direction	0.5	3.5	7.5	12.5	18.5	24.5	36.0	Total
N	63	158	19	1	0	0	2	243
NNE	92	228	72	0	0	0	0	392
NE	100	253	29	0	0	0	0	382
ENE	107	235	13	0	0	0	0	355
E	135	295	25	0	0	0	0	455
ESE	154	295	8	1	0	0	0	458
SE	110	543	20	0	0	0	0	673
SSE	33	406	42	1	0	0	1	483
S	11	121	22	1	0	0	0	155
SSW	7	42	7	0	0	0	0	56
SW	5	16	8	0	0	0	0	29
WSW	10	16	12	2	0	0	0	40
W	20	62	5	0	0	0	0	87
WNW	37	56	0	0	0	0	1	94
NW	45	85	18	0	0	0	1	149
NNW	47	135	17	0	0	0	0	199
Total	976	2946	317	6	0	0	5	4250

4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

Table 4.1-19
PAVAN Input:
Joint Frequency Distribution for 2000-2004
Stability Class: G Extremely Stable

Elevation: 10m

	Wind Speed (mph)							
Wind Direction	0.5	3.5	7.5	12.5	18.5	24.5	36.0	Total
N	131	121	6	0	0	0	0	258
NNE	221	324	27	0	0	0	0	572
NE	319	423	22	0	0	0	0	764
ENE	242	256	2	0	0	0	0	500
E	169	218	5	0	0	0	0	392
ESE	125	158	1	0	0	0	0	284
SE	77	158	1	0	0	0	0	236
SSE	26	79	6	0	0	0	0	111
S	12	14	0	0	0	0	0	26
SSW	6	3	0	0	0	0	0	9
SW	7	1	0	0	0	0	0	8
WSW	6	4	0	0	0	0	0	10
W	18	54	6	0	0	0	0	78
WNW	95	97	0	0	0	0	0	192
NW	95	105	8	0	0	0	0	208
NNW	104	121	4	0	0	0	0	229
Total	1653	2136	88	0	0	0	0	3877

4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

Table 4.1-20
Summary of PAVAN Input:
Joint Frequency Distribution for 2000-2004
Summary of All Stability Classes

Elevation: 10m

	Maximum Wind Speed (mph)							
Wind Direction	0.5	3.5	7.5	12.5	18.5	24.5	36.0	Total
N	298	916	1082	788	101	3	3	3191
NNE	408	1300	1142	306	12	3	0	3171
NE	545	1502	1043	208	1	1	0	3300
ENE	457	1146	712	147	2	0	0	2464
E	403	1208	751	324	22	4	0	2712
ESE	377	1232	1073	608	54	4	0	3348
SE	248	1954	2177	1164	123	2	0	5668
SSE	103	1757	2760	1414	106	2	1	6143
S	65	1030	3034	1054	37	0	0	5220
SSW	35	675	872	139	3	0	0	1724
SW	36	282	416	52	2	0	0	788
WSW	37	131	133	26	1	0	0	328
W	71	275	66	20	0	0	0	432
WNW	197	395-	93	29	1	0	1	716
NW	222	533	262	176	25	0	1	1219
NNW	245	785	771	520	115	9	0	2445
Total	3747	15121	16387	6975	605	28	6	42869

4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

Table 4.1-21
Relative Concentration (χ/Q) Values (sec/m^3) Versus Averaging Time
@EAB

Downwind Sector	0-2 hours	2-8 hours	8-24 hours	1-4 days	4-30 days	Annual Average	Hrs/yr 0-2hr χ/Q is Exceeded in Sector
S	1.22E-4	5.49E-5	3.69E-5	1.56E-5	4.53E-6	9.96E-7	39.4
SSW	1.23E-4	5.81E-5	3.99E-5	1.77E-5	5.49E-6	1.31E-6	29.5
SW	1.28E-4	6.20E-5	4.31E-5	1.96E-5	6.34E-6	1.59E-6	43.7
WSW	1.25E-4	5.81E-5	3.97E-5	1.73E-5	5.28E-6	1.23E-6	35.3
W	1.19E-4	5.57E-5	3.80E-5	1.66E-5	5.07E-6	1.19E-6	23.5
WNW	1.15E-4	5.38E-5	3.68E-5	1.62E-5	4.95E-6	1.16E-6	17.5
NW	1.12E-4	5.45E-5	3.80E-5	1.74E-5	5.66E-6	1.43E-6	9.3
NNW	1.02E-4	4.94E-5	3.44E-5	1.57E-5	5.06E-6	1.27E-6	27.3
N	5.53E-5	2.71E-5	1.89E-5	8.71E-6	2.86E-6	7.30E-7	1.7
NNE	3.96E-5	1.78E-5	1.20E-5	5.02E-6	1.45E-6	3.15E-7	1.0
NE	2.85E-5	1.21E-5	7.86E-6	3.10E-6	8.12E-7	1.58E-7	1.3
ENE	1.21E-5	5.43E-6	3.64E-6	1.52E-6	4.35E-7	9.38E-8	1.0
E	3.92E-5	1.63E-5	1.05E-5	4.04E-6	1.03E-6	1.92E-7	1.9
ESE	9.55E-5	3.91E-5	2.50E-5	9.49E-6	2.36E-6	4.30E-7	29.2
SE	1.04E-4	4.36E-5	2.83E-5	1.10E-5	2.86E-6	5.48E-7	30.3
SSE	1.12E-4	4.90E-5	3.24E-5	1.32E-5	3.65E-6	7.55E-7	14.9
Max χ/Q	1.28E-4				Total hours around Site:		306.8
SRP 2.3.4	2.02E-4	9.07E-5	6.08E-5	2.55E-5	7.32E-6	1.59E-6	
Site Limit	1.44E-4	6.84E-5	4.71E-5	2.10E-5	6.58E-6	1.59E-6	

DISTANCE:	1430 m
WIND SENSORS HEIGHT:	10 m
TYPE OF RELEASE:	Ground-level Release
DELTA-T HEIGHTS:	10.0 – 60.0 m

4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

Table 4.1-22
Relative Concentration (χ/Q) Values (sec/m^3) Versus Averaging Time
@ LPZ

Downwind Sector	0-2 hours	2-8 hours	8-24 hours	1-4 days	4-30 days	Annual Average	Hrs/yr 0-2hr χ/Q is Exceeded in Sector
S	3.54E-5	1.48E-5	9.57E-6	3.71E-6	9.52E-7	1.80E-7	21.2
SSW	4.59E-5	1.93E-5	1.25E-5	4.88E-6	1.27E-6	2.43E-7	29.5
SW	5.27E-5	2.24E-5	1.46E-5	5.75E-6	1.51E-6	2.95E-7	43.7
WSW	4.79E-5	1.98E-5	1.28E-5	4.90E-6	1.24E-6	2.30E-7	35.3
W	4.12E-5	1.74E-5	1.13E-5	4.43E-6	1.16E-6	2.23E-7	23.5
WNW	3.62E-5	1.55E-5	1.02E-5	4.07E-6	1.09E-6	2.17E-7	17.5
NW	3.31E-5	1.49E-5	9.98E-6	4.19E-6	1.21E-6	2.64E-7	9.3
NNW	2.64E-5	1.21E-5	8.14E-6	3.47E-6	1.02E-6	2.28E-7	12.1
N	1.32E-5	6.13E-6	4.18E-6	1.82E-6	5.50E-7	1.27E-7	1.6
NNE	9.23E-6	3.97E-6	2.60E-6	1.04E-6	2.79E-7	5.57E-8	0.8
NE	6.09E-6	2.50E-6	1.60E-6	6.08E-7	1.52E-7	2.77E-8	1.1
ENE	2.10E-6	9.47E-7	6.36E-7	2.69E-7	7.80E-8	1.72E-8	0.9
E	8.55E-6	3.46E-6	2.20E-6	8.27E-7	2.02E-7	3.62E-8	1.9
ESE	2.72E-5	1.04E-5	6.41E-6	2.25E-6	5.02E-7	7.99E-8	14.9
SE	2.94E-5	1.15E-5	7.21E-6	2.61E-6	6.05E-7	1.01E-7	15.4
SSE	3.28E-5	1.33E-5	8.44E-6	3.16E-6	7.72E-7	1.38E-7	14.9
Max χ/Q	5.27E-5			Total hours around Site:			243.7
SRP 2.3.4 Site Limit	6.39E-5 4.29E-5	2.63E-5 1.88E-5	1.68E-5 1.25E-5	6.42E-6 5.10E-6	1.61E-6 1.42E-6	2.95E-7 2.95E-7	

DISTANCE:	4800 m
WIND SENSORS HEIGHT:	10 m
TYPE OF RELEASE:	Ground-level Release
DELTA-T HEIGHTS:	10.0 – 60.0 m

4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

Table 4.1-23
χ/Q Values Based on DT(60M-10M) Stability Data and 10 Meter Winds

January 1, 2000 – December 31, 2004

Averaging Time[8]	Distance	Maximum Sector[9]	5 Percent Overall	50 Percent Overall	Worst Case[10]
1-hour 2-hour	Minimum Exclusion Area Boundary (EAB)- 1430 meters	1.28×10^{-4} (SW)	2.02×10^{-4}	2.19×10^{-5}	8.59×10^{-4} (ESE,SE,S,NNW)
1-hour 2-hour	Low Population Zone (LPZ)- 4800 meters	5.27×10^{-5} (SW)	6.40×10^{-5}	3.67×10^{-6}	3.18×10^{-4} (ESE,SE,S,NNW)
8-hour	Low Population Zone (LPZ)- 4800 meters	2.24×10^{-5} (SW)	2.63×10^{-5}		
16-hour	Low Population Zone (LPZ)- 4800 meters	1.46×10^{-5} (SW)	1.68×10^{-5}		
72-hour	Low Population Zone (LPZ)- 4800 meters	5.75×10^{-6} (SW)	6.42×10^{-6}		
624-hour	Low Population Zone (LPZ)- 4800 meters	1.51×10^{-6} (SW)	1.61×10^{-6}		

The directions for the sectors given above are the directions of the "Affected Sectors" (i.e., wind from the east will affect a west sector).

Table 4.1-24
χ/Q Values for Radiological Dose Calculations – EAB and LPZ
(sec/m^3)

	EAB (1430m)		LPZ (4800m)	
Time Interval	CLB	Updated Met Data (PAVAN)	CLB	Updated Met Data (PAVAN)
0-2 hrs	1.3E-4	1.44E-4	3.8E-5	5.27E-5
2-8 hrs	N/A	N/A	1.6E-5	2.24E-5
8-24 hrs	N/A	N/A	1.1E-5	1.46E-5
1-4 days	N/A	N/A	4.3E-6	5.75E-6
4-30 days	N/A	N/A	1.2E-6	1.51E-6

[8] The 1-hour value was calculated; the 2-hour value is assumed to be equal to the 1-hour value.
[9] Maximum sector values are the highest 0.5 percent Sector χ/Q values.
[10] Worst case values are the highest calculated 1-hour values, all sectors considered.

4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

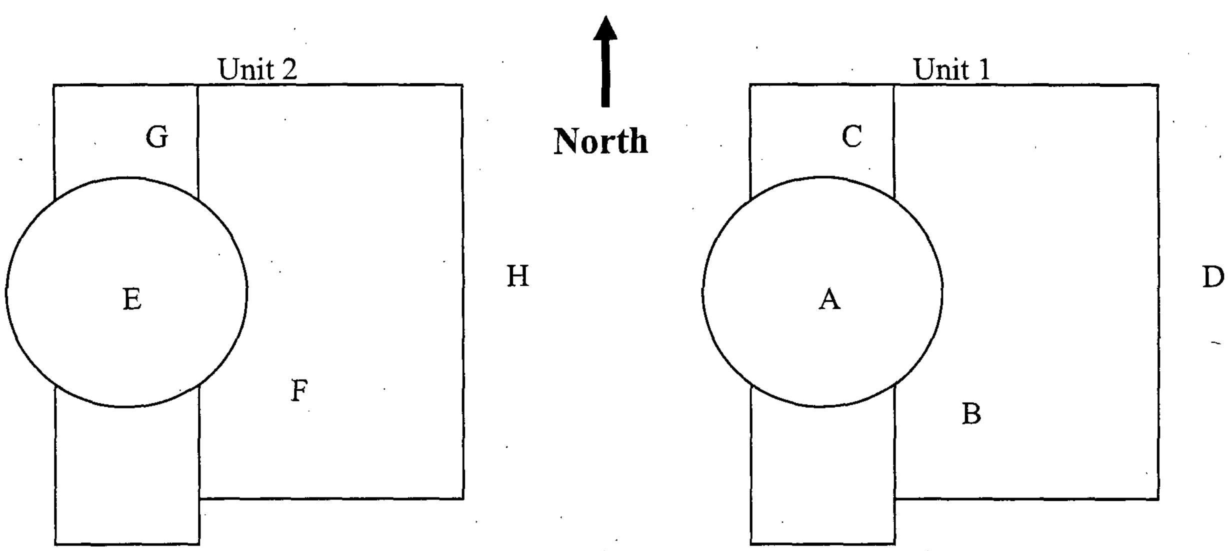
4.1.3 Control Room and Technical Support Center χ/Q Analyses

For each unit at STP, there are three release points:

- the containment building outer wall surface;
- the Plant Vent; and,
- the SG power operated relief valve (PORV) nearest the Control Room intake (this is also the area of the steam release for a postulated MSLB)

These points are illustrated in Figure 4.1-13.

Figure 4.1-13 Simplified Plot Plan with Release Points and Receptors



4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

Table 4.1-25
Key to Figure 4.1-13: Release and Receptor Locations

Release Source	Figure 4.1-13 Label	Applicable Accidents
Unit 1 Containment	A	LOCA, CREA
Unit 1 Plant Vent	B	LOCA (ESF leakage) LOCA (supplemental purge) FHA in FHB or RCB
Unit 1 East PORV/IVC	C	MSLB, SGTR, LRA
Unit 1 Control Room/TSC Intake	D	All
Unit 2 Containment	E	LOCA, CREA
Unit 2 Plant Vent	F	LOCA (ESF leakage) LOCA (supplemental purge) FHA in FHB or RCB
Unit 2 East PORV/IVC	G	MSLB, SGTR, LRA
Unit 2 Control Room/TSC Intake[11]	H	All

For all postulated accidents, steam releases from the secondary system (including the MSLB) are all assumed to occur in the Isolation Valve Cubicle (IVC), located between the containment building and the turbine building. This structure houses the main steam lines, the safety relief valves and the SG PORVs. The distance from the closest SG PORV to the control room HVAC emergency intake was used as the basis for the PORV-to-CRE χ/Q. Since this maximizes the χ/Q, a χ/Q for each PORV, steam line, or safety relief valve was not generated. The PORV-to-CRE χ/Q is used for all secondary system steam releases.

Releases from the Fuel Handling Building (for the Fuel handling Accident and the LOCA ESF leakage) are vented to the atmosphere via the Plant Vent. The RCB normal and supplemental purge is also via the same Plant Vent. Therefore, for the FHA releases and the LOCA supplemental purge release, the Plant-Vent-to-Control Room χ/Q is used. Releases from the RCB Personnel Airlock are also exhausted via this Plant Vent. The Plant-Vent-to- Control Room χ/Q also bounds a release from the RCB Equipment Hatch opening since the Plant Vent is much closer to the Control room air intake than the Equipment Hatch (which is located on the southwest quadrant of the RCB).

Each unit at STP has two associated receptors, the Control Room Emergency Makeup Air Intake and the Electrical Auxiliary Building Air Intake. The Control Room Emergency Makeup Air Intake is the air intake for both the Control Room and the Technical Support Center (TSC) HVAC systems.

[11] On each unit, the EAB HVAC Intake is immediately South of the Control Room/TSC intake.

4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

The control room and TSC are both wholly contained within the Electrical Auxiliary Building. Therefore, unfiltered in-leakage entering either the Control Room or the TSC would come from the Electrical Auxiliary Building atmosphere. Since the Electrical Auxiliary Building Air Intake is adjacent to the Control Room Emergency Makeup Air Intake, the χ/Q values calculated for the Control Room/TSC are also used for the Electrical Auxiliary Building and the unfiltered in-leakage entering either the Control Room or the TSC.

The postulation of a loss of offsite power does not change the location of release points or receptor locations. Steam releases from the secondary side are conservatively assumed to be released through the PORVs in the IVC. This is closer to the CR/TSC HVAC intake than any release points in the Turbine Generator Building (TGB).

Updated Control Room χ/Q values for releases from the containment, from the plant vent, and from the PORV area were calculated using the computer code ARCON96 (Reference 18) using the methods of Regulatory Guide 1.194. The STP meteorological databases for the five-year period (2000 – 2004) were used in the ARCON96 modeling analysis. Wind measurements were taken at 10 m and the vertical temperature difference was measured between 60 m and 10 m. The minimum wind speed (i.e., wind threshold) was set to the ARCON96 default value of 0.5 m/sec in accordance with Regulatory Guide 1.194, Table A-2.

ARCON96 requires the direction from the receptor to the source. Because plant north and true north are aligned, there is no need to correct directions. Using guidance from Section 3.2.4.5 of Regulatory Guide 1.194, the containment surface releases are taken to be on the surface of the containment at the horizontal location closest to the receptor. The release elevation for containment surface releases, using the Section 3.2.4.5 of Regulatory Guide 1.194, is the vertical center of the above-grade portion of the containment projected on a plane tangent to the containment surface and perpendicular to the line of sight from the containment center to the intake. Accordingly, the elevation of the containment leakage is determined to be 129.5 feet.

Using Sections 3.2.4.4 and 3.2.4.5 of Regulatory Guide 1.194, the initial sigmas for the containment surface source are:

$$\sigma_{Y0} = 26'\ 4" = 8.03\ \text{m},$$

and

$$\sigma_{Y0} = 33'\ 10" = 10.31\ \text{m}.$$

To determine the building area, the guidance in Regulatory Guide 1.194, Table A-2, was followed. The area to be used for each release point was chosen to be the vertical cross-section of the building that has the largest impact on the building wake for the release point. For all of the release points considered, the largest impact on the building wake is the containment building. The containment is treated as a right cylinder surmounted by half of a spheroid with horizontal radius equal to the cylinder radius and vertical radius equal to the height difference between the containment spring line and the top of the containment. The grade elevation is

4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

28'0", the containment spring line is 153'0", and the top of the containment is 231'0". The containment radius is 79'0". The resulting area of the containment is 29,429 ft^2.

The plant grid system is used to place release and receptor locations on a Cartesian coordinate grid. With the grid data, distances are computed in two dimensions (x,y) only. Distances between release locations and receptor points are presented in Table 4.1-26. All releases are treated as point sources, with the exception of containment leakage, which is treated as a diffuse source. The height of these release points are all less than 2.5 times the height of their adjacent buildings and therefore, in accordance with Regulatory Guide 1.194, are modeled as "ground level" releases. Buoyancy or mechanical jets of high energy releases are not credited in the χ/Q analyses.

A χ/Q value was determined from each release point in both units to each receptor in that unit. The maximum χ/Q value for a release point/receptor point pair in one unit was used in the analyses (for example, the χ/Q value from the PORV in Unit 2 to the Control Room in Unit 2, and the χ/Q value from the RCB in Unit 1 to the Control Room in Unit 1). The 0-2 hr χ/Q is used for the worst 2-hour doses.

Tables 4.1-26 through 4.1-29 present data that was used to develop the ARCON96 analyses. Table 4.1-30 through 4.1-35 present ARCON96 results. A summary of the resulting χ/Qs for each source/receptor pair is presented in Table 4.1-36. Table 4.1-37 presents a summary of the ARCON96 results used in the radiological analyses.

Table 4.1-26
Geometric Relationships Between Release Locations and Receptors

Release Location	Control Room for Unit	Distance (m)	Direction to Source (°)	Release Height (m)	Receptor Height (m)
U1 RCB Leakage	1	62.44	274.45	30.94	16.46
U1 Plant Vent	1	62.50	240.81	21.03	16.46
U1 East PORV	1	84.39	292.11	20.73	16.46
U2 RCB Leakage	2	62.14	274.46	30.94	16.46
U2 Plant Vent	2	62.50	240.81	21.03	16.46
U2 East PORV	2	84.11	292.19	20.73	16.46

4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

Table 4.1-27
Data Used to Generate ARCON96 Inputs

Parameter	Value
Containment diameter	158 feet
Containment height	203 feet
Area of containment	29,429 ft^2
Surface roughness length	0.2m
Minimum wind speed	0.5 m/s
Wind direction window	90°
Averaging sector width	4.3
Distances between Release points and Receptors	See Table 4.1-26

Number of hours in the averages and the minimum number of hours:

ARCON96 defaults:

Hours	Minimum
1	1
2	2
4	4
8	8
12	11
24	22
96	87
168	152
360	324
720	648

4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

Table 4.1-28
ARCON96 Input:
Unit 1 Releases to Unit 1 Control Room/TSC

Parameter	Release Source		
	RCB	Plant Vent	East PORV
Height of lower wind speed instrument (m)	10		
Height of upper wind speed instrument (m)	60		
Wind speed units	Miles per hour		
Release type	Ground level		
Release Height (m)	30.9	21.0	20.7
Building Area (m^2)	2734.0		
Effluent vertical velocity (m/s)	0.0		
Vent or Stack Flow (m^3/s)	0.0		
Vent or Stack radius (m)	0.0		
Direction: Intake to Source (deg)	274	241	292
Wind Direction Sector Width (deg)	90		
Wind Direction Window (deg)	229 – 319	196-286	247-337
Distance to Intake (m)	62.4	62.5	84.4
Intake Height (m)	16.5		
Terrain Elevation Difference (m)	0.0		
Minimum Wind Speed (m/s)	0.5		
Surface Roughness Length (m)	0.20		
Sector Averaging Constant	4.3		
Initial value of Sigma Y	8.03	0.0	0.0
Initial value of Sigma Z	10.31	0.0	0.0

4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

Table 4.1-29
ARCON96 Input:
Unit 2 Releases to Unit 2 Control Room/TSC

Parameter	Release Source		
	RCB	Plant Vent	East PORV
Height of lower wind speed instrument (m)	10		
Height of upper wind speed instrument (m)	60		
Wind speed units	Miles per hour		
Release type	Ground level		
Release Height (m)	30.9	21.0	20.7
Building Area (m^2)	2734.0		
Effluent vertical velocity (m/s)	0.0		
Vent or Stack Flow (m^3/s)	0.0		
Vent or Stack radius (m)	0.0		
Direction: Intake to Source (deg)	274	241	292
Wind Direction Sector Width (deg)	90		
Wind Direction Window (deg)	229-319	196-286	247-337
Distance to Intake (m)	62.1	62.5	84.1
Intake Height (m)	16.5		
Terrain Elevation Difference (m)	0.0		
Minimum Wind Speed (m/s)	0.5		
Surface Roughness Length (m)	0.20		
Sector Averaging Constant	4.3		
Initial value of Sigma Y	8.03	0.0	0.0
Initial value of Sigma Z	10.31	0.0	0.0

Table 4.1-30
ARCON96 Results
Unit 1 Containment to Unit 1 Control Room/TSC

Total number of hours of data processed	43848	Hours elevated plume w/dir. in window	0
Hours of missing data	885	Hours of calm wind	123
Hours direction in window	2391	Hours direction not in window or calm	40449

Averaging Period (hours)

	1	2	4	8	12	24	96	168	360	720
Upper Limit	1.00E-3	1.00E-3	1.00E-3	1.00E-3	1.00E-3	1.00E-3	1.00E-3	1.00E-3	1.00E-3	1.00E-3
Lower Limit	1.00E-7	1.00E-7	1.00E-7	1.00E-7	1.00E-7	1.00E-7	1.00E-7	1.00E-7	1.00E-7	1.00E-7
Above range	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
In Range	2514.	3295.	4573.	6680.	8556.	13048.	27768.	34351.	38630.	38820.
Below Range	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
Zero	40449.	39543.	38071.	35668.	33915.	29186.	13589.	6481.	1890.	52.
Total χ/Qs	42963.	42838.	42644.	42348.	42471.	42234.	41357.	40832.	40520.	38872.
% non-Zero	5.85	7.69	10.72	15.77	20.15	30.89	67.14	84.13	95.34	99.87
95%-ile χ/Q	2.16E-4	2.12E-4	1.76E-4	1.57E-4	1.26E-4	9.30E-5	5.41E-5	4.39E-5	3.46E-5	2.71E-5

95% χ/Q for standard averaging intervals

0 to 2 hours	2.16E-4
2 to 8 hours	1.37E-4
8 to 24 hours	6.11E-5
1 to 4 days	4.11E-5
4 to 30 days	2.30E-5

Hourly Value Range

	Max χ/Q	Min χ/Q
Centerline	*5.67E-4*	*7.60E-5*
Sector-Average	3.30E-4	4.43E-5

Table 4.1-31
ARCON96 Results
Unit 1 Plant Vent to Unit 1 Control Room/TSC

Total number of hours of data processed	43848	Hours elevated plume w/dir. in window	0
Hours of missing data	885	Hours of calm wind	123
Hours direction in window	2987	Hours direction not in window or calm	39853

Averaging Period (hours)

	1	2	4	8	12	24	96	168	360	720
Upper Limit	1.00E-2	1.00E-2	1.00E-2	1.00E-2	1.00E-2	1.00E-2	1.00E-2	1.00E-2	1.00E-2	1.00E-2
Lower Limit	1.00E-6	1.00E-6	1.00E-6	1.00E-6	1.00E-6	1.00E-6	1.00E-6	1.00E-6	1.00E-6	1.00E-6
Above range			0.	0.	0.	0.	0.	0.	0.	0.
In Range	3310.	3988.	5435.	7816.	9936.	14604.	28578.	34340.	38553.	38871.
Below Range			0.	0.	0.	0.	0.	0.	495.	1.
Zero	39853.	38850.	37209.	34532.	32535.	27630.	12779.	6492.	1472.	0.
Total χ/Qs	42963.	42838.	42644.	42348.	42471.	42234.	41357.	40832.	40520.	38872.
% non-Zero	7.24	9.31	12.75	18.46	23.39	34.58	69.10	84.10	96.37	100.00
95%-ile χ/Q	6.76E-4	7.12E-4	6.20E-4	5.74E-4	4.58E-4	3.27E-4	2.03E-4	1.78E-4	1.45E-4	1.12E-4

95% χ/Q for standard averaging intervals

0 to 2 hours	7.12E-4
2 to 8 hours	5.28E-4
8 to 24 hours	2.04E-4
1 to 4 days	1.61E-4
4 to 30 days	9.76E-5

Hourly Value Range

	Max χ/Q	Min χ/Q
Centerline	2.05E-3	1.62E-4
Sector-Average	1.19E-3	9.47E-5

Table 4.1-32
ARCON96 Results
Unit 1 East PORV to Unit 1 Control Room/TSC

Total number of hours of data processed	43848	Hours elevated plume w/dir. in window	0
Hours of missing data	885	Hours of calm wind	123
Hours direction in window	3481	Hours direction not in window or calm	39359

Averaging Period (hours)

	1	2	4	8	12	24	96	168	360	720
Upper Limit	1.00E-2	1.00E-2	1.00E-2	1.00E-2	1.00E-3	1.00E-3	1.00E-3	1.00E-3	1.00E-3	1.00E-3
Lower Limit	1.00E-6	1.00E-6	1.00E-6	1.00E-6	1.00E-7	1.00E-7	1.00E-7	1.00E-7	1.00E-7	1.00E-7
Above range			0.	0.	0.	0.	0.	0.	0.	0.
In Range	3604.	4543.	6063.	8422.	10472.	15255.	30100.	35798.	39341.	38821.
Below Range			0.	0.	0.	0.	0.	0.	0.	0.
Zero	39359.	38295.	36581.	33926.	31999.	26979.	11257.	5034.	1179.	51.
Total χ/Qs	42963.	42838.	42644.	42348.	42471.	42234.	41357.	40832.	40520.	38872.
% non-Zero	8.39	10.61	14.22	19.89	24.66	36.12	72.78	87.67	97.09	99.87
95%-ile χ/Q	6.08E-4	4.71E-4	4.40E-4	3.96E-4	3.22E-4	2.35E-4	1.34E-4	1.09E-4	9.07E-5	7.97E-5

95% χ/Q for standard averaging intervals

0 to 2 hours	6.08E-4
2 to 8 hours	3.26E-4
8 to 24 hours	1.54E-4
1 to 4 days	1.00E-4
4 to 30 days	7.13E-5

Hourly Value Range

	Max χ/Q	Min χ/Q
Centerline	1.15E-3	7.69E-5
Sector-Average	6.71E-4	4.48E-5

Table 4.1-33
ARCON96 Results
Unit 2 Containment to Unit 2 Control Room/TSC

Total number of hours of data processed	43848	Hours elevated plume w/dir. in window	0
Hours of missing data	885	Hours of calm wind	123
Hours direction in window	2391	Hours direction not in window or calm	40449

Averaging Period (hours)

	1	2	4	8	12	24	96	168	360	720
Upper Limit	1.00E-3	1.00E-3	1.00E-3	1.00E-3	1.00E-3	1.00E-3	1.00E-3	1.00E-3	1.00E-3	1.00E-3
Lower Limit	1.00E-7	1.00E-7	1.00E-7	1.00E-7	1.00E-7	1.00E-7	1.00E-7	1.00E-7	1.00E-7	1.00E-7
Above range			0.	0.	0.	0.	0.	0.	0.	0.
In Range	2514.	3295.	4573.	6680.	8556.	13048.	27768.	34351.	38630.	38820.
Below Range			0.	0.	0.	0.	0.	0.	0.	0.
Zero	40449.	39543.	38071.	35668.	33915.	29186.	13589.	6481.	1890.	52.
Total χ/Qs	42963.	42838.	42644.	42348.	42471.	42234.	41357.	40832.	40520.	38872.
% non-Zero	5.85	7.69	10.72	15.77	20.15	30.89	67.14	84.13	95.34	99.87
95%-ile χ/Q	2.17E-4	2.13E-4	1.77E-4	1.57E-4	1.27E-4	9.34E-5	5.44E-5	4.40E-5	3.49E-5	2.72E-5

95% χ/Q for standard averaging intervals

0 to 2 hours	2.17E-4
2 to 8 hours	1.37E-4
8 to 24 hours	6.15E-5
1 to 4 days	4.14E-5
4 to 30 days	2.30E-5

Hourly Value Range

	Max χ/Q	Min χ/Q
Centerline	5.70E-4	7.64E-5
Sector-Average	3.32E-4	4.45E-5

Table 4.1-34
ARCON96 Results
Unit 2 Plant Vent to Unit 2 Control Room/TSC

Total number of hours of data processed	43848	Hours elevated plume w/dir. in window	0
Hours of missing data	885	Hours of calm wind	123
Hours direction in window	2987	Hours direction not in window or calm	39853

Averaging Period (hours)

	1	2	4	8	12	24	96	168	360	720
Upper Limit	1.00E-2	1.00E-2	1.00E-2	1.00E-2	1.00E-2	1.00E-2	1.00E-2	1.00E-2	1.00E-2	1.00E-2
Lower Limit	1.00E-6	1.00E-6	1.00E-6	1.00E-6	1.00E-6	1.00E-6	1.00E-6	1.00E-6	1.00E-6	1.00E-6
Above range			0.	0.	0.	0.	0.	0.	0.	0.
In Range	3110.	3988.	5435.	7816.	9936.	14604.	28578.	34340.	38553.	38871.
Below Range			0.	0.	0.	0.	0.	0.	495.	1.
Zero	39853.	38850.	37209.	34532.	32535.	27630.	12779.	6492.	1472.	0.
Total χ/Qs	42963.	42838.	42644.	42348.	42471.	42234.	41357.	40832.	40520.	38872.
% non-Zero	7.24	9.31	12.75	18.46	23.39	34.58	69.10	84.10	96.37	100.00
95%-ile χ/Q	6.76E-4	7.12E-4	6.20E-4	5.74E-4	4.58E-4	3.27E-4	2.03E-4	1.78E-4	1.45E-4	1.12E-4

95% χ/Q for standard averaging intervals	
0 to 2 hours	7.12E-4
2 to 8 hours	5.28E-4
8 to 24 hours	2.04E-4
1 to 4 days	1.61E-4
4 to 30 days	9.76E-5

Hourly Value Range	Max χ/Q	Min χ/Q
Centerline	2.05E-3	1.62E-4
Sector-Average	1.19E-3	9.47E-5

Table 4.1-35
ARCON96 Results
Unit 2 East PORV to Unit 2 Control Room/TSC

Total number of hours of data processed	43848	Hours elevated plume w/dir. in window	0
Hours of missing data	885	Hours of calm wind	123
Hours direction in window	3481	Hours direction not in window or calm	39359

Averaging Period (hours)

	1	2	4	8	12	24	96	168	360	720
Upper Limit	1.00E-2	1.00E-2	1.00E-2	1.00E-2	1.00E-3	1.00E-3	1.00E-3	1.00E-3	1.00E-3	1.00E-3
Lower Limit	1.00E-6	1.00E-6	1.00E-6	1.00E-6	1.00E-7	1.00E-7	1.00E-7	1.00E-7	1.00E-7	1.00E-7
Above range			0.	0.	0.	0.	0.	0.	0.	0.
In Range	3604.	4543.	6063.	8422.	10472.	15255.	30100.	35798.	39341.	38821.
Below Range			0.	0.	0.	0.	0.	0.	0.	0.
Zero	39359.	38295.	36581.	33926.	31999.	26979.	11257.	5034.	1179.	51.
Total χ/Qs	42963.	42838.	42644.	42348.	42471.	42234.	41357.	40832.	40520.	38872.
% non-Zero	8.39	10.61	14.22	19.89	24.66	36.12	72.78	87.67	97.09	99.87
95%-ile χ/Q	6.13E-4	4.73E-4	4.43E-4	3.98E-4	3.23E-4	2.36E-4	1.34E-4	1.10E-4	9.09E-5	8.02E-5

95% χ/Q for standard averaging intervals	
0 to 2 hours	6.13E-4
2 to 8 hours	3.27E-4
8 to 24 hours	1.55E-4
1 to 4 days	1.01E-4
4 to 30 days	7.18E-5

Hourly Value Range	Max χ/Q	Min χ/Q
Centerline	1.16E-3	7.74E-5
Sector-Average	6.76E-4	4.51E-5

4.1 METEOROLOGY AND ATMOSPHERIC DISPERSION

The 95%-ile χ/Qs from ARCON96 are summarized in Table 4.1-36. The maximum χ/Q for a release location was chosen for use in the dose analyses.

Table 4.1-36
Control Room/TSC 95 Percentile χ/Qs
(sec/m^3)

	Time Interval				
Release Location	0-2 hours	2-8 hours	8-24 hours	1-4 days	4-30 days
U1 RCB Leakage	2.16E-04	1.37E-04	6.11E-05	4.11E-05	2.30E-05
U1 Plant Vent	7.12E-04	5.28E-04	2.04E-04	1.61E-04	9.76E-05
U1 East PORV	6.08E-04	3.26E-04	1.54E-04	1.00E-04	7.13E-05
U2 RCB Leakage	2.17E-04	1.37E-04	6.15E-05	4.14E-05	2.30E-05
U2 Plant Vent	7.12E-04	5.28E-04	2.04E-04	1.61E-04	9.76E-05
U2 East PORV	6.13E-04	3.27E-04	1.55E-04	1.01E-04	7.18E-05

Table 4.1-37
Summary of Control Room and TSC χ/Q Values
(sec/m^3)

	Containment		Plant Vent		PORV	
Time Interval	CLB	ARCON96	CLB	ARCON96	CLB[12]	ARCON96
0-2 hrs	1.06E-3	2.17E-4	1.29E-2	7.12E-4	N/A	6.13E-4
2-8 hrs	1.06E-3	1.37E-4	1.29E-2	5.28E-4	N/A	3.27E-4
8-24 hrs	7.03E-4	6.15E-5	8.55E-3	2.04E-4	N/A	1.55E-4
1-4 days	4.45E-4	4.14E-5	5.42E-3	1.61E-4	N/A	1.01E-4
4-30 days	1.91E-4	2.30E-5	2.32E-3	9.76E-5	N/A	7.18E-5

[12] The CLB does not have Control Room or TSC doses for the MSLB, SGTR, CREA, or LRA analyses.

4.2 ANALYTICAL MODELS

4.2 Analytical Models

The RADTRAD code (Version 3.0.3, Reference 19) was used to determine offsite doses and doses to Control Room and Technical Support Center personnel. However, the Fuel Handling Accidents used a simplified spreadsheet technique as discussed in Section 4.4.

No credit for personal protective equipment or prophylactic drugs is taken in the analyses.

The 0-2 hr χ/Q is used for the worst 2-hour doses for offsite, control room, and TSC dose analyses.

4.2.1 Offsite Dose Model

The analytical equation for determining the offsite doses is described in Section 2.3.1 of the RADTRAD documentation. The following is a summary of that discussion.

The dose to the hypothetical individual is calculated using the specified χ/Qs and the amount of each nuclide released during the exposure period. The air immersion dose from each nuclide, *n*, in an environmental compartment is calculated as:

$$D_{c,n}^{env} = A_n \left(\chi/Q \right) DCF_{c,n}$$

where $D_{c,n}^{env}$ = air immersion (cloudshine) dose due to nuclide *n* in the environment compartment (Sv)

$DCF_{c,n}$ = FGR 11 and 12 (References 20 and 21) air immersion (cloudshine) dose conversion factor for nuclide n as discussed in Section 1.4.3.3 of the RADTRAD documentation. (Sv m^3/Bq s)

χ/Q = atmospheric relative concentration (s/m^3)

A_n = released activity of nuclide *n* (Bq).

The inhalation dose from each nuclide, *n*, in an environmental compartment is calculated as:

$$D_{i,n}^{env} = A_n \left(\chi/Q \right) BR * DCF_{i,n}$$

where $D_{i,n}^{env}$ = inhalation dose commitment due to nuclide *n* in the environment compartment (Sv)

BR = breathing rate (m^3 / s)

$DCF_{i,n}$ = inhalation dose conversion factor for nuclide n as discussed in Section 1.4.3.3 of the RADTRAD documentation (Sv/Bq)

χ/Q = atmospheric relative concentration (s/m^3)

A_n = released activity of nuclide *n* (Bq).

4.2 ANALYTICAL MODELS

The breathing rates used in the offsite analyses are presented in Table 4.2-1.

Table 4.2-1
Offsite Breathing Rates
(m^3/sec)

Time	LPZ and EAB[13]
0 – 8 hours	3.5×10^{-4}
8 – 24 hours	1.8×10^{-4}
1 – 4 days	2.3×10^{-4}
4 – 30 days	2.3×10^{-4}

The TEDE is determined at the EAB for the limiting 2-hour period and at the outer boundary of the LPZ. No correction is made for depletion of the effluent plume by deposition on the ground.

4.2.2 Control Room Analytical Model

To determine the dose to personnel in the control room, the RADTRAD code and the built-in control room model are used. The analytical equation for determining the control room and TSC doses is described in Section 2.3.2 of the RADTRAD documentation. The following is a summary of that discussion.

The dose to a hypothetical individual in the control room is calculated based on the time-integrated concentration in the control room compartment. The air immersion dose in the control room is:

$$D_{c,n}^{CR} = C_n(t)\,dt\left(DCF_{c,n} / G_F\right)$$

Where $C_n(t)$ is the instantaneous concentration of nuclide n in the compartment. The Murphy-Campe (Reference 22) geometric factor, G_F, relates the dose from an infinite cloud to the dose from a cloud of volume V as:

$$G_F = \frac{1173}{V^{0.338}}$$

[13] Reference 3, page 16

4.2 ANALYTICAL MODELS

The inhalation dose in the control room is

$$D_{i,n}^{CR} = C_n(t)\,dt\left(\frac{BR * OF * DCF_{i,n}}{G_F}\right)$$

where OF = occupancy factor.

No credit is taken for the use of personal protective equipment or prophylactic drugs in the accident analyses when calculating dose consequences to the control room operator.

The control room envelope is located at elevation 35 ft and in two heating, ventilating, and air conditioning (HVAC) rooms at elevations 10 ft and 60 ft. in the Electrical Auxiliary Building as shown in Figure 4.2-1 (from Figure 6.4-1 of the Updated Final Safety Analysis Report (UFSAR)).

The Control Room HVAC system is designed to maintain the control room envelope at a minimum of 0.125-inch water gauge (wg) positive pressure relative to the surrounding area, following postulated accidents (other than hazardous chemical/smoke releases) and/or Loss-of-Offsite Power (LOOP), by introducing makeup air equivalent to the expected exfiltration air during plant emergency conditions (Engineered Safety Features [ESF] signal and/or high radiation in outside air). The design outside makeup air is 2,000 ft^3/min and drawn from a single intake on the east side of the Electrical Auxiliary Building at elevation 80 ft-0 in. Additionally, during postulated accident conditions, on detection of high radiation in the outside air or safety injection (SI) signal, outside makeup air for the control room envelope is automatically routed through makeup air units and cleanup units containing charcoal filters. The control room air is also automatically recirculated partially (i.e., 10,000 ft^3/min) through control room air cleanup units containing charcoal filters. This arrangement provides cleanup of the control room air.

The control room envelope HVAC system is not connected to other areas or HVAC systems where the potential for radioactivity exists, except for sharing common air intake and exhaust with the remaining Electrical Auxiliary Building.

The Control Room HVAC model schematic is given by Figure 4.2-2. The mathematical model used to represent the system uses a single outside air intake and a filtered make-up inflow which mixes with part of the recirculating air in the Control Room Envelope. The combined recirculating air and make-up air stream is then filtered before being supplied to the air-handling unit along with the remaining recirculating air. The air handling unit supplies the conditioned air to the control room envelope. A summary of these parameters is presented in Table 4.2-2. The assumed unfiltered in-leakage into the control room envelope has been revised to 100 cfm for the AST analyses as the result of control room in-leakage testing as described below.

Unless otherwise noted, the analyses assume there is an emergency diesel failure and that only two trains of HVAC are in operation. The make-up flow rate for two trains of emergency HVAC

4.2 ANALYTICAL MODELS

operation is 2000 cfm. The make-up flow is assumed to operate at +10% of design flow (2200 cfm). The flow rate for two trains of Control Room HVAC recirculation flow is 8600 cfm (4300[14] cfm per train). The Control Room HVAC exhaust flow rate (Label G) is 2300 cfm. The Control Room HVAC exhaust flow rate is the sum of the make-up flow (Label A) and the unfiltered in-leakage (Label F). Also, each of the three trains of Control Room HVAC system contains 2 sets of 2-inch charcoal filters. The first 2-inch filter is the make-up filter. Filtered make-up air is then combined with recirculated air and then passes through the 2-inch recirculation filter before entering the Control Room.

In the CLB, if one train of control room HVAC is not functioning, for example due to diesel generator failure, not all of the makeup air would be filtered twice before it is introduced into the control room envelope. In the worst case, 235 cfm of the makeup air is filtered by the makeup units, but not by the recirculation units, before it is introduced into the control room envelope. In the revised AST analyses, this assumption is not needed since credit is not taken for filtration of the make-up air.

In contrast to the CLB, the revised AST analyses assume that all makeup flow is unfiltered (e.g., removing the 4 inches of filtration per train, 2 inches for the makeup filters and 2 inches of the cleanup filters, for make-up air in the CLB). Only the recirculation filtration is credited. Hence, the assumed make-up air flow (Label A on Figure 4.2-2) on Table 4.2-2 is assumed to be 0 cfm. The 2200 cfm make-up flow is added to the 100 cfm unfiltered in-leakage value (which includes the contribution from door pumping action from Control Room ingress and egress) and a total of 2300 cfm is assumed to directly enter the Control Room without filtration (Label F on Figure 4.2-2). No credit is taken for the use of non-ESF ventilation systems during the Design Basis Accident. In summary, Table 4.2-2 reflects the air flow with 2 trains operating while Table 4.2-3 reflects the flows used in the analyses.

The Control Room recirculation clean-up filter efficiencies are assumed to have 95% removal efficiency for elemental iodine and organic iodine and 99% removal efficiency for particulates.

The assumption of 100 cfm unfiltered in-leakage is validated by in-leakage testing conducted in Unit 1 in March 2004 and in Unit 2 in March 2007. The testing was conducted using the tracer gas method described in ASTM E741-00 (Reference 23). The test results for Unit 1 were reported in Reference 24 in response to NRC Generic Letter 2003-01, "Control Room Habitability." The limiting train combination test results were 9.4 +/- 50 scfm in Unit 1 and 64 +/- 8 scfm in Unit 2. Therefore, an unfiltered in-leakage assumption of 100 cfm is conservative.

The calculated control room volume is 304000 ft^3. Approximately 10% of this volume is occupied by walls and equipment. The volume used in dose analyses is 274080 ft^3.

[14] Per plant procedures, the acceptance criteria for the surveillance testing of the make-up flow and make-up+clean-up flow is 1000 cfm +/- 10% and 6000 cfm +/- 10%, respectively. Therefore, it is acceptable to have a recirculation flow rate of 4300 cfm ([6000 cfm x 0.9] – [1000 cfm x 1.1]) = 5400 cfm – 1100 cfm = 4300 cfm. The +/- 10% band on the flow rages is based on the acceptance criteria of TS Surveillance 4.7.7.c.3.

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Note that the Fuel Handling Accident analysis does not credit either the make-up or recirculation filters. The Control Room internal air is assumed to be in equilibrium with the air outside the Control Room HVAC intake. Therefore, the Control Room is not assumed to be pressurized during the accident, nor are any assumptions made as to the functioning of the Control Room HVAC systems.

4.2 ANALYTICAL MODELS

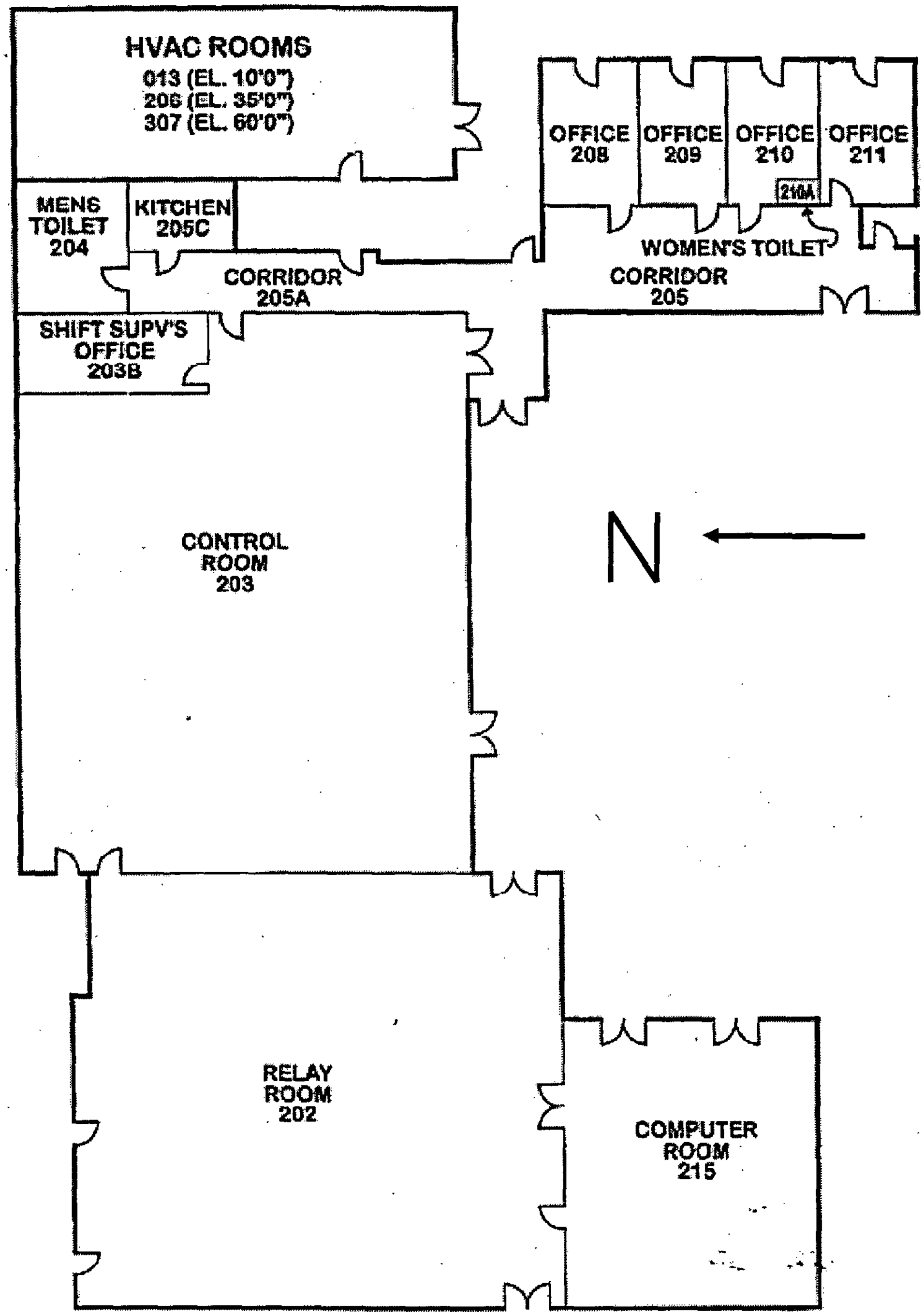

Figure 4.2-1 Control Room Envelope

4.2 ANALYTICAL MODELS

Figure 4.2-2: Control Room HVAC Analytical Model

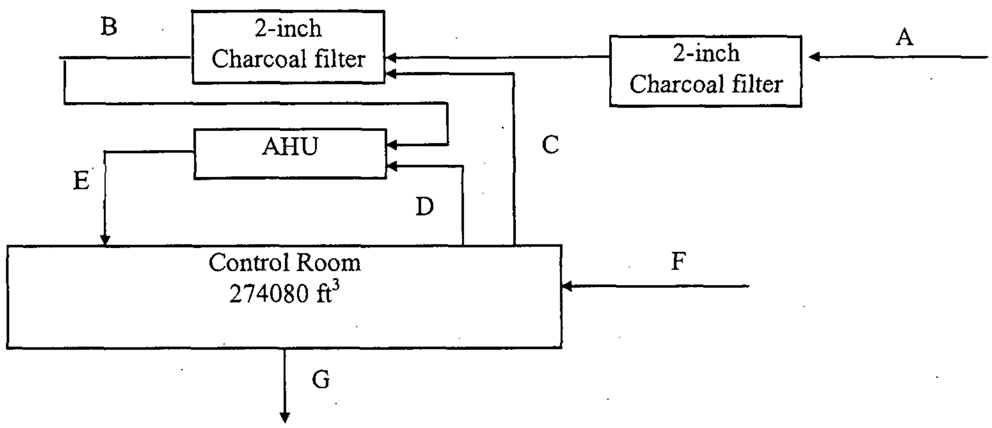

Table 4.2-2
Control Room HVAC Flow Rates
(2 trains)

Flow Path	Label	CLB Flow Rate (cfm)	AST Flow Rate (cfm)
Make-up	A	2200	2200[15]
Clean-up (or recirc)	B	11,700	10,800
Clean-up (or recirc)	C	9500	8600
A/C Intake	D	21,660	24,760
A/C + Clean-up Exhaust	E	33,360	35,560
Unfiltered In-leakage	F	10	100[16]
Exhaust Flow	G	2210	2300

[15] Set to 0 cfm in the analytical model. See Table 4.2-3.
[16] Set to 2200+100=2300 cfm in the analytical model. See Table 4.2-3.

4.2 ANALYTICAL MODELS

Table 4.2-3
Parameters Used in Modeling the Control Room[17]

Parameter	CLB	AST
Pressurization (makeup) flow (Label A)	2200 cfm	0 cfm[18]
Pressurization (makeup) 2" filter efficiencies[19]:		
inorganic (elemental)	95%	0%
organic	95%	0%
particulate	99%	0%
Clean-up (recirculation) flow (Label C)	9500 cfm	8600 cfm
Clean-up (recirculation) filter efficiencies (2, 2" filters):		
inorganic (elemental)	95%	95%
organic	95%	95%
particulate	99%	99%
Free Volume	274,080 ft^3	
Unfiltered In-leakage (Label F)	10 cfm (from door pumping action)	2300 cfm total (including door pumping action)
Portion of above make-up flow that bypasses Control Room recirculation clean-up filters	235 cfm[20] (two trains)	0 cfm
All χ/Q's	Table 4.1-37	
Control Room Occupancy Factors		
0-24 hrs	100%	
1-4 days	60%	
4-30 days	40%	
Breathing Rate	3.5E-4 m^3/sec	

4.2.2.1 CRE Unfiltered In-leakage and Possible "Sneak" Paths

The unfiltered in-leakage into the CRE is assumed to be 100 cfm for all accidents. The Control Room and the TSC are enclosed in the Electrical Auxiliary Building and the surrounding spaces are supplied by the Electrical Auxiliary Building HVAC system. The intake of the Electrical Auxiliary Building HVAC system is located just south of the Control Room/TSC HVAC intakes on the east wall of the Electrical Auxiliary Building (points D/H on Figure 4.1-13). Since the two intakes are very close, the Control Room/TSC χ/Q's are used for the air entering the Electrical Auxiliary Building HVAC, and, therefore, for the unfiltered in-leakage.

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17 This table is based upon the current UFSAR Table 6.4-2, *Control Room Dose Analysis*

18 For AST, all makeup flow is assumed to be unfiltered, bypassing the 2" recirculation filters.

19 For the CLB, 1765 cfm is filtered through makeup and recirculation filters; 235 cfm is filtered through makeup filters only. The effective filter efficiencies for 2000 cfm were used.

20 Only receives filtration from the 2" makeup filters

4.2 ANALYTICAL MODELS

Since the spaces surrounding the Control Room are in the Electrical Auxiliary Building, the chances for a more direct, unanalyzed, path (i.e., a "sneak" path) for airborne contaminants to enter the Control Room are minimized. The largest potential source of a "sneak" path is the Electrical Penetration area which is directly between the Control Room Envelope and the containment building (on the bottom of the Control Room Envelope, west of the Relay Room and Computer Room, as depicted in Figure 4.2-1) for the LOCA and Control Rod Ejection accidents. However, the possibility of leakage from the containment into the penetration area and finally into the Relay Room and Control Room Envelope is minimized by the presence of double doors between the Relay Room and the penetration area (partially shown on Figure 4.2-1). In addition, there is no equipment located in the penetration area that must be manipulated or observed in a post-accident scenario. Therefore, traffic through the doors would be minimal, if any. In consideration of the above, leakage from the penetration area into the Control Room Envelope is not considered credible.

4.2.3 Technical Support Center (TSC) Analytical Model

To determine the dose to personnel in the TSC, RADTRAD is used and the control room node in the code is used as the TSC. The analytical model of the TSC is identical to the one discussed for the control room in Section 4.2.2, above. No credit is taken for the use of personal protective equipment or prophylactic drugs in the accident analyses when calculating dose consequences to TSC personnel. A description of the TSC HVAC model is given below.

It is assumed that walls and equipment occupy 25% of the TSC volume measured from exterior dimensions. The TSC volume used in radiological dose analysis is 48170 ft^3. The TSC HVAC make-up flow passes through two 2-inch carbon filters in series. However, for conservatism, this analysis assumes that all makeup flow is unfiltered. A portion of the recirculation flow from the TSC passes through the carbon filters. The remainder of the recirculation flow combines with the make-up flow prior to entering the air-handling unit. The TSC HVAC model schematic is given by Figure 4.2-3.

4.2 ANALYTICAL MODELS

Figure 4.2-3 TSC HVAC Analytical Model

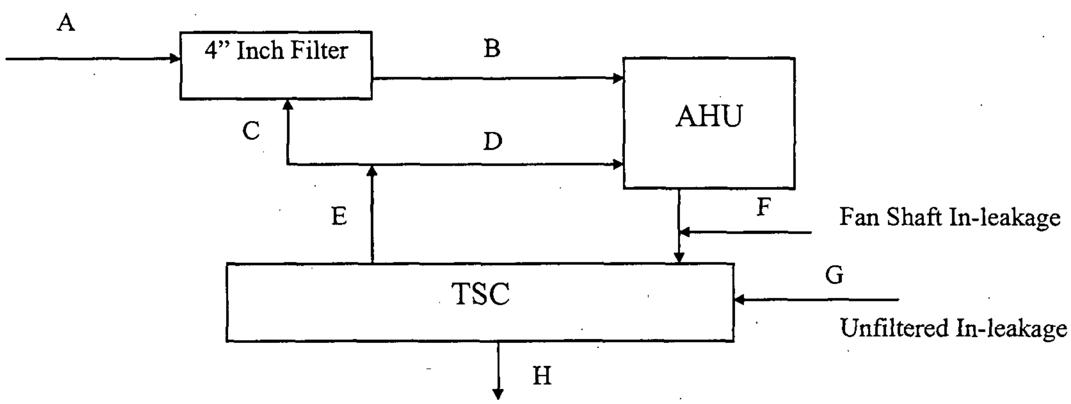

Table 4.2-4
TSC HVAC Flow Rates

Flow Path	Label	Design Flow (cfm)	Assumed Flow Rate (cfm) CLB	Assumed Flow Rate (cfm) AST
Make-up	A	1200	1210	1210[21]
Clean-up (or recirc) + Make-up	B	6200	5960	5960
Clean-up (or recirc)	C	5000	4750	4750
A/C Intake	D	-	5225	5225
A/C + Clean-up	E	-	9975	9975
Fan Shaft In-leakage	F	0	5	5
Unfiltered In-leakage	G	-	10	10[22]
Exhaust Flow	H	1200	1225	1225

[21] Set to 0 cfm in the analysis. See Table 4.2-5.
[22] Set to 1210+10+5=1225 cfm in the analysis. See Table 4.2-5.

4.2 ANALYTICAL MODELS

TSC filter efficiencies are based on two 2-inch filters in series.

Table 4.2-5
Parameters Used in Modeling the TSC

Parameter	CLB	AST
Pressurization (makeup) flow (cfm) (Label A)	1210	0[23]
Clean-up (recirculation) flow (cfm) (Label C)	4750	
Filter efficiencies:		
inorganic (elemental)	99%	
organic	99%	
particulate	99%	
Free Volume	48,167 ft^3	
Unfiltered In-leakage (cfm) (Labels F & G)	15	1225
All χ/Q's	Table 4.1-37	
Control Room/TSC Occupancy Factors		
0-24 hrs	100%	
1-4 days	60%	
4-30 days	40%	
Breathing Rate	3.5E-4 m^3/sec	

The TSC HVAC make-up flow rate is 1100 cfm. The TSC HVAC make-up flow rate (Label A) operates at +10% off design (1210 cfm). The recirculation flow rate is 5000 cfm. The recirculation flow rate (Label C of Figure 4.2-3) operates at –5% off design (4750). The TSC HVAC exhaust flow rate (Label H) is 1225 cfm. The fan shaft in-leakage (Label F) is 5 cfm. The unfiltered in-leakage (Label G) is 10 cfm. The TSC HVAC exhaust flow rate is the sum of the make-up flow (Label A), the fan shaft in-leakage (Label F) and the unfiltered in-leakage (Label G).

The TSC HVAC system is non-safety; therefore, no single failures are assumed.

[23] All AST makeup flow is assumed to be unfiltered, bypassing the 4" of filtration used for the makeup and recirculation pathways.

4.2 ANALYTICAL MODELS

4.2.4 Radiological Source Terms

4.2.4.1 Dose Conversion Factors and Physical Parameters

The dose conversion factors (DCF) used in the LOCA and FHA are the default RADTRAD values (Reference 19, Tables 1.4.3.3-1 and -2), with slight modifications for parents with short-lived daughters (I-135,Cs-137, Te-129m, Te-131m, Ru-103, Ru-106, Zr-97, Ce-144).

The DCFs used in the MSLB, SGTR, CREA, and LRA analyses are presented in Table 4.2-6. The CLB DCFs are based on ICRP-30 (Reference 25). The AST DCFs for external exposure (EDE) and inhalation (CEDE) are from the Federal Guidance Report No. 12, "External Exposure to Radionuclides in Air, Water, and Soil" (Reference 21) and the Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," respectively.

Table 4.2-6
Dose Conversion Factors

	CLB			AST	
Isotope	Thyroid (rem/Ci)	Beta-Skin (rem-m^3/Ci-sec)	Whole Body (rem-m^3/Ci-sec)	EDE (Sv-m^3/Bq-sec)	CEDE (Sv/Bq)
I-131	1.080E+06	4.087E-02	6.734E-02	1.82E-14	8.89E-09 (3.29E+4 rem/Ci)
I-132	6.438E+03	1.617E-01	4.144E-01	1.12E-13	1.03E-10 (3.81E+2 rem/Ci)
I-133	1.798E+05	1.032E-01	1.088E-01	2.94E-14	1.58E-09 (5.85E+3 rem/Ci)
I-134	1.066E+03	2.011E-01	4.810E-01	1.30E-13	3.55E-11 (1.31E+2 rem/Ci)
I-135	3.130E+04	1.153E-01	2.953E-01	7.98E-14	3.32E-10 (1.32E+3 rem/Ci)
Kr-83m	N/A	1.547E-05	5.550E-06	1.50E-18	0
Kr-85m	N/A	5.468E-02	2.768E-02	7.48E-15	0
Kr-85	N/A	4.843E-02	4.403E-04	1.19E-16	0
Kr-87	N/A	3.482E-01	1.524E-01	4.12E-14	0
Kr-88	N/A	1.221E-01	3.774E-01	1.02E-13	0
Kr-89	N/A	3.981E-01	3.232E-01	N/A	N/A
Sr-89	N/A	N/A	N/A	7.73E-17	1.12E-08
Xe-131m	N/A	1.544E-02	1.439E-03	3.89E-16	0
Xe-133m	N/A	3.227E-02	5.069E-03	1.37E-15	0
Xe-133	N/A	1.145E-02	5.772E-03	1.56E-15	0
Xe-135m	N/A	3.144E-02	7.548E-02	2.04E-14	0
Xe-135	N/A	7.066E-02	4.403E-02	1.19E-14	0

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Table 4.2-6
Dose Conversion Factors

	CLB			AST	
Isotope	Thyroid (rem/Ci)	Beta-Skin (rem-m^3/Ci-sec)	Whole Body (rem-m^3/Ci-sec)	EDE (Sv-m^3/Bq-sec)	CEDE (Sv/Bq)
Xe-137	N/A	4.642E-01	3.026E-02	N/A	N/A
Xe-138	N/A	1.728E-01	2.135E-01	5.77E-14	0
Rb-86	N/A	N/A	N/A	4.81E-15	1.79E-09
Rb-87	N/A	N/A	N/A	1.82E-18	8.74E-10
Rb-88	N/A	N/A	N/A	3.36E-14	2.26E-11
Rb-89	N/A	N/A	N/A	1.06E-13	1.16E-11
Cs-134	N/A	N/A	N/A	7.57E-14	1.25E-08
Cs-135	N/A	N/A	N/A	5.65E-19	1.23E-09
Cs-136	N/A	N/A	N/A	1.06E-13	1.98E-09
Cs-137	N/A	N/A	N/A	7.74E-18	8.63E-09
Cs-138	N/A	N/A	N/A	1.21E-13	2.74E-11
Ba-137m	N/A	N/A	N/A	2.88E-14	0

The LOCA and FHA analyses use the default RADTRAD isotopic data and progeny data (Reference 19, Table 1.4.3.2-2). Table 4.2-7 presents physical data for the isotopes of interest for the MSLB, SGTR, CREA, and LRA analyses. The half life data is from Reference 26. The progeny and decay fractions are from RADTRAD (Reference 19, Table 1.4.3.2-2). Some meta-stable isotopes of xenon not contained in RADTRAD are assumed to always decay to the ground state of the same isotope.

Table 4.2-7
Isotopic Half Lifes, Parent-to-Daughter Decay Isotopes and Fractions

Isotope	$T_{1/2}$ (sec)	Daughter 1	Fraction 1	Daughter 2	Fraction 2
I-131	6.947E+05	Xe-131m	0.1100E-01	-	-
I-132	8.208E+03	-	-	-	-
I-133	7.488E+04	Xe-133m	0.2900E-01	Xe-133	0.9700E+00
I-134	3.156E+03	-	-	-	-
I-135	2.365E+04	Xe-135m	0.1500E+00	Xe-135	0.8500E+00
Kr-83m	6.696E+03	-	-	-	-
Kr-85m	1.613E+04	Kr-85	0.2100E+00	-	-
Kr-85	3.386E+08	-	-	-	-
Kr-87	4.572E+03	Rb-87	0.1000E+01	-	-
Kr-88	1.022E+04	Rb-88	0.1000E+01	-	-
Kr-89	1.890E+02	Rb-89	0.1000E+01	-	-
Sr-89	4.365E+06	-	-	-	-

4.2 ANALYTICAL MODELS

Table 4.2-7
Isotopic Half Lifes, Parent-to-Daughter Decay Isotopes and Fractions

Isotope	$T_{1/2}$ (sec)	Daughter 1	Fraction 1	Daughter 2	Fraction 2
Xe-131m	1.028E+06	-	-	-	-
Xe-133m	1.892E+05	Xe-133	0.1000E+01	-	-
Xe-133	4.530E+05	-	-	-	-
Xe-135m	9.180E+02	Xe-135	0.9940E+00	Cs-135	0.6000E-03
Xe-135	3.276E+04	Cs-135	0.1000E+01	-	-
Xe-137	2.292E+02	-	-	-	-
Xe-138	8.460E+02	-	-	-	-
Rb-86	1.610E+06	-	-	-	-
Rb-87	1.515E+18	-	-	-	-
Rb-88	1.062E+03	-	-	-	-
Rb-89	9.240E+02	Sr-90	0.1000E+01	-	-
Cs-134	6.510E+07	-	-	-	-
Cs-135	7.250E+13	-	-	-	-
Cs-136	1.140E+06	-	-	-	-
Cs-137	9.510E+08	Ba-137m	0.9500E+00	-	-
Cs-138	1.932E+03	-	-	-	-
Ba-137m	1.531E+02	-	-	-	-

4.2.4.2 Reactor Core Source Terms

The basic source terms used in the Current Licensing Basis for the reactor core and the reactor coolant system were taken from Revision 4 of the Westinghouse *Radiation Analysis Design Manual* (Reference 27). This document was based on the 1973 ORIGEN (Reference 28) computer code. Revision 5 of the *Radiation Analysis Design Manual* (Reference 29), based upon ORIGEN 2.1 (Reference 30), has been used for all AST analyses.

Table 4.2-8 provides a comparison of the major parameters used to determine the source terms in the *Radiation Analysis Design Manual.* The major difference in the two revisions of the analysis is the different versions of ORIGEN used. Also, the difference in the assumed reactor coolant system (RCS) cleanup flow rate (letdown rate) lowers the calculated isotopic inventory in the RCS (see Table 4.2-14). Since there is no purging of the volume control tank (VCT), the gases reach equilibrium in the VCT and RCS. Since the analyses are performed at 1% failed fuel, both sets of data bound actual plant operation. The iodine concentrations resulting from the 1% failed fuel assumption are much greater than those at the Technical Specification maximum of 1 μCi/gm.

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Table 4.2-8
Comparison of Revisions to the *Radiation Analysis Design Manual*

Parameter	Rev 4/CLB	Rev 5/AST
ORIGEN version	0	2.1
Reactor Power	4100 MWt	4100 MWt
Core Burnup (EOL, 3 region equilibrium core)	20K/40K/60K MWD/MTU Or 509, 1018, and 1527 EFPD	20K/40K/60K MWD/MTU Or 509, 1018, and 1527 EFPD
Reactor Coolant Volume	13,521 ft^3	13,521 ft^3
% Failed Fuel	1%	1%
RCS Letdown Rate	100 gpm @130°F and 2250 psia	140 gpm @130°F and 2250 psia
Volume Control Tank purge rate	0 cfm	0 cfm

The AST values used in this analysis were derived using guidance outlined in Regulatory Guide 1.183. The ORIGEN 2.1 code was used to calculate plant-specific fission product inventories for use in the dose analyses. The assumed period of irradiation was sufficient (three-region equilibrium cycle core at end of life with the three regions having operated at 39.31 MW/MTU for 509, 1018, and 1527 EFPD, respectively) to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. The reactor core inventory is presented in Table 4.2-9.

Table 4.2-9
Comparison of CLB and AST Reactor Core Sources
(Ci)

Isotope	CLB[24]	AST[25]	% Difference
Kr83m	1.40E+07	1.40E+07	0.0%
Kr85m	3.00E+07	2.90E+07	-3.3%
Kr85	1.20E+06	1.20E+06	0.0%
Kr87	5.50E+07	5.50E+07	0.0%
Kr88	7.90E+07	7.80E+07	-1.3%
Kr89	9.70E+07	9.50E+07	-2.1%
Xe131m	7.70E+05	1.10E+06	42.9%
Xe133m	3.30E+07	6.80E+06	-79.4%
Xe133	2.30E+08	2.20E+08	-4.3%

[24] Reference 27
[25] Reference 29

4.2 ANALYTICAL MODELS

Table 4.2-9
Comparison of CLB and AST Reactor Core Sources
(Ci)

Isotope	CLB[24]	AST[25]	% Difference
Xe135m	4.60E+07	4.20E+07	-8.7%
Xe135	6.50E+07	5.50E+07	-15.4%
Xe137	2.00E+08	1.90E+08	-5.0%
Xe138	1.90E+08	1.80E+08	-5.3%
I131	1.14E+08	1.06E+08	-7.0%
I132	1.64E+08	1.52E+08	-7.3%
I133	2.40E+08	2.20E+08	-8.3%
I134	2.60E+08	2.40E+08	-7.7%
I135	2.20E+08	2.00E+08	-9.1%
Cs134	3.30E+07	2.20E+07	-33.3%
Cs136	9.30E+06	6.30E+06	-32.3%
Cs137	1.40E+07	1.30E+07	-7.1%
Sb129	3.70E+07	3.40E+07	-8.1%
Te129m	9.50E+06	5.00E+06	-47.4%
Te129	3.50E+07	3.30E+07	-5.7%
Te131m	1.70E+07	1.50E+07	-11.8%
Ba137m	1.30E+07	1.20E+07	-7.7%
Ba140	2.00E+08	1.90E+08	-5.0%
Ru103	1.80E+08	1.60E+08	-11.1%
Ru105	1.20E+08	1.10E+08	-8.3%
Ru106	5.80E+07	5.50E+07	-5.2%
Y91	1.40E+08	1.40E+08	0.0%
Y92	1.50E+08	1.40E+08	-6.7%
Y93	1.70E+08	1.60E+08	-5.9%
Zr95	1.90E+08	1.80E+08	-5.3%
Zr97	1.90E+08	1.80E+08	-5.3%
Nb95	2.00E+08	1.30E+08	-35.0%
La140	2.10E+08	1.90E+08	-9.5%
La142	1.80E+08	1.70E+08	-5.6%
Pr143	1.70E+08	1.60E+08	-5.9%
Nd147	7.40E+07	7.10E+07	-4.1%
Ce141	1.90E+08	1.80E+08	-5.3%
Ce143	1.80E+08	1.70E+08	-5.6%
Ce144	1.40E+08	1.40E+08	0.0%

4.2 ANALYTICAL MODELS

Table 4.2-9
Comparison of CLB and AST Reactor Core Sources
(Ci)

Isotope	CLB[24]	AST[25]	% Difference
Sr89	1.10E+08	1.10E+08	0.0%
Sr90	1.00E+07	9.70E+06	-3.0%
Sr91	1.40E+08	1.30E+08	-7.1%
Sr92	1.50E+08	1.40E+08	-6.7%

The non-LOCA design bases analyses used the Departure from Nucleate Boiling Ratio (DNBR) as a fuel damage criterion.

4.2.4.2.1 Peak Pin Evaluation for non-LOCA Fuel Gap Inventory

Footnote 11 for Regulatory Guide 1.183, Table 3, *Non-LOCA Fraction of Fission Product Inventory in Gap* states that the release fractions for Table 3 are "acceptable for use with currently approved LWR fuel with a peak burnup of 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kW/ft peak rod average power for burnups exceeding 54 GWD/MTU" (the "54/6.3" criteria).

Westinghouse's design code, ANC (Reference 31), was used to calculate the best estimate pin power and pin burnup for a fuel cycle. For the purpose of this evaluation, the code is used to calculate and edit the limiting relative power and the limiting pin burnup of the assembly for Unit 1 Cycles 13 and 14 Unit 2 Cycle 12. The 3 cycles evaluated are typical 18-month cycles that were designed for about 500 EFPD hot full power energy plus an additional 30 EFPD of coastdown operation.

The evaluation selected the limiting relative pin power and the limiting pin burnup of the assembly. This assessment approach is conservatively bounding as it assumes that the maximum power rod is also the maximum burnup rod of the assembly. At hot full power condition (3853 Mwth), the average linear power density of a fuel pin is 5.4 Kw/ft. Therefore, the 6.3 Kw/ft pin power limit corresponds to a "relative" pin power value of 1.167 (normalized to an average of 1.0).

The Unit 2 Cycle 12 maximum relative pin power for burnup exceeding 54 GWD/MTU is 1.055, well below the 1.167 limit. The maximum pin burnup for the cycle, including 30 EFPD of coastdown, is 58,433 MWD/MTU. The Unit 1 Cycle 13 limiting pin burnup remains below 54 GWD/MTU at the end of hot full power (cycle burnup about 19350 MWD/MTU). The maximum pin burnup slightly exceeds 54 GWD/MTU at extended coastdown (20700 MWD/MTU). However, since the limiting burnup assemblies are located on the core periphery, the relative pin power is only 0.901, well below the 1.167 limit.

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The Unit 1 Cycle 14 limiting pin burnup exceeds 54 GWD/MTU at the middle of the cycle (about 9,000 MWD/MTU). Assemblies having high pin burnup are located on the core periphery. The relative pin powers for these assemblies are less than 0.7. As the core depletes, eight of the in-board assemblies have maximum pin burnup exceeding 54 GWD/MTU near the end of the cycle. The maximum relative pin power for these assemblies reaches 1.042 at the end of hot full power (about 18380 MWD/MTU). The maximum pin burnup remains below 58 GWD/MTU. Continued operation with power coastdown does not show an increase in the relative pin power. The highest pin burnup for the cycle, including a power coastdown to 19500 MWD/MTU, is 60,588 MWD/MTU. The relative pin power for the assembly is 0.703. All parameters are well below the limits specified in the Regulatory Guide 1.183.

The above evaluation of STP's typical cycle designs shows that the "54/6.3" criteria is met with significant margin. The highest relative pin power for pin burnup greater than 54 GWD/MTU is 1.055, which corresponds to a linear heat rate of about 5.7 Kw/ft. STP uses a low-low core leakage design, placing high burnup fuel on the core periphery, to improve the fuel economy. As a result, the high burnup fuel assemblies typically have a low relative power. In some instances, a limited number of twice-burned fuel assemblies may be placed in inboard locations to optimize the core power peaking behavior. In this case, the assembly will be driven to have a higher power. Unit 2 Cycle 12, for instance, has a peak pin power of 1.055 for burnups greater than 54 GWD/MTU. Inboard placements of these assemblies are usually planned one cycle in advance and are evaluated during the cycle design to make sure the power peaking and pin burnup are not outside the norm. STP plans to continue to use this design approach for future core designs, therefore, it is expected that the "54/6.3" criteria will continue to be met with adequate margin.

Currently, the licensed limit for the maximum burnup of a fuel pin is 62,000 MWD/MTU. However, the 6.3 Kw/ft pin power limit for burnup greater than 54 GWD/MTU is not currently a requirement for reload cycle design verification. To ensure this criterion is met in future cycles, the procedure used to check the adequacy of a core design has been revised to include an evaluation on the pin power/burnup of the design core.

In summary, an evaluation was performed to determine the best estimate fuel rod average burnup and power for STP's typical 18-month cycle designs. The evaluation shows that the Regulatory Guide "54/6.3" criteria for the application of Alternative Source Term are met with significant margin. The highest relative pin power for pin burnup greater than 54 GWD/MTU is 1.055, which corresponds to a linear heat rate of 5.7 Kw/ft. STP plans to continue to use a low-low leakage core design approach and it is expected that the "54/6.3" criteria will continue to be met.

4.2 ANALYTICAL MODELS

4.2.4.3 Dose Equivalent I-131 and Coolant Activity

Coolant activity limits are specified in terms of dose equivalent (DE) I-131. This is the I-131 concentration that would provide the same dose response as the combined concentration of all iodine isotopes in the coolant. In the CLB, thyroid dose response is used as the measure of equivalency; and the actual isotopic concentrations representing an equivalent concentration of DE I-131 for the CLB are as follows (60 μCi/gm DE I-131 is presented as the example; other concentrations would scale proportionately):

Table 4.2-10
Isotopic Concentrations for 60 μCi/gm
Representing An Equivalent Concentration of DE I-131
in the Current Licensing Basis

	μCi/gm	Thyroid DCF (thyroid rem/Ci)	Product
I-131	46	1.08E+06	5.0E+07
I-132	52	6.44E+03	3.4E+05
I-133	72	1.80E+05	1.3E+07
I-134	10	1.07E+03	1.1E+04
I-135	40	3.13E+04	1.3E+06
		Sum divided by I-131 DCF	6.0E+01

In Table 4.2-10, the activity (second column) is multiplied by the dose conversion factor (DCF) (third column) to obtain a product (fourth column). These are summed and the sum is divided by the I-131 DCF to obtain the DE I-131. The combined concentrations are equivalent in terms of dose response to 60 μCi/gm of I-131 as intended.

It is readily evident that any number of concentration combinations of these five iodine isotopes can be equivalent to 60 μCi/gm of I-131. However, once the ratio of each of the four isotopes I-132 through I-135 is established relative to I-131, then only a single set of concentrations will correspond to 60 μCi/gm of I-131 (or any other given concentration) in terms of dose equivalency.

Independent of adoption of the AST, this submittal also adopts Reference 29 as the basis for iodine dose equivalency. Therefore, there are two parts to the change in iodine dose equivalency proposed in this amendment request: (1) the use of CEDE DCFs for consistency with the dose basis for AST and (2) the adoption of the relative iodine isotopic concentrations from Reference 29.

The I-131 dose equivalency from Reference 29 based on thyroid dose (ICRP-30 DCFs) is shown in Table 4.2-11A.

4.2 ANALYTICAL MODELS

Table 4.2-11A
Isotopic Concentrations
Representing An Equivalent Concentration of DE I-131
Using Updated Iodine RCS Concentrations and Thyroid DCFs

	μCi/gm	Thyroid DCF (thyroid rem/Ci)	Product
I-131	42.5	1.08E+06	4.59E+07
I-132	60	6.44E+03	3.86E+05
I-133	70	1.80E+05	1.26E+07
I-134	13	1.07E+03	1.39E+04
I-135	190	3.13E+04	5.95E+06
		Sum divided by I-131 DCF	6.0E+01

Repeating the Reference 29 calculation using CEDE DCFs based on the RADTRAD AST default file (Table 1.4.3.3-2, "Dose Conversion Factors for NUREG-1465 Nuclides" from Reference 9), the comparison in Table 4.2-11B is obtained.

Table 4.2-11B
Isotopic Concentrations
Representing An Equivalent Concentration of DE I-131
Using Updated Iodine RCS Concentrations and CEDE DCFs

	μCi/gm	CEDE DCF (thyroid rem/Ci)	Product
I-131	42.5	3.29E+04	1.40E+06
I-132	60	3.81E+02	2.29E+04
I-133	70	5.85E+03	4.10E+05
I-134	13	1.31E+02	1.70E+03
I-135	190	1.23E+03	2.34E+05
		Sum divided by I-131 DCF	6.3E+01

This means that defining dose equivalency based on CEDE DCFs and the individual radionuclide concentrations from Reference 29 would result in a CEDE dose response that would exceed that for 60 μCi/gm of I-131 by about 5%. Therefore, for analysis purposes, the isotopic concentrations in Table 4.2-12 is proposed as the dose equivalency to I-131 considering both CEDE DCFs and the relative iodine isotopic concentrations from Reference 29.

4.2 ANALYTICAL MODELS

Table 4.2-12
Proposed Isotopic Concentrations
Representing An Equivalent Concentration of DE I-131

	μCi/gm	CEDE DCF (thyroid rem/Ci)	Product
I-131	40.6	3.29E+04	1.34E+06
I-132	57	3.81E+02	2.17E+04
I-133	67	5.85E+03	3.92E+05
I-134	12	1.31E+02	1.57E+03
I-135	182	1.23E+03	2.24E+05
		Sum divided by I-131 DCF	6.0E+01

This result confirms that dose calculations performed using the specific concentrations shown above will produce the same dose result as 60 μCi/gm of I-131 when CEDE is the measure of dose consequence.

To make a relevant comparison for TEDE, one must assume a breathing rate. In the limit, for a case with a very high breathing rate or one in which substantial shielding reduces the external exposure dose to a very low level, the comparison would be similar to that for CEDE. Therefore, to make the TEDE comparison, the minimum breathing rate from RG 1.183 has been used; i.e., 1.75E-4 m^3/sec. For such a case, a pseudo-DCF for TEDE can be established where the TEDE DCF is the sum of the CEDE DCF multiplied by the assumed breathing rate and the effective dose equivalent (EDE) external exposure DCF, both values being taken from the RADTRAD AST default file. Then, the calculation can be repeated once again, with the results in Table 4.2-13.

Table 4.2-13
Proposed Isotopic Concentrations
Representing An Equivalent Concentration Of DE I-131
Using a TEDE DCF

	μCi/gm	TEDE DCF	Product
I-131	40.6	5.82E+00	2.36E+02
I-132	57	4.81E-01	2.74E+01
I-133	67	1.13E+00	7.57E+01
I-134	12	5.04E-01	6.05E+00
I-135	182	5.11E-01	9.30E+01
		Sum divided by I-131 DCF	7.5E+01

This shows the conservatism of using the CEDE DCFs to define DE I-131 for the purpose of making TEDE calculations. If the given coolant concentrations of I-131 through I-135 are used to make AST dose calculations, the calculated TEDE dose could be equivalent to as much as 75 μCi/gm of I-131 but not less than 60 μCi/gm depending on shielding and breathing rate. Therefore, the proposed definition of DE I-131 can be used conservatively for AST dose analyses and for the AST licensing basis. If the original Reference 29 iodine isotopic concentrations are used, the results will be even more conservative (by about 5% as discussed above).

4.2.4.4 Reactor Coolant System Source Terms

4.2.4.4.1 RCS at 1% Failed Fuel

The Reactor Coolant System source terms for 1% failed fuel are presented in Table 4.2-14.

Table 4.2-14
Comparison of CLB and AST Reactor Coolant Sources
@ 1% Failed Fuel
(μCi/gm)

Isotope	CLB[26]	AST[27]	% Difference
Kr83m	3.8E-01	3.7E-01	-2.6%
Kr85m	1.6E+00	1.5E+00	-6.3%
Kr85	7.7E+00	7.6E+00	-1.3%
Kr87	1.0E+00	9.8E-01	-2.0%
Kr88	2.9E+00	2.8E+00	-3.4%
Kr89	8.4E-02	8.4E-02	0.0%
Xe131m	1.9E+00	2.8E+00	47.4%
Xe133m	1.6E+01	4.2E+00	-73.8%
Xe133	2.4E+02	2.4E+02	0.0%
Xe135m	4.5E-01	4.0E-01	-11.1%
Xe135	8.5E+00	7.6E+00	-10.6%
Xe137	1.7E-01	1.6E-01	-5.9%
Xe138	5.9E-01	5.8E-01	-1.7%
I131	2.4E+00	1.7E+00	-29.2%
I132	2.7E+00	2.4E+00	-11.1%
I133	3.7E+00	2.8E+00	-24.3%
I134	5.5E-01	5.2E-01	-5.5%
I135	2.1E+00	7.6E+00	261.9%
Rb86	2.4E-02	1.7E-02	-29.2%

[26] Reference 27
[27] Reference 29

4.2 ANALYTICAL MODELS

Table 4.2-14
Comparison of CLB and AST Reactor Coolant Sources
@ 1% Failed Fuel
(μCi/gm)

Isotope	CLB[26]	AST[27]	% Difference
Rb88	3.6E+00	3.7E+00	2.8%
Rb89	2.4E-01	1.7E-01	-29.2%
Cs134	3.0E+00	1.4E+00	-53.3%
Cs136	3.6E+00	2.5E+00	-30.6%
Cs137	1.6E+00	1.1E+00	-31.3%
Te129m	1.6E-02	6.3E-03	-60.6%
Te129	1.6E-02	1.0E-02	-37.5%
Te131m	2.3E-02	1.6E-02	-30.4%
Te132	2.6E-01	1.8E-01	-30.8%
Ba137m	1.5E+00	1.0E+00	-33.3%
Ba140	3.5E-03	2.5E-03	-28.6%
Mo99	6.6E-01	4.6E-01	-30.3%
Tc99m	6.0E-01	4.2E-01	-30.0%
Ru103	5.0E-04	3.3E-04	-34.0%
Ru106	1.6E-04	1.1E-04	-31.3%
Y91	4.7E-04	3.2E-04	-31.9%
Y92	9.2E-04	7.8E-04	-15.2%
Y93	3.1E-04	2.5E-04	-19.4%
Zr95	5.5E-04	3.8E-04	-30.9%
Nb95	5.5E-04	3.8E-04	-30.9%
La140	1.1E-03	7.0E-04	-36.4%
Pr143	5.1E-04	3.6E-04	-29.4%
Ce143	4.2E-04	3.1E-04	-26.2%
Ce144	4.3E-04	2.8E-04	-34.9%
Sr89	3.3E-03	2.4E-03	-27.3%
Sr90	1.8E-04	1.2E-04	-33.3%
Sr91	6.9E-03	3.8E-03	-44.9%
Sr92	1.1E-03	1.0E-03	-9.1%

The CLB used only the iodine, krypton, and xenon isotopes. The AST analyses use these and the cesium and rubidium isotopes (unless otherwise noted).

4.2 ANALYTICAL MODELS

4.2.4.4.2 RCS Iodines at Normal Tech Spec Limit of 1 μCi/gm

Since the iodine concentrations at 1% failed fuel bound the concentrations for the normal Technical Specification limit of 1 μCi/gm DE I-131, the concentrations corresponding to the 1% failed fuel condition are used instead of the 1 μCi/gm DE I-131 concentrations in the revised analyses.

4.2.4.4.3 RCS Iodines at Spiking Tech Spec Limit of 60 μCi/gm

The initial iodine concentration in the reactor coolant is based on 60 μCi/gm DE I-131. Equation 1 shows the formulation for calculating DE I-131.

$$\chi_{131} + \chi_{132} \times \frac{DCF_{132}}{DCF_{131}} + \chi_{133} \times \frac{DCF_{133}}{DCF_{131}} + \chi_{134} \times \frac{DCF_{134}}{DCF_{131}} + \chi_{135} \times \frac{DCF_{135}}{DCF_{131}} = 60 \qquad \text{(Eq 1)}$$

where,

where,	χ_{131}	=	concentration of I-131
	χ_{132}	=	concentration of I-132
	χ_{133}	=	concentration of I-133
	χ_{134}	=	concentration of I-134
	χ_{135}	=	concentration of I-135
	DCF_{131}	=	I-131 dose conversion factor
	DCF_{132}	=	I-132 dose conversion factor
	DCF_{133}	=	I-133 dose conversion factor
	DCF_{134}	=	I-134 dose conversion factor
	DCF_{135}	=	I-135 dose conversion factor

The relative abundance of each isotope in the RCS is used in conjunction with Equation 1 to solve for the five concentrations. The concentration of each isotope in the RCS, based on 1% failed fuel, is presented in Table 4.2-14. The dose conversion factors are also included in Table 4.2-6. These dose conversion factors are the thyroid conversions from Reference 20.

Table 4.2-15 shows the calculation for the Reactor Coolant System (RCS) iodine concentration, based on Thyroid DCFs, for 1% failed fuel.

4.2 ANALYTICAL MODELS

Table 4.2-15
RCS Iodine Concentrations for 1% Failed Fuel

Isotope	CLB[28] (µCi/gm)	AST[29] (µCi/gm)	% Difference
I-131	2.4	1.7	-29.2%
I-132	2.7	2.4	-11.1%
I-133	3.7	2.8	-24.3%
I-134	0.55	0.52	-5.5%
I-135	2.1	7.6	261.9%

Table 4.2-16
RCS Iodine Concentrations and DCFs

Isotope	Concentration[30] (µC/gm)	Thyroid DCF[31] (Sv/Bq)
I-131	1.7	2.92E-7
I-132	2.4	1.74E-9
I-133	2.8	4.86E-8
I-134	0.52	2.88E-10
I-135	7.6	8.46E-9

The following relationships are based on the concentrations in Table 4.2-16.

$$\chi_{132} = \left(2.4/1.7\right)\chi_{131} \qquad \chi_{133} = \left(2.8/1.7\right)\chi_{131} \qquad \chi_{134} = \left(0.52/1.7\right)\chi_{131} \qquad \chi_{135} = \left(7.6/1.7\right)\chi_{131}$$

The relationships above are substituted in Equation 1 and this equation is solved for χ_{131}.

A summary of the RCS iodine concentrations is provided in Table 4.2-17.

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[28] Reference 27
[29] Reference 29
[30] Reference 29, page 5.34
[31] Reference 20, page 136

4.2 ANALYTICAL MODELS

Table 4.2-17
RCS Iodine Concentrations for a Pre-existing Iodine Spike to 60 μCi/gm

Isotope	CLB (μCi/gm)	AST (μCi/gm)	% Difference
I-131	46	42.5	-7.6%
I-132	52	60.0	15.4%
I-133	72	70.0	-2.8%
I-134	10	13.0	30.0%
I-135	40	190.0	375.0%

4.2.4.4.4 RCS Cs and Rubidium Concentrations

The RCS initial Cs and Rb concentrations are given in Table 4.2-14. However, a preexisting iodine spike is conservatively assumed to cause an increase in Cs and Rb activities, along with the increase in iodine concentrations. Table 4.2-18 shows the total activities from a pre-accident spike.

Table 4.2-18
Total RCS Cs and Rb Activity for a Pre-Accident
Iodine Spike
(Ci)

Isotope	CLB	AST
Rb-86	N/A	1.36E+2
Rb-88	N/A	2.95E+4
Rb-89	N/A	1.34E+3
Cs-134	N/A	1.12E+4
Cs-136	N/A	1.99E+4
Cs-137	N/A	8.77E+3

For cases involving an accident-induced spike, the activities are shown accident-dependant and provided in the respective accident discussion.

4.2 ANALYTICAL MODELS

4.2.4.5 Secondary System Source Terms

4.2.4.5.1 Secondary System Iodine Concentrations

The initial iodine concentration in the secondary systems is based on the Technical Specification limit of 0.10 μCi/gm DE I-131. Equation 2 shows the formulation for calculating DE I-131.

$$\chi_{131} + \chi_{132} \times \frac{DCF_{132}}{DCF_{131}} + \chi_{133} \times \frac{DCF_{133}}{DCF_{131}} + \chi_{134} \times \frac{DCF_{134}}{DCF_{131}} + \chi_{135} \times \frac{DCF_{135}}{DCF_{131}} = 0.10 \qquad \text{(Eq 2)}$$

where,

χ_{131}	=	concentration of I-131
χ_{132}	=	concentration of I-132
χ_{133}	=	concentration of I-133
χ_{134}	=	concentration of I-134
χ_{135}	=	concentration of I-135
DCF_{131}	=	I-131 dose conversion factor
DCF_{132}	=	I-132 dose conversion factor
DCF_{133}	=	I-133 dose conversion factor
DCF_{134}	=	I-134 dose conversion factor
DCF_{135}	=	I-135 dose conversion factor

The relative abundance of each isotope in the RCS is used in conjunction with Equation 2 to solve for the five concentrations. The concentration of each isotope in the RCS, based on 1% failed fuel, is presented in Table 4.2-16. The dose conversion factors are also included in Table 4.2-16.

The following relationships are based on the concentrations in Table 4.8-19.

$$\chi_{132} = \left(2.4/1.7\right)\chi_{131} \qquad \chi_{133} = \left(2.8/1.7\right)\chi_{131} \qquad \chi_{134} = \left(0.52/1.7\right)\chi_{131} \qquad \chi_{135} = \left(7.6/1.7\right)\chi_{131}$$

The relationships above are substituted in Equation 2 and this equation is solved for χ_{131}.

A summary of the secondary iodine concentrations is provided in Table 4.2-19.

4.2 ANALYTICAL MODELS

Table 4.2-19
Secondary Iodine Concentrations at 0.1 μCi/gm
(μCi/gm)

Isotope	CLB	AST	% Difference
I-131	7.5E-02	7.08E-02	-5.6%
I-132	8.8E-02	1.00E-01	13.6%
I-133	1.2E-01	1.17E-01	-2.5%
I-134	1.8E-02	2.17E-02	20.6%
I-135	6.6E-02	3.17E-01	380.3%

The large increase in I-135 is attributable to the change in relative DCFs from the CLB to the AST/TEDE analysis.

4.2.4.5.2 Secondary System Noble Gas Concentrations

The noble gas concentrations and the organic iodine concentration are determined as a function of the primary-to-secondary leak rate and the steam flow rate (1.574E+07 lbm/hr). The RCS concentrations are taken from Reference 29. The secondary concentrations are calculated using the equation below.

$$Secondary\ Concentration = \frac{RCS\ Concentration \times (Primary - to - Secondary\ Leakrate)}{Steam\ Flow\ Rate}$$

The initial RCS and secondary activities are presented in Table 4.2-20. The RCS mass used for calculating the activities is 2.658E+8 gm. The secondary mass is 659,412 lbm (2.991E+8 gm). This results in the secondary side concentration of a nuclide being a factor of 3.18E-5 that of the primary side concentration.

4.2 ANALYTICAL MODELS

Table 4.2-20
Initial RCS (@60 μCi/gm) and Secondary Concentrations (@ 0.1 μCi/gm DEI)
(Noble Gases based on 1% Failed Fuel)

Isotope	RCS[32] (μCi/gm)	Secondary[33] (μCi/gm)	RCS (Ci)	Secondary[34] (Ci)
I-131	4.25E+01	7.08E-02	1.1E+04	2.1E+01
I-132	6.00E+01	1.00E-01	1.6E+04	3.0E+01
I-133	7.00E+01	1.17E-01	1.9E+04	3.5E+01
I-134	1.30E+01	2.17E-02	3.5E+03	6.5E+00
I-135	1.90E+02	3.17E-01	5.1E+04	9.5E+01
Kr-83m	3.7E-01	1.2E-05	9.8E+01	3.6E-03
Kr-85m	1.5E+00	4.8E-05	4.0E+02	1.4E-02
Kr-85	7.6E+00	2.4E-04	2.0E+03	7.2E-02
Kr-87	9.8E-01	3.1E-05	2.6E+02	9.3E-03
Kr-88	2.8E+00	8.9E-05	7.4E+02	2.7E-02
Kr-89	8.4E-02	2.7E-06	2.2E+01	8.1E-04
Rb-86	1.7E-02	5.4E-07	4.5E+00	1.6E-04
Rb-88	3.7E+00	1.2E-04	9.8E+02	3.6E-02
Rb-89	1.7E-01	5.4E-06	4.5E+01	1.6E-03
Xe-131m	2.8E+00	8.9E-05	7.4E+02	2.7E-02
Xe-133m	4.2E+00	1.3E-04	1.1E+03	3.9E-02
Xe-133	2.4E+02	7.6E-03	6.4E+04	2.3E+00
Xe-135m	4.0E-01	1.3E-05	1.1E+02	3.9E-03
Xe-135	7.6E+00	2.4E-04	2.0E+03	7.2E-02
Xe-137	1.6E-01	5.1E-06	4.3E+01	1.5E-03
Xe-138	5.8E-01	1.8E-05	1.5E+02	5.4E-03
Cs-134	1.4E+00	4.5E-05	3.7E+02	1.3E-02
Cs-136	2.5E+00	8.0E-05	6.6E+02	2.4E-02
Cs-137	1.1E+00	3.5E-05	2.9E+02	1.0E-02
Cs-138	8.9E-01	2.8E-05	2.4E+02	8.4E-03

[32] Table 4.2-17 for iodine data and Reference 29 for balance of data.
[33] Table 4.2-19 for iodine data, balance of data determined from the previous equation.
[34] Last digit is subject to round-off changes in individual analyses.

4.2 ANALYTICAL MODELS

4.2.5 Iodine Species Released from Steam Generators

For the applicable accidents, the release of iodines from the fuel (and RCS) is modeled in accordance with Regulatory Guide 1.183 (Appendix E.4: 4.85% elemental, 0.15% organic, and 95% particulate. Appendix E also states that the iodine release from the SG should be 97% elemental and 3% organic. This is a result of not releasing the particulates that comprise 95% of the RCS flow mixing into the bulk SG water. However, Section 5.5.4 allows for a partition factor of 100 for iodines and states that "[t]he retention of particulate radionuclides in the steam generators is limited by the moisture carryover for the steam generators." The contradiction is that in Appendix E, Section 4, the particulates are seemingly not released and in Section 5.5.4 there is some guidance on handling particulates.

The STP analysis, therefore, make two assumptions:

1. Organic iodines are released without the reduction of 100 afforded by the partition factor granted in Appendix E, Section 5.5.4; and
2. Release of iodine particulates will be modeled, in seeming contradiction to Appendix E, Section 4, but using the partition factor of 100.

Therefore, the 4.85/0.015/95 split from the RCS becomes

$$\frac{4.85}{100} \ / \ \frac{0.15}{1} \ / \ \frac{95}{100}$$

when the partition factors are applied. The resulting split is then 0.485/0.15/0.95 among the iodine species. Renormalizing, the fractions are 4.2% elemental, 13.1% organic, and 82.7% particulate.

Based on the above, the STP analyses use an elemental/organic/particulate species split of 4.2%/13.1%/82.7% in lieu of the Regulatory Guide 1.183 split of 97%/3%/0%. Note that the number of curies of iodines released are greater than that required by Regulatory Guide 1.183 (particulates are released and no partition factor is used to reduce the amount of organics released).

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4.3 Loss of Coolant Accident Radiological Assessment

4.3.1 Methodology Overview

The LOCA is modeled as a release of nuclides from the reactor core into the containment building. Subsequent releases to the environment are as follows:

- Leakage through the containment walls, at the allowed Technical Specification leakage rate of 0.3% for the first 24 hours and one half that value after 24 hours
- The (pre-clad rupture) activity in the reactor coolant system through the containment supplemental purge system, terminating when the supplemental purge system isolation valves close (automatically upon receipt of the safety injection signal)
- Leakage via Engineered Safety Features (ESF) components in the Fuel Handling Building, at an assumed rate of 8280 cc/hr (double the allowed leakage rate of 4140 cc/hr).

Credit for containment spray is taken to reduce the amount of radionuclides available for leakage from the containment.

The radiological source term characteristics and release timing are based on the Alternative Source Term (AST) methodology in Regulatory Guide 1.183 and from NUREG-1465.

Atmospheric dispersion factors from Section 4.1, above, are used in this analysis.

Doses to the public at the Exclusion Area Boundary (EAB) and the Low Population Zone (LPZ) and to operators in the Control Room and the Technical Support Center (TSC) are determined.

4.3.2 Radiological Source Term

For conservatism, the LOCA core source terms are those associated with a DBA power level of 4100 MWth compared to the licensed power level of 3853 MWth with a 0.6% measurement uncertainty.

The AST values used in this analysis were derived using guidance outlined in Regulatory Guide 1.183. The ORIGEN 2.1 code was used to calculate plant-specific fission product inventories for use in the DBA LOCA dose analyses. The assumed period of irradiation was sufficient (three-region equilibrium cycle core at end of life with the three regions having operated at 39.31 MW/MTU for 509, 1018, and 1527 EFPD, respectively) to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. Certain radionuclides appearing in the default list of radionuclides for the RADTRAD 3.03 computer code but not appearing in the summary of the ORIGEN analysis were taken from the PWR default .NIF file for RADTRAD. These include Ba139, La141, and Np239 (used as-is from the PWR default .NIF file) and Am241, Cm242, Cm244, Pu238, Pu239, Pu240, and Pu241 (used with activities

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increased by a factor of three for conservatism because of their half-lives being greater than 100 days).

In addition to the radionuclides appearing in the RADTRAD list, Kr83m, Xe131m, Xe133m, and Xe135m were added for dose analysis purposes based on their inclusion in TID-14844. Xe138 was also added. Co58 and Co60 were deleted from the list because only 63 radionuclides can be used. A study indicated that omitting Co58 and Co60 decreased the control room dose by about 0.01 percent while adding the noble gas isotopes increased the control room dose by about 0.1 percent.

Fission product activities were calculated for immediately after shutdown and decayed for the required times. The shutdown values are shown in Table 4.3-1 (values are from Table 4.2-9, except they are expressed in terms of Ci/MWt for RADTRAD). The CLB analyses assumed 100% of the noble gases, 50% of the iodines, and 1% of the core solids were released from the core, per TID-14844. For the CLB offsite, TSC, and Control Room doses, only the iodines and noble gases were considered.

Table 4.3-1
LOCA: Reactor Core Fission Product Inventory @ t=0
(Ci/MWt)

Isotope[35]	CLB (TID)[36]	AST	% Difference
Kr83m	3.41E+03	3.41E+03	-0.1%
Kr85m	7.32E+03	7.07E+03	-3.4%
Kr85	2.93E+02	2.93E+02	0.1%
Kr87	1.34E+04	1.34E+04	-0.1%
Kr88	1.93E+04	1.90E+04	-1.4%
Kr89	2.37E+04	***2.32E+04***	-1.9%
Xe131m	1.88E+02	2.68E+02	42.7%
Xe133m	8.05E+03	1.66E+03	-79.4%
Xe133	5.61E+04	5.37E+04	-4.3%
Xe135m	1.12E+04	1.02E+04	-9.1%
Xe135	1.59E+04	1.34E+04	-15.5%
Xe137	4.88E+04	***4.63E+04***	-5.1%
Xe138	4.63E+04	4.39E+04	-5.3%
I131	2.78E+04	2.59E+04	-6.9%
I132	4.00E+04	3.71E+04	-7.3%
I133	5.85E+04	5.37E+04	-8.3%
I134	6.34E+04	5.85E+04	-7.8%

[35] The three isotopes in bold italics were only used in the STARDOSE (Reference 34) confirmatory analyses.
[36] Derived from Table 5-9, Reference 27

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Table 4.3-1
LOCA: Reactor Core Fission Product Inventory @ t=0
(Ci/MWt)

Isotope[35]	CLB (TID)[36]	AST	% Difference
I135	5.37E+04	4.88E+04	-9.1%
Rb86	-	9.92E+01	-
Cs134	8.05E+03	5.37E+03	-33.3%
Cs136	2.27E+03	1.54E+03	-32.1%
Cs137	3.41E+03	3.17E+03	-7.2%
Sb127	-	3.05E+03	-
Sb129	9.02E+03	8.29E+03	-8.1%
Te127m	-	4.32E+02	-
Te127	-	3.05E+03	-
Te129m	2.32E+03	1.22E+03	-47.3%
Te129	8.54E+03	8.05E+03	-5.7%
Te131m	4.15E+03	3.66E+03	-11.7%
Te132	3.90E+04	3.82E+04	-2.1%
Ba137m	3.17E+03	***2.93E+03***	-7.6%
Ba139	-	4.98E+04	-
Ba140	4.88E+04	4.63E+04	-5.1%
Mo99	5.12E+04	4.83E+04	-5.7%
Tc99m	4.39E+04	4.07E+04	-7.3%
Ru103	4.39E+04	3.90E+04	-11.2%
Ru105	2.93E+04	2.68E+04	-8.4%
Ru106	1.41E+04	1.34E+04	-5.3%
Rh105	-	3.05E+04	-
Y90	-	3.56E+03	-
Y91	3.41E+04	3.41E+04	-0.1%
Y92	3.66E+04	3.41E+04	-6.8%
Y93	4.15E+04	3.90E+04	-5.9%
Zr95	4.63E+04	4.39E+04	-5.3%
Zr97	4.63E+04	4.39E+04	-5.3%
Nb95	4.88E+04	4.32E+04	-11.4%
La140	5.12E+04	4.63E+04	-9.6%
La141	-	4.62E+04	-
La142	4.39E+04	4.15E+04	-5.5%
Pr143	4.15E+04	3.90E+04	-5.9%
Nd147	1.80E+04	1.73E+04	-4.1%
Am241	-	2.75E+00	-
Cm242	-	1.05E+03	-
Cm244	-	6.17E+01	-
Ce141	4.63E+04	4.39E+04	-5.3%

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Table 4.3-1
LOCA: Reactor Core Fission Product Inventory @ t=0
(Ci/MWt)

Isotope[35]	CLB (TID)[36]	AST	% Difference
Ce143	4.39E+04	4.15E+04	-5.5%
Ce144	3.41E+04	3.41E+04	-0.1%
Np239	-	5.12E+05	-
Pu238	-	8.71E+01	-
Pu239	-	1.96E+01	-
Pu240	-	2.48E+01	-
Pu241	-	4.17E+03	-
Sr89	2.68E+04	2.68E+04	-0.1%
Sr90	2.44E+03	2.37E+03	-2.8%
Sr91	3.41E+04	3.17E+04	-7.2%
Sr92	3.66E+04	3.41E+04	-6.8%

4.3.3 Radiological Releases

4.3.3.1 Radiological Releases from the Containment

Activity released to the containment is apportioned to the sprayed and unsprayed regions according to volume, 0.8 to the sprayed region and 0.2 to the unsprayed region based on the relative volumes.

Containment spray removal coefficients continue to be based on Standard Review Plan 6.5.2, "Containment Spray as a Fission Product Cleanup System," Revision 2, December 1998, with "particulate" removal coefficients applied to "aerosols." Spray timing is adjusted slightly to reflect AST-caused differences in time to reach decontamination factor (DF) credit limits.

The assumed containment leak rate directly to the environment is the same as the CLB. For the first 24 hours following the accident, the leak rate is assumed to be at the Containment Leakage Testing Program (Technical Specification 6.8.3.j) limit of 0.30% per day, while for the remainder of the 30-day period the leak rate is assumed to be 0.15% per day.

Primary containment leakage to the environment is modeled as a diffuse area source in conformance with Regulatory Guide 1.194.

Containment leakage through electrical penetrations into the electrical penetration area is very limited (0.025 cfm out of 7.02 cfm containment leakage). It is held up in the electrical penetration area as a source of gamma shine dose to the Control Room. A discussion of releases

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into the electrical penetration area as a source for a "sneak path" of airborne contaminants into the Control Room is provided in Section 4.2.2.1.

4.3.3.1.1 Release from the Containment Supplemental Purge Subsystem

Containment leakage via open supplemental purge lines occurs for the first 23 seconds of the onset of the accident (following the Containment Pressure-High 1 signal and including valve closing time, Standby Diesel Generator startup time and signal and sequencer delays). The assumed volumetric flow rate is found in Table 4.3-11. This leakage is released to the environment via the plant vent.

During this time period, fuel failure has not occurred (see Table 4.3-10). This release consists of reactor coolant blowing down into the containment. The flow rate out of the supplemental purge line is assumed to be at maximum choke flow. The flow is doubled to account for flow in both the intake and exhaust lines. The purge system exhausts via the Plant Vent.

The reactor coolant concentrations are based on 1% failed fuel, which is greater than the values corresponding to the 1 μCi/gm DE I-131 Technical Specification limit. Accordingly, these values also bound the Regulatory Guide 1.183, Appendix A, Section 3.8, position that iodine concentrations corresponding to 1.0 μCi/gm DE I-131 should be used.

The CLB uses the reactor coolant concentrations for xenons and kryptons for 1% failed fuel (Table 4.2-14). A pre-existing iodine spike to 60 μCi/gm DE I-131 was modeled by using a value of 60 μCi/gm for I-131.

4.3.3.1.2 Containment Sump pH and Iodine Re-evolution

An evaluation of containment sump pH was conducted to ensure that the particulate iodine deposited into the containment water during the DBA LOCA does not re-evolve beyond the amount recognized in the DBA LOCA analysis. The objective of the analysis was to determine the transient containment sump pH so that the removal of elemental and particulate iodine (cesium iodide - CsI) from the containment atmosphere in the course of the DBA LOCA would not be overstated. The analysis credits the pH buffering effect of trisodium phosphate (TSP) stored in the containment sump.

4.3.3.1.2.1 Determination of Sump pH

The calculation methodology for containment sump pH control is based on the approach outlined in NUREG-1465 and NUREG/CR-5950, (Reference 35). Specifically, credit is taken for TSP dissolution in the containment water as a result of released reactor coolant and injected spray water coming in contact with the stored TSP in the lower elevation of containment.

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The pH of the containment sump water was then calculated using the STARpH code (Reference 36). The STARpH computer code is used for determining the pH of the containment sump (PWR) or suppression pool (BWR). It has been used in the following AST applications that have received a satisfactory NRC Safety Evaluation:

- Perry
- Hope Creek
- Browns Ferry
- Vermont Yankee
- Waterford-3

The amount of cable insulation in containment was determined by performing a survey of design documents and determining the total volume of cable insulation and jacket materials. As would be expected, STP has a larger mass of cable insulation than most plants. STP has three safety trains instead of the traditional two. Also, the Residual Heat Removal System (RHR) is located inside the containment. Additionally, the containment is relatively large at 158 feet in diameter. Cable data is presented in Table 4.3-2.

The design inputs were conservatively established to maximize the post-LOCA production of acids and to minimize the post-LOCA production and/or addition of bases.

In calculating the sump pH, the three major contributors to strong acid production are considered: boric acid from the reactor coolant system, the accumulators, and the refueling water storage tank (RWST); nitric acid from radiolysis of water; and, hydrochloric acid from radiolysis of chloride-bearing cable jacket/insulation. Production of organic acid from coatings is also evaluated. For South Texas, this contribution was found to be negligible.

Major assumptions used in the sump pH analysis are:

1. Per the Technical Specifications, the containment contains a minimum of 11,500 lbm of trisodium phosphate (TSP). Trisodium phosphate is stored in baskets located on the containment floor where they would be submerged in the event of a LOCA. During each refueling outage, a surveillance is performed to verify that the six trisodium phosphate storage baskets are in place, have maintained their integrity, and are filled with trisodium phosphate such that the level is within the specified range.
2. For cables without specific dimension data, the fraction of the cable cross section that is insulation is assumed to be 0.6 and all the insulation and jacket material is assumed to be Hypalon.
3. The as-built thickness of cable insulation is assumed to be a maximum of 10% larger and the jacket material is assumed to be a maximum of 25% larger than the design specification value.
4. Cable insulation quantities are increased by 5% to bound future modifications that add cable to the containment building.

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5. The fraction of the aerosol source term in the sump is 0.9.

   Spray removal of activity will wash a large fraction of the activity from the atmosphere into the containment sump. Most of the aerosol activity will be released during the 1.3 hour in-vessel release phase (Regulatory Guide 1.183), giving a maximum release rate of 1/1.3= 0.77 per hour. With a spray removal rate of 6.9 per hour in the sprayed region and a total containment volume about 1.25 times greater than the sprayed region, the effective lambda is about 6.9/1.25=5.5 per hour. The maximum equilibrium fraction of aerosol airborne during the release phase would be about 0.77/5.5=0.14. Beyond the end of the release phase, this fraction would rapidly decrease, most likely approaching zero. Even though some of the release would effectively remain airborne as transient spray droplets and wetted surfaces, greater than 0.9 would be expected to be waterborne, and it is, therefore, conservative to use this fraction.

6. Organic acid from radiolysis of organic materials dissolved from the containment surface coatings in contact with the pool can be neglected.

   The $[H^+]$ from production of organic acid in the containment sump is expected to be a small fraction of the total $[H^+]$ from nitric and hydrochloric acid calculated to be produced from the radiolysis of water and cables. The bases for this assertion are:
   a) The $[H^+]$ from organic acid produced in the RTF experiments with painted surfaces varies from 4.2E-07 moles/liter for Epoxy and Polyurethane paints to 1.7E-05 mol/L for vinyl paint with paints cured 3 months (Reference 37), whereas the total $[H^+]$ calculated from the production of nitric and hydrochloric acids is 1.04E-03 mol/L;
   b) The dissolution of organics from paints (the controlling mechanism for the radiolytic production of organic acid) decreases with the age of the paint (reduction factor of approximately 4 from 10 to 100 days and an additional factor of 2 from 100 days to 1000 days) (Reference 37), whereas the STP containment surfaces were originally coated with organic materials prior to reactor startup in 1988 (Unit 1) and 1989 (Unit 2) (over 6000 days of aging) and only limited touchup has been done during outages since that time; and,
   c) The painted surfaces in contact with the containment sump are coated with epoxy paints.

   Thus, the [H+] from organic acids will be 4.2E-07/1.04E-03, or 0.039% of the $[H^+]$ produced from nitric and hydrochloric acids.

7. No credit is taken for basic alkali metal compounds that result from fission products co-released with the iodine.
8. Cesium compounds are not credited in the long-term pH analyses and the determination of the final (i.e., 30 day) pH value.
9. The favorable impact of fission product chemistry on sump pH is largely ignored. Although some HI may be formed, the amount of HI would be overwhelmed by the favorable impact of Cs compounds, in particular CsOH.

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10. For conservatism, 10% of non-noble gas activity is assumed to remain airborne for the full 30 days, even in the presence of sprays. All of the noble gas activity is assumed to remain airborne. This increases the amount of radiation exposure to cables.
11. For HCl formation, a factor of two reduction is credited for beta shielding of cable in trays. This is conservative since cables are usually layered in trays, providing a significant amount of self shielding. A factor of ten is credited for the 16% of cable that is estimated to be in conduit.

The inputs for the pH evaluation are presented in Table 4.3-2.

Table 4.3-2

Containment Sump pH Control Inputs

Input/Assumption	Current Licensing Basis	AST
Mass of water in post-accident sump	2.44E+09 grams	2.44E+09 grams
Boron concentration (as boric acid) in sump	3060 ppm	3060 ppm
Mass of TSP dodecahydrate	11,500 lbm	11,500 lbm
Initial sump pH after TSP dissolution	7.01	7.01
Containment volume	N/A	3.38E+06 ft^3
Volume of Hypalon in containment	N/A	1.76E+07 cc
Mass of Hypalon in containment	N/A	2.73E+07 grams
Density of Hypalon	N/A	1.55 gm/cc
Representative thickness of cable jacket	N/A	64.5 mils
Representative cable outside diameter	N/A	0.65 inches
Percent of cable in conduit	N/A	16%
Conduit outside diameter	N/A	1.94 inches
Conduit thickness	N/A	0.153 inches
Fraction of exposed cables in trays	N/A	100%

The STARpH code was used to determine the amount of [HNO_3] in the sump water generated by radiolysis of water. Organic acids from the containment surfaces coated with organic materials was neglected. A water density of 1.0 gm/ml was used to minimize the volume and maximize the HNO_3 concentration. The cumulative amount of HNO_3 in the containment sump as a function of time is provided in Table 4.3-3.

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The STARpH code was used to determine the amount of [HCl] in the sump water generated by radiolysis of cable insulation. STARpH uses the methodology described in Appendix B of NUREG/CR-5950. Using the formulation in Appendix B of NUREG/CR-5950, the rate of HCl formulation is given by

$$R = R_{\gamma H} + R_{\beta H}$$

where R = the total rate of HCl generation rate (gm-mols/sec)
$R_{\gamma H}$ = the HCl generation rate due to γ radiolysis (gm-mols/sec)
$R_{\beta H}$ = the HCl generation rate due to β radiolysis (gm-mols/sec)

$R_{\gamma H}$ is determined to be 3.00E-15 gm-mols/sec and $R_{\beta H}$ is determined to be 1.75E-15 gm-mols/sec. After correcting for beta shielding, $R_{\beta H}$ is 1.52E-15 gm-mols/sec. The cumulative amount of HCl in the containment sump as a function of time is provided in Table 4.3-3.

To determine the sump pH the mass of boron and TSP are also used. The sump temperature at 22 hours is 172°F (78°C). After which, it will decrease with long-term cooling of the sump. Based on curve fitting and extrapolation of dissociation constant data as a function of temperature (0 to 50°C) from Dean (Reference 38), at 78°C the dissociation constant (K_A) for boric acid is 9.04E-10 and that for PO_4 (K_{A2}) is 5.01E-08. These parameters were used in STARpH to yield the pH values as a function of time presented in Table 4.3-3.

The initial effects on post-accident containment sump pH is from rapid fission product transport and the formation of cesium compounds, which results in increasing the containment sump pH. The buffering effect of TSP within a few hours is sufficient to offset the effects of these acids that are transported to the sump and maintain containment sump pH at or above 7.0 for the first day.

The impact of HCl formation from cable radiolysis is about four times greater than the impact of nitric acid formation from water radiolysis.

As radiolytic production of nitric acid and hydrochloric acid proceeds and these acids are transported to the pool over the first days of the event, the pH becomes more acidic. After the first day, the containment sump pH will begin to decrease, reaching 6.8 by the end of the 30-day duration of the radiological consequence analysis for the DBA LOCA, and the impact of that decrease has been reflected in the Control Room and offsite doses.

Although the results of this analysis indicate that the sump pH drops slightly below 7.0, in reality there should be little impact on the actual iodine re-evolution due to the conservatisms in the analysis:

- Conservative estimates on cable dimensions and materials were made to increase the cable insulation mass and its effect on sump pH;

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- Cesium compounds are not credited in the long-term pH analyses and the determination of the final (i.e., 30 day) pH value;
- No credit is taken for basic alkali metal compounds that result from fission products co-released with the iodine;
- Conservative assumptions were made to retain 10% of non-noble gas activity as airborne activity for the full 30 days, even in the presence of sprays, and all of the noble gas activity is assumed to remain airborne (increasing the amount of radiation exposure to cables); and
- Conservative assumptions were made concerning the vulnerability of cables to beta radiation.

The above conservatisms are consistent with conservatisms in other Polestar sump pH analyses. In addition, further conservatisms are incorporated into the determination of iodine re-evolution, discussed in Section 4.3.3.1.2.2.

Table 4.3-3
Sump Concentrations and pH as a Function of Time

End of Time Interval	[HNO_3]	[HCl]	[H^+]	pH
1 hour	8.19E-06	2.70E-05	3.52E-05	7.0
2 hours	1.13E-05	4.42E-05	5.55E-05	7.0
5 hours	1.77E-05	8.00E-05	9.77E-05	7.0
12 hours	2.82E-05	1.29E-04	1.58E-04	7.0
1 day	4.21E-05	1.84E-04	2.27E-04	7.0
3 days	8.13E-05	3.34E-04	4.16E-04	6.9
10 days	1.53E-04	6.10E-04	7.64E-04	6.9
20 days	1.99E-04	7.48E-04	9.47E-04	6.9
30 days	2.29E-04	8.12E-04	1.04E-03	6.8

4.3.3.1.2.2 Iodine Re-evolution

The STP DBA LOCA analysis assumes iodine removal from the containment atmosphere by both containment sprays and natural diffusion to walls. This will lead to a large fraction of activity being deposited in the containment sump. The sump water will also retain soluble gaseous and soluble fission products such as iodides and cesium, but not noble gases. Once deposited, the iodine will remain in solution as long as the containment sump pH is maintained at or above 7.0. An analysis of the associated iodine DF for containment iodine removal and retention was also performed.

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When evaluating the impact of pH being below 7 in the long term (i.e., the elemental iodine DF for spray and natural removal), the maximum sump temperature is used in conjunction with the lowest pH, even though the first condition occurs in the first minutes of the accident and the latter condition occurs at the very end of the 30-day dose assessment period. This is a very conservative treatment of the impact of sump pH on iodine DF.

Major assumptions used to determine the elemental iodine DF in the containment are:

1. The containment sump has a pH of 6.8 at the time of the maximum sump temperature;
2. The containment sump reaches its maximum temperature of 266°F at 1600 seconds into the LOCA (based on STP analyses);
3. The mass of I-127 is 6.2 kg (48.8 moles);
4. The mass of I-129 is 3.2 kg (164.4 moles);
5. Assume a release fraction of 0.4 for I-127 and I-129; and,
6. The sump volume is 2.44E+06 liters (2.44E+09 grams at a density of 1 gm/cc).

To determine the maximum DF for elemental iodine, both the fraction of total iodine in the sump water that is in elemental form and how much elemental iodine would have to be airborne to be in equilibrium with the remaining elemental iodine in the water must be determined.

To determine the mass fraction of dissolved iodine that is in elemental form, Equation 24 of Reference 39 is used, along with constants "a" and "b" from Table 5 of Reference 39. Rearranging the equation yields

$$2[I_2] / [I^-] = 2([H^+]2[I^-]) / (a + (b[H^+])$$

where 2 = the mass fraction is 2x the mole fraction,
a = $(6.05 \pm 1.83) \times 10^{-14}$,
b = 1.47E-09.

Assuming the mass fraction is very small, $[I^-] \approx [I]$. Also, using the smaller value of "a" to maximize the elemental iodine mass fraction, and the masses for I-127 and I-129 and the sump volume assumed above, yields

$$2[I_2] / [I] = 2(3.5E\text{-}05[H^+]^2) / (4.22E\text{-}14 + 1.47E\text{-}09[H^+]).$$

A plot of the total iodine that is in elemental form in the sump water as a function of pH (where $[H+] = 10^{-pH}$) is presented in Figure 4.3-1.

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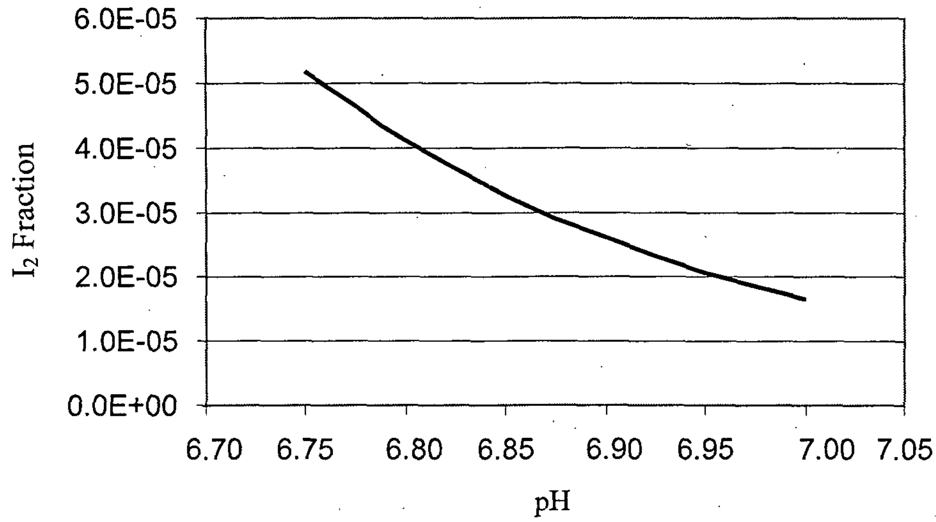
Figure 4.3-1: I_2 Fraction vs pH



The relative concentration of the elemental iodine in the sump water to that in the atmosphere is developed from Reference 39, page 55:

$$P = 10^{(6.29-0.0149T)}$$

for T in degrees Kelvin.

The same expressions for iodine speciation in the sump and partitioning between the sump and containment atmosphere appear in NUREG/CR-5950 as well as Reference 39.

Equilibrium is reached when

$$[I_2]_L / [I_2]C_G = P = I_{2L}\, V_G / I_{2G} V_L$$

where V_L = volume of liquid
V_G = volume of gas
I_{2L} = mass of I_2 in the liquid
I_{2G} = mass of I_2 in the gas.

The mass of I_2 in the gas is then

$$I_{2G} = I_{2L}\, V_G / P\, V_L$$

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Using the volume of the containment and the containment sump, the relative iodine concentration is

$$I_{2G} / I_{2L} = 39.2/P$$

Therefore, the fraction of released iodine that may be airborne, as a function of temperature is

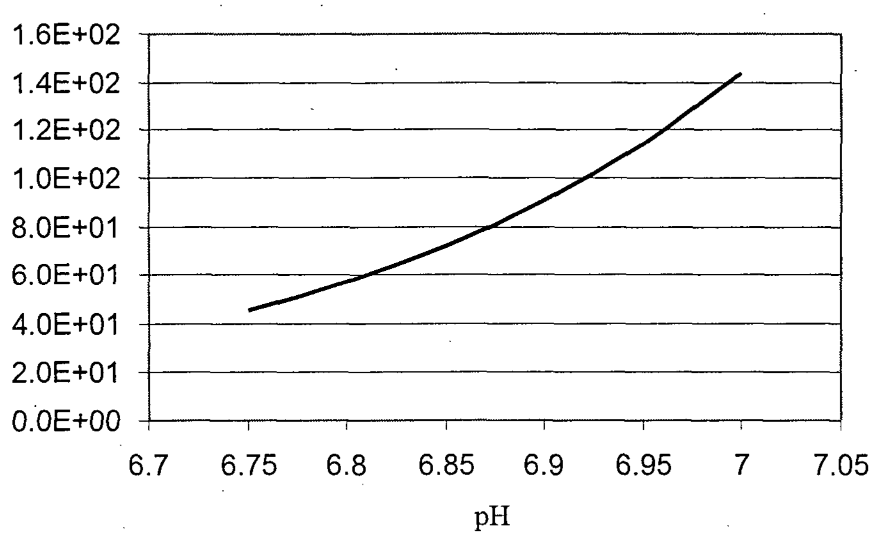
$$I_{2G} / I_{2L} = (39.2 * 10^{(6.29-0.0149T)})(2)(3.5E\text{-}05[H^+]^2) / (4.22E\text{-}14 + 1.47E\text{-}09[H^+]).$$

Since the fraction of released iodine that is airborne initially as elemental iodine is 0.0485 (Regulatory Guide 1.183), the DF will be the initial fraction divided by the fraction at any time, or

$$DF = 0.0485 / [(39.2 * 10^{(6.29-0.0149T)})(2)(3.5E\text{-}05[H^+]^2) / (4.22E\text{-}14 + 1.47E\text{-}09[H^+])]$$

Assuming the maximum containment temperature (265.8°F), the DF as a function of pH is shown in Figure 4.3-2.

Figure 4.3-2: Iodine Decontamination Factor as a Function of pH



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A DF of 60, corresponding to a pH of 6.8, is to be used in the dose analysis, even though the calculated value of pH at 30 days is just below 6.85. Note that at a pH of 7.0, the DF approaches 150. The calculation is very conservative in that (1) the highest sump temperature is used and (2) the lowest pH is assumed throughout the duration of the accident. The DF of 60 will be exceeded at all times since early in the accident the sump pH is greater than 6.8 and later the sump temperature is much less than the maximum value.

Iodine Re-evolution from ESF Leakage

The percentage of iodine in ESF system coolant leakage outside containment that becomes airborne may be assumed to be 10% as long as the pH is equal to or greater than 7.0. However, this is only true for the first day per Table 4.3-3. From Figure 4.3-1, at a pH of 6.8, the I_2 fraction is about 2.5 times greater than at a pH of 7.0; and at a pH of 6.9, the I_2 fraction is about 1.6 times greater than at a pH of 7.0.

Therefore, to account for the impact of a pH less than 7 on iodine re-evolution from ESF leakage, the 10% re-evolution fraction (for a pH > 7, Regulatory Guide 1.183) is increased by the ratio of elemental iodine abundance at pH = f(t) to that at pH = 7. From t = 24 hours to t = 20 days a factor of 1.6 (16%) (corresponding to pH = 6.9) is used, and from t = 20 days to t = 30 days a factor of 2.5 (corresponding to pH = 6.8) is used.

Impact of Using a Transient DF

To judge the conservatism of using a DF value of 60 at the sump's minimum pH and maximum temperature, the equation above for DF as a function of temperature and pH was used, along with the sump temperature over time, to generate Figure 4.3-3. The sump pH was assumed to follow Table 4.3-3. (The step changes for the pH(t) plot are due to the step decreases in sump pH, per Table 4.3-3.) If iodine is allowed to re-evolve in the dose analysis according to what is implied above, the doses would certainly be lower than the current results. However, the dose reduction is probably not significant. Note the ratio of about 1.6 between the two plots from about one day to about 20 days (480 hours), and then the increase to 2.5 by 30 days (720 hours). These are the ratios that were used to increase the 10% iodine re-evolution fraction for ESF leakage. So the greater degree of precision resulting from applying the time-dependent DF concept would have no effect on the ESF leakage analysis.

Also, 0.15% of the iodine remains airborne as organic, so when the DF of 60 is applied to the elemental, the 4.85% elemental is reduced to 0.08%. The organic is still twice as great. If the DF is tracked, especially over the first eight hours when control room χ/Qs are high, there would certainly be some improvement with respect to the 0.08%. However, during approximately the first two hours of that eight hours, the DF is not an issue because there is still a source, and from two to four hours, the time-dependent DF is still in the range of 200 (i.e., about 0.025% elemental airborne). So for the case of a DF of 60, the first two hours of the important eight-hour period would have a gaseous iodine airborne fraction of about 0.3% (elemental is equivalent to organic with a source still present), and the next six hours would have a total gaseous iodine airborne fraction of about 0.23% (using an integrated percent fraction versus time metric to gauge importance yields a total of 2hr *0.3% + 6hr * 0.23% = 2%-hour).

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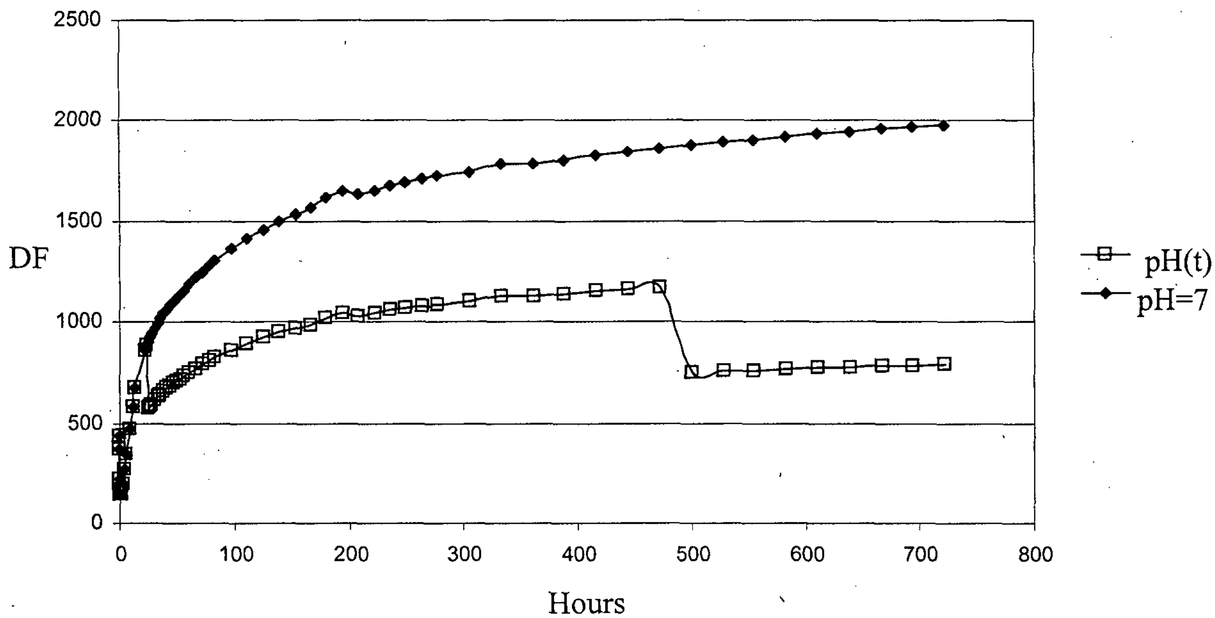
For the case of the transient DF, the first two hours would be the same, the next two hours would have about 0.175% gaseous iodine airborne, and the last four hours would have about 0.15% gaseous iodine airborne (i.e., essentially organic only) (for a total of about 1.6 %-hour). Therefore, the transient DF would decrease the gaseous iodine containment leakage control room dose contribution by about a factor of 1.25, and would not decrease the gaseous iodine ESF leakage contribution. Overall, it is estimated that the control room dose would be reduced by less than 0.1 rem TEDE if the transient DF were modeled.

Using a transient DF is too complex, considering the slight reduction in doses; therefore a DF of 60 was selected.

Figure 4.3-3: Iodine Decontamination Factor in the Containment Sump



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4.3.3.2 Radiological Releases from ESF Equipment

ECCS leakage is controlled in accordance with Technical Specification 6.8.3.a, "Primary Coolant Sources Outside Containment." As described in UFSAR Table 15.6-12, "Maximum Potential Recirculation Loop Leakage External to Containment," the maximum permitted recirculation loop leakage (i.e., ECCS leakage) is 4,140 cc/hr. These values are assumed in the CLB analysis of dose assessment to the Control Room, with a safety factor of two applied in accordance with Regulatory Guide 1.183. The ECCS leakage rate assumed in the AST LOCA analysis is the same as that for the CLB.

For determination of the dose contribution from ESF leakage, all radionuclides assumed to be released from the core (except noble gases) are assumed to be instantaneously and homogeneously mixed in the containment sump. Actual leakage from the RCS sump through ESF equipment would not start until after the recirculation phase of the accident begins. However, for conservatism, and to decouple the dose analyses from the actual calculated recirculation start time, ESF leakage is assumed to begin at t=0.

Because the pH of the containment sump falls below 7.0 after one day, a fractional iodine release for ESF leakage greater than 10% must be considered per Regulatory Guide 1.183. A discussion of the pH used for iodine re-evolution from ESF leakage was presented in the preceding section 4.3.3.1.2.2. Using the relative DF as an indication of iodine volatility, a release fraction of 16% of the iodine in the ESF leakage is assumed to be released for pH = 6.9 (24 hours to 480 hours) and 25% for pH = 6.8 (480 hours to 720 hours).

This leakage is released directly into the Fuel Handling Building (FHB) from the leaked reactor coolant. It is then assumed to be released instantaneously into the environment without benefit of filtration via the FHB Exhaust Air System units.

A calculation was performed to determine the impact of a substantially degraded leakage condition (i.e., degraded condition leak rates assumed to be approximately 10 times greater than the design leak rate) for the ECCS isolation valves, thus allowing a greater than design leakage to migrate back to the Refueling Water Storage Tank (RWST). The leakage values used in the analysis ranged from 180 to 480 cc/hr. The RWST suction line isolation valves, low head/high head safety injection pumps' recirculation line isolation valves, and the containment spray pump's test line isolation valves for the three safety trains are considered in this analysis. The analysis concluded:

1. The motive force for leakage in the containment sump suction line is the high pressure in the containment resulting from the large break LOCA. This pressure is reduced, within the first 3.36 hours of an accident, below a pressure capable of forcing water into the RWST. No contaminated sump water will reach the RWST via this leak path.

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2. The containment spray pumps may be secured up to 13.4 days after initiation of a DBA LOCA and containment water will not reach the RWST via this leak path.

3. The minimum time for leakage from valves assumed to have the degraded leak rate to reach the RWST following the initiation or the recirculation phase of the DBA LOCA is 44.1 days. At this point in time the leakage into the RWST would be 1200 cc/hr.

Therefore, this potential ECCS leakage path to the RWST does not impact the LOCA analyses results.

The surveillance criteria for ESF leakage outside containment accounts for accident leakage. During normal operations and ECCS testing, leakage is at room temperature. The total ESF leakage for the unit is compared to the 4140 cc/hr limit used in the LOCA analysis. During an accident, ESF leakage would be at a maximum temperature of 212°F. Also, at the time the injection phase of the accident ended and the recirculation phase begins (minimum of 1000 seconds into the accident, UFSAR Table 6.2.1.1-10, Revision 13), the containment building pressure would have dropped from its peak pressure of about 42 psig to about 28 psig. The LOCA analysis assumes the leakage is at room temperature.

The ESF leakage surveillance is part of the STP Contaminated System Leakage Program. During the surveillance, the ESF leakage is room temperature and under a static head from the RWST and possibly an additional dynamic head from SI pump operation (for leakage from mini-flow valves and isolation valves). Under accident conditions, the fluid will be at 212°F and under a diminishing static head from the RWST during the injection phase and a static head from the RCB sump and RCB pressure during the recirculation phase. During both the injection and recirculation phases some leakage will also be under an additional dynamic head from SI pump operation.

To correct surveillance results to accident conditions, a correction factor will be used. The analysis used a constant leak rate of 8280 cc/hr for 30 days (i.e., per Regulatory Guide 1.183, two times the sum of the simultaneous leakage limit from all components in the ESF recirculation systems established by Technical Specifications program requirement 6.8.3.a). A correction factor that considers the time-dependent accident pressure in the RCB plus the RCB emergency sump head to the RWST head during test conditions will be used to correct surveillance conditions to accident conditions.

4.3.4 Radiological Dose Models

The RADTRAD 3.03 code was used to calculate the immersion and inhalation dose contributions for both the onsite and the offsite radiological dose consequences. Eight models were needed; four for the Control Room dose analysis and four for the TSC dose analysis. The offsite doses generated for each set are identical. The RADTRAD models for DBA-LOCA are graphically presented on Figures 4.3-4 through 4.3-11. The Case 1 series is for Control Room dose consequence analysis (as well as for offsite), and the Case 2 series is for TSC dose

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consequence analysis (as well as for offsite). Table 4.3-4 describes the RADTRAD cases constructed for the LOCA analyses. All of these pathways are part of the CLB with the exception of the electrical penetration pathway.

Table 4.3-4
RADTRAD Models for LOCA

Case Designators	Figure	Purpose
1	4.3-4	General Containment Leakage
2	4.3-8	
1pen	4.3-5	Containment leakage into the electrical penetration area
2pen	4.3-9	(primarily to obtain airborne activity for Control Room shine as discussed below)
1esf	4.3-6	ESF leakage into the fuel handling building (FHB) with no
2esf	4.3-10	hold-up or filtration of the re-evolved iodine release
1pur	4.3-7	Initial containment purge flow path dose contribution
2pur	4.3-11	

In all of these models, the same basic structure is used. In order to set up a model for a specific pathway, certain junctions are actuated (solid lines) and certain junctions are closed (dashed lines). The key junctions which constitute the release path to the environment or from the environment to the Control Room/TSC are shown in heavy solid lines. For a given model graphic, a certain control volume (CV) or junction as shown may represent one of two actual junctions or CVs. The CV or junction ID for the CV or junction applicable to the model in question is "boxed" to show that applicability. The "intermediate" CV represents the electrical penetration area for Cases 1/2 and for Cases 1/2pen and the FHB for Cases 1/2esf. It is ignored for Cases 1/2pur. The applicable χ/Q set is also identified for each model.

The computer code STARDOSE was used to check the RADTRAD results for the DBA LOCA. The RADTRAD and STARDOSE programs are radiological consequence analysis codes used to determine post-accident doses at offsite and control room locations due to immersion and inhalation. The STARDOSE code is the proprietary property of Polestar Applied Technology, Inc.

Figure 4.3-4 - RADTRAD Model for Case 1
Heavy lines = active pathways to environment or CR/TSC, dashed lines = closed pathways, boxed text = active control volume/junction where multiples shown

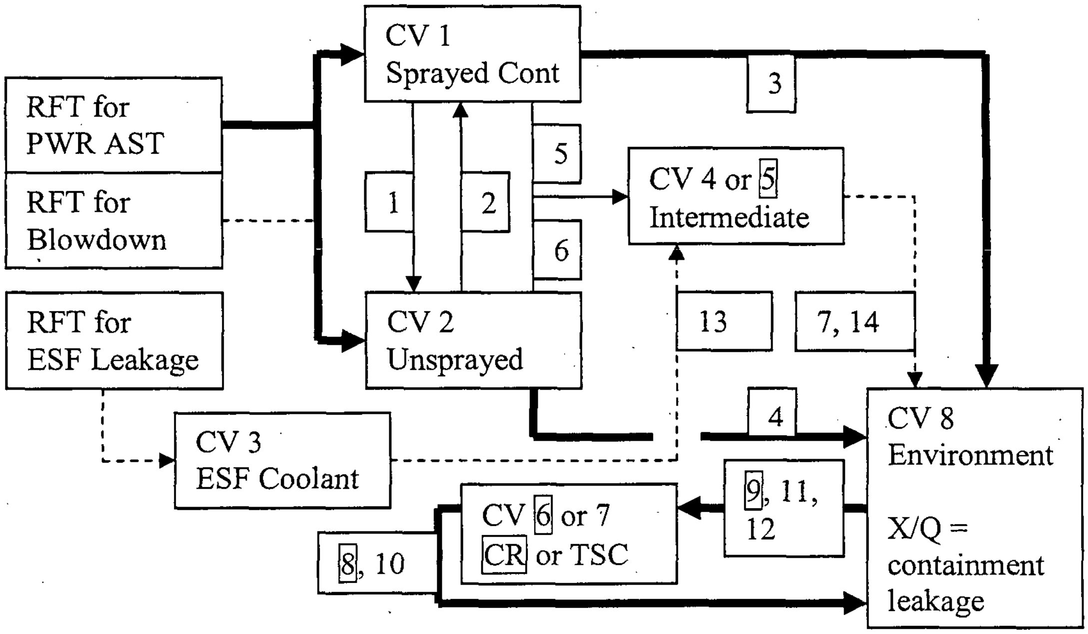


Figure 4.3-5 - RADTRAD Model for Case 1pen
Heavy lines = active pathways to environment or CR/TSC, dashed lines = closed pathways, boxed text = active control volume/junction where multiples shown

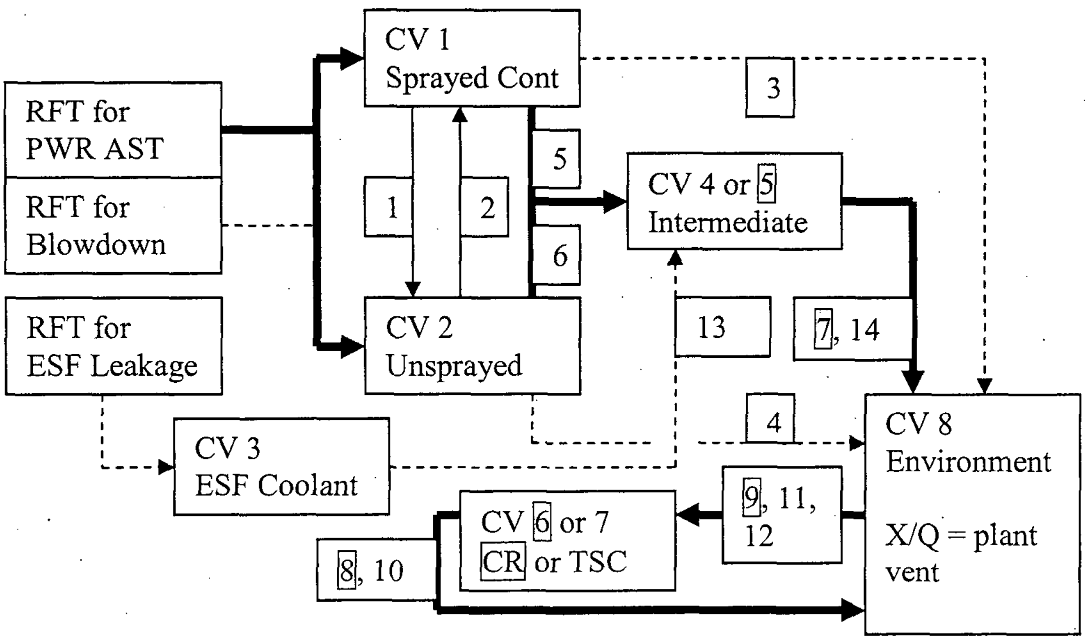


Figure 4.3-6 - RADTRAD Model for Case 1esf
Heavy lines = active pathways to environment or CR/TSC, dashed lines = closed pathways, boxed text = active control volume/junction where multiples shown

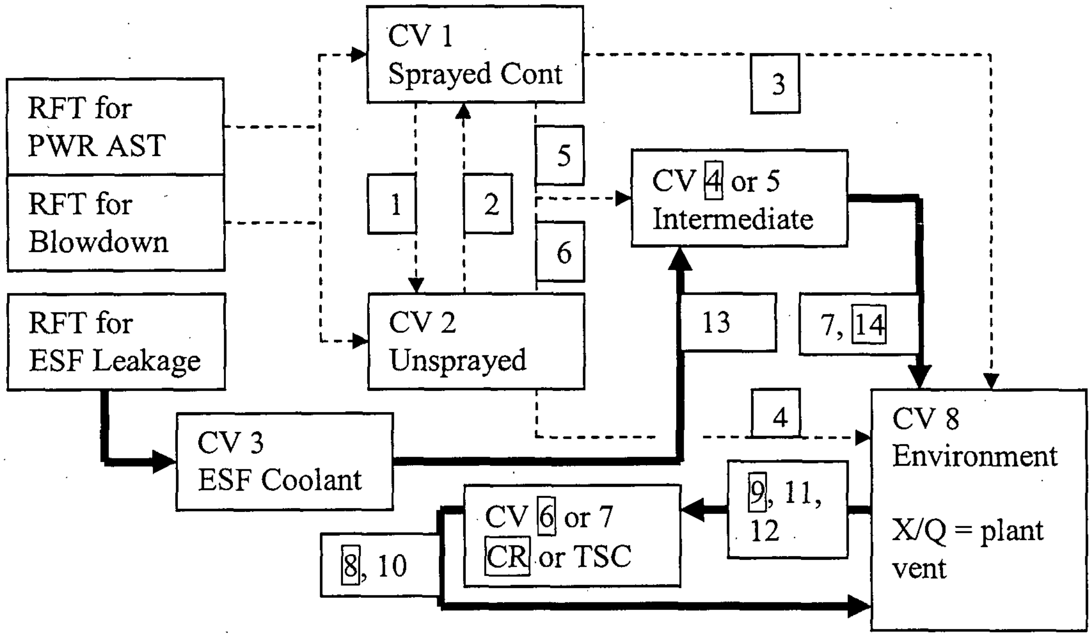


Figure 4.3-7 - RADTRAD Model for Case 1pur
Heavy lines = active pathways to environment or CR/TSC, dashed lines = closed pathways, boxed text = active control volume/junction where multiples shown

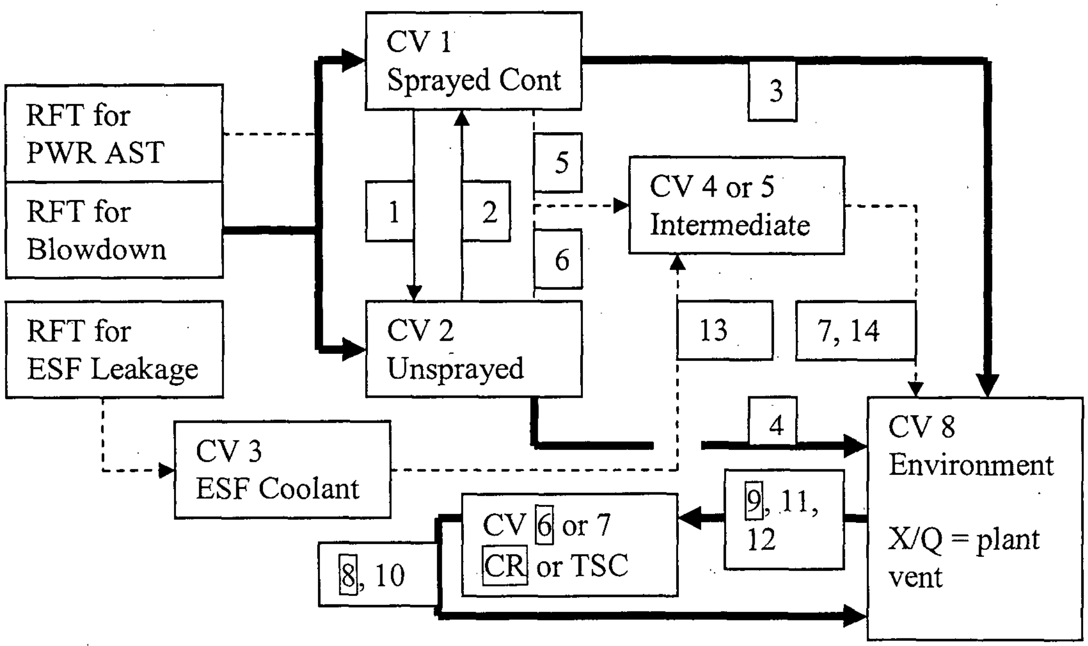


Figure 4.3-8 - RADTRAD Model for Case 2
Heavy lines = active pathways to environment or CR/TSC, dashed lines = closed pathways, boxed text = active control volume/junction where multiples shown

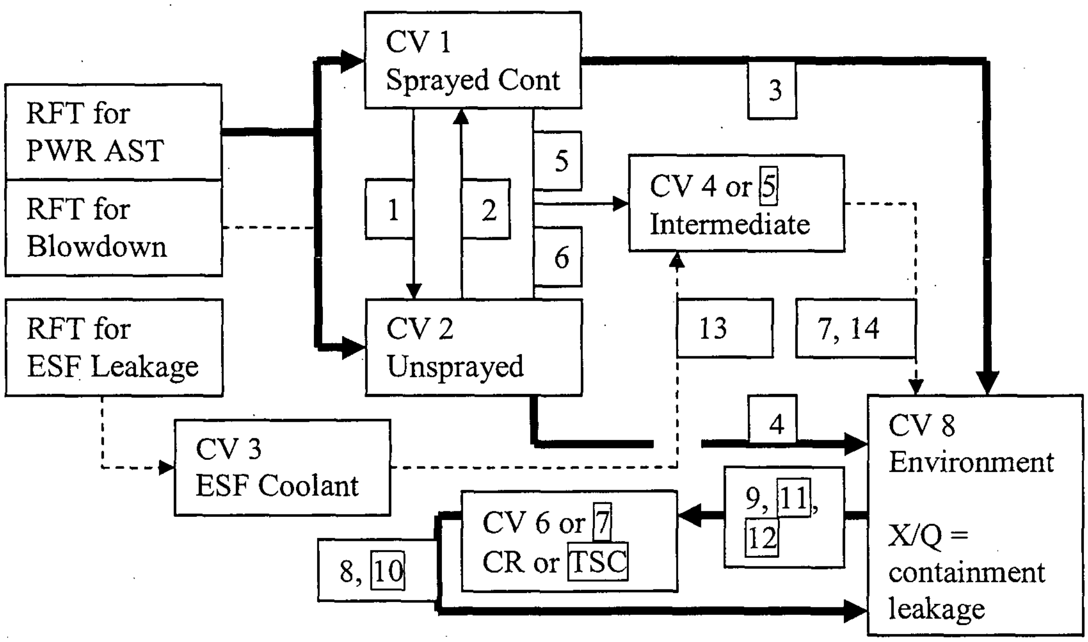


Figure 4.3-9 - RADTRAD Model for Case 2pen
Heavy lines = active pathways to environment or CR/TSC, dashed lines = closed pathways, boxed text = active control volume/junction where multiples shown

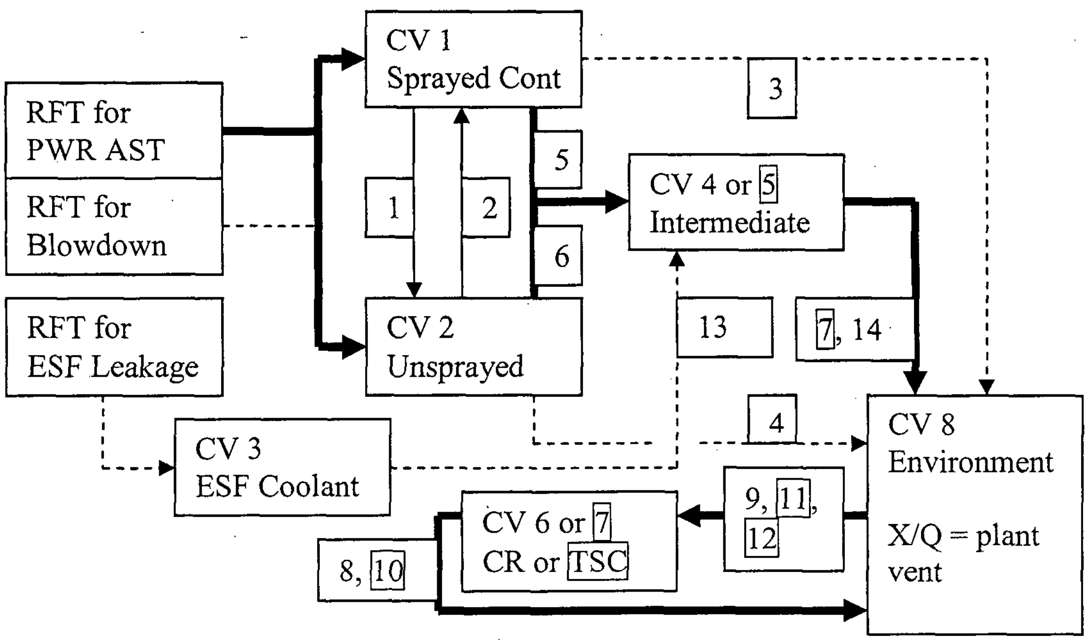


Figure 4.3-10 - RADTRAD Model for Case 2esf
Heavy lines = active pathways to environment or CR/TSC, dashed lines = closed pathways, boxed text = active control volume/junction where multiples shown

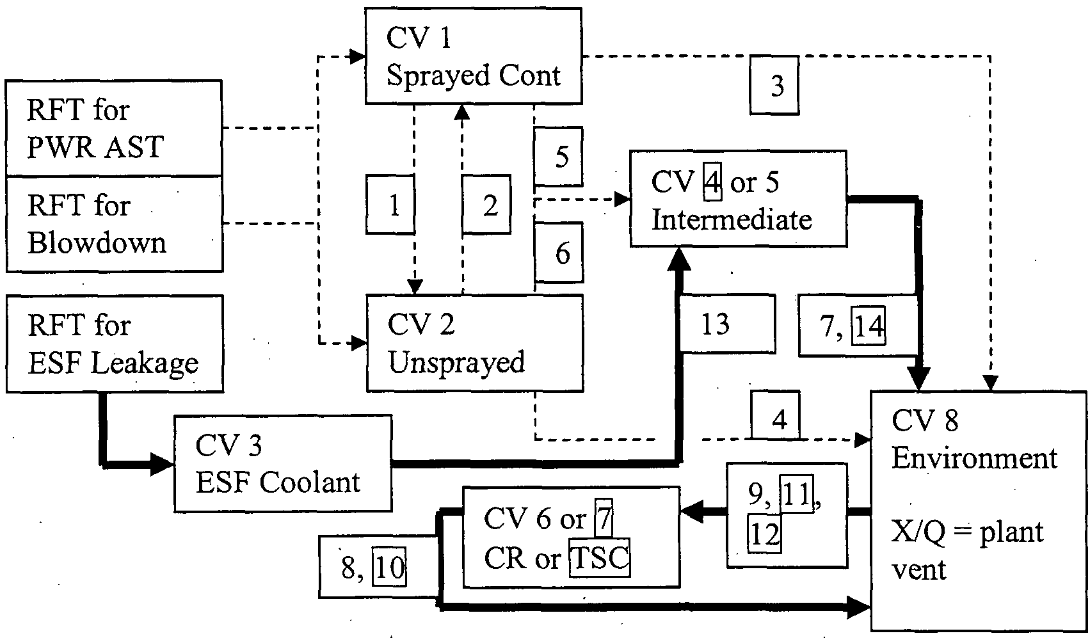
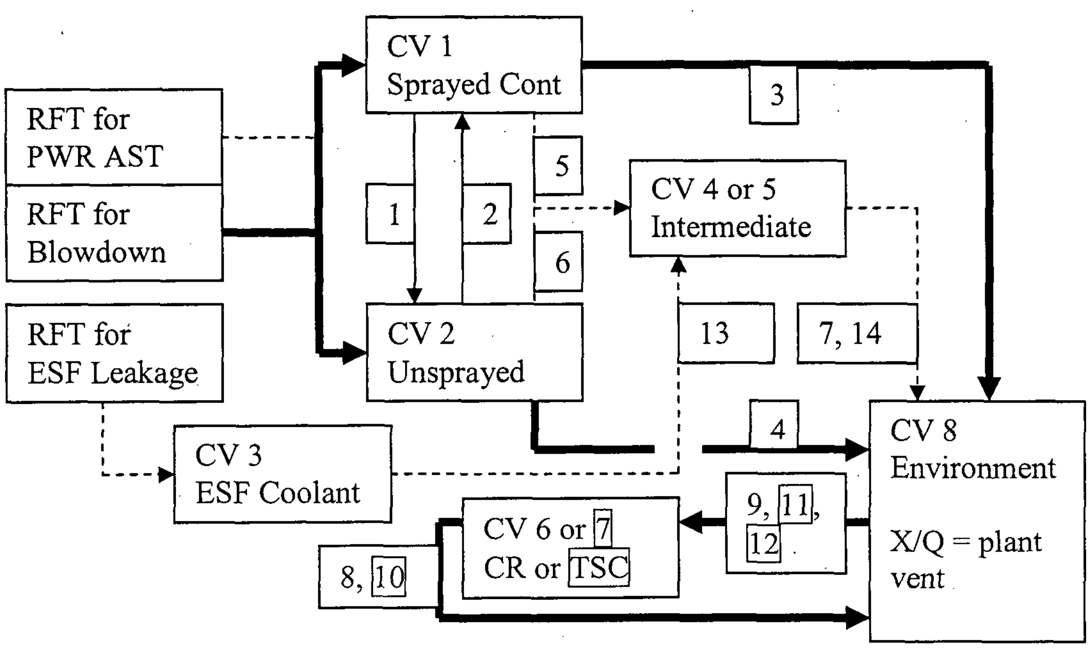


Figure 4.3-11 - RADTRAD Model for Case 2pur
Heavy lines = active pathways to environment or CR/TSC, dashed lines = closed pathways, boxed text = active control volume/junction where multiples shown



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4.3.4.1 Control Room and TSC

Dose to operators in the Control Room and to TSC personnel are from two main pathways:

- Dose from airborne contaminants in the Control Room/TSC
- Dose from gamma sources outside the Control Room/TSC

These sources are discussed in the following sections.

Consideration of Single Failure

Without credit being taken for the FHB filters or for the Control Room make-up filters (and the associated heaters to control intake humidity), the single-failure assessment becomes much simpler for application of the AST than that of the CLB. For the AST DBA LOCA, an electrical division electrical failure is assumed as a single failure to minimize containment mixing via the containment fan-coolers. This assumption maximizes dose. Only two out of three trains of containment ventilation are assumed to operate, and one reactor containment fan-cooler on one of the operating trains is assumed to be out of service, as well. The spray removal lambdas used are also consistent with the loss of one spray train, as are the assumptions regarding Control Room ventilation and filtration.

4.3.4.1.1 CR/TSC Doses from Airborne Contaminants

The analytical models used for the Control Room and TSC are described in Sections 4.2.2 and 4.2.3, respectively. Releases are assumed to be drawn into the Control Room/TSC HVAC intakes (points D/H on Figure 4.1-13). The atmospheric dispersion factors are developed in Section 4.1.3. Unfiltered in-leakage and possible "sneak" paths into the CRE are addressed in Section 4.2.2.1.

4.3.4.2 CR/TSC Doses from Gamma Shine

The gamma shine dose contribution consists of four parts:

- Gamma shine from the containment airborne activity
- Gamma shine from airborne activity in the electrical penetration area (CR only)
- Gamma shine from activity in the external radioactive cloud surrounding the plant structures
- Gamma shine from trapped activity on filters

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The DBA LOCA radiation dose to personnel in the Control Room includes the gamma shine from the primary containment airborne activity, from airborne activity in the electrical penetration area, from activity in the radioactive cloud surrounding the plant structures, and from trapped activity on filters. Of these four contributors, all but the shine from the electrical penetration area are in the CLB. A tabulation of the dose components for the gamma shine doses is presented in Table 4.3-9.

Gamma shine from the containment airborne activity
For shine from the containment, a comparison of source gamma power was made to either justify the use of the calculated dose as-is or to adjust the CLB value for AST application.

It was determined that the time-integrated activity of radioiodine airborne in the containment would be approximately an order of magnitude lower for the AST than for the STP CLB. The airborne noble gas is comparable for both the AST and the STP CLB. While the AST involves the airborne release of significant quantities of additional non-iodine activity in particulate form, this activity is readily removed by filtration and plate-out. The external gamma dose from non-iodine airborne particulate for the AST is only 10% of that for the iodine. Because of this behavior, it is evident that radiation from the containment will be less for the AST than for the STP CLB.

The STP Control Room CLB shine dose due to activity airborne in the containment is 0.101 rem. By a comparison of the basis for the 0.101 rem (in terms of transient airborne activity) to the transient airborne activity for the AST, this value was determined to be bounding. Therefore, it has been used as-is for the AST application as a dose increment for the Control Room dose consequence analysis. In a similar fashion, the CLB shine dose to the TSC due to the RCB airborne activity of 0.004 rem is bounding for the AST analysis.

Gamma shine from airborne activity in the electrical penetration area
The Electrical Penetration area is directly between the Control Room Envelope and the containment building (on the bottom of the Control Room Envelope, west of the Relay Room and Computer Room, as depicted in Figure 4.2-1). This exposure source was not considered in the CLB.

For Control Room shine from the electrical penetration area just outside containment, a compartment was added to the plant model for both the RADTRAD and the STARDOSE DBA LOCA analyses. The maximum post-LOCA containment temperature and pressure listed in Table 4.3-11 were used to convert the electrical penetration tested mass leak rate (expressed in sccm) to a volumetric leak rate for use in the dose analysis model. The transient airborne activity within this compartment was calculated using these models. From this transient airborne activity, a dose calculation was performed for shine dose in the Control Room. This calculation was performed using the MicroShield code (Reference 40). MicroShield is a point kernel integration code used for general-purpose gamma shielding analysis. The dose contribution from

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this source is 0.0174 rem. This value has been used for the AST application as a dose increment for the Control Room dose consequence analysis.

Gamma shine from activity in the external cloud surrounding the plant structures
For shine from the radioactive cloud, the 30-day shine dose increment for the LPZ (no shielding protection considered and no occupancy factor credited) was adjusted by the ratio of the maximum onsite χ/Q value to that for the LPZ for each χ/Q averaging period. The result was then reduced by a shielding attenuation factors for the Control Room and TSC (Table 4.3-5). The final shine dose was obtained by adding the increments for each averaging period. The dose contribution from this source is 0.014 rem to the Control Room and 0.212 rem to the TSC.

Gamma shine from trapped activity on filters
Similar to the treatment of the shine from the containment, a comparison of source gamma power was made to either justify the use of the calculated dose as-is or to adjust the CLB value for AST application.

It was determined that the time-integrated activity of radioiodine airborne in the containment would be approximately an order of magnitude lower for the AST than for the STP CLB. The airborne noble gas is comparable for both the AST and the STP CLB. While the AST involves the airborne release of significant quantities of additional non-iodine activity in particulate form, this activity is readily removed by filtration and plate-out. The external gamma dose from non-iodine airborne particulate for the AST is only about 10% of that for the iodine. Because of this behavior, it is evident that radiation from the activity trapped on filters will be less for the AST than for the STP CLB.

This source of exposure is part of the CLB for both the Control Room and the TSC. Three of the filter shine contributions assessed as part of the CLB are negligible: (1) Control Room shine dose due to Control Room make-up filters, (2) Control Room shine dose due to TSC make-up/recirculation clean-up filter, and (3) TSC shine dose due to TSC make-up/recirculation clean-up filter. The Control Room shine dose due to the Control Room recirculation make-up filters is 0.00218 rem, and the TSC shine dose due to the Control Room make-up filters is 0.844 rem.

The CLB activity trapped on one Control Room make-up filter is shown on Table 4.3-6. The gamma power due to activity on two Control Room make-up filters (the basis for the CLB TSC shine dose contribution) is shown on Table 4.3-7.

For the CLB, it is assumed that the activity trapped on the Control Room recirculation clean-up filters is one-tenth of that on the Control Room make-up filters (i.e., one-tenth of Table 4.3-6) due to the iodine removal by the makeup filters. The CLB Control Room shine dose due to the Control Room recirculation clean-up filters is based on one-tenth of Table 4.3-7.

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The AST gamma power due to activity trapped on two Control Room make-up filters is presented in Table 4.3-7, and the AST gamma power due to activity trapped on two Control Room recirculation clean-up filters is presented in Table 4.3-8.

Note that in the AST analysis, the Control Room inhalation and immersion doses were calculated without benefit of the Control Room make-up filters. The Control Room recirculation filter loading was taken from Table 4.3-11. A separate calculation for the Control Room make-up filter loading was done solely for the purpose of evaluating the gamma shine contribution to the TSC. This second analysis assumed 100% filter efficiency for the Control Room make-up filters.

As can be seen from Table 4.3-7, the CLB Control Room make-up filter loading is about an order of magnitude greater than that for the AST. On this basis, the CLB 0.844 rem Control Room make-up filter shine dose contribution to the TSC has been used as-is for the AST application as a dose increment for the TSC dose consequence analysis.

As can be seen from a comparison of one-tenth of Tables 4.3-7 and 4.3-8, the maximum ratio of AST Control Room recirculation clean-up filter loading to the CLB Control Room recirculation clean-up filter loading is 1.74 at 720 hours. On this basis, the CLB 0.00218 rem Control Room recirculation clean-up filter shine dose contribution to the Control Room has been increased by a factor of 1.74 to 0.0038 rem for the AST application as a dose increment for the Control Room dose consequence analysis.

Table 4.3-5
CR and TSC Gamma Shine Dose Analysis Inputs for DBA LOCA

Input/Assumption	CLB Analysis	AST Analysis
Attenuation factor for Control Room shine from atmospheric activity	1.03E-3	1.03E-3
Attenuation factor for TSC shine from atmospheric activity	1.03E-3	1.56E-2
Activity on one Control Room make-up filter	Table 4.3-6	N/A
Gamma power due to activity on two Control Room make-up filters	Table 4.3-7	Table 4.3-7

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Table 4.3-5
CR and TSC Gamma Shine Dose Analysis Inputs for DBA LOCA

Input/Assumption	CLB Analysis	AST Analysis
Gamma power by photon energy due to activity on Control Room recirculation clean-up filters	One-tenth of Table 4.3-7	One-tenth of Table 4.3-7
AST gamma power by photon energy due to activity on Control Room recirculation clean-up filters	N/A	Table 4.3-8
Maximum ratio of one-tenth of Table 4.3-7 to Table 4.3-8	N/A	1.74 at 720 hours

Table 4.3-6
Current Licensing Basis Activity on One Control Room Make-Up Filter
(Ci)

Time (hours)	Radionuclide				
	I-131	I-132	I-133	I-134	I-135
0.05	8.76E-2	1.26E-1	1.88E-1	1.96E-1	1.71E-1
0.7882	4.41E-1	5.10E-1	9.23E-1	5.51E-1	8.00E-1
8	1.65	2.19E-1	2.78	6.96E-3	1.44
24	3.24	3.57E-3	3.41	-	5.61E-1
44	3.63	-	2.10	-	8.31E-2
64	3.97	-	1.28	-	1.19E-2
84	4.22	-	7.54E-1	-	1.68E-3
96	4.36	-	5.45E-1	-	5.14E-4
150	4.07	-	1.04E-1	-	-
400	2.58	-	-	-	-
720	1.18	-	-	-	-

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Table 4.3-7
Gamma Power from Activity on Two Control Room Make-Up Filters
(MeV/sec)

Time (hours)	CLB	AST	% Difference
0.05	9.03E+10	2.78E+09	-96.9%
0.7882	3.41E+11	2.16E+10	-93.7%
8	3.8E+11	4.84E+10	-87.3%
24	3.1E+11	2.82E+10	-90.9%
44	2.1E+11	2.21E+10	-89.5%
64	1.7E+11	1.97E+10	-88.4%
84	1.5E+11	1.84E+10	-87.7%
96	1.5E+11	1.79E+10	-88.1%
150	1.2E+11	1.58E+10	-86.8%
400	7.3E+10	1.11E+10	-84.8%
720	3.3E+10	7.39E+09	-77.6%

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Table 4.3-8
Gamma Power from Activity on Two Control Room Cleanup Filters
(Mev/sec)

Time (hours)	CLB	AST	% Difference
0.05	4.14E+10	2.19E+08	-99.5%
0.7882	1.57E+11	7.54E+09	-95.2%
8	1.76E+11	3.66E+10	-79.2%
24	1.45E+11	2.12E+10	-85.4%
44	9.68E+10	1.65E+10	-82.9%
64	8.02E+10	1.48E+10	-81.5%
84	7.23E+10	1.39E+10	-80.8%
96	6.96E+10	1.36E+10	-80.5%
150	5.67E+10	1.21E+10	-78.7%
400	3.45E+10	8.66E+09	-74.9%
720	1.58E+10	5.79E+09	-63.4%

Table 4.3-9
Gamma Shine Component Doses
(rem)

Source	Receptor	CLB	AST
RCB Activity	CR	0.101	0.101
External Cloud	CR	0.464	0.014
CR Makeup Filters	CR	Negligible	Negligible
CR Recirc Filters	CR	0.00218	0.004
Electrical Penetration Room	CR	N/A	0.017
	TOTAL	0.567	0.136
RCB Activity	TSC	0.004	0.004
External Cloud	TSC	0.464	0.212
CR Makeup Filters	TSC	0.844	0.844
TSC Makeup/Recirc Filters	TSC	Negligible	Negligible
Electrical Penetration Room	TSC	N/A	Negligible
	TOTAL	1.312	1.060

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4.3.5 Inputs and Assumptions

The following assumptions are made in the LOCA analyses.

Release into Containment

1. The source term is based upon a power level of 4100 MW thermal, 5 w/o enrichment, and a three region core with equilibrium cycle core at end of life. The three regions have operated at a specific power of 39.3 MW/MTU for 509, 1018, and 1527 EFPD, respectively. The assumed power level is greater than the Rated Thermal Power of 3853 MWth plus a 0.6% measurement uncertainty. The AST requires the consideration of additional radionuclides to ensure that the TEDE dose (which considers organs other than thyroid) is properly calculated. For the DBA LOCA, all fuel assemblies in the core are assumed to be affected, and the core average inventory is used.

2. A total of 100 percent of the core noble gas inventory and 50 percent of the core iodine inventory is assumed to be immediately available for leakage from the Containment.

3. For the AST, in accordance with Regulatory Guide 1.183, of the radioiodine released from the reactor core, 95 percent of the iodine released is assumed to be particulate in the form of cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. This includes releases from the gap and the fuel pellets. For the CLB, in accordance with Regulatory Guide 1.4 (Reference 41), of the iodine activity released to the Containment, it is assumed that 95.5 percent is in the elemental form, 2 percent is in the organic or methyl iodine from, and 2.5 percent is in particulate form.

4. The radioactivity released from the fuel is assumed to mix instantaneously and homogeneously throughout the containment air space as it is released. The distribution is not adjusted for internal compartment effects.

Release via Containment Supplemental Purge

5. The Containment Supplementary Purge System is assumed to be in operation and the purge is assumed to be isolated within 23 seconds of the generation of the Containment Pressure-High 1 signal. This includes the signal and sequencer delays, Standby Diesel Generator startup time, and the valve closing time. This time dose not include the 1.2 seconds between the postulated instantaneous break and the containment pressure reaching the High-1 setpoint. However, the constant value used for the choke flow through the ventilation system bounds the effect of neglecting these 1.2 seconds. During normal power operation, the Containment Supplementary Purge System vents the containment at 4,500 ft^3/min. However, for this analysis, the maximum flow rate due to the pressure spike inside the Containment was used (83,200 ft^3/min for each purge line, intake and exhaust).

6. The coolant activity in the AST analysis does not include iodine spiking (per the guidance of Regulatory Guide 1.183, Appendix A, Section 3.8). However, the RCS

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concentrations are based on 1% failed fuel, which are greater than those corresponding to the 1.0 μCi/gm Technical Specification limit on DE I-131. The CLB assumes the Containment airborne iodine inventory available for release is the flashed portion of the total primary coolant iodine inventory based upon a preexisting iodine spike level of 60 μCi/g dose equivalent I-131. For noble gases, 100 percent of the primary coolant inventory based upon 1% failed fuel is assumed to be available for release. In both cases, no failed fuel due to the accident is assumed to have occurred because isolation occurs prior to the core reaching a temperature that could cause a fuel failure.

Fission Product Removal Inside Containment

7. A two-volume model of the Containment is used to represent sprayed and unsprayed regions of the Containment.

8. Of the six Reactor Containment Fan Cooler (RCFC) units, only three are assumed to function. Two are failed due to the assumed failure of a Standby Diesel Generator to start upon Loss of Offsite Power and one is assumed down for maintenance.

9. The transfer rate between the sprayed and unsprayed regions is assumed to be limited to the forced convection induced by the RCFC units. The assumed minimum flow rate conservatively neglects the effects of natural convection, steam condensation, and diffusion, although these effects are expected to enhance the mixing rate between the sprayed and unsprayed volumes. The majority of the RCFC air supply, except a small portion discharged to the dome, is discharged to the space within the secondary shield wall, where it is relieved to the balance of the Containment volume through the vent areas.

10. For fission products other than iodine, the only removal processes considered are radioactive decay and leakage. Iodine is assumed to be removed by radioactive decay and leakage, plateout, and also by the Containment Spray System (CSS).

11. The AST analyses change the ultimate DF to 60 instead of a value of 100 as used in the CLB. This change is based on pH decreasing below 7.0 at 24 hours.

12. Since the AST elemental iodine activity release to the containment is different in magnitude and timing as compared to the CLB, the elemental iodine DF of 60 is reached at a different time. Also, once the elemental iodine DF of 60 is reached, all elemental iodine removal, both natural removal and removal by spray, is terminated. For the CLB, a spray removal rate of 20 per hour is assumed until the airborne elemental iodine is reduced by a factor of 60. After this time, the elemental spray removal rate is assumed to be zero.

13. For the AST, the natural removal rate in the containment for elemental iodine is changed to 4.5 per hour. This is due to the application of an ultimate DF of 60 (instead of 100) based on the sump pH decreasing below 7.0 after 24 hours. The CLB uses 3.59 per hour

4.3 LOSS OF COOLANT ACCIDENT

for the sprayed region and 0.91 per hour for the unsprayed region. This is based on partitioning the 4.50 per hour total removal rate between the regions based on relative volumes. However, if it is assumed that the volumes have the same surface area to volume ratio, then 4.5 may be used for both volumes. For the CLB, the deposition removal rate for elemental iodine is assumed to be 4.5 per hour which is reduced to 5% of this value once a DF of 100 is reached and no additional credit is taken for deposition after a DF of 200 is reached.

14. For the CLB, for particulate iodine, a spray removal rate of 6.9 per hour is assumed until a DF of 50 is reached and it is then reduced to 10% of this value until a DF of 1000 is reached.

15. Since the AST particulate activity release to the containment is different in magnitude and timing as compared to the CLB, the particulate DF of 50 (the time at which the spray removal rate is reduced by a factor of ten) is reached at a different time. For the CLB, the analysis assumes containment spray for 380 minutes for removal of particulate iodine.

Release from Containment

16. The Containment leak rate to the atmosphere used in the analysis is the design-basis leak rate indicated in the Technical Specifications. For the first 24 hours following the accident, the leak rate is assumed to be 0.30 percent per day, while for the remainder of the 30-day period the leak rate is assumed to be 0.15 percent per day. This Containment leakage is assumed to leak directly to the environment.

17. To support the revised analysis consideration of the gamma "shine" dose to the Control Room, leakage from the containment into the adjacent electrical penetration room is assumed based on the relative number of penetrations in the penetration area.

ESF Leakage Release

18. The amount of water in the Containment sumps at the start of recirculation is the total of the RCS water and the water added due to operation of the engineered safeguards, i.e., the ECCS and CSS. This amount has been calculated to be 512,494 gallons. This value is conservatively low to maximize iodine concentration in the sump water.

19. Since most of the radioiodine released during the LOCA would be retained by the Containment sump water due to operation of the CSS and the ECCS, it is conservatively assumed that 50 percent of the core iodine inventory is introduced to the sump water to be recirculated through the external piping systems. Because noble gases are assumed to be available for leakage from the Containment atmosphere and are not readily entrained in water, the noble gases are not assumed to be part of the source term of this contribution to the total LOCA dose.

20. For the fractional release of iodine from the ESF leakage, the Regulatory Guide 1.183 recommended value of 10% is used only for the first 24 hours. Beyond that time, the

4.3 LOSS OF COOLANT ACCIDENT

release fraction is increased to 16% (24 hours to 480 hours) and then to 25% (480 hours to 720 hours) based on the calculated volatility of the iodine for the pH values over those intervals relative to the volatility for a pH of 7.0. The CLB assumes that 10% of the iodine is released into the Fuel Handling Building.

21. The maximum potential recirculation loop leakage is 4140 cc/hr. This value represents expected leakages from ESF equipment and is the total leakage from all three trains of ESF equipment. The radiological dose model does not distinguish between the specific source, component, or train of the ESF leakage. The radiological dose model conservatively uses twice the total leakage.

22. The iodine activity released from the ESF leakage, once released to the atmosphere of the FHB, is assumed to be quickly transported by the ventilation system through the exhaust filters and released to the environment at ground level. The AST analysis assumes no filtration of the iodines released to the environment. The CLB assumes the iodine filtration efficiency to be 95 percent.

Control Room HVAC

23. The Control Room ventilation system is assumed to automatically transfer to the emergency mode of operation after the initiation of safety injection.

24. The AST analyses use the nominal Control Room HVAC flow rates plus uncertainties to more conservatively model the Control Room HVAC system. The Control Room make-up flow is increased from nominal 2000 cfm to 2200 cfm to allow for tolerances, and the Control Room recirculation flow is decreased from 9500 cfm to 8600 cfm to allow for tolerances. The Control Room make-up filter is conservatively ignored except for determining the filter shine dose to the TSC. 100 cfm of unfiltered in-leakage is assumed in addition to the 2200 cfm of make-up flow that is assumed to experience no filtration at all. This yields a total of 2300 cfm of unfiltered in-leakage.

Miscellaneous

25. Offsite Power is lost. One Standby Diesel Generator fails to start. This causes the loss of one train of Control Room emergency HVAC and the loss of one train of RCFC's (two RCFC units).

26. For determination of offsite doses, all activity is released to the environment with no consideration given to cloud depletion by ground deposition during transport to the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ).

Input parameters used for the LOCA analysis are given in Table 4.3-11. Conformance with Regulatory Guide 1.183 guidance addressing LOCA analysis is provided in Attachment 6, Tables A and B.

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Table 4.3-10
LOCA Time-Dependent Release Fractions

Time Period (sec)		Fraction of core inventory[37]
0 - 30		No Release
30 - 1830	Gases	Xe, Kr – 0.1/hr (0.05 total) Elemental I – 4.9E-3/hr (2.4E-3 total) Organic I – 1.5E-4/hr (7.5E-5 total)
	Aerosols	I, Br – 0.095/hr (0.0475 total) Cs, Rb – 0.1/hr (0.05 total)
1830 - 6510	Gases	Xe, Kr – 0.73/hr (0.95 total) Elemental I – 1.3E-2/hr (1.7E-2 total) Organic I – 4.0E-4/hr (5.3E-4 total)
	Aerosols	I, Br – 0.256/hr (0.3325 total) Cs, Rb – 0.192/hr (0.25 total) Te Group – 0.038/hr (0.05 total) Ba, Sr – 0.015/hr (0.02 total) Noble Metals – 1.9E-3/hr (2.5E-3 total) La Group – 1.5E-4/hr (2E-4 total) Ce Group – 3.8E-4/hr (5E-4 total)

[37] From RG 1.183 Table 2 considering the chemical form described in RG 1.183, Section 3.5.

4.3 LOSS OF COOLANT ACCIDENT

Table 4.3-11
Dose Analysis Inputs for LOCA

Input/Assumption	Current Licensing Basis	Proposed AST
Core power level (for radiological source terms)	4100 MWt	
Core power level (for RCS steam releases for supplemental purge)	3876 MWt (3853 MWt + 0.6%)	
Core inventory per MWt	Table 4.3-1	
Activity in coolant blowdown (1% failed fuel)	Table 4.2-14	
Coolant blowdown mass	(parameter not used)	9.3E5 lbm
Activity release from overheated fuel	Table 4.2-9 (Iodines and noble gases only)	Table 4.2-9
Volume of containment sprayed region	2.7E6 ft^3	
Volume of containment unsprayed region	6.8E5 ft^3	
Volume of water in containment sump	61,486 ft^3	
Volume of electrical penetration area	N/A	101,477 ft^3
Volumetric flowrate due to open purge valves	142,000 cfm	
Duration of flow through open purge valves	23 seconds	
Volumetric flowrate between sprayed and unsprayed regions of containment[38]	152,475 cfm (3 of 6 coolers)	
Volumetric leakrate from containment	0.3%/day, first 24 hours 0.15%/day, 24-720 hours	
ESF leakrate	4140 cc/hr (analyzed at 8280 cc/hr)	

[38] Values given are for full flow. Less than 10% of this total will recirculate in the unsprayed region, and dose is insensitive to mixing flow bypass of this magnitude.

4.3 LOSS OF COOLANT ACCIDENT

Table 4.3-11
Dose Analysis Inputs for LOCA

Input/Assumption	Current Licensing Basis	Proposed AST
Fraction of radioiodine released from ESF leakage	10%	10% (0-24 hours) 16% (24-480 hours) 25% (480 hours-720 hours)
Fraction of core iodine inventory released to RCB	50%	See Table 4.3-10
Fraction of Iodines released into the RCB which is available for release	50%	See Table 4.3-10
Iodine Species for the Iodines Released to RCB (elemental/organic/particulate)	91%/4%/5%	4.85%/0.15%/95%
Iodine Species in ESF Leakage (elemental/organic/particulate)	97% / 3% / 0%	
Containment electrical penetration leakrate	N/A	100 sccm per penetration
Number of containment electrical penetrations	N/A	18
Ventilation exhaust rate for electrical penetration area	N/A	833 cfm
Spray start time	2.34 minutes	
Maximum post-LOCA containment pressure	N/A	41.2 psig
Maximum post-LOCA containment temperature	N/A	330 F
Assumed FHB exhaust rate (for ESF leakage)	Infinite	
Assumed FHB filter efficiency Elemental Organic Particulate	 95% 95% 99%	 0% 0% 0%
Dose Conversion Factors	Table 4.2-6	
Decay Constants and Decay Daughter Fractions	Table 4.2-7	
Offsite breathing rates	Table 4.2-1	

4.3 LOSS OF COOLANT ACCIDENT

Table 4.3-11
Dose Analysis Inputs for LOCA

Input/Assumption	Current Licensing Basis	Proposed AST
Offsite χ/Q's	Table 4.1-24	
Control Room HVAC Parameters	Table 4.2-3	
Control Room HVAC Flow Rates	Table 4.2-2	
TSC HVAC Parameters	Table 4.2-5	
TSC HVAC Flow Rates	Table 4.2-4	
Control Room and TSC χ/Q's	Table 4.1-37	

Table 4.3-12
CLB Spray Removal Parameters

Event	Time (hr)	Elemental[39]		Particulate/Aerosol	
		Sprayed Region	Unsprayed Region	Sprayed Region	Unsprayed Region
Break	0.0000	3.59	0.91	0.0	0.0
CSS Start	0.039	23.59	0.91	6.9	0.0
Elemental DF of 60 Reached	0.2808	3.59	0.91	6.9	0.0
Elemental DF of 100 Reached	0.62	0.0	0.0	6.9	0.0
Particulate DF of 50 Reached	0.7559	0.0	0.0	0.7	0.0
Particulate DF of 1000 Reached	6.335	0.0	0.0	0.0	0.0
30 days	720.	0.0	0.0	0.0	0.0

Table 4.3-13
AST Spray Removal Parameters

Event	Time (hr)	Elemental[40]		Particulate/Aerosol	
		Sprayed Region	Unsprayed Region	Sprayed Region	Unsprayed Region
Break	0.0000	4.5	4.5	0.0	0.0
CSS Start	0.039	24.5	4.5	6.9	0.0
Elemental DF of 60 Reached	1.855	0.0	0.0	6.9	0.0
Particulate DF of 50 Reached	2.185	0.0	0.0	0.7	0.0
30 days	720.	0.0	0.0	0.0	0.0

[39] Includes removal by deposition
[40] Includes removal by deposition

4.3 LOSS OF COOLANT ACCIDENT

4.3.6 Summary and Conclusions

Radiological doses resulting from a design-basis LOCA to a Control Room operator and a person located at the EAB or LPZ are to be less than the regulatory dose limits given 10CFR50.67. Table 4.3-14 presents the results of the LOCA radiological consequence analysis.

Table 4.3-14
LOCA Dose Results
(rem TEDE)

	EAB (worst 2 hour)		LPZ		Control Room/TSC (30 days)		
Dose Component	Result	Limit	Result	Limit	Control Room	TSC	Limit
Containment Leakage[41]	5.49		2.52		1.93	0.11	
Elec. Penetration Room	0.01		0.01		0.02	Negligible	
ESF Leakage	0.10		0.27		1.57	0.04	
Supplemental RCB Purge	0.02		0.01		0.02	Negligible	
Shine dose	N/A		N/A		0.14	1.06	
TOTAL	5.62	25	2.81	25	3.68	1.21	5

The worst 2-hour dose at the EAB is between t=0 and 3.7 hours and is less than 5.55 rem TEDE. This is developed from the worst 2-hour EAB dose from four separate RADTRAD cases. The time frame represents the earliest start of a worst 2-hour interval and the latest end of a worst 2-hour interval from these four runs.

[41] The containment release is direct to the environment.

4.4 FUEL HANDLING ACCIDENT

4.4 Fuel Handling Accident (FHA) Radiological Assessment

4.4.1 Methodology Overview

This postulated refueling accident involves the drop of a fuel assembly on top of other fuel assemblies during refueling operations. The mechanical part of the analysis remains unchanged from the CLB; the total number of failed fuel rods is 314 (out of 50952 for an entire core). The depth of water over the damaged fuel is not less than 23 feet.

Following reactor shutdown, decay of short-lived fission products greatly reduces the fission product inventory present in irradiated fuel. The proposed amendment takes credit for the normal decay of irradiated fuel rather than crediting certain active mitigative systems (e.g., ventilation filtration systems). Since radioactive decay is a natural phenomenon, it has a reliability of 100 percent in reducing the potential radiological release from the fuel assemblies.

In addition, the water level that covers the fuel assemblies is another natural method that provides an adequate barrier to a significant radiological release. This defense-in-depth method will continue to be enforced by Technical Specification controls. Technical Specifications 3/4.9.10 and 3/4.9.11 control water level over irradiated fuel assemblies.

The analysis is fully compliant with Regulatory Guide 1.183. The analysis was performed for 42 hours after shutdown assuming a ground-level release through the plant vent. An accident in either the containment or the FHB would involve a release via the plant vent or possibly directly from the containment. However, a release directly from the containment would experience more favorable atmospheric dispersion on the path to the Control Room and TSC air intake than a release from the plant vent because of the greater distance involved.

4.4.2 Analytical Model

A simple analytical model is used for the proposed analysis. The activity released from the fuel assembly is scrubbed by the pool water and then immediately released into the environment. No credit is taken for building holdup or HVAC filtration. The activity release is multiplied by the appropriate χ/Q (EAB, LPZ or Control Room/TSC air intake) to obtain the activity concentration at that location. The offsite χ/Qs and Control Room/TSC χ/Qs are provided in Tables 4.1-24 and 4.1-37, respectively.

4.4 FUEL HANDLING ACCIDENT

The air concentrations are multiplied by the DCFs taken from the RADTRAD AST default file for CEDE and EDE WB (Effective/Inhaled Chronic and Effective/Cloudshine, respectively, with the appropriate conversion from Sv/Bq to rem/Ci) and by the assumed breathing rate of 3.5E-4 m^3/sec from Regulatory Guide 1.183 for the CEDE. The CEDE and EDE WB doses are combined for each location to obtain the TEDE. For the Control Room, the EDE WB is reduced by the finite volume correction factor described in Section 4.2.7 of Regulatory Guide 1.183 prior to calculating the TEDE. Because this correction factor reduces the dose and varies with (volume ratio)$^{0.338}$ and because the TSC volume is smaller than that of the Control Room, the FHA Control Room dose is limiting for the TSC.

No credit is taken for filtration by the FHB filters or for hold-up in either the containment or the FHB. No credit is taken for any filtration (make-up or recirculation clean-up) for either the Control Room or the TSC. The only benefit afforded by the Control Room or TSC envelope is the finite volume EDE WB dose correction. Because of the simplicity of this model, it applies to a FHA both in the containment and in the FHB.

4.4.3 Radiological Source Term

Following accident initiation at 42 hours after shutdown, the radionuclide inventory from the damaged fuel pins is assumed to leak out to the environment instantaneously (even though releases to the environment could be assumed to occur over a 2-hour period according to the NRC regulatory guidance). The core radial peaking factor used is 1.7 (the CLB value). This is a conservatively high peaking factor which bounds past operating experience at STP and is expected to bound future core designs as well. The cycle-specific peaking factor limits are stated in each cycle's Core Operating Limits Report (COLR).

The ORIGEN 2.1 code was used to calculate plant-specific fission product inventories for use in the FHA dose analyses. The fraction of the core that is damaged is assumed to be one fuel assembly (264 fuel pins) plus an additional 50 pins in an impacted assembly (314 pins total) out of 50,952 pins in the core. A peaking factor of 1.7 was applied to the fission product inventory of these pins. This peaking factor value is a practical bounding value for the peaking factors found in the cycle-specific Core Operating Limits Repot (COLR), based on previous core design history and future projections. The gap fractions of Table 3 of Regulatory Guide 1.183 were also applied. This result is the activity release from the damaged fuel.

The core inventory of relevant radionuclides is shown on Table 4.4-1. The CLB activities are based on the discharge batch of a three-region core. The AST activities are based on the core average inventory. Note that the AST activities bound the CLB activities. The values in the table reflect a core average gap inventory and do not include the 1.7 power peaking factor, the 314/50952 pin fraction, or any effects from pool water scrubbing. Since alkali metal releases (as particulates) are assumed to experience an infinite DF due to the water submergence (per Regulatory Guide 1.183), no alkali metals (e.g., Cs and Rb) are included.

4.4 FUEL HANDLING ACCIDENT

Table 4.4-1

Base Fission Product Gap Inventory for the FHA[42]

	CLB		AST	
Isotope	Ci/MWt @t = 0 (shutdown)	Ci/MW @t = 42 hr (accident)	Ci/MWt @t = 0 (shutdown)	Ci/MW @t = 42 hr (accident)
Kr83m	-	-	3.41E+03	5.05E-04
Kr85m	-	-	7.07E+03	9.26E+00
Kr85	1.2E+02	1.23E+02	5.86E+02	5.86E+02
Kr87	-	-	1.34E+04	1.40E-06
Kr88	-	-	1.90E+04	5.77E-01
Kr89	-	-	2.32E+04	0
Xe131m	2.0E+01	1.81E+01	2.68E+02	2.79E+02
Xe133m	8.2E+02	4.73E+02	1.66E+03	9.79E+02
Xe133	5.6E+03	4.43E+03	5.37E+04	4.87E+04
Xe135m	1.2E+03	3.08E-47	1.02E+04	1.08E+02
Xe135	1.4E+03	5.76E+01	1.34E+04	3.83E+03
Xe137	-	-	4.63E+04	0
Xe138	-	-	4.39E+04	9.77E-41
I131	3.5E+03	2.98E+03	4.14E+04	3.56E+04
I132	4.2E+03	1.19E-02	3.71E+04	1.38E-01
I133	5.6E+03	1.38E+03	5.37E+04	1.33E+04
I134	-	-	5.85E+04	1.33E-10
I135	5.1E+03	6.09E+01	4.88E+04	6.46E+02

[42] Reflects core inventory without 1.7 peaking factor or pool DFs applied.

4.4 FUEL HANDLING ACCIDENT

4.4.4 Radiological Releases

The analysis assumes 23 feet of water above damaged fuel. This value corresponds to the minimum depth of water coverage over the top of irradiated fuel assemblies seated in the spent fuel pool racks as required by TS 3/4.9.11, Water Level – Storage Pools Spent Fuel Pool. Twenty-three feet of water is also assumed for an assembly drop in the core. TS 3/4.9.10, "Water Level - Refueling Cavity", requires maintaining at least 23 feet of water above the top of the reactor vessel flange during movement of irradiated fuel assemblies within containment. This assumption is consistent with Regulatory Guide 1.183. Due to the submergence of the damaged fuel, the iodine release is assumed to experience a DF of 200 per Regulatory Guide 1.183. The assumed iodine chemical form after decontamination by the water pool is 43% organic and 57% elemental. No DF is applied to the noble gas. As previously noted, the DF for particulates is assumed to be infinite.

Releases from the Fuel Handling Building are vented to the atmosphere via the Plant Vent. The RCB purge is also from the same Plant Vent. Therefore, for the FHA releases, the Plant-Vent-to-Control-Room χ/Q is used. Releases from the RCB Personnel Airlock are also exhausted via this Plant Vent. The Plant-Vent-to-Control-Room χ/Q also bounds a release from the RCB Equipment Hatch opening since the Plant Vent is much closer to the Control Room air intake than the Equipment Hatch.

4.4.5 Assumptions and Inputs

Assumptions and inputs utilized in the analysis are:

1. The bounding core inventory is based on a DBA power level of 4100 MWth compared to the Rated Thermal Power (RTP) level of 3853 MWth with a 0.6% measurement uncertainty.
2. The release consists of the gap activity in the 264 fuel pins in the dropped assembly and 50 pins in an impacted fuel assembly, for a total of 314 fuel pins. Since there are 193 fuel assemblies in the core, there are 50,952 fuel pins in a core.
3. The dropped assembly and the impacted assembly are assumed to have peaking factors of 1.7.
4. A water depth above the damaged fuel of 23 feet is the limiting case.
5. The activity is assumed to be released directly to the Control Room HVAC intake from the Plant Vent (using the Plant Vent to Control Room χ/Q). The Control Room internal air is assumed to be in equilibrium with the air outside the Control Room HVAC intake. Therefore, the Control Room is not assumed to be pressurized during the accident, nor are any assumptions made as to the functioning of the Control Room HVAC systems.

Input parameters used for the FHA analysis are given in Table 4.4-2. Conformance with Regulatory Guide 1.183 guidance addressing FHA analysis is provided in Attachment 6, Tables A and C.

4.4 FUEL HANDLING ACCIDENT

Table 4.4-2
Fuel Handling Accident Inputs

Input/Assumption	CLB	AST
Previous Reactor Power	4100 MWt	
Fuel Decay Period	42 hrs	
Radial Peaking Factor	1.7	
Release Fractions		
Noble Gases (except Kr-85)	10%	5%
Kr-85	30%	10%
Iodines (except I-131)	10%	5%
I-131	12%	8%
Number of Failed Rods (Equivalent Assemblies)	314 of 50952	
Minimum water depth over damaged fuel	23 feet	
Pool Water Iodine Decontamination Factor	100	200
Release Period	2hr	Instantaneous
Release Location	Plant Vent	
Credit for Filtration on Release	Accident in FHB: Yes Accident in RCB: No	No
Credit for Control Room Filtration	Yes	No
Dose Conversion Factors	Table 4.2-6	
Decay Constants and Decay Daughter Fractions	Table 4.2-7	
Offsite χ/Q's	Table 4.1-24	
Offsite breathing rates	Table 4.2-1	
Control Room and TSC χ/Q's	Table 4.1-37	

4.4 FUEL HANDLING ACCIDENT

4.4.6 Summary and Conclusions

The radiological consequences of the design-basis refueling accident were analyzed using the simplified and conservative assumptions described above. A spreadsheet calculation was carried out to obtain the results for 42 hours of decay and the results are presented in Table 4.4-3. The spreadsheet results were verified with STARDOSE code. The dose agreement between the spreadsheet and STARDOSE was excellent.

Control Room and EAB doses are explicitly calculated. Because the release occurs within two hours, the 0-2 hour EAB dose is bounding for the 0-8 hour LPZ dose. The Control Room dose bounds that for the TSC since the TSC uses the same air intake as the Control Room, and the TSC has a smaller volume. A smaller finite volume correction factor gives a smaller immersion dose.

Table 4.4-3
Fuel Handling Accident Dose Results
(rem TEDE)

Receptor	Dose (0-2 hours)	Limit[43]
EAB	0.83	6.3
LPZ	0.30	6.3
Control Room	3.39	5
TSC	3.39	5

Radiological doses to a Control Room operator and a person located at the EAB or LPZ resulting from a design basis FHA are less than the regulatory dose limits given in 10CFR50.67.

4.4.7 Core Alterations

CORE ALTERATIONS are defined as the movement of any fuel, sources, or reactivity control components (excluding rod cluster control assemblies locked out in the integrated head package) within the reactor vessel with the reactor head removed and fuel in the vessel. As described in TSTF-51, Revision 2, accidents postulated to occur during core alterations include inadvertent criticality (due to control rod removal error or continuous rod withdrawal error during refueling or boron dilution), fuel handling accident, and the loading of a fuel assembly or control component in an incorrect location. Generically, it was concluded that of these off-normal

[43] 10CFR50.67 for offsite and 10CFR50.67, as modified by Regulatory Guide 1.183 in Table 6, page 1.183-20, for the Control Room and TSC.

occurrences, only the fuel handling accident results in cladding damage and potential radiological release. Consequently, it is being proposed that the APPLICABLE MODE "during core alterations" be deleted from TS 3/4.3.2, Table 3.3-3, Functional Unit 3.b.4), TS 3/4.3.2, Table 4.3-2, Functional Unit 3.b.4), and TS 3/4.7.7. Functional Unit 3.b.4) is the Containment Ventilation Isolation RCB Purge Radioactivity instrument. In addition, the ACTION to "immediately suspend core alterations" if the Limiting Condition for Operations is not met is deleted from TS 3/4.3.2, Table 3.3-3, Functional Unit 10.d and from TS 3.7.7, Modes 5 and 6. The affected system by TS 3/4.7.7 is the Control Room Makeup and Cleanup Filtration System.

TS Limiting Conditions for Operation (LCO) requirements remain unaffected for other Technical Specifications that are needed to prevent or mitigate CORE ALTERATION events other than the fuel handling accident. This includes Technical Specifications such as the required boron concentration for refueling operations (Specification 3/4.9.1), and the required nuclear instrumentation for refueling operations (Specification 3/4.9.2).

The LCO APPLICABILITY requirements for operations with a potential for draining the reactor vessel are unaffected by the proposed changes. Also, APPLICABILITY requirements are unaffected for decay heat removal systems during shutdown condition specifications, and for specifications that require maintenance of high water levels over irradiated fuel.

4.4.8 Shutdown Safety Assessment/Defense-in-Depth

In previous amendments for similar relaxations at other facilities, the NRC staff requested that licensees make appropriate commitments to implement administrative controls to facilitate restoration of containment or fuel building closure, and to provide a filtered and monitored release path as a defense-in-depth measure to mitigate the consequences of a postulated FHA. TSTF-51, Revision 2, requires licensees that incorporate this generic change to commit to NUMARC 93-01, Revision 3, Section 11.2.6, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants", subheading "Containment – Primary (PWR)/Secondary (BWR)". The commitment in TSTF-51, Revision 2, is based on a draft version of NUMARC 93-01, Revision 3. When NUMARC 93-01, Revision 3, was approved in July 2000, the guidelines referred to in TSTF-51, Revision 2, were designated as Section 3.6.5. Section 3.6.5 of NUMARC 93-01 states:

"...for plants which obtain license amendments to utilize shutdown safety administrative controls in lieu of Technical Specification requirements on primary or shutdown containment operability and ventilation system operability during fuel handling or core alterations, the following guidelines should be included in the assessment of systems removed from service:

- *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 RCS decays 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*

4.4 FUEL HANDLING ACCIDENT

radiation monitor availability is to reduce doses even further below that provided by the natural decay, and to avoid unmonitored releases.

- *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. The purpose is to enable ventilation systems to draw the release from a postulated fuel handling accident in the proper direction such that is can be treated and monitored.*

The NUMARC 93-01 guidance is built upon two basic premises: avoiding unmonitored releases and using available (although not necessarily "Technical Specification OPERABLE") filtration capabilities to reduce doses below those achieved from the decay of the source term and the scrubbing of the water.

STP License amendments 69 and 139 for Unit 1 and 58 and 128 for Unit 2 were approved based on the premise that administrative controls will be in place to shut a Personnel Airlock Door and the Equipment Hatch in the Reactor Containment Building in the event of a fuel handling accident. Administrative controls will be in place to close penetrations providing direct access from the containment atmosphere to the outside atmosphere and entrances to the Fuel Handling Building in the event of a FHA when integrity of this building is not required. These controls do not need to result in completely blocking the penetration or being capable of resisting pressure.

To support the purpose stated in the preceding paragraph, additional administrative controls will be in place for controlling the removal from service of ventilation filtration and radiation monitoring systems. These controls will be in place so that ventilation filtration and radiation monitoring remains "available" (not necessarily OPERABLE) in the Containment, the Control Room and Fuel Handling Buildings whenever handling irradiated fuel or loads over irradiated fuel to ensure that the release is treated and monitored. If for any reason the ventilation requirements can not be met, fuel movement within the affected building shall be discontinued until the flow path(s) become available. Attachment 4 provides a description of the planned changes to the STP Technical Requirements Manual to close containment penetrations and to maintain ventilation systems available so that releases from a FHA can be treated and monitored. Attachment 5 provides a List of Commitments to maintain these systems available.

4.5 MAIN STEAM LINE BREAK

4.5 Main Steam Line Break Radiological Assessment

4.5.1 Methodology Overview

The Main Steam Line Break (MSLB) accident is postulated as a break of one of the large steam lines outside the containment leading from a SG. For the three intact SGs loops, primary-to-secondary coolant leakage transfers activity into the secondary coolant. This makes it available for release into the environment via steaming through the SG PORV. For the coolant loop with the broken steam line (i.e., faulted SG), primary-to-secondary coolant leakage is assumed to be released from the RCS directly into the environment without passing through any secondary coolant. This is due to assumed "dry-out" conditions in the faulted SG.

Consistent with Regulatory Guide 1.183, two reactor transients that maximize the radioactivity available for release were modeled.

Pre-accident Iodine Spike
A pre-accident iodine spike raises the primary coolant iodine concentration to the Technical Specification maximum 60 μCi/gm assumed DE I-131 value at full power operations. It is assumed that all of the spike activity is homogeneously mixed in the primary coolant, prior to accident initiation.

Note that the equilibrium secondary coolant system iodine activity must also be evaluated. This consists of the 0.1 μCi/gm DE I-131 equilibrium secondary coolant activity concentration, as allowed by the TS. This activity is used to determine the dose contribution that results from the initial blowdown of all fluid in the faulted SG, and the SG PORV release of secondary coolant through the intact SGs.

In addition, the MSLB analysis was performed to determine the release path through the above seat main steam line orifices. This steam release rate is assumed to be 1.93 lbm/second per SG orifice and conservatively continues for 36 hours.

Accident-Initiated Concurrent Iodine Spike
It is assumed that the MSLB event causes a primary reactor system transient concurrent with the release of fluid from the primary and secondary coolant systems. This transient, in turn, is associated with an iodine spike which assumes that the iodine release rate from the fuel rods to the primary coolant increases to a value 500 times greater than the release rate corresponding to the 1.0 μCi/gm DE I-131 RCS equilibrium iodine concentration. The elemental and particulate iodines release rate spike is assumed to occur for a duration of eight hours. Since no partitioning is assumed for the organic iodines, they are released, along with the noble gases, as an instantaneous release.

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As described for the Pre-accident Iodine Spike case above, the dose due to the equilibrium secondary coolant system iodine activity (0.1 μCi/gm DE I-131) must also be determined. The release path through the above seat main steam line orifices is also modeled.

4.5.1.1 Comparison of Modeling with Regulatory Guide 1.183, Appendix E

Regulatory Guide 1.183 Appendix E, Position 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.*

Treatment for the MSLB analysis:

In the faulted SG, all of the primary-to-secondary leakage is assumed to flash to vapor and be released to the environment with no mitigation. In the intact SGs, the primary-to-secondary leakage is assumed to mix with the secondary water without flashing. The SG tubes in the intact SGs are assumed to not be uncovered during the accident.

Regulatory Guide 1.183 Appendix E, Position 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.

Treatment for the MSLB analysis:

This assumption is not used. It is assumed that the primary-to-secondary leakage does not flash in the intact SGs. The nuclides in the primary-to-secondary leakage are added to the bulk fluid in the intact SGs.

Regulatory Guide 1.183 Appendix E, Position 5.5.3:

The leakage that does not immediately flash is assumed to mix with the bulk water.

Treatment for the MSLB analysis:

It is assumed that the primary-to-secondary leakage does not flash in the intact SGs. The nuclides in the primary-to-secondary leakage are added to the bulk fluid in the intact SGs.

4.5 MAIN STEAM LINE BREAK

Regulatory Guide 1.183 Appendix E, Position 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.*

Treatment for the MSLB analysis:

> A partition coefficient of 100 is assumed for elemental and particulate iodines released from the intact steam generators. Organic iodine is not partitioned. Organic iodine is assumed to migrate directly to the steam space and become immediately available for release.

Regulatory Guide 1.183 Appendix E, Position 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.*

Treatment for the MSLB analysis:

> Tube uncovery does not occur in the intact SGs following this event and the subsequent reactor trip.

4.5.2 Analytical Model

The RADTRAD computer code is used to determine the MSLB accident doses at the EAB, LPZ, and Control Room, consistent with Regulatory Guide 1.183. For each spiking scenario, two models were designed for two different release paths, i.e., the intact SG PORVs and the broken steam line. (Note that all releases from the PORVs of different SGs and from a broken steam line are postulated to occur at the location of the PORV closest to the Control Room HVAC intake.)

Following a main steam line break, auxiliary feedwater to the faulted loop is isolated and the steam generator is allowed to steam dry. Thus, radionuclides carried from the primary coolant to the faulted steam generator via leaking tubes are assumed to be released directly to the environment. Radionuclides released from the generators in the intact loops are assumed to be mixed with the secondary coolant and partitioned between the generator liquid and steam before releasing to the environment. The intact steam generator iodine partition coefficient (PC) is 100. The iodine partition is modeled using a reduced release flow rate. The steam release to the environment through relief valves is assumed to last for 8 hours. The steam release through the above seat valve is conservatively assumed to last for 36 hours. For the radiological evaluation of the postulated MSLB, the following two scenarios were considered:

4.5 MAIN STEAM LINE BREAK

1) A pre-existing iodine spike has raised the concentration in the RCS to 60 μCi/gm DEI 131.
2) An accident-induced iodine spike which increases the release rate to the RCS to a value 500 times greater than the release rate corresponding to an RCS iodine concentration of 1 μCi/gm DEI 131.

The CLB uses the ICRP-30 dose conversion factors. For the application of AST and TEDE dose criteria, dose conversion factors from Federal Guidance Reports 11 and 12 are used. A schematic of the analytical model is provided in Figure 4.5-1.

Iodine Spike Release Model:
The reactor trip and the primary system depressurization associated with the MSLB create an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spike model which assumes that the iodine release rate from the fuel rods to the primary coolant increases to a value 500 times greater than the release rate corresponding to the iodine concentration of 1 μCi/gm dose equivalent I-131 in the RCS. The release rate is calculated using the following equation.

$$P_i = N_i\left(\lambda_i + \frac{f_L\eta + LR}{M_{RCS}}\right)$$

P_i	=	Production Rate for Nuclide i (μCi/gm-sec)
λ_i	=	Radioactive Decay Constant for Nuclide i (sec^{-1})
N_i	=	Concentration of Nuclide i (μCi/gm)
f_L	=	Letdown Flow (gm/sec)
M_{RCS}	=	RCS volume (gm)
η	=	Letdown Demineralizer Efficiency/100 (unitless)
LR	=	Rate of Reactor Coolant System Identified and unidentified Leakage (as allowed by plant Technical Specifications). (gm/sec)

This is the same modeling technique as used in the Current Licensing Basis.

Per Appendix E (Section 2.2) of Regulatory Guide 1.183, the assumed iodine spike duration is 8 hours. For conservatism and simplicity of RADTRAD modeling, the 8 hours of total radioactivity release is assumed to be instantaneously released at the beginning of the event. The spike is assumed to also increase the RCS concentration of Alkali metals (Cs and Rb).

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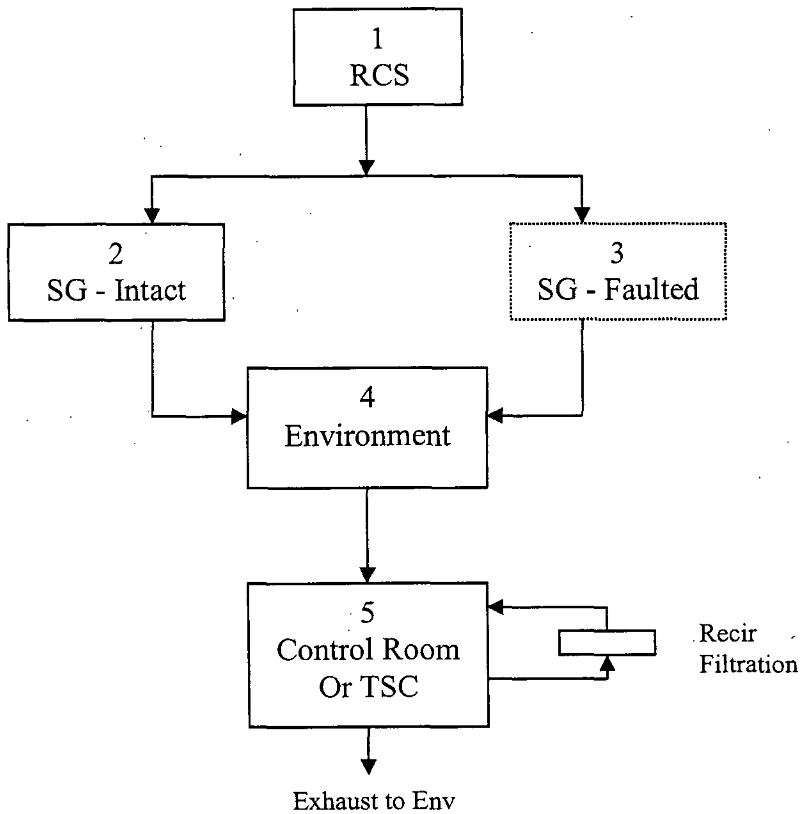
Figure 4.5-1 MSLB RADTRAD Model



4.5 MAIN STEAM LINE BREAK

4.5.3 Radiological Source Term

For this analysis only the iodine and noble gas activities, which are conservatively characterized by operation with 1% core fuel defects and the equilibrium and spiked release rates from that fuel, define the source terms. RADTRAD uses these activities, in curies per megawatt, and then applies nuclide release fractions and a specified core power to calculate the source term for a given case. The AST release fractions associated with iodines and noble gases are assumed to be 100%, and are released to the reactor coolant system.

No additional fuel damage is assumed due to this accident. Two different cases of iodine spiking are analyzed, in accordance with regulatory guidance as previously described.

4.5.3.1 Reactor Coolant System Source Term

4.5.3.1.1 RCS Iodine Concentrations

Table 4.2-14 shows the calculation for the Reactor Coolant System (RCS) iodine concentration, based on thyroid DCFs, for 1% failed fuel. Table 4.2-17 shows the calculation for the RCS iodine concentration, based on Thyroid DCFs, for a Pre-existing Iodine Spike.

For the accident-induced iodine spike, the iodine release rates corresponding to a RCS concentration of 1 μCi/gm are calculated using methodology described in Section 4.5.2. The release rates are then multiplied by the RCS mass and a factor of 500 to yield a release rate in units of Ci/minute. For conservatism, the modeling assumes that the total iodines released from the gap during the 8 hour period are instantaneously released to the RCS (i.e. puff release) following the initiation of the event. Table 4.5-1 shows the total iodine spike activity.

Table 4.5-1
RCS Iodine Inventory Due to Accident-Induced Spike
(500x Release Rate)

Isotope	CLB (Ci)	AST (Ci)	% Difference
I-131	1.86E+05	1.73E+05	-6.7%
I-132	4.82E+05	5.62E+05	16.5%
I-133	3.31E+05	3.23E+05	-2.4%
I-134	1.84E+05	2.35E+05	27.9%
I-135	2.36E+05	1.12E+06	375.3%

4.5.3.1.2 RCS Noble Gas Concentrations

Table 4.2-14 shows the calculation for the RCS noble gas concentration for 1% failed fuel. However, Kr-89 and Xe-137 were not used in the AST analysis.

4.5.3.1.3 RCS Cesium and Rubidium Concentrations

Iodine spikes are conservatively assumed to cause an increase in Cesium and Rubidium activities, along with the increase in iodine concentrations. Table 4.2-18 shows the total activities released from the pre-accident spike. Table 4.5-2 shows the activities for an accident-induced spike. For the MSLB, the spike is modeled as an instantaneous release at time 0 of the total number of curies that would be released into the RCS over an 8-hour period.

Table 4.5-2
Total RCS Cs and Rb Activity for an Accident-Induced Iodine Spike
(Ci)

Isotope	CLB	AST
Rb-86	N/A	2.06E+03
Rb-88	N/A	5.07E+06
Rb-89	N/A	2.65E+05
Cs-134	N/A	1.68E+05
Cs-136	N/A	3.04E+05
Cs-137	N/A	1.32E+05

4.5.3.2 Secondary System Source Terms

4.5.3.2.1 Secondary System Iodine Concentrations

The secondary systems iodine concentrations corresponding to the Technical Specification limit of 0.10 μCi/gm are given in Table 4.2-19.

4.5.3.2.2 Secondary System Noble Gas Concentrations

The secondary systems noble gas concentrations corresponding to 1.0% failed fuel are given in Table 4.2-20.

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4.5.4 Radiological Releases

The activity release model is consistent with the model given on Figure E-1 of Regulatory Guide 1.183. Activity that originates in the RCS is released to the secondary coolant by means of the primary-to- secondary coolant leak rate. This design basis leak rate value is 0.65 gpm from the three intact SGs, and 0.35 gpm for the faulted SG with the broken steam line.

Primary-to-secondary coolant leakage through the faulted SG conservatively goes directly to the environment, without mixing with any secondary coolant. Therefore, under the assumed dry-out conditions, no partitioning of any nuclides is expected to occur in this release pathway.

For all post-accident releases through the PORVs of the intact SG loops, the mechanism for release to the environment is steaming of the secondary coolant. Because of this release dynamic, Regulatory Guide 1.183 allows for a reduction in the amount of activity released to the environment based on partitioning of nuclides between the liquid and gas states of water. For iodine, Regulatory Guide 1.183 allows a partition coefficient of 100 for all iodines. However, organic iodines are assumed to be released directly to the environment. Reviewing the specified AST release fractions, it is concluded that the only nuclides other than iodines to be released from the core source term are noble gas nuclides. Because of their volatility, 100% of the noble gases are assumed to be released.

The methodology used to model steaming of activity through PORVs following the postulated MSLB event assumes an average cumulative release rate through the SG PORVs. The partition factors are applied to these release rates. This data was then converted using the assumption of cooled liquid conditions (i.e., 62.4 lbm/ft^3), as specified by the applicable guidance of Regulatory Guide 1.183. The steaming release and primary-to-secondary coolant leakage is postulated to end at 8 hours when the RCS and secondary loop have reached equilibrium. Steam release through the above seat main steam line orifices is assumed to conservatively continue for 36 hours.

Per Appendix E of Regulatory Guide 1.183, the chemical form of radionuclide released from the fuel is assumed to be 95% cesium iodine (CsI), 4.85% elemental iodine, and 0.15% organic iodine. This analysis assumes that iodine released from the steam generators to the environment is 4.2% elemental, 13.1% organic, and 82.7% particulate (see Section 4.2.5).

Different forms of radionuclide have different transport behaviors. Based on Regulatory Guide 1.183, the particulate form of radioiodine (CsI) is not released from the steam generator to the environment. However, for conservatism, particulate iodine released from the intact steam generators is assumed to have the same partition coefficient as elemental iodine. Also, RCS leakage to the faulted steam generator is assumed to be directly released to the environment. This release includes cesium and rubidium particulates and all chemical forms of radioiodine.

All releases from the SG PORVs (i.e., from the intact SGs) and the faulted SG are considered ground releases.

4.5 MAIN STEAM LINE BREAK

4.5.5 Assumptions and Inputs

The following inputs and assumptions were used in the MSLB analysis.

1. The source term is based upon a power level of 4100 MW thermal, 5 w/o enrichment, and a three region core with equilibrium cycle core at end of life. The three regions have operated at a specific power of 39.3 MW/MTU for 509, 1018, and 1527 EFPD, respectively. The assumed power level is greater than the Rated Thermal Power of 3853 MWth plus a 0.6% measurement uncertainty.
2. The equilibrium secondary activity before the accident is based upon a pre-accident primary-to-secondary leakage of 1 gpm. This is conservative since the Technical Specifications limit the pre-accident leakage to 150 gpd per steam generator or 600 gpd (0.42 gpm) total. The secondary coolant activity is based on 0.1 μCi/gm of dose equivalent I-131. Noble gas activity in the secondary coolant is based on 1% failed fuel.
3. Primary-to-secondary leakage through the steam generator tubes prior to the accident and during the first 8 hours following the transient is 1 gpm. Eight hours after the accident, the residual heat removal system starts and primary-to-secondary leakage is stopped. Primary-to-secondary leakage is modeled as 0.65 gpm for the three intact steam generators and at 0.35 gpm for the faulted steam generator.
4. No fuel failures are assumed to be caused by the main steam line break.
5. For a pre-accident iodine spike, the activity in the reactor coolant is based upon an iodine spike which has raised the reactor coolant concentration to 60 μCi/gm of dose equivalent I-131. Noble gas activity is based on 1% failed fuel.
6. For an accident-induced iodine spike, the accident initiates an iodine spike in the RCS which increases the iodine release rate from the fuel to a value 500 times greater than the release rate corresponding to a RCS concentration of 1 μCi/gm dose equivalent I-131. Iodine is assumed to be released at this rate for 8 hours into the RCS. The iodine activity released from the fuel to the RCS is conservatively assumed to mix instantaneously and uniformly in the RCS. The accident-induced spike is modeled as an instantaneous release at t=0 of the 0-8 hour integrated iodine release.

   Since Regulatory Guide 1.183 specifies that the chemical form of particulate iodine is cesium iodide (CsI), the spike is also assumed to increase the Alkali metal (Cs and Rb) in the RCS in relative amounts. Noble gas activity is conservatively based on 1% failed fuel.
7. Following the rupture, auxiliary feedwater to the faulted loop is isolated and the steam generator is allowed to steam dry. Thus, the iodine partition factor for the faulted steam generator is 1.
8. The activity released from the fuel from the gap is assumed to be instantaneously mixed with the reactor coolant within the pressure vessel per Regulatory Guide 1.183.

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9. Tube uncovery does not occur in the three intact SGs. Primary-to-secondary leakage in these SGs is added to the bulk fluid in the SGs and does not flash directly to the environment.
10. A partition coefficient of 100 is assumed for elemental iodine released from the intact steam generators (Regulatory Guide 1.183, Appendix E, Section 5.5.4). Organic iodine is not partitioned. Organic iodine is assumed to migrate directly to the steam space and become immediately available for release.
11. Similar to the CLB, operator action is taken to isolate the faulted SG within 30 minutes of the event. The total release from the faulted SG is 214,000 lbm initially plus a subsequent release of 385,000 lbm from the Main Feedwater System and the Auxiliary Feedwater System, for a total of 599,000 lbm.
12. Steam releases from the faulted and the intact SGs are assumed to occur at a constant rate for the time period of interest.
13. Eight hours after the accident, the residual heat removal system is in operation and no further steam containing radionuclides is released from steam generators to the environment except the leakage through the MSIV above seat orifices. The release from the orifices continues until 36 hours after the start of the accident. This is conservative since all releases would terminate in less than 8 hours when the RHR system is in operation.
14. The break and the above-seat drain releases occur in the Isolation Valve Cubicle next to the PORVs. Therefore, the PORV-to-Control Room χ/Qs are used for the Control Room and TSC dose analyses.
15. Offsite Power is lost. The condensers are unavailable for steam dump.
16. The Control Room ventilation system is assumed to automatically transfer to the emergency mode of operation after the initiation of safety injection.
17. All activity is released to the environment with no consideration given to cloud depletion by ground deposition during transport to the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ).
18. Reactor coolant density is 8.33 lbs/gal.

Input parameters used for the MSLB analysis are given in Table 4.5-3.

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Table 4.5-3
Inputs for MSLB Analysis

Parameter	CLB	AST
Core power (for radiological source terms)	4100MWt	
Core power (for steam releases)	3876 MWt (3853MWt + 0.6%)	
RCS density	8.33 lbm/gallon	
RCS Mass[44]	2.658E+8 gm	
SG Node Volume Intact Faulted	 5.937E+4 gal 1.979E+4 gal	 5.937E+4 gal 7.94E+3 ft^3 1.979E+4 gal 2.65E+3 ft^3
Primary-to-Secondary Leakage Intact Faulted	 0.65 gpm 0.35 gpm	
Release from Intact SGs 0-2 hours 2-8 hours	 452,000 lbm 1,080,000 lbm	
Release from Faulted SG[45] 0-0.5 hours	 599,000 lbm	
Release from Above (MSIV) Seat Drains Intact SGs Faulted SG	 5.79 lbm/sec 1.93 lbm/sec	
Iodine appearance rate into the RCS for the accident-induced spike	500	
Iodine Species Released from the RCS (%) (elemental/organic/particulate)	91/4/5	4.85/0.15/95
Iodine Species Released from flashed RCS primary-to-secondary leakage flow to the environment (%) (elemental/organic/particulate)	91/4/5	4.85/0.15/95
Iodine Partition Factors for Releases from the Secondary Side (elemental/organic/particulate)	100/100/100	100/1/100

[44] For the CLB analyses, a high value of 2.658E+8 gm was used to maximize the total activity in the RCS and a smaller value of 2.6E+8 was used to determine the RCS volume.
[45] The total release from the faulted SG is 214,000 lbm initially plus a subsequent release of 385,000 lbm from the Main Feedwater System and the Auxiliary Feedwater System, for a total of 599,000 lbm. Operator action is taken to isolate feedwater to the faulted SG within 30 minutes of the event.

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Table 4.5-3
Inputs for MSLB Analysis

Parameter	CLB	AST
Resulting Iodine Species Released from the Secondary Side to Environment % (elemental/organic/particulate)	91/4/5	4.2/13.1/82.7[46]
Steam Flow rate	1.574E+7 lbm/hr	
Dose Conversion Factors	Table 4.2-6	
Decay Constants and Decay Daughter Fractions	Table 4.2-7	
Offsite breathing rates	Table 4.2-1	
Offsite χ/Q's	Table 4.1-24	
Control Room HVAC Parameters	Table 4.2-3	
Control Room HVAC Flow Rates	Table 4.2-2	
TSC HVAC Parameters	Table 4.2-5	
TSC HVAC Flow Rates	Table 4.2-4	
Control Room and TSC χ/Q's	Table 4.1-37	

[46] See Section 4.2.5

4.5 MAIN STEAM LINE BREAK

4.5.6 Summary and Conclusions

Table 4.5-4 provides the results from the analyses.

Table 4.5-4
MSLB Dose Results
(rem TEDE)

Receptor	Pre-Existing Iodine Spike		Accident-induced Iodine Spike	
	Result	Limit	Result	Limit
EAB (worst 2 hour)	0.052	25[47]	0.85	2.5[48]
LPZ	0.041	25[47]	0.66	2.5[48]
Control Room	0.109	5[47]	1.70	5[47]
TSC	0.106	5	1.65	5

All doses are well below their respective acceptance criteria, so it is verified that this design basis MSLB accident is sufficiently mitigated.

[47] 10CFR50.67
[48] 10CFR50.67 as modified by Regulatory Guide 1.183 in Table 6 on Page 1.183-20.

4.6 STEAM GENERATOR TUBE RUPTURE

4.6 Steam Generator Tube Rupture Radiological Assessment

4.6.1 Methodology Overview

The Steam Generator Tube Rupture (SGTR) accident is postulated as a complete severance of a single SG tube. This is a conservative assumption because tube material is a highly ductile metal alloy, and the most probable mode of failure would be one or more minor tube leaks of varying sizes and undetermined origin.

The tube rupture results in the release of radioactive material from the containment. For the three intact SGs, primary-to-secondary coolant leakage continues to transfer activity into the secondary coolant side. This makes it available for release into the environment via steaming through the SG PORVs. For the SG with the ruptured tube, referred to as the ruptured SG, coolant release will take two forms:

1. Break Flow - un-flashed release of RCS coolant directly into the secondary loop, and made available for steaming release to the environment through the PORV.

2. Flashed Break Flow - RCS coolant that flashes directly to steam when released from the ruptured tube, and is sent through the PORV to the environment.

Operators are assumed to identify the ruptured steam generator and attempt to close the PORV on the ruptured steam generator in 10 minutes. However, the PORV is assumed to fail open (the single failure for this accident scenario) at that time. It is assumed that the failed PORV is isolated by manually closing the PORV block valve within 15 minutes of the PORV failure. Therefore, the steam release via the ruptured steam generator's PORV is assumed to continue for a total of 25 minutes. These assumptions are consistent with the current licensing basis.

Consistent with Regulatory Guide 1.183, two reactor transients (i.e., cases) that maximize the radioactivity available for release were modeled.

Pre-accident Iodine Spike

A pre-accident iodine spike raises the primary coolant iodine concentration to the Technical Specification maximum 60 μCi/gm assumed DE I-131 value at full power operations. It is assumed that all of the spike activity is homogeneously mixed in the primary coolant, prior to accident initiation.

The equilibrium secondary coolant system iodine activity must also be evaluated. The total activity available for release from both the intact SGs and ruptured SG is the Technical Specification limit of 0.10 μCi/gm dose equivalent of I-131.

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Accident-Initiated Concurrent Iodine Spike
It is assumed that the SGTR event causes a primary reactor system transient concurrently with the release of fluid from the primary and secondary coolant systems. This transient, in turn, is associated with an iodine spike which assumes that the iodine release rate from the fuel rods to the primary coolant increases to a value 335 times greater than the release rate corresponding to the 1.0 μCi/gm DE I-131 RCS equilibrium iodine concentration. The elemental and particulate iodines release rate spike is assumed to occur for a duration of eight hours. Since no partitioning is assumed for the organic iodines, they are released, along with the noble gases, as an instantaneous release.

The doses due to the equilibrium secondary coolant system iodine activity (0.1 μCi/gm DE I-131) and the release path through the above seat main steam line orifices are also determined.

4.6.1.1 Comparison of Modeling with Regulatory Guide 1.183, Appendix E

Regulatory Guide 1.183 Appendix E, Position 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.*

Treatment for the SGTR analysis:

In the ruptured SG, the portion of the break flow that is assumed to flash to vapor is released to the environment with no mitigation. The unflashed portion and the primary-to-secondary leakage are assumed to mix with the secondary water without flashing. In the intact SGs, the primary-to-secondary leakage is assumed to mix with the secondary water without flashing. The SG tubes in the intact SGs are assumed to not be uncovered during the accident.

Regulatory Guide 1.183 Appendix E, Position 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.

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Treatment for the SGTR analysis:

This assumption is not used. It is assumed that the primary-to-secondary leakage does not flash in the intact SGs. The nuclides in the primary-to-secondary leakage are added to the bulk fluid in the intact SGs.

Regulatory Guide 1.183 Appendix E, Position 5.5.3:

The leakage that does not immediately flash is assumed to mix with the bulk water.

Treatment for the MSLB analysis:

In the ruptured SG, the unflashed portion and the primary-to-secondary leakage are assumed to mix with the bulk SG water without flashing. In the intact SGs, the primary-to-secondary leakage is assumed to mix with the bulk SG water without flashing.

Regulatory Guide 1.183 Appendix E, Position 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.

Treatment for the SGTR analysis:

A partition coefficient of 100 is assumed for elemental iodine released from the intact steam generators. Organic iodine is not partitioned. Organic iodine is assumed to migrate directly to the steam space and become immediately available for release.

Regulatory Guide 1.183 Appendix E, Position 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.

Treatment for the SGTR analysis:

Tube uncovery does not occur in the intact SGs following this event and the subsequent reactor trip.

4.6 STEAM GENERATOR TUBE RUPTURE

4.6.2 Analytical Model

The RADTRAD computer code is used to determine the accident doses, consistent with Regulatory Guide 1.183.

For the analysis of radionuclides other than noble gas, calculation of doses in the accident-initiated (AI) iodine spike case, the appearance rate of the nuclides from the reactor core is a significant factor in the analysis. Each nuclide has a unique appearance rate based on preaccident production and spiking assumptions. Complicating this case are modeling limitations in the RADTRAD code that do not allow the code to either explicitly model appearance rates, source distribution, and application of partition factors to nodes or nuclides. In order to accurately model these behaviors in a reasonably limited number of computer calculations, some simplifying assumptions were made:

1) The I-131 appearance rate is arbitrarily selected as the base core to RCS appearance rate. The core is modeled as a 1 cubic foot volume, and the core to RCS flow rate corresponded to the I-131 appearance rate. This only applies to the AI spike cases.
2) The AI spike source terms for non-noble gas nuclides was scaled by the ratio of the nuclide's appearance rate to the I-131 appearance rate. This modified the nuclide source so that the curie count appearing for release remains correct despite the difference between the actual and modeled flow from fuel to RCS to release.
3) The organic iodine available for release is calculated based on each nuclide's appearance rate and the 8-hour time period. This iodine was modeled with the noble gases as a puff release because neither set of nuclides are assumed to be partitioned.
4) For pre-existing (PE) iodine spiking, the source term in the RCS is calculated directly and appearance rates from the fuel were not a part of that model.
5) Because the code does not allow multiple sources, the primary source was assumed present in an initial node, with additional activity assigned via a "fraction" term. For core release sources (AI spikes), this fraction for the RCS and SGs is based on the core activity level (with the core assigned a fraction of ~1.0). For all RCS release sources (both AI and PE spikes), the source fractions in the steam generators are based on the conservatively high PE iodine spike source distribution.

These modeling assumptions allow the control room and offsite doses for the either iodine spike to be modeled in three computer runs:

- Iodine, cesium, rubidium modeled from core to RCS to SG to environment with steam generator flows partitioned;
- Noble gases and organic iodine eight-hour puff release from RCS to SG to environment without partitioned steam generator flows; and,

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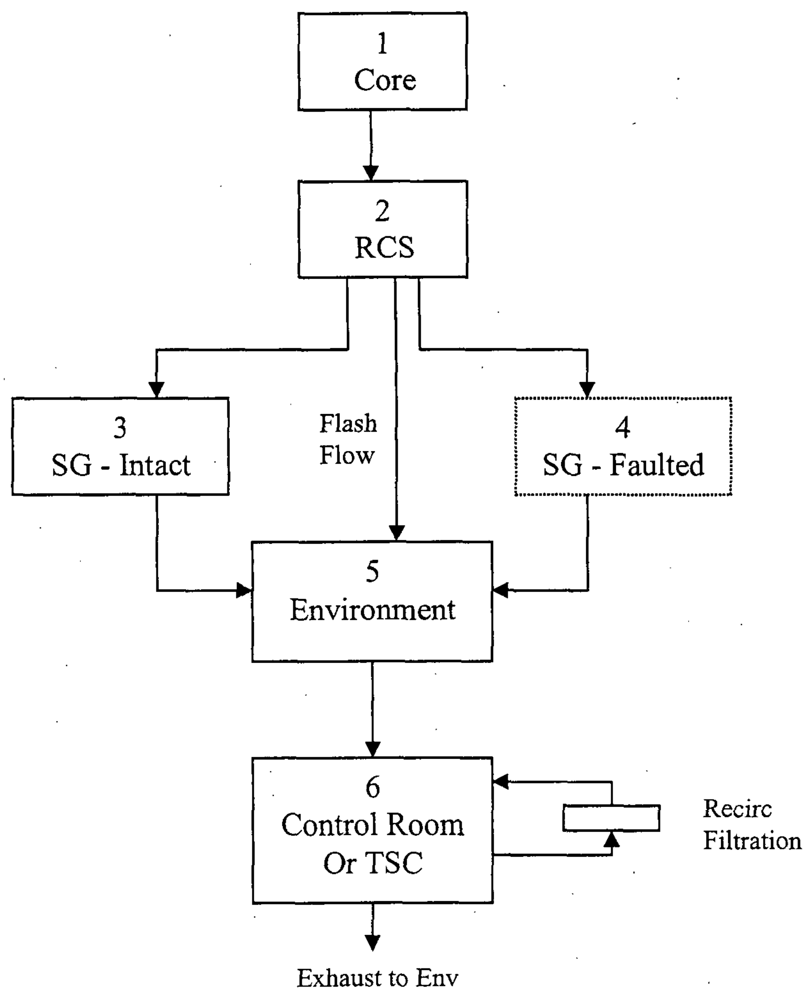
- Core model with spike-increased isotopes (335x) to RCS (I, Cs, & Rb) directly released from the core to the RCS to the environment without flow partitioning.

These models were run separately for Control Room and TSC modeling, and the analysis was performed for the accident-initiated iodine spike and the preexisting iodine spike.

A schematic of the analytical model is provided in Figure 4.6-1.

Figure 4.6-1: SGTR RADTRAD Model



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4.6.3 Radiological Source Term

For this analysis, only the iodine and noble gas activities, which are conservatively characterized by operation with 1% core fuel defects and the equilibrium and spiked release rates from that fuel, define the source terms. RADTRAD uses these activities, in curies per megawatt, and then applies nuclide release fractions and a specified core power to calculate the source term for a given case. The AST release fractions associated with iodines and noble gases are assumed to be 100%, and are released to the reactor coolant.

No additional fuel damage is assumed due to this accident. Two different cases of iodine spiking are analyzed, in accordance with regulatory guidance as previously described.

4.6.3.1 Reactor Coolant System Source Term

4.6.3.1.1 RCS Iodine Concentrations

Table 4.2-14 shows the calculation for the Reactor Coolant System (RCS) iodine concentration, based on Thyroid DCFs, for 1% failed fuel that was used in this analysis. Table 4.2-17 shows the calculation for the RCS iodine concentration, based on Thyroid DCFs, for a Pre-existing Iodine Spike.

For the Accident-induced iodine spike, the iodine release rates corresponding to a RCS concentration of 1 μCi/gm are calculated using the methodology described in Section 4.5.2. The release rates are then multiplied by the RCS mass, and a factor of 335 to yield a release rate in units of Ci/minute. The iodines are assumed to be 95% particulate, 4.85% elemental, and 0.15% organic. The elemental and particulate iodines are released to the RCS at the I-131 release rate. For conservatism, no partition factor is assigned to organic iodines and the modeling assumes that the total organic iodines released from the gap during the 8 hours period is instantaneously released to the RCS (i.e. puff release) following the initiation of the event. Table 4.6-1 shows the total iodine spike activity.

Table 4.6-1
RCS Iodine Inventory Due to an 8-Hour Accident-Induced Spike
(335x Release Rate)

Isotope	CLB@ 500x (Ci)	AST @ 335x (Ci)	% Difference
I-131	1.84E+05	1.16E+05	-37.0%
I-132	4.95E+05	3.77E+05	-23.8%
I-133	3.30E+05	2.16E+05	-34.5%
I-134	1.95E+05	1.57E+05	-19.4%
I-135	2.33E+05	7.44E+05	219.1%

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4.6.3.1.2 RCS Noble Gas Concentrations

Table 4.2-14 shows the calculation for the RCS noble gas concentration for 1% failed fuel. However, Kr-89 and Xe-137 were not used in the AST analyses. Because noble gases are not subject to spiking, the same source terms are used in both spiking cases. Also, the noble gases released from the RCS are modeled as an 8 hour integrated puff release.

4.6.3.1.3 RCS Cesium and Rubidium Concentrations

Iodine spikes are conservatively assumed to cause an increase in Cesium and Rubidium activities, along with the increase in iodine concentrations. Table 4.2-18 shows the activity in the RCS due to a pre-accident spike. The Cs and Rb activity released due to an accident-induced spike are modeled as a time release into the RCS at the rate of the I-131 release rate.

4.6.3.2 Secondary System Source Terms

4.6.3.2.1 Secondary System Iodine Concentrations

The secondary systems iodine concentrations corresponding to the Technical Specification limit of 0.10 μCi/gm are given in Table 4.2-19

4.6.3.2.2 Secondary System Noble Gas Concentrations

The secondary systems noble gas concentrations corresponding to 1.0% failed fuel are given in Table 4.2-20.

4.6.4 Radiological Releases

The activity release model is consistent with the model given on Figure E-1 of Regulatory Guide 1.183. Activity that originates in the RCS is released to the secondary coolant by means of the RCS break flow and the primary-to- secondary coolant leak rate.

Activity that originates in the RCS is released to the secondary coolant by means of the primary-to-secondary coolant leak rate and the break flow in the ruptured SG. The total primary-to-secondary coolant leak rate is assumed to be 1 gpm. After the accident, 0.35 gpm of the primary-to-secondary coolant leakage is assumed to occur in the ruptured SG and 0.65 gpm in the intact SGs. This leakage continues for 8 hours.

The flashed portion of the ruptured tube break flow in the ruptured SG conservatively goes directly to the environment, without mixing with any secondary coolant. Therefore, no partitioning of any nuclides is expected to occur in this release pathway. The unflashed portion of the break flow and the 0.35 gpm normal primary-to-secondary leakage are assumed to mix in the bulk water of the SG.

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The intact SGs do not experience tube bundle uncovery. Therefore, primary-to-secondary coolant leakage into the intact SGs mixes with the bulk water in the SG and no flashing to the environment is assumed to occur.

For all post-accident releases through the PORVs of the intact SG loops, the mechanism for release to the environment is steaming of the secondary coolant. Because of this release dynamic, Regulatory Guide 1.183 allows for a reduction in the amount of activity released to the environment based on partitioning of nuclides between the liquid and gas states of water. For iodine, Regulatory Guide 1.183 allows a partition coefficient of 100 for all iodines. However, organic iodines are assumed to be released directly to the environment. Reviewing the specified AST release fractions, it is concluded that the only nuclides other than iodines to be released from the core source term are noble gas nuclides. Because of their volatility, 100% of the noble gases are assumed to be released.

The methodology used to model steaming of activity through PORVs following the postulated SGTR event assumes an average cumulative release rate through the SG PORVs. The partition factors are applied to these release rates. This data was then converted using the assumption of cooled liquid conditions (i.e., 62.4 lbm/ft3), as specified by the applicable guidance of Regulatory Guide 1.183. The steaming release and primary-to-secondary coolant leakage is postulated to end at 8 hours, when the RCS and secondary loop have reached equilibrium. Steam release through the above seat main steam line orifices is assumed to conservatively continue for 36 hours.

Per Appendix E of Regulatory Guide 1.183, the chemical form of radionuclide released from the fuel is assumed to be 95% cesium iodine (CsI), 4.85% elemental iodine, and 0.15% organic iodine. This analysis assumes that iodine released from the steam generators to the environment is 4.2% elemental, 13.1% organic, and 82.7% particulate (see Section 4.2.5).

Different forms of radionuclide have different transport behaviors. Based on Regulatory Guide 1.183, the particulate form of radioiodine (CsI) is not released from the steam generator to the environment. However, for conservatism, particulate iodine released from the intact steam generators is assumed to have the same partition coefficient as elemental iodine.

All releases from the SG PORVs (i.e., from the intact SGs) and the ruptured SG are considered ground releases from the nearest PORV to the Control Room HVAC intake.

4.6.4.1 Thermal/Hydraulic Analysis of the SGTR

The sequence of events for the thermal hydraulics model is provided in Table 4.6-2.

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Table 4.6-2
Thermal Hydraulics Analysis Sequence of Events
(seconds)

Event	Time
SG Tube Rupture	0
Reactor Trip	66.5
SI Actuation	544.5
Ruptured SG Isolated	607
Ruptured SG PORV Fails Open	611
Ruptured SG PORV Block Valve Closed	1507
RCS Cool-down Initiated	1800
Two Charging Pumps Started	1800
Break Flow Stops Flashing	2027
RCS Cool-down Terminated	2610.6
RCS Depressurization Initiated	3150.6
RCS Depressurization Terminated	3305.8
SI Terminated	3425.8
Excess Charging Flow Eliminated	3425.8
Break Flow Terminated	5128

From this detailed timeline for the accident, the dose analysis uses the nine events in Table 4.6-3 in the model. The last three entries in the table below are taken from the description of the transient. These times are also used in the CLB analysis.

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Table 4.6-3
Thermal Hydraulics Analysis Time Points used in the SGTR
Dose Analysis
(seconds)

Event	T&H Time
SG Tube Rupture	0
Reactor Trip	66.5
Ruptured SG Isolated	607
Ruptured SG PORV Block Valve Closed	1507
RCS Flashing in Faulted SG ends	2027
Break Flow Terminated	5128
Dose Model Parameter Changes	7200
RHR Entry	28800
End of Orifice Releases	129600

The mass flows to and from the SGs for both the CLB and the AST analyses are presented in Tables 4.6-4 through 4.6-7.

Table 4.6-4
Thermal Hydraulics Analysis Total Break Flow

Time Period (sec)			Flow During Period (lbm)
0	to	66.5	3941
66.5	to	607	25368
607	to	1507	50333
1507	to	2027	26649
2027	to	5128	97094
5128	to	7200	0
7200	to	28800	0
28800	to	1.296E+5	0

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Table 4.6-5
Thermal Hydraulics Analysis Flashed Break Flow

Time Period (sec)			Flow During Period (lbm)
0	to	66.5	617
66.5	to	607	1696
607	to	1507	6900
1507	to	2027	2208
2027	to	5128	0
5128	to	7200	0
7200	to	28800	0
28800	to	1.296E+5	0

Table 4.6-6
Thermal Hydraulics Analysis Total Intact SG Steam Flow to Atmosphere

Time Period (sec)			Flow During Period (lbm)
0	to	66.5	240000
66.5	to	607	37085
607	to	1507	7369
1507	to	2027	120353
2027	to	5128	269795
5128	to	7200	227041
7200	to	28800	1158465
28800	to	1.296E+5	0

The iodine partition factor of 100 for liquid and steam phase for steam generator releases is modeled by reducing the flow out of the SG by a factor of 100. The steam released from the condenser (during 0-66.5 seconds) has a total iodine DF of 10,000 from changing phase in the SG and exiting through the condenser. After 66.5 seconds, the condenser is longer available due to the assumed loss of offsite power.

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Table 4.6-7
Thermal Hydraulics Analysis Total Ruptured SG Steam Flow to Atmosphere

Time Period (sec)			Total Ruptured SG Flow During Period (lbm)
0	to	66.5	84000
66.5	to	607	10066
607	to	1507	131365
1507	to	2027	0
2027	to	5128	0
5128	to	7200	0
7200	to	28800	50040
28800	to	1.296E+5	0

4.6.4.2 Modification of the Thermal/Hydraulic Data for Dose Analysis

The time intervals determined by the thermal/hydraulic analyses were arbitrarily increased to provide additional margin in the dose analyses. The adjusted times listed below are based on the data in Table 4.6-3.

Table 4.6-8
Modified Time Sequence of Events for SGTR Dose Analysis

Event	Time (sec)
SG Tube Rupture	0
Reactor Trip	66.5
Ruptured SG Isolated	607
Ruptured SG PORV Block Valve Closed	1507
RCS Flashing in Ruptured SG ends	2087
Break Flow Terminated	5248
Dose Model Parameter Changes	7380
RHR Entry	28980
End of Orifice Releases	129800

Flow rates are calculated using the time periods taken from Table 4.6-3. These flow rates are arbitrarily increased by 40% and the increased flow rates are assumed to exist during the longer phase intervals with the adjusted times in Table 4.6-9. This results in a larger integrated mass release. The mass releases used in the SGTR are presented in Tables 4.6-9 through 4.6-12. In actual use in RADTRAD these mass releases are converted to cfm using a cold water density of 8.33 lbm/gal.

Tables 4.6-9 through 4.6-12 report a total mass released during each time period. Although the analyses use a volumetric flow rate, the total flow values are presented to illustrate the

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conservatisms built into the analyses. The values for the "Dose Analysis" columns are calculated as:

$$Dose\ Analysis\ Value = \frac{T\,\&\,H\,Flow, Table\,4.6-5}{T\,\&\,H\,\Delta t, Table\,4.6-4} * 1.4 * Dose\,\Delta t\ \ from\,Table\,4.6-9$$

Table 4.6-9
Total Break Flow used in SGTR Dose Analysis

Phase	Ending Time (sec)	Total Flow During Period (lbm) T&H Analysis	Dose Analysis	% Difference
SG Tube Rupture	0	-	-	-
Reactor Trip	66.5	3941	5517	40%
Ruptured SG Isolated	607	25368	35515	40%
Ruptured SG PORV Block Valve Closed	1507	50333	70466	40%
RCS Flashing in Faulted SG Stops	2087	26649	41613	56%
Break Flow Terminated	5248	97094	138562	43%
Dose Model Parameter Changes	7380	0	0	0%
RHR Entry	28980	0	0	0%
End of Orifice Releases	129800	0	0	0%

Table 4.6-10
Total Flashed Break Flow used in SGTR Dose Analysis

Phase	Ending Time (sec)	Total Flow During Period (lbm) T&H Analysis	Dose Analysis	% Difference
SG Tube Rupture	0	-	-	-
Reactor Trip	66.5	617	864	40%
Ruptured SG Isolated	607	1696	2374	40%
Ruptured SG PORV Block Valve Closed	1507	6900	9660	40%
RCS Flashing in Faulted SG Stops	2087	2208	3448	56%
Break Flow Terminated	5248	0	0	0%
Dose Model Parameter Changes	7380	0	0	0%
RHR Entry	28980	0	0	0%
End of Orifice Releases	129800	0	0	0%

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Table 4.6-11
Total Intact SG Flow to Atmosphere Used in SGTR Dose Analysis

Phase	Time (sec)	Total Flow During Period (lbm) T&H Analysis	Dose Analysis	% Difference
SG Tube Rupture	0	-	-	-
Reactor Trip	66.5	240000	336000	40%
Ruptured SG Isolated	607	37085	51919	40%
Ruptured SG PORV Block Valve Closed	1507	7369	10317	40%
RCS Flashing in Faulted SG Stops	2087	120353	187936	56%
Break Flow Terminated	5248	269795	385021	43%
Dose Model Parameter Changes	7380	227041	327062	44%
RHR Entry	28980	1158465	1621851	40%
End of Orifice Releases	129800	0	0	0%

Table 4.6-12
Total Ruptured SG Flow to Atmosphere Used in SGTR Dose Analysis

Phase	Time (sec)	Total Flow During Period (lbm) T&H Analysis	Dose Analysis	% Difference
SG Tube Rupture	0	-	-	-
Reactor Trip	66.5	84000	117600	40%
Ruptured SG Isolated	607	10066	14092	40%
Ruptured SG PORV Block Valve Closed	1507	131365	183911	40%
RCS Flashing in Faulted SG Stops	2087	0	0	0%
Break Flow Terminated	5248	0	0	0%
Dose Model Parameter Changes	7380	0	0	0%
RHR Entry	28980	50040	70056	40%
End of Orifice Releases	129800	0	0	0%

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4.6.5 Assumptions and Inputs

The following inputs and assumptions are used in the SGTR analysis.

1. The source term is based upon a power level of 4100 MW thermal, 5 w/o enrichment, and a three region core with equilibrium cycle core at end of life. The three regions have operated at a specific power of 39.3 MW/MTU for 509, 1018, and 1527 EFPD, respectively. The assumed power level is greater than the Rated Thermal Power of 3853 MWth plus a 0.6% measurement uncertainty.
2. The equilibrium secondary activity before the accident is based upon a pre-incident primary-to-secondary leakage of 1 gpm. This is conservative since the Technical Specifications limits the pre-accident leakage to 150 gpd per steam generator or 600 gpd (0.42 gpm) total. The secondary coolant activity is based on 0.1 μCi/gm of dose equivalent I-131. Noble gas activity in the secondary coolant is based on 1% failed fuel.
3. No fuel failures are assumed to be caused by the SGTR.
4. Total primary-to-secondary leakage through the steam generator tubes prior to the accident and during the first 8 hours following the transient is 1 gpm. Eight hours after the accident, the residual heat removal system starts and primary-to-secondary leakage is stopped. Primary-to-secondary leakage is conservatively modeled at 0.65 gpm for the three intact steam generators and at 0.35 gpm for the ruptured steam generator.
5. The intact SGs do not experience tube bundle uncovery. Therefore, primary-to-secondary coolant leakage into the intact SGs mixes with the bulk water in the SG and no flashing to the environment is assumed to occur.
6. For a Pre-accident iodine spike, the activity in the reactor coolant is based upon an iodine spike which has raised the reactor coolant concentration to 60 μCi/gm of dose equivalent I-131. Noble gas activity is based on 1% failed fuel.
7. For an Accident-induced iodine spike, the accident initiates an iodine spike in the RCS which increases the iodine release rate from the fuel to a value 335 times greater than the release rate corresponding to a RCS concentration of 1 μCi/gm dose equivalent I-131. Iodines, Cs, and Rb are assumed to be released at this rate for 8 hours (Cs and Rb are released at the rate of I-131). The iodine activity released from the fuel to the RCS is conservatively assumed to mix instantaneously and uniformly in the RCS. Since Regulatory Guide 1.183 specifies that the chemical form of particulate iodine is (CsI), the spike is also assumed to relatively increase the Alkali metal (Cs and Rb) in the RCS. Noble gas activity is conservatively based on 1% failed fuel.
8. The activity released from the fuel gap is assumed to be instantaneously mixed with the reactor coolant within the pressure vessel per Regulatory Guide 1.183.
9. A partition coefficient of 100 is assumed for elemental iodine released from the steam generators. (Regulator Guide 1.183, Appendix E, Section 5.6) Organic iodine is not partitioned. Organic iodine is assumed to migrate directly to the steam space and become immediately available for release.

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10. Operators are assumed to identify the ruptured steam generator and attempt to close the PORV on the ruptured steam generator in 10 minutes. However, the PORV is assumed to fail open (the single failure for this accident scenario) at that time. It is assumed that the failed PORV is isolated by manually closing the PORV block valve within 15 minutes of the PORV failure. Therefore, the steam release via the ruptured steam generator's PORV is assumed to continue for a total of 25 minutes.
11. Eight hours after the accident, the residual heat removal system is in operation and no further steam containing radionuclides is released from steam generators to the environment except the leakage through the MSIV above-seat drain orifices. The release through the orifices continues until 36 hours after the start of the accident. (These orifice releases occur in the Isolation Valve Cubicle next to the PORVs. Therefore, the PORV-to-Control Room χ/Qs are used for the Control Room and TSC dose analyses.) This is conservative since all releases would terminate in less than 8 hours when the RHR system is in operation.
12. The SG releases are via the PORVs and ruptured SG safety valves. The above-seat drain releases occur in the Isolation Valve Cubicle next to the PORVs. Therefore, the PORV-to-Control Room χ/Qs are used for the Control Room and TSC dose analyses.
13. Offsite Power is lost. After 66.5 seconds, the condensers are unavailable for steam dump.
14. The Control Room ventilation system automatically transfers to the emergency mode of operation after the initiation of safety injection. This is assumed to happen at t=0 instead of upon reactor trip at 66.5 seconds. Since the mass releases are increased by about 40%, the time difference is negligible.
15. All activity is released to the environment with no consideration given to cloud depletion by ground deposition during transport to the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ).
16. Reactor coolant density is 8.33 lbs/gal.

Input parameters used for the SGTR analysis are given in Table 4.6-13. Conformance with Regulatory Guide 1.183 guidance addressing the SGTR analysis is provided in Attachment 6, Tables A and F.

4.6 STEAM GENERATOR TUBE RUPTURE

Table 4.6-13
Inputs for SGTR Analysis

Parameter	CLB	AST
Core power (for radiological source terms)	4100MWt	
Core power level (for steam releases)	3876 MWt (3853MWt + 0.6%)	
RCS & Secondary density	8.33 lbm/gallon	
RCS Mass	2.658E+8 gm	
SG Node Volume Intact Ruptured	 5.94E+4 gal 1.98E+4 gal	 5.937E+4 gal 1.979E+4 gal
Secondary Mass	659,412 lbm	
Primary-to-Secondary Leakage Intact Ruptured	 0.42 gpm 0 gpm	 0.65 gpm 0.35 gpm
Accident Time line	Table 4.6-8	
Operator Action Times diagnose SGTR close PORV block valve on ruptured SG	 @10 minutes @25 minutes	
Total Break Flow	Table 4.6-9	
Total Flashed Break Flow	Table 4.6-10	
Total Intact SG Flow to Atmosphere	Table 4.6-11	
Total Ruptured SG Flow to Atmosphere	Table 4.6-12	
Release from Above (MSIV) Seat Drains Intact SGs Ruptured SG	 5.79 lbm/sec 1.93 lbm/sec	
Steam Flow rate	1.574E+7 lbm/hr	
DF in condenser (before LOOP)	10,000	
Iodine appearance rate into the RCS for the accident-induced spike	500	335
Iodine Species Released from the RCS (%) (elemental/organic/particulate)	91/4/5	4.85/0.15/95
Iodine Species for Flashed RCS Break Flow to the environment (%) (elemental/organic/particulate)	91/4/5	4.85/0.15/95
Iodine Partition Factors for Releases from the Secondary Side (elemental/organic/particulate)	100/100/100	100/1/100

4.6 STEAM GENERATOR TUBE RUPTURE

Table 4.6-13
Inputs for SGTR Analysis

Parameter	CLB	AST
Resulting Iodine Species Released from the Secondary Side to Environment % (elemental/organic/particulate)	91/4/5	4.2/13.1/82.7[49]
Dose Conversion Factors	Table 4.2-6	
Decay Constants and Decay Daughter Fractions	Table 4.2-7	
Offsite breathing rates	Table 4.2-1	
Offsite χ/Q's	Table 4.1-24	
Control Room HVAC Parameters	Table 4.2-3	
Control Room HVAC Flow Rates	Table 4.2-2	
TSC HVAC Parameters	Table 4.2-5	
TSC HVAC Flow Rates	Table 4.2-4	
Control Room and TSC χ/Q's	Table 4.1-37	

[49] See Section 4.2.5

4.6 STEAM GENERATOR TUBE RUPTURE

4.6.6 Summary and Conclusions

Table 4.6-14 below provides the results for the SGTR scenarios.

Table 4.6-14
SGTR Dose Results
(rem TEDE)

Receptor	Pre-Existing Iodine Spike		Accident-induced Iodine Spike	
	Result	Limit	Result	Limit
EAB (worst 2 hour)	2.37	25[50]	1.08	2.5[51]
LPZ	0.92	25[50]	0.44	2.5[51]
Control Room	2.15	5[50]	1.00	5[50]
TSC	2.09	5	0.98	5

For the cases analyzed in this calculation, it is shown that a SGTR that involves a pre-accident 60 μCi/gm iodine spike, which instantaneously releases activity into the RCS prior to initiating SGTR releases, would be the bounding SGTR accident scenario. All doses are well below their respective acceptance criteria; therefore, this design-basis SGTR accident is sufficiently mitigated.

[50] 10CFR50.67

[51] 10CFR50.67 as modified by Regulatory Guide 1.183 in Table 6 on Page 1.183-20.

4.7 CONTROL ROD EJECTION ACCIDENT

4.7 Control Rod Ejection Radiological Assessment

The Control Rod Ejection Accident (CREA) is defined as the mechanical failure of a control rod mechanism pressure housing resulting in the ejection of an RCCA and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

4.7.1 Methodology Overview

An analysis of the effects of a postulated rod ejection accident is performed using the assumptions of Regulatory Guide (RG) 1.183. For the analysis, it is assumed that prior to the postulated accident, the plant is operating at an equilibrium level of radioactivity in the primary and secondary systems as a result of coincident fuel defects and SG tube leakage. Following a postulated rod ejection accident, two activity release paths contribute to the total radiological consequences of the accident. The first release path is via Containment leakage resulting from release of activity from the primary coolant to the Containment. The second path is the contribution of steam in the secondary system dumped through the SG PORVs and safety valves since offsite power is assumed to be lost.

4.7.1.1 Comparison of Modeling with Regulatory Guide 1.183, Appendix E

Regulatory Guide 1.183 Appendix E, Position 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.*

Treatment for the CREA analysis:

The SGs do not experience tube uncovery. The primary-to-secondary leakage is assumed to mix with the secondary water without flashing.

Regulatory Guide 1.183 Appendix E, Position 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.

4.7 CONTROL ROD EJECTION ACCIDENT

Treatment for the CREA analysis:

This assumption is not used. It is assumed that the primary-to-secondary leakage does not flash in the SGs. The nuclides in the primary-to-secondary leakage are added to the bulk fluid in the SGs.

Regulatory Guide 1.183 Appendix E, Position 5.5.3:

The leakage that does not immediately flash is assumed to mix with the bulk water.

Treatment for the CREA analysis:

It is assumed that the primary-to-secondary leakage does not flash in the SGs. The nuclides in the primary-to-secondary leakage are added to the bulk fluid in the SGs.

Regulatory Guide 1.183 Appendix E, Position 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.

Treatment for the CREA analysis:

A partition coefficient of 100 is assumed for elemental iodine released from the steam generators. Organic iodine is not partitioned. Organic iodine is assumed to migrate directly to the steam space and become immediately available for release.

Regulatory Guide 1.183 Appendix E, Position 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.

Treatment for the CREA analysis:

Tube uncovery does not occur following this event and the subsequent reactor trip.

4.7.2 Analytical Model

It is assumed that prior to the accident the plant has been operating with simultaneous fuel defects and SG tube leakage for a period of time sufficient to establish equilibrium levels of activity in the primary and secondary coolant. The model for the activity available for leakage from the Containment assumes that the activity in the fuel pellet-clad gap and the activity released due to fuel melting are instantaneously mixed in the Containment and available for release. All of the gap activity of the fuel rods failed by accident is assumed released to the Containment. Of the fuel melted, 100 percent of the noble gases and 25 percent of the iodines are

4.7 CONTROL ROD EJECTION ACCIDENT

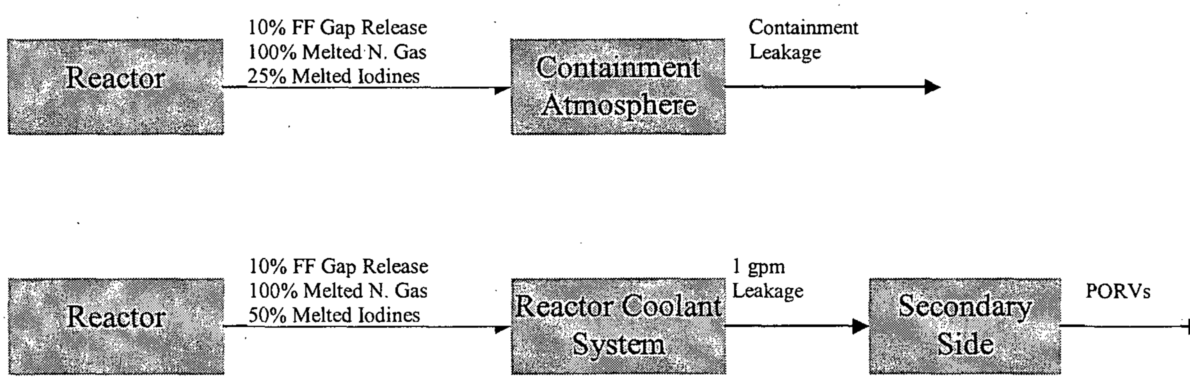
assumed available for leakage from the Containment. The only removal processes considered for the Containment are radioactive decay and leakage.

The model for the activity available for release to the atmosphere from the safety valves assumes that the release consists of the activity in the secondary coolant prior to the accident plus that activity leaking from the primary coolant through the SG tubes following the accident. The primary coolant activity after the accident is assumed to be composed of the equilibrium activity prior to the accident, plus 100 percent of the noble gases and iodines released by fuel failed during the accident, plus 100 percent of the noble gases and 50 percent of the iodines released by fuel melted by the accident. The leakage of primary coolant to the secondary side of the SG is assumed to continue at its initial rate, assumed to be the same rate as the leakage prior to the accident, until the pressures in the primary and secondary systems are equalized. No mass transfer from the primary system to the secondary system is assumed thereafter.

Since a coincident loss of offsite power is assumed, activity is assumed to be released to the atmosphere through the SG PORVs and not the main condenser.

The two release pathways are shown below.

Figure 4.7-1: CREA Analysis Model



4.7.3 Radiological Source Term

The sudden rod ejection and localized temperature spike associated with the CREA results in 10% core damage. Only 2.5% of the damaged core releases melted fuel activity (i.e., 0.25% of the total core melts). Therefore for both cases, the source term available for release is associated with this fraction of melted fuel and the fraction of core activity existing in the gap.

Release fractions and transport fractions conform to Regulatory Guide 1.183, Appendix H and Table 3. To conform to this regulatory guidance, 10% of the core inventory of iodine and noble

4.7 CONTROL ROD EJECTION ACCIDENT

gas is assumed to be in the fuel-clad gap. Additionally, Table 3 of Regulatory Guide 1.183 shows that 12% of the core cesium and rubidium should be assumed to be in the fuel-clad gap and should be released in its entirety from the damaged 10% of the total core.

With regard to the fraction released from melted fuel, it is assumed that 90% of the core inventory of iodine and noble gas, and 88% of the core cesium and rubidium remain available for release due to melting (i.e., these are the remaining fractions of activity that are not in the fuel-clad gap).

One hundred percent of the noble gases and iodines in the clad gaps of the fuel rods experiencing clad damage (assumed to be 10 percent of the rods in the core) is assumed released. The accident evaluation conservatively assumes this activity to be released twice: to the Containment for leakage to the atmosphere and to the primary coolant for leakage to the secondary system.

The fraction of fuel melting is assumed to be 0.25 percent of the core as determined by the following method:

1. A conservative upper limit of 50 percent of rods experiencing clad damage may experience centerline melting (a total of 5 percent of the core).
2. Of the rods experiencing centerline melting, only a conservative maximum of the innermost 10 percent of the volume actually melts (0.5 percent of the core could experience melting).
3. A conservative maximum of 50 percent of the axial length of the rod would experience melting due to the power distribution (0.5 of the 0.5 percent of the core = 0.25 percent of the core).

4.7 CONTROL ROD EJECTION ACCIDENT

4.7.3.1 Reactor Core Releases

4.7.3.1.1 Release from Cladding Failures

Table 4.7-1 provides the radionuclides released from the gap of the 10% failed fuel.

Release fractions and transport fractions are consistent with Regulatory Guide 1.183, Appendix G and Table 3. To conform with this regulatory guidance, 5% of the core inventory of iodine and noble gas is assumed to be in the fuel-clad gap, excluding I-131 and Kr-85, where 8% and 10% are assumed, respectively. Additionally, Table 3 of Regulatory Guide 1.183 shows that 12% of the core cesium and rubidium should be assumed to be in the fuel-clad gap.

Table 4.7-1
10% Failed Fuel Gap Release Source
(Ci)

Isotope	CLB	AST	% Difference
I-131	1.10E+06	8.50E+05	-22.7%
I-132	1.60E+06	7.60E+05	-52.5%
I-133	2.30E+06	1.10E+06	-52.2%
I-134	2.50E+06	1.20E+06	-52.0%
I-135	2.10E+06	1.00E+06	-52.4%
Kr-83m	1.40E+05	7.00E+04	-50.0%
Kr-85m	3.00E+05	1.50E+05	-50.0%
Kr-85	3.70E+04	1.20E+04	-67.6%
Kr-87	5.50E+05	2.80E+05	-49.1%
Kr-88	7.90E+05	3.90E+05	-50.6%
Kr-89	9.70E+05	4.80E+05	-50.5%
Rb-88	-	9.50E+05	-
Rb-89	-	1.20E+06	-
Xe-131m	7.70E+03	5.50E+03	-28.6%
Xe-133m	3.30E+05	3.40E+04	-89.7%
Xe-133	2.30E+06	1.10E+06	-52.2%
Xe-135m	4.60E+05	2.10E+05	-54.3%
Xe-135	6.50E+05	2.80E+05	-56.9%
Xe-137	2.00E+06	9.50E+05	-52.5%
Xe-138	1.90E+06	9.00E+05	-52.6%
Cs-134	-	2.6E+05	-
Cs-136	-	7.6E+04	-
Cs-137	-	1.6E+05	-
Cs-138	-	2.4E+06	-

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4.7.3.1.2 Release from Fuel Melt

The material released as a result of the fuel melt is presented in Table 4.7-2. These are based upon the inventory in Table 4.2-9.

Table 4.7-2
0.25% Core Melt Source

Isotope	CLB	AST	% Difference
I-131	2.75E+05	2.8E+05	1.8%
I-132	4.00E+05	3.8E+05	-5.0%
I-133	5.75E+05	5.5E+05	-4.3%
I-134	6.25E+05	6.0E+05	-4.0%
I-135	5.25E+05	5.0E+05	-4.8%
Kr-83m	3.50E+04	3.5E+04	0.0%
Kr-85m	7.50E+04	7.3E+04	-2.7%
Kr-85	3.08E+03	3.0E+03	-2.6%
Kr-87	1.38E+05	1.4E+05	1.4%
Kr-88	1.98E+05	2.0E+05	1.0%
Kr-89	2.43E+05	2.4E+05	-1.2%
Rb-88	-	2.0E+05	-
Rb-89	-	2.5E+05	-
Xe-131m	1.93E+03	2.8E+03	45.1%
Xe-133m	8.25E+04	1.7E+04	-79.4%
Xe-133	5.75E+05	5.5E+05	-4.3%
Xe-135m	1.15E+05	1.1E+05	-4.3%
Xe-135	1.63E+05	1.4E+05	-14.1%
Xe-137	5.00E+05	4.8E+05	-4.0%
Xe-138	4.75E+05	4.5E+05	-5.3%
Cs-134	-	5.5E+04	-
Cs-136	-	1.6E+04	-
Cs-137	-	3.3E+04	-
Cs-138	-	5.0E+05	-

4.7.3.2 Reactor Coolant System Source Terms

4.7.3.2.1 RCS Iodine Concentrations

The initial RCS concentrations are assumed to be at a pre-existing iodine spike level of 60 μCi/gm as shown in Table 4.2-17.

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4.7.3.2.2 RCS Noble Gas Concentrations

The initial RCS noble gas concentrations corresponding to 1% failed fuel are given in Table 4.2-14.

4.7.3.2.3 RCS Cesium and Rubidium Concentrations

The RCS cesium and rubidium concentrations corresponding to a 1% failed fuel (Table 4.2-14). The Cs and Rb is assumed not to spike along with the iodines. Since the Cs and Rb are bound into particulate iodines, and since the iodines do not leave the water in the SGs or are appreciably from the RCB, the impact of this assumption is negligible.

4.7.3.3 Secondary System Source Terms

Releases from the secondary systems are only modeled for the Release from Secondary Systems scenario – not for the Release from Containment Building scenario.

4.7.3.3.1 Secondary System Iodine Concentrations

The initial secondary systems concentrations are assumed to be at the Technical Specification limit for the secondary side of 0.1 μCi/gm as shown in Table 4.2-19.

4.7.3.3.2 Secondary System Noble Gas Concentrations

The secondary systems noble gas concentrations corresponding to 1.0% failed fuel are given in Table 4.2-20.

4.7.3.3.3 Secondary System Cesium and Rubidium Concentrations

Cesium and rubidium are assumed to be bound with iodines as particulates. Therefore, there is no release of Cs or Rb from water in the steam generators.

4.7.4 Radiological Releases

In a CRE accident, nuclides released to the RCS from the fuel would be available for release to the environment through two pathways: into the containment and subsequent leakage to the environment; or, leakage to the environment via primary-to-secondary leakage and then steaming from the SGs. In order to bound the resultant doses from this accident, two cases are considered when analyzing the radioactive release:

- Scenario 1: RCB Leakage
  For this scenario, the ejected control rod is assumed to breach the reactor pressure vessel (RPV), effectively causing the equivalent of a small break loss of coolant accident. In this case, all activity from damaged fuel that has been mixed with the

4.7 CONTROL ROD EJECTION ACCIDENT

primary coolant of the reactor coolant system (RCS) leaks directly to the containment volume. This flashed release is assumed to instantaneously and homogeneously mix with the containment atmosphere and subsequently be available for release to the environment via an assumed containment leak rate limit. Credit for mitigation of the release by containment spray is not taken.

- Scenario 2: Steam Generator PORV Release
  All of the activity from damaged fuel is mixed with the RCS. The combined RCS activity then leaks to the secondary side through the steam generator (SG) tubes at a conservative rate of 1.0 gpm total leakage. The activity is then available for release to the environment by steaming of the SG PORVs and safeties.

This methodology maximizes the release of activity released to the environment. Therefore, certain secondary aspects of each scenario, which would be modeled if the scenarios were stand-alone methodologies, are not modeled. Specifically, the dose contribution of primary-to-secondary leakage and subsequent release to the environment via SG releases is not modeled for the Containment Building leakage scenario. Similarly, for the release through the Secondary Side scenario, the dose contribution of any release into the containment building and subsequent leakage to the environment is not modeled.

In reality, the release path would probably be a combination of these two release pathways, but the radiological consequences would be limited by the total doses determined using the independent scenarios.

4.7.4.1 Release from Containment Building Scenario

For this scenario, the ejected control rod is assumed to breach the reactor pressure vessel (RPV), effectively causing the equivalent of a small break loss of coolant accident. In this case, all activity from damaged fuel that has been mixed with the primary coolant of the reactor coolant system (RCS) leaks directly to the containment volume. This flashed release is assumed to instantaneously and homogeneously mix with the containment atmosphere and subsequently be available for release to the environment via an assumed containment leak rate limit.

The nuclides released to the containment are a mix of the core source term (Tables 4.7-1 and 4.7-2) and the curies contained in the RCS fluid released from the reactor pressure vessel. The total activity released to the containment and available for release is presented in Table 4.7-3.

No releases from the secondary side are assumed in this scenario.

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Table 4.7-3
Release From the RCB Scenario:
Total Activity Released into the RCB
(Ci)

Isotope	CLB	AST	% Difference
I-131	1.18E+06	9.3E+05	-21.2%
I-132	1.71E+06	8.8E+05	-48.5%
I-133	2.46E+06	1.2E+06	-51.2%
I-134	2.66E+06	1.4E+06	-47.4%
I-135	2.24E+06	1.2E+06	-46.4%
Kr-83m	1.75E+05	1.1E+05	-37.1%
Kr-85m	3.75E+05	2.2E+05	-41.3%
Kr-85	4.21E+04	1.7E+04	-59.6%
Kr-87	6.88E+05	4.2E+05	-39.0%
Kr-88	9.89E+05	5.9E+05	-40.3%
Kr-89	1.12E+06	7.2E+05	-35.7%
Rb-86	-	4.5E+00	-
Rb-88	-	1.2E+06	-
Rb-89	-	1.5E+06	-
Xe-131m	1.01E+04	9.0E+03	-10.9%
Xe-133m	4.17E+05	5.2E+04	-87.5%
Xe-133	2.94E+06	1.8E+06	-38.8%
Xe-135m	5.75E+05	3.2E+05	-44.3%
Xe-135	8.15E+05	4.2E+05	-48.5%
Xe-137	2.50E+06	1.4E+06	-44.0%
Xe-138	2.38E+06	1.4E+06	-41.2%
Cs-134	-	3.2E+05	-
Cs-136	-	9.3E+04	-
Cs-137	-	1.9E+05	-
Cs-138	-	2.9E+06	-

4.7.4.2 Release via Secondary Side Scenario

Activity that originates in the RCS is released to the secondary coolant by means of the primary-to-secondary coolant leak rate. This design basis leak rate value is 1.0 gpm for all SGs.

Releases to the environment are associated with the secondary coolant steaming from the SGs. Because of the release dynamic of the activity from the SG PORVs, Regulatory Guide 1.183 allows for a reduction in the amount of activity released to the environment based on partitioning of nuclides between the liquid and gas states of water for this release path. For iodine, the partition factor of 100 was taken directly from the suggested guidance. No particulates are

4.7 CONTROL ROD EJECTION ACCIDENT

assumed to be released. Because of their volatility, 100% of the noble gases are assumed to be released. The total activity released to the steam generators is presented in Table 4.7-4.

Table 4.7-4
Release From the Secondary Side Scenario:
Total Activity in the Steam Generators (RCS+SG)
(Ci)

Isotope	CLB	AST	% Difference
I-131	1.25E+06	1.0E+06	-20.0%
I-132	1.81E+06	9.7E+05	-46.4%
I-133	2.61E+06	1.4E+06	-46.4%
I-134	2.82E+06	1.5E+06	-46.8%
I-135	2.37E+06	1.4E+06	-40.9%
Kr-83m	1.75E+05	1.1E+05	-37.1%
Kr-85m	3.75E+05	2.2E+05	-41.3%
Kr-85	4.21E+04	1.7E+04	-59.6%
Kr-87	6.88E+05	4.2E+05	-39.0%
Kr-88	9.89E+05	5.9E+05	-40.3%
Kr-89	1.21E+06	7.2E+05	-40.5%
Rb-86	-	4.5E+00	-
Rb-88	-	1.2E+06	-
Rb-89	-	1.5E+06	-
Xe-131m	1.01E+04	9.0E+03	-10.9%
Xe-133m	4.17E+05	5.2E+04	-87.5%
Xe-133	2.94E+06	1.8E+06	-38.8%
Xe-135m	5.75E+05	3.2E+05	-44.3%
Xe-135	8.15E+05	4.2E+05	-20.0%
Xe-137	2.50E+06	1.4E+06	-46.4%
Xe-138	2.38E+06	1.4E+06	-46.4%
Cs-134	-	3.2E+05	-
Cs-136	-	9.3E+04	-
Cs-137	-	1.9E+05	-
Cs-138	-	2.9E+06	-

The methodology used to model steaming of activity through intact SG PORVs following the postulated CREA event assumes an average cumulative release rate through the SG PORVs that, for simplicity and conservatism, is reduced in steps. The steaming release from the PORVs and primary-to-secondary coolant leakage is postulated to end at 8 hours, when the RCS and secondary loop have reached equilibrium. Leakage via the MSIV above-seat drain orifices is assumed to continue for 36 hours. Table 4.7-5 shows the time steps and associated release rates.

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Table 4.7-5
Steam Released to the Environment
(lbm)

Time (Hours)	CLB	AST			% Difference
	PORV	PORV	Above Seat Drains	Total	
0 – 1.25	15,526,178	15,535,885	34,740	15,570,625	0.3%
1.25 - 36	0	0	965,772	965,772	
36 - 720	0	0	0	0	
Total	15,526,178	15,535,885	1,000,512	16,536,397	6.5%

A constant 7.72 lbm/sec leakage from the MSIV above seat drain orifices is assumed in the revised analyses (0 lbm/sec was assumed in the CLB).

4.7.5 Assumptions and Inputs

The following inputs and assumptions were used in the CREA analysis.

Assumptions Applicable to both Scenarios

1. The source term is based upon a power level of 4100 MW thermal, 5 w/o enrichment, and a 3 region core with equilibrium cycle core at end of life. The three regions have operated at a specific power of 39.3 MW/MTU for 509, 1018, and 1527 EFPD, respectively. The assumed power level is greater than the Rated Thermal Power of 3853 MWth plus a 0.6% measurement uncertainty.
2. The clad of 10% of the fuel is damaged during the initiation of this accident, and is assumed to have failed. Therefore, 10% of the core inventory of noble gases and iodines are released from the fuel gap (Regulatory Guide 1.183, Appendix H). Release fractions of other nuclide groups contained in the fuel gap are detailed in Table 3 of Regulatory Guide 1.183.
3. The amount of fuel melt is 0.25%. The 0.25% of the core is determined by the following method: a) A conservative upper limit of 50% of rods experiencing clad damage may experience centerline melting (a total of 5% of the core); b) Of the rods experiencing centerline melting, only a conservative maximum of the innermost 10 percent of the volume actually melts (0.5% of the core could experienced melting); and c) A conservative maximum of 50% of the axial length of the rod would experience melting due to the power distribution (half of the 0.5% of the core = 0.25% of the core)..
4. The initial RCS iodine concentrations are based on a pre-existing iodine spike to the Technical Specification limit of 60 μCi/gm and the initial Secondary system concentrations are based on a pre-existing iodine spike to the Technical Specification limit of 0.1 μCi/gm. Noble Gas concentrations are based on 1% failed fuel.
5. The Control Room ventilation system is assumed to transfer to the emergency mode of operation immediately upon the receipt of the safety injection signal (at t=0).
6. All releases to the atmosphere are assumed to be at ground level.

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7. The RCS density is 8.33 lbm/gal.

Assumptions Specific to the Release via Containment Leakage Scenario

8. One hundred percent of the noble gases and iodines in the gap of the fuel failed by the accident, plus 100% of noble gases and 25% of the iodines contained in the melted fuel fraction are assumed to be released to the containment in accordance with Appendix H of Regulatory Guide 1.183.
9. The containment free volume is 3.41E+6 ft^3 (+0.1% / -0.85%) or 3.38E+6 ft^3 to 3.41E+6 ft^3. A value of 3.38E+6 ft^3 is utilized for the dilution volume in containment and 3.41E+6 ft^3 is used for the leakage determination. Utilizing the minimum containment free volume conservatively maximizes the radioactive concentration in containment and using the maximum value for determining the containment leakage conservatively maximizes the containment leakage.
10. The activity released to the containment through the rupture in the reactor vessel head is assumed to mix instantaneously throughout the containment. No credit is assumed for removal of iodine in the containment due to containment sprays.
11. For the containment leakage case, all leakage is assumed to be at the Technical Specification limit of 0.3 percent per day for the first 24 hours and 0.15% per day thereafter.
12. Iodines released to the containment (from the fuel and RCS) are assumed to be 95% particulate, 4.85% elemental, and 0.15% organic (Regulatory Guide 1.183, Appendix H, position 4).

Assumptions Specific to the Release via the Secondary Side Scenario

13. One hundred percent of the noble gases and iodines in the gap of the fuel failed by the accident, plus 100% of noble gases and 50% of the iodines contained in the melted fuel fraction are assumed to be released to the reactor coolant in accordance with Appendix H of Regulatory Guide 1.183. Fractions of other nuclides released from the melted fuel are used from Table 2 of Regulatory Guide 1.183. Though these are described as LOCA values for fuel melt release, they are conservatively used for the other nuclide groups.
14. The activity released from the fuel from either the gap or from fuel pellets is assumed to be instantaneously mixed with the reactor coolant within the pressure vessel.
15. Primary-to-secondary leakage is conservatively modeled at a total of 1 gpm for all steam generators. Primary-to secondary leakage stops at 8 hours when the RCS and SG pressures are equalized.
16. Iodines released to the Secondary side (from the fuel and RCS) are assumed to be 95% particulate, 4.85% elemental, and 0.15% organic (Regulatory Guide 1.183, Appendix H, position 4).
17. This analysis assumes that iodine released from the steam generators to the environment is 4.2% elemental, 13.1% organic, and 82.7% particulate (see Section 4.2.5).
18. A partition coefficient of 100 is assumed for iodine, cesium, and rubidium released from the steam generators. (Regulatory Guide 1.183, Appendix G, Section 5.6) Organic iodine is not partitioned. Organic iodine is assumed to migrate directly to the steam space and become immediately available for release.

4.7 CONTROL ROD EJECTION ACCIDENT

19. Upon loss of offsite power, a total of 1.56×10^7 pounds of steam is discharged from the secondary system through the safety valves or PORVs for 4500 seconds following the accident. Steam release is terminated after this time. The minimum time to release the initial steam generator mass is 191 seconds. The rate of release necessary to release the total steam generator mass of 659,412 pounds in 191 seconds is 207,000 lbm/min. Assuming this flow rate is constant for 4500 seconds yields a total mass release of 1.56×10^7 pounds. Note that the total mass released is very conservative in relation to the initial SG mass.
20. Steam continues to be released from the orifices that replaced the MSIV above-seat isolation valves until 36 hours.
21. All releases are via the PORVs or safeties and the above-seat drains. These releases occur in the Isolation Valve Cubicle next to the PORVs. Therefore, the PORV-to-Control Room χ/Qs are used for the Control Room and TSC dose analyses.

Input parameters used for the CREA analysis are given in Tables 4.7-6 and 4.7-7. Conformance with Regulatory Guide 1.183 guidance addressing CREA analysis is provided in Attachment 6, Tables A and F.

Table 4.7-6
Inputs for CREA Analysis
Release from the RCB Scenario

Parameter	CLB	AST
Core power (for radiological source terms)	4100 MWt	
Core power (for steam releases)	3876 MWt (3853MWt + 0.6%)	
RCS density	8.33 lbm/gallon	
RCS Volume	2.6E+8 gm	2.658E+8 gm
Initial RCS Activities		
Iodines	Pre-existing spike to Tech Spec limit of 60 µCi/gm	
Noble Gases	1% Failed Fuel	
Initial Secondary Side Activities		
Iodines	Pre-existing spike to Tech Spec limit of 0.10 µCi/gm	
Noble Gases	1% Failed Fuel	
Fuel Melted by Accident	0.25% of core	
Fuel Clad Damage	10% of core	

4.7 CONTROL ROD EJECTION ACCIDENT

Table 4.7-6
Inputs for CREA Analysis
Release from the RCB Scenario

Parameter	CLB	AST
Iodine Species Released to Containment (elemental/organic/particulate)	91/4/5	4.85/0.15/95
Iodine Species Released From Containment[52] (elemental/organic/particulate)	91/4/5	4.85/0.15/95
Containment Free Volume For dilution of radionuclides For leakage rate	 3.38E+6 ft^3 3.41E+6 ft^3	 3.38E+6 ft^3 3.41E+6 ft^3
Containment Leak Rate 0-24 hrs 24hrs – 30 days	 0.3%/day 0.15%/day	
Dose Conversion Factors	Table 4.2-6	
Decay Constants and Decay Daughter Fractions	Table 4.2-7	
Offsite breathing rates	Table 4.2-1	
Offsite χ/Q's	Table 4.1-24	
Control Room HVAC Parameters	Table 4.2-3	
Control Room HVAC Flow Rates	Table 4.2-2	
TSC HVAC Parameters	Table 4.2-5	
TSC HVAC Flow Rates	Table 4.2-4	
Control Room and TSC χ/Q's	Table 4.1-37	

[52] Containment sprays are not used in this analysis.

4.7 CONTROL ROD EJECTION ACCIDENT

Table 4.7-7
Inputs for CREA Analysis
Release from the Secondary Side Scenario

Parameter	CLB	AST
Core power (for radiological source terms)	4100 MWt	
Core power level (for steam releases)	3876 MWt (3853MWt + 0.6%)	
RCS density	8.33 lbm/gallon	
RCS Mass	2.6E+8 gm	2.658E+8 gm
SG Mass	6.59E+5 lbm	659,412 lbm
Initial RCS Activities		
Iodines	Pre-existing spike to Tech Spec limit of 60 µCi/gm	
Noble Gases	1% Failed Fuel	
Initial Secondary Side Activities		
Iodines	Pre-existing spike to Tech Spec limit of 0.10 µCi/gm	
Noble Gases	1% Failed Fuel	
Primary-to-Secondary Leakage	1 gpm	
Fuel Melted by Accident	0.25% of core	
Fuel Clad Damage	10% of core	
Minimum time to release initial SG mass	191 seconds	
Steam Flow Rate to release initial SG mass	2.07E+5 lbm/min	
Maximum time for primary to secondary side pressure equilibrium	4500 seconds	
Steam Releases	Table 4.7-5	
Iodine Species Released to RCS (elemental/organic/particulate)	91%/4%/5%	4.85%/0.15%/95%
Iodine Partition Factors for Releases from the Secondary Side (elemental/organic/particulate)	100/100/100	100/1/100
Resulting Iodine Species Released from the Secondary Side to Environment % (elemental/organic/particulate)	91/4/5	4.2/13.1/82.7[53]
Steam Flow rate	1.574E+7 lbm/hr	
Dose Conversion Factors	Table 4.2-6	

[53] See Section 4.2.5

4.7 CONTROL ROD EJECTION ACCIDENT

Table 4.7-7
Inputs for CREA Analysis
Release from the Secondary Side Scenario

Parameter	CLB	AST
Decay Constants and Decay Daughter Fractions	Table 4.2-7	
Offsite breathing rates	Table 4.2-1	
Offsite χ/Q's	Table 4.1-24	
Control Room HVAC Parameters	Table 4.2-3	
Control Room HVAC Flow Rates	Table 4.2-2	
TSC HVAC Parameters	Table 4.2-5	
TSC HVAC Flow Rates	Table 4.2-4	
Control Room and TSC χ/Q's	Table 4.1-37	

4.7.6 Summary and Conclusions

Tables 4.7-8 through 4.7-10 below provides the analysis results. Per Standard Review Plan 15.4.8 (Reference 42), doses resulting from both release pathways are provided.

Table 4.7-8
CREA Doses from Containment Leakage
(rem TEDE)

Receptor	Dose
EAB (worst 2 hour)	0.86
LPZ	1.7
Control Room	2.4
TSC	2.3

Table 4.7-9
CREA Doses from Secondary Side Release
(rem TEDE)

Receptor	Dose
EAB (worst 2 hour)	0.55
LPZ	0.20
Control Room	0.41
TSC	0.40

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Table 4.7-10
Total CREA Dose Results
(rem TEDE)

Receptor	Dose	Limits
EAB (worst 2 hour)	1.4	6.3
LPZ	1.9	6.3
Control Room	2.8	5
TSC	2.7	5

The actual doses for the CREA would be a composite of the doses computed for the independent release paths via the containment building and through the secondary system releases. The primary-to-secondary leakage used in the CREA analyses bound the primary-to-secondary leakage limit in the Technical Specifications. Since the doses resulting from the secondary side release path are below the acceptance criteria for the CREA, the Technical Specification limit on primary-to-secondary leakage is acceptable.

Also, the CREA analyses do not take credit for containment sprays to mitigate the release of radionuclides from the containment building. Since the doses resulting from the containment leakage pathway are below the acceptance criteria for the CREA, a reduction of the pressure setpoint for actuation of the containment sprays is not necessary to obtain credit for spray removal of fission products.

Radiological doses resulting from a design basis CREA for a Control Room operator and a person located at the EAB or LPZ are less than the regulatory dose limits as given in 10CFR50.67 for the Control Room and TSC and in 10CFR50.67, as modified by Regulatory Guide 1.183 in Table 6 on Page 1.183-20, for the EAB and LPZ.

4.8 LOCKED ROTOR ACCIDENT

4.8 Locked Rotor Accident Radiological Assessment

4.8.1 Methodology Overview

The Locked Rotor Accident (LRA) analysis postulates the instantaneous seizure of a reactor coolant pump (RCP) rotor, where the reactor is tripped on the subsequent low flow signal. Following the trip, heat stored in fuel rods continues to pass into the reactor coolant, causing the coolant to expand. At the same time, heat transfer to the shell side of the SG is reduced, first because the reduced flow results in a decreased tube side film coefficient, and then because the reactor coolant in the tubes cools down while the shell side temperature increases (turbine steam flow is reduced to zero upon plant trip). The rapid expansion of the coolant in the reactor core, combined with the reduced heat transfer in the SGs, causes an insurgence of coolant into the pressurizer and a pressure increase throughout the RCS. This insurgence into the pressurizer causes a pressure increase, which in turn actuates the automatic spray system, opens the pressurizer PORVs, and also opens the pressurizer safety valves.

The pressurizer PORVs are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism, their pressure reducing effect and the pressure reducing effect of the spray is not included in the analysis.

This evaluation of the radiological consequences of a postulated seizure of a RCP rotor, i.e., an LRA, assumes that the reactor has been operating with a small percent of defective fuel (i.e., 1%) and leaking SG tubes (1.0 gpm total). Tube uncovery, due to failure of a feedwater isolation valve, is assumed for the steam generator in the loop with the locked rotor with a primary-to-secondary leak rate of 0.35 gpm of the 1.0 gpm total. The reactor is assumed to have been operating in this condition for sufficient time to establish equilibrium concentrations of radionuclides in the reactor coolant and secondary coolant.

It is conservatively assumed that, as a result of the postulated LRA, 10% of the fuel rods in the core undergo sufficient clad damage to result in the release of their gap activity.

As a result of this accident, radionuclides carried by the primary coolant to the SGs, via leaking SG tubes, are released to the environment via SG PORVs. A failure of the feedwater system is assumed to occur which causes tube uncovery in one SG.

The LRA dose assessment is modeled to calculate the doses due to the activity that was instantaneously released into the RCS from the postulated damaged fuel fraction, and the activity resulting from a pre-accident 60 μCi/gm DE I-131 spike. Leakage and steaming rates through the SG PORVs are used to model the transport of activity from the RCS to the environment. Prior to the accident, a secondary coolant specific activity equal to the Technical Specification limit of 0.1 μCi/gm DE I-131 equilibrium activity is assumed.

4.8 LOCKED ROTOR ACCIDENT

4.8.1.1 Comparison of Modeling with Regulatory Guide 1.183, Appendix E

Regulatory Guide 1.183 Appendix E, Position 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.*

Treatment for the LRA analysis:

In one SG assumed to experience tube uncovery, all of the primary-to-secondary leakage is assumed to flash to vapor and be released to the environment with no mitigation. In the other SGs, the primary-to-secondary leakage is assumed to mix with the secondary water without flashing.

Regulatory Guide 1.183 Appendix E, Position 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.

Treatment for the LRA analysis:

This assumption is not used. It is assumed that the primary-to-secondary leakage does not flash in the SGs that do not experience tube uncovery. The nuclides in the primary-to-secondary leakage are added to the bulk fluid in these SGs.

Regulatory Guide 1.183 Appendix E, Position 5.5.3:

The leakage that does not immediately flash is assumed to mix with the bulk water.

Treatment for the LRA analysis:

It is assumed that the primary-to-secondary leakage does not flash in the SGs that do not experience tube uncovery. The nuclides in the primary-to-secondary leakage are added to the bulk fluid in the intact SGs.

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Regulatory Guide 1.183 Appendix E, Position 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.*

Treatment for the LRA analysis:

A partition coefficient of 100 is assumed for elemental iodine released from the bulk fluid in the steam generators. Organic iodine is not partitioned. Organic iodine is assumed to migrate directly to the steam space and become immediately available for release.

Regulatory Guide 1.183 Appendix E, Position 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.*

Treatment for the LRA analysis:

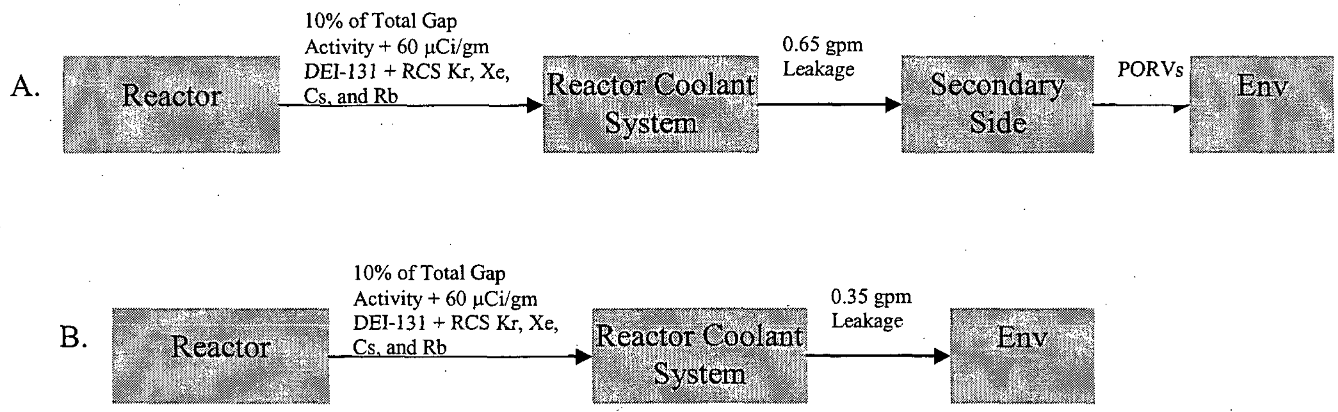
Tube uncovery does not occur following this event and the subsequent reactor trip. However, a single failure in the feedwater system that isolated the feedwater entering one SG is assumed to occur. Tube uncovery is assumed for the steam generator in the loop with the feedwater isolation valve malfunction. Primary-to-secondary leakage from this steam generator is assumed to be 0.35 gpm of the total 1.0 gpm. This leakage is assumed to flash and be immediately released to the environment.

4.8.2 Analytical Model

The RADTRAD computer code is used to calculate the offsite, Control Room, and TSC doses. The first model (A, in Figure 4.8-1) calculates the dose resulting from secondary side steam releases due to primary-to-secondary leakage to the three steam generators that have covered tubes. The second model (B) calculates the dose due to primary-to-secondary leakage in the steam generator with uncovered tubes. Section 5.5.1 in Appendix E Regulatory Guide 1.183 states that during periods of steam generator dryout; all primary-to-secondary leakage will flash and be released directly to the environment. The release paths are shown below.

4.8 LOCKED ROTOR ACCIDENT

Figure 4.8-1: LRA RADTRAD Model



4.8.3 Radiological Source Term

For conservatism, the LRA core source terms are those associated with a DBA power level of 4100 MWth, which is greater than the RTP of 3853 MWth plus a 0.6% measurement uncertainty.

The instantaneous seizure of the RCP rotor associated with the LRA results in a small percentage of fuel damage. The dose analysis for this event conservatively assumes 10% fuel damage. The design basis of this accident assumes that no fuel melt is postulated to occur. Therefore, the source term available for release is associated with this fraction of damaged fuel and the fraction of core activity existing in the gap, plus the iodine in the RCS due to a design basis pre-accident 60 μCi /gm DE I-131 spike, and the noble gas activity associated with assumed 1% fuel defects.

Release fractions and transport fractions are consistent with Regulatory Guide 1.183, Appendix G and Table 3. To conform with this regulatory guidance, 5% of the core inventory of iodine and noble gas is assumed to be in the fuel-clad gap, excluding I-131 and Kr-85, where 8% and 10% are assumed, respectively. Additionally, Table 3 of Regulatory Guide 1.183 shows that 12% of the core cesium and rubidium should be assumed to be in the fuel-clad gap.

The source term model also consists of the 0.1 μCi/gm DE I-131 equilibrium secondary coolant activity concentration, consistent with the TS requirements.

4.8 LOCKED ROTOR ACCIDENT

4.8.3.1 Reactor Coolant System Source Term

4.8.3.1.1 RCS Iodine Concentrations

The RCS iodine concentrations for a pre-existing iodine spike to 60 μCi/gm are given in Table 4.2-17.

4.8.3.1.2 RCS Noble Gas Concentrations

The RCS noble gas concentrations for 1% failed fuel are given in Table 4.2-14.

4.8.3.1.3 RCS Cesium and Rubidium Concentrations

No spiking of Cs or Rb is assumed. The RCS Cs and Rb concentrations corresponding to 1% failed fuel are used (Table 4.2-20).

4.8.3.2 Secondary System Source Terms

4.8.3.2.1 Secondary System Iodine Concentrations

The secondary systems iodine concentrations corresponding to the Technical Specification limit of 0.10 μCi/gm are given in Table 4.2-19.

4.8.3.2.2 Secondary System Noble Gas Concentrations

The secondary systems noble gas concentrations corresponding to 1.0% failed fuel are given in Table 4.2-20.

4.8.3.2.3 Secondary System Cesium and Rubidium Concentrations

The secondary system Cs and Rb concentrations corresponding to 1% failed fuel are used (Table 4.2-20). Cesium and rubidium are assumed to be bound with iodines as particulates. Therefore, there is no release of Cs or Rb from water in the steam generators.

4.8.3.3 Fuel Pin Gap Source

The accident release inventory is derived from the core isotopic inventory. This inventory is corrected to the total gap inventory in order to calculate the release from a failure of 10% of the fuel rods. The gap fractions are from Regulatory Guide 1.183, Table 3. The gap release source is presented in Table 4.8-1.

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Table 4.8-1
10% Gap Release Source
(Ci)

Isotope	CLB	AST	% Difference
I-131	1.10E+06	8.5E+05	-22.7%
I-132	1.60E+06	7.6E+05	-52.5%
I-133	2.30E+06	1.1E+06	-52.2%
I-134	2.50E+06	1.2E+06	-52.0%
I-135	2.10E+06	1.0E+06	-52.4%
Kr-83m	1.40E+05	7.0E+04	-50.0%
Kr-85m	3.00E+05	1.5E+05	-50.0%
Kr-85	3.70E+04	1.2E+04	-67.6%
Kr-87	5.50E+05	2.8E+05	-49.1%
Kr-88	7.89E+05	3.9E+05	-50.6%
Kr-89	9.70E+05	4.8E+05	-50.5%
Rb-86		0	
Rb-88		9.5E+05	
Rb-89		1.2E+06	
Xe-131m	7.70E+03	5.5E+03	-28.6%
Xe-133m	3.30E+05	3.4E+04	-89.7%
Xe-133	2.30E+06	1.1E+06	-52.2%
Xe-135m	4.60E+05	2.1E+05	-54.3%
Xe-135	6.50E+05	2.8E+05	-56.9%
Xe-137	2.00E+06	9.5E+05	-52.5%
Xe-138	1.90E+06	9.0E+05	-52.6%
Cs-134		2.6E+05	
Cs-136		7.6E+04	
Cs-137		1.6E+05	
Cs-138		2.4E+06	

The values in Table 4.8-1 are based on the reactor core sources in Table 4.2-9. The AST values use the gap fractions from Regulatory Guide 1.183, Table 3.

4.8 LOCKED ROTOR ACCIDENT

4.8.3.4 Total Source Available for Release

The total source available for release is presented in Table 4.8-2.

Table 4.8-2
Total Source Available for Release (RCS+Sec)
(Ci)

Isotope	CLB	AST	% Difference
I-131	1.11E+06	8.6E+05	-22.5%
I-132	1.61E+06	7.8E+05	-51.6%
I-133	2.32E+06	1.1E+06	-52.6%
I-134	2.50E+06	1.2E+06	-52.0%
I-135	2.11E+06	1.1E+06	-47.9%
Kr-83m	1.40E+05	7.0E+04	-50.0%
Kr-85m	3.00E+05	1.5E+05	-50.0%
Kr-85	3.91E+04	1.4E+04	-64.2%
Kr-87	5.50E+05	2.8E+05	-49.1%
Kr-88	7.91E+05	3.9E+05	-50.7%
Kr-89	9.70E+05	4.8E+05	-50.5%
Rb-86		4.5E+00	
Rb-88		9.5E+05	
Rb-89		1.2E+06	
Xe-131m	8.21E+03	6.2E+03	-24.5%
Xe-133m	3.34E+05	3.5E+04	-89.5%
Xe-133	2.36E+06	1.2E+06	-49.2%
Xe-135m	4.60E+05	2.1E+05	-54.3%
Xe-135	6.52E+05	2.8E+05	-57.1%
Xe-137	2.00E+06	9.5E+05	-52.5%
Xe-138	1.90E+06	9.0E+05	-52.6%
Cs-134		2.6E+05	
Cs-136		7.7E+04	
Cs-137		1.6E+05	
Cs-138		2.4E+06	

The chemical form of the iodine in the RCS is 95% CsI, 4.85% elemental, and 0.15% organic. The chemical form of the iodine released from the secondary side is 4.2% elemental, 13.1% organic, and 82.7% particulate (Section 4.2.5).

4.8 LOCKED ROTOR ACCIDENT

4.8.4 Radiological Releases

Activity that originates in the RCS is released to the secondary coolant by means of the primary-to-secondary coolant leak rate. This design basis leak rate value is 1.0 gpm for all SGs. For the SG on the loop with the locked RCP rotor, tube uncovery is assumed due to a feedwater isolation valve malfunction. Primary-to-secondary leakage from this steam generator is assumed to be 0.35 gpm of the total.

Releases to the environment are associated with the secondary coolant steaming from the SGs. Because of the release dynamic of the activity from the SG PORVs, Regulatory Guide 1.183 allows for a reduction in the amount of activity released to the environment based on partitioning of nuclides between the liquid and gas states of water for this release path. For iodine, the partition factor of 100 was taken directly from the suggested guidance. No particulates are assumed to be released. Because of their volatility, 100% of the noble gases are assumed to be released.

The methodology used to model steaming of activity through intact SG PORVs following the postulated LRA event assumes an average cumulative release rate through the SG PORVs that, for simplicity and conservatism, is reduced in steps. The steaming release from the PORVs and primary-to-secondary coolant leakage is postulated to end at 8 hours, when the RCS and secondary loop have reached equilibrium. Leakage via the MSIV above-seat drain orifices is assumed to continue for 36 hours. Table 4.8-3 below shows the time steps and associated release rates.

Table 4.8-3
Steam Released to the Environment
(lbm)

Time (Hours)	CLB	AST			% Difference
	PORV	PORV	Above Seat Drains	Total	
0 – 2	455,047	640,000	55,584	695,584	52.9%
2 – 8	1,137,757	1,120,000	166,752	1,286,752	13.1%
8 – 12	0	0	111,168	111,168	-
12 – 36	0	0	667,008	667,008	-
36 – 720	0	0	0	0	-

A constant 7.72 lbm/sec leakage from the MSIV above seat drain orifices is assumed in the revised analyses (0 lbm/sec was assumed in the CLB).

4.8 LOCKED ROTOR ACCIDENT

4.8.5 Assumptions and Inputs

The following inputs and assumptions were used in the LRA analysis.

1. The source term is based upon a power level of 4100 MW thermal, 5 w/o enrichment, and a three region core with equilibrium cycle core at end of life. The three regions have operated at a specific power of 39.3 MW/MTU for 509, 1018, and 1527 EFPD, respectively. The assumed power level is greater than the Rated Thermal Power of 3853 MWth plus a 0.6% measurement uncertainty.
2. The initial activity in the reactor coolant is based upon an iodine spike which has raised the reactor coolant concentration to 60 μCi/gm of dose equivalent I-131. Noble gas activity is based on 1% failed fuel.
3. Prior to the accident, the secondary coolant specific activity is equal to the Technical Specification limit of 0.10 μCi/gm dose equivalent I-131. This DEI activity is given in Table 4.2-19.
4. Ten percent (10%) fuel failure is assumed to occur. The activity released from the pellet-to-clad gap of the failed fuel is assumed to be instantaneously mixed with the reactor coolant system, per Regulatory Guide 1.183. No fuel melting occurs.
5. A feedwater system malfunction caused by the closure of a feedwater isolation valve is postulated, resulting in tube uncovery in that SG.
6. Primary-to-secondary leakage through the steam generator tubes prior to the accident and during the first 8 hours following the transient is 1 gpm. Eight hours after the accident, the residual heat removal system starts and primary-to-secondary leakage is stopped. Primary-to-secondary leakage is conservatively modeled at 0.65 gpm for the three steam generators with covered tubes and at 0.35 gpm for the steam generator with uncovered tubes.
7. A partition coefficient of 100 is assumed for elemental iodine released from the steam generators. (Regulatory Guide 1.183, Appendix G, Position 5.5.4) Organic iodine is not partitioned. Organic iodine is assumed to migrate directly to the steam space and become immediately available for release.
8. The Control Room and TSC ventilation systems are assumed to transfer to the emergency mode of operation immediately after the initiation of this accident. This assumption is countered by the assumption of an additional (second) single failure of a train of the Control Room Emergency HVAC system, specifically the clean-up (recirculation) filters.
9. Offsite power is lost; Main Steam condensers are not available for steam dump.
10. Eight hours after the accident, the residual heat removal system is in operation and no further steam containing radionuclides are released from steam generators to the environment except the leakage through the MSIV above seat drain orifices. The release through the orifices continues until 36 hours after the start of the accident.
11. All releases occur via the PORVs or safeties and the above-seat drain orifices in the Isolation Valve Cubicle next to the PORVs. Therefore, the PORV-to-Control Room χ/Qs are used for the Control Room and TSC dose analyses.
12. The reactor coolant density is 8.33 lbm/gal (14.7 psia, 70 °F).

4.8 LOCKED ROTOR ACCIDENT

Input parameters used for the LRA analysis are given in Table 4.8-4. Conformance with Regulatory Guide 1.183 guidance addressing LRA analysis is provided in Attachment 6, Tables A and G.

Table 4.8-4
Inputs for LRA Analysis

Parameter	CLB	AST
Core power (for radiological source terms)	4100MWt	
Core power (for steam releases)	3876 MWt (3853MWt + 0.6%)	
RCS density	8.33 lbm/gallon	
RCS Mass	2.658E+8 gm	
SG Mass	659,412 lbm 2.991E+08 gm	
Primary-to-Secondary Leakage SGs w/o tube uncovery SG w/tube uncovery	 1.0 gpm N/A	 0.65 gpm 0.35 gpm
Release from SGs	Table 4.8-3	
Release from Above (MSIV) Seat Drains SGs w/o tube uncovery SG w/tube uncovery	 N/A N/A	 5.79 lbm/sec 1.93 lbm/sec
Steam Flow rate	1.574E+7 lbm/hr	
Iodine Partition Factors for Releases from the Secondary Side (elemental/organic/particulate)	100/100/100	100/1/100
Resulting Iodine Species Released from the Secondary Side to Environment % (elemental/organic/particulate)	91/4/5	4.2/13.1/82.7[54]
Dose Conversion Factors	Table 4.2-6	
Decay Constants and Decay Daughter Fractions	Table 4.2-7	
Offsite breathing rates	Table 4.2-1	
Offsite χ/Q's	Table 4.1-24	
Control Room HVAC Parameters	Table 4.2-3	
Control Room HVAC Flow Rates	Table 4.2-2	

[54] See Section 4.2.5

4.8 LOCKED ROTOR ACCIDENT

Table 4.8-4
Inputs for LRA Analysis

Parameter	CLB	AST
TSC HVAC Parameters	Table 4.2-5	
TSC HVAC Flow Rates	Table 4.2-4	
Control Room and TSC χ/Q's	Table 4.1-37	

4.8.6 Summary and Conclusions

Radiological doses resulting from a design basis LRA for a Control Room operator and a person located at the EAB or LPZ are to be less than the regulatory dose limits as given in 10CFR50.67 for the Control Room and TSC and in 10CFR50.67, as modified by Regulatory Guide 1.183 in Table 6 on Page 1.183-20, for the EAB and LPZ.

Table 4.8-5 provides the results for the LRA analysis.

Table 4.8-5
LRA Dose Results
(rem TEDE)

Receptor	Dose	Limits
EAB (worst 2 hour)	1.9	2.5
LPZ	1.5	2.5
Control Room	3.9	5
TSC	3.7	5

These calculated doses above are well below their respective acceptance criteria, so it is verified that the LRA is sufficiently mitigated.

4.9 NUREG-0737 Evaluations

As part of the DBA LOCA analysis, radiation levels from contained sources (containment structure and Control Room and Technical Support Center (TSC) filters) were evaluated. These evaluations were used to determine if an impact on the following areas covered by NUREG-0737 would occur as a result of an increase in the associated radiation levels:

- CLB radiological dose analyses for post-accident vital area access and post-accident sampling (NUREG-0737, Item II.B.2 and Item II.B.3),
- CLB radiological dose analyses for the post-accident containment high range radiation monitors (NUREG-0737, Item II.F.1), and
- CLB control room post-accident radiological dose analyses for emergency support facility upgrades and control room habitability (NUREG-0737, Items III.A.1.2 and III.D.3.4).

Evaluations

- Post Accident Vital Area Access and Sampling - Post-accident personnel missions resulting in mission doses (including post-accident sampling) have been previously identified. The implementation of the AST methodology does not result in any new operator missions. Plant calculations used in support of plant post-accident vital area access (prepared in accordance with NUREG-0737, Items II.B.2 and II.B.3) were judged to be unaffected based on an assessment of AST vs. TID-14844 contained sources.

  The results of the assessment of post-accident shine due to contained sources in various plant locations is that the current calculated doses (based on TID-14844 source terms) bound the corresponding doses that would be calculated based on the AST. This conclusion is reached on the basis of (1) a comparison of the post-LOCA containment airborne source terms (MeV/sec as a function of time for different photon energy groups) using STPEGS-specific airborne activity removal rates, (2) a general comparison of the potential for post-LOCA waterborne source terms (total MeV/sec as a function of time as well as MeV/sec as functions of time for photons with energies greater than 1.5 MeV), and (3) a comparison of the post-SGTR source term (MeV/sec for different photon energy groups) for the location and time where post-SGTR access would be required.

- Post Accident Sampling System - The requirements of NUREG 0737 for Post Accident Sampling System (PASS) were deleted as part of Amendment No. 133 to Facility Operating License No. NPF-76 and Amendment No. 122 to Facility Operating License No. NPF-80 issued November 7, 2001 via Document ST-AE-NOC-01000894 South Texas Projects, Units 1 and 2 - Issuance of Amendments on the Elimination of Requirements for Post Accident Sampling (TAC Nos. MB2900 and MB2904).

- Post-Accident Radiation Monitor - The CLB analysis for the containment high range radiation monitors used to monitor post-accident primary containment radiation levels use a source term different from either TID-14844 or the AST. Therefore, there is no impact of AST implementation on the containment high range monitor evaluation. (NUREG-0737, Item II.F.1)

- Control Room Radiation Protection - The doses to Control Room operators were specifically calculated using AST for the Design Basis Accidents described in this submittal. Results are presented with each respective accident description. (NUREG-0737, Item III.D.3.4).
- Technical Support Center Radiation Protection - The doses to TSC personnel were specifically calculated using AST for the Design Basis Accidents described in this submittal. Results are presented with each respective accident description. (NUREG-0737, Item III.A.1.2).
- Radioactive Sources Outside the Primary Containment - The DBA LOCA Control Room/TSC dose analysis, as well as that for offsite doses, considers the effects of ESF leakage outside the primary containment and (for the Control Room and TSC dose analyses only) the shine contribution from the containment and other source term bearing systems and/or components (NUREG - 0737, Item III.D.1.1).

4.10 Conclusion

The proposed changes provide a source term for STP that will result in a more accurate assessment of the DBA radiological doses. The revised radiological dose to the control room operator allows for a revised air unfiltered in-leakage assumption that provides a conservative margin over that determined by air in-leakage testing. Changes related to the applicability requirements during movement of irradiated fuel assemblies use insights from TSTF-51.

Adequate defense-in-depth is maintained by the requirements for radioactive decay and water level. Technical Specification requirements for systems needed for decay heat removal, or to mitigate potential reactor vessel drain down events, or the requirements to maintain high water levels over irradiated fuel are not impacted by the proposed amendment.

The proposed amendment also deletes the APPLICABILITY requirements of CORE ALTERATIONS for selected TS, since the only accident postulated to occur during CORE ALTERATIONS that results in radioactive release is the fuel handling accident.

Shutdown safety controls are provided during periods when fuel is being handled. These controls address (1) procedures to assess the impact of removing systems from service during shutdown conditions, (2) the ability to implement prompt methods to close both the Reactor Containment Building and/or the Fuel Handling Building(s) in the event of a FHA, and (3) controls to avoid unmonitored releases.

Implementation of the AST as the plant radiological consequence analyses licensing basis requires a licensing amendment request pursuant to the requirements of 10 CFR 50.67. Radiological dose analyses were performed for the DBA LOCA, FHA, MSLB, SGTR, CREA, and LRA using conservative assumptions. Doses calculated with the AST for accidents involving damaged fuel reflect delayed and/or reduced activity releases (relative to those of the TID-14844-based CLB) to the containment and to the FHB, as applicable. Offsite, Control Room, and TSC doses remain well below regulatory requirements.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

5.1.1 Overview

On December 23, 1999, the NRC issued the Final Rule on "Use of Alternate Source Terms at Operating Reactors." The Final Rule, issued under 10 CFR 50.67, "Accident Source Term", allows holders of operating licenses issued prior to January 10, 1997, to voluntarily replace the traditional source term used in design basis accident analyses with alternative source terms. This action would allow interested licensees to pursue cost beneficial licensing actions to reduce unnecessary regulatory burden without compromising the margin of safety of the facility.

Based on the above rule and in accordance with 10 CFR 50.67 and 10 CFR 50.90, "Application for amendment of license or construction permit." STPNOC is requesting an amendment to Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF-76 and NPF-80 for STP, Units 1 and 2. The proposed changes are requested to support application of an alternative source term (AST) methodology, with the exception that Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," will continue to be used as the radiation dose basis for equipment qualification. The proposed AST methodology conforms to the guidance in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000, except where alternate methods for complying with the specified portions of the NRC's regulations have been used as allowed by RG 1.183. The AST analyses were also performed in accordance with the guidance in Standard Review Plan Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms."

In support of a full-scope implementation of the AST methodology, STP has performed radiological consequence analyses for the following six design basis accidents (DBAs) that result in control room and offsite exposure as specified in RG 1.183.

- Loss of Coolant Accident (LOCA)
- Fuel Handling Accident (FHA)
- Main Steam Line Break (MSLB)
- Steam Generator Tube Rupture (SGTR)
- Control Rod Ejection Accident (CREA)
- Locked Rotor Accident (LRA)

The proposed changes related to the applicability requirements during movement of irradiated fuel assemblies are based on insights from Technical Specification Task Force Traveler (TSTF)-51, "Revise Containment Requirements During Handling of Irradiated Fuel and Core Alterations," Revision 2. The NRC approved TSTF-51 on July 31, 2003. TSTF-51 changes the TS operability requirements for engineered safety features such that they are not required to be operable after sufficient radioactive decay has occurred to ensure that offsite doses remain within limits.

Proposed changes to the current licensing basis, justified by the AST analyses, include the following items:

- The use of updated meteorological data to calculate onsite and offsite atmospheric dispersion
- Relies on less filtration
  - ➢ No credit taken for Fuel Handling Building Exhaust Air Ventilation filtration
  - ➢ No credit taken for Control Room Ventilation makeup filtration
  - ➢ No credit taken for either Control Room Ventilation makeup or recirculation cleanup filtration for the Fuel Handling Accident
- Containment isolation capability is no longer required to mitigate a FHA
- Analysis of only a single limiting FHA rather than one analysis for an FHA inside containment and a second analysis for an FHA in the fuel handling building (FHB)
- Revised control room unfiltered in-leakage assumption.

5.1.2 Criteria

According to 10 CFR 50.92, "Issuance of amendment," paragraph (c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or
(2) Create the possibility of a new or different kind of accident from any accident previously evaluated, or
(3) Involve a significant reduction in a margin of safety.

In support of this determination, an evaluation of each of the three criteria set forth in 10 CFR 50.92 is provided below regarding the proposed license amendment.

1. **The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.**

   The implementation of AST assumptions has been evaluated in revisions to the analyses of the following limiting DBAs.

   - Loss-of-Coolant Accident
   - Fuel Handling Accident
   - Control Rod Ejection Accident

- Locked Rotor Accident
- Main Steam Line Break Accident
- Steam Generator Tube Rupture Accident

Based upon the results of these analyses and evaluations, it has been demonstrated that, with the requested changes, the dose consequences of these limiting events satisfies the dose limits in 10 CFR 50.67 and are within the regulatory guidance provided by the NRC for use with the AST methodology. The AST is an input to calculations used to evaluate the consequences of an accident and does not affect the plant response or the actual pathway of the activity released from the fuel. Therefore, it is concluded that AST does not involve a significant increase in the consequences of an accident previously evaluated.

Implementation of AST provides for elimination of the Fuel Handling Building ventilation system filtration TS requirements and elimination of Control Room ventilation filtration TS requirements in Modes 5 or 6. It also eliminates containment integrity TS requirements while handling irradiated fuel and during core alterations. The equipment affected by the proposed changes is mitigative in nature and relied upon after an accident has been initiated. The affected systems are not accident initiators; and application of the AST methodology is not an initiator of a design basis accident.

Elimination of the requirement to suspend operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or required boron concentration if the control room ventilation system is inoperable in Modes 5 or 6 does not increase the probability of an accident because the proposed change does not affect the design and operational controls to prevent dilution events. These same design and operational controls prevent a loss of SHUTDOWN MARGIN or a boron dilution event so that radiological consequences from these events are precluded.

The proposed changes do not involve physical modifications to plant equipment and do not change the operational methods or procedures used for moving irradiated fuel assemblies. The proposed changes do not affect any of the parameters or conditions that could contribute to the initiation of any accidents. Relaxation of operability requirements during the specified conditions will not significantly increase the probability of occurrence of an accident previously analyzed. Since design basis accident initiators are not being altered by adoption of the AST, the probability of an accident previously evaluated is not affected.

Administrative changes to delete a footnote from Technical Specification surveillance requirement 4.7.7.e.3) and a note from ACTION 20 of Technical Specification Table 3.3-3, in which the provisions of the notes have expired, does not impact the probability or consequences of an accident previously evaluated.

Based on the above discussion, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated**

The proposed changes do not involve a physical change. The change will allow the automatic start feature of systems no longer credited in the accident analyses for mitigation to be disabled through the STPNOC modification process. Implementation of AST provides increased operating margins for filtration system efficiencies. Application of AST provides for relaxation of certain Control Room ventilation system filtration requirements. The Fuel Handling Building filtration and holdup is no longer credited in the AST analyses. Therefore, the Fuel Handling Building Exhaust Air Ventilation system is no longer required in the Technical Specifications. It also relaxes containment integrity requirements while handling irradiated fuel and during core alterations.

Elimination of the requirement to suspend operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or required boron concentration if the control room ventilation system is inoperable in Mode 5 or Mode 6 does not create the possibility of a new or different kind of accident because these events have already been analyzed in the safety analysis with a conclusion that adequate measures exist to prevent these events.

Similarly, the proposed changes do not require any physical changes to any structures, systems or components involved in the mitigation of any accidents. Therefore, no new initiators or precursors of a new or different kind of accident are created. New equipment or personnel failure modes that might initiate a new type of accident are not created as a result of the proposed changes.

Administrative changes to delete a footnote from Technical Specification surveillance requirement 4.7.7.e.3) and a note from ACTION 20 of Technical Specification Table 3.3-3, in which the provisions of the notes have expired, does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Based on the above discussion, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **The proposed change does not involve a significant reduction in a margin of safety.**

Approval of a change from the original source term methodology (i.e., TID 14844) to an AST methodology, consistent with the guidance in RG 1.183, will not result in a significant reduction in the margin of safety. The safety margins and analytical conservatisms associated with the AST methodology have been evaluated and were found acceptable. The results of the revised DBA analyses, performed in support of the proposed changes, are subject to specific acceptance criteria as specified in RG 1.183. The dose consequences of these DBAs remain within the acceptance criteria presented in 10 CFR 50.67 and RG 1.183.

Elimination of the requirement to suspend operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or required boron concentration if the control room ventilation system is inoperable in Mode 5 or Mode 6 does not result in a reduction in a margin to safety because adequate measures exist to preclude radiological consequences from these events.

The proposed changes continue to ensure that the doses at the exclusion area boundary (EAB) and low population zone boundary (LPZ), as well as the Control Room and Technical Support Center, are within the specified regulatory limits.

Administrative changes to delete a footnote from Technical Specification surveillance requirement 4.7.7.e.3) and a note from ACTION 20 of Technical Specification Table 3.3-3, in which the provisions of the notes have expired, does not impact the margin of safety.

Therefore, based on the above discussion, the proposed changes do not involve a significant reduction in a margin of safety.

Conclusion

Based on the above discussion, it has been determined that the requested TS changes do not involve a significant increase in the probability or consequences of an accident previously evaluated; or create the possibility of a new or different kind of accident from any accident previously evaluated; or involve a significant reduction in a margin of safety. Therefore, the requested license amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c).

5.2 Applicable Regulatory Requirements/Criteria

The NRC's traditional methods for calculating the radiological consequences of design basis accidents (i.e., prior to adopting the AST methodology) are described in a series of Regulatory Guides (RGs) and Standard Review Plan (SRP) chapters. That guidance was developed to be consistent with the TID-14844 source term and the whole body and thyroid dose guidelines stated in 10 CFR 100.11. Many of those analysis assumptions and methods are inconsistent with the AST methodology and with the Total Effective Dose Equivalent (TEDE) criteria provided in 10 CFR 50.67. RG 1.183 provides assumptions and methods that are acceptable to the NRC staff for performing design basis radiological analyses using an AST approach. This guidance supersedes corresponding radiological analysis assumptions provided in the previous Regulatory Guides and SRP chapters when used in conjunction with an approved AST methodology and the TEDE criteria provided in 10 CFR 50.67.

Due to the comprehensive nature of RG 1.183, Attachment 6, "Regulatory Guide Conformance Tables," were developed to show how each section of the RG 1.183 guidance is being addressed.

The NRC also published a new SRP section to address AST; i.e., SRP Section 15.0.1, Revision 0, "Radiological Consequence Analyses Using Alternative Source Terms." This SRP section is consistent with the guidance found in RG 1.183. The plant-specific information provided in this license amendment request is also consistent with the guidance found in SRP 15.0.1.

10 CFR 50.36, "Technical Specifications

10 CFR 50.36 specifies the items that should be included in the Technical Specifications. Specifically, Part 50.36(c)(2)(ii) states that a technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:

(a) *Criterion 1.* Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

The systems proposed for removal from the Technical Specifications are not used to detect degradation of the reactor coolant pressure boundary. The systems are not an initial condition of a design basis accident or transient analysis that either assumes the failure of, or presents a challenge to the integrity of a fission product barrier.

(b) *Criterion 2.* A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The systems proposed for removal from the Technical Specifications are not process variables, design features, or pose any operating restrictions that are an initial condition of a design basis accident or transient analysis that either assumes the failure of, or presents a challenge to the integrity of a fission product barrier.

(c) *Criterion 3.* A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The results of the revised accident analyses based on the alternative source term no longer credits the items proposed for removal from the Technical Specifications as accident mitigation features. The items proposed for removal from the Technical Specifications do not present a challenge to the integrity of a fission product barrier. These systems are not primary success path for mitigation of the DBA.

(d) *Criterion 4.* A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The requirements proposed for relocation from the TS, in the Modes specified, do not contribute to the conditional probability of core damage or conditional probability of a large release. The requirements being relocated do not contain constraints of prime importance in limiting the likelihood or severity of the accident sequences that are commonly found to dominate risk. The operability of the system is not risk significant.

Safety Margins and Defense in Depth

Regulatory Guide 1.183 states that proposed uses of an AST and the associated proposed facility modifications and changes to procedures should be evaluated to determine whether the proposed changes are consistent with the principle that sufficient safety margins are maintained, including a margin to account for analysis uncertainties. Specific values and limits contained in the technical specifications and the response times for the safety system assumed in the accident analyses are not changed. Caution has been taken to ensure that the dose analyses have not been "tuned" to a specific set of accident progression assumptions so that the assumptions remain conservative for the accident sequences considered. The dose consequence results of the accident analyses remain well below regulatory limits.

Regulatory Guide 1.183 states that proposed uses of an AST and the associated proposed facility modifications and changes to procedures should be evaluated to determine whether the proposed changes are consistent with the principle that adequate defense in depth is maintained to compensate for uncertainties in accident progression and analysis data. System redundancy, independence, and diversity features are not changed for those safety systems credited in the accident analyses. No new programmatic compensatory activities or reliance on manual operator actions is required to implement this change. For those systems that are no longer credited in the accident analyses for mitigation, programmatic controls will be used to provide an additional layer of defense-in-depth to align these systems to ensure that any release from a fuel handling accident will be filtered (not required to be met to meet the dose consequence results) and monitored.

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

6.0 ENVIRONMENTAL CONSIDERATION

On December 23, 1999, the NRC issued the Final Rule on "Use of Alternate Source Terms at Operating Reactors." The Final Rule, issued under 10 CFR 50.67, "Accident source term, "allows holders of operating licenses issued prior to January 10, 1997, to voluntarily replace the traditional source term used in design basis accident analyses with alternative source terms. This action would allow interested licensees to pursue cost beneficial licensing actions to reduce unnecessary regulatory burden without compromising the margin of safety of the facility.

Based on the above rule and in accordance with 10 CFR 50.67 and 10 CFR 50.90, "Application for amendment of license or construction permit," STPNOC is requesting an amendment to Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF-76 and NPF-80 for South Texas Project, Units 1 and 2. The proposed changes are requested to support application of an alternative source term (AST) methodology, with the exception that Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," will continue to be used as the radiation dose basis for equipment qualification. The proposed AST methodology conforms to the guidance in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000, except where alternate methods for complying with the specified portions of the NRC's regulations have been used as allowed by RG 1.183. The AST analyses were also performed in accordance with the guidance in Standard Review Plan Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms."

STPNOC has evaluated the proposed changes against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21, "Criteria for and identification of licensing and regulatory actions requiring environmental assessments." STPNOC has determined that the proposed changes meet the criteria for a categorical exclusion as set forth in 10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," paragraph (c)(9), and as such, has determined that no irreversible consequences exist in accordance with 10 CFR 50.92, "Issuance of amendment," paragraph (b). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," which changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20," Standards for Protection Against Radiation," or that changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria.

(i) The amendment involves no significant hazards consideration.

As demonstrated in Section 5.1 above, the proposed changes do not involve a significant hazards consideration.

(ii) There is no significant change in **the types or significant increase in the amounts of any effluents that may be released offsite.**

STPNOC meets the radiological criteria described in 10 CFR 50.67 for the exclusion area boundary (EAB) and the low population zone (LPZ).

Adoption of the AST methodology and TS changes which implement certain conservative assumptions in the AST analyses will not result in physical changes to the plant that could significantly alter the type or amounts of effluents that may be released offsite. Changes to operational parameters that could affect effluent releases have been demonstrated through analysis to satisfy regulatory requirements.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

STPNOC meets the radiological criteria described in 10 CFR 50.67 for the Control Room and Technical Support Center. Control Room and Technical Support Center exposure to operators is less than the five rem total effective dose equivalent (TEDE) over 30 days for all accidents.

The implementation of the AST methodology has been evaluated in revisions to the analyses of the limiting design basis accidents at the South Texas Project, Units 1 and 2. These accidents include the loss of coolant accident, the fuel handling accident, the control rod ejection accident, locked rotor accident, main steam line break accident, and steam generator tube rupture accident. Based upon the results of these analyses, it has been demonstrated that, with the proposed changes, the dose consequences of these limiting events are within the regulatory guidance provided by the NRC for use with the alternative source term approach (i.e., 10 CFR 50.67 and RG 1.183). Thus, there will be no significant increase in either individual or cumulative occupational radiation exposure.

7.0 REFERENCES

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2. NUREG-0800, Standard Review Plan, SRP-15.0.1, Rev. 0, "Radiological Consequence Analyses Using Alternative Source Terms," USNRC.
3. NRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors", USNRC, July 2000.
4. [Not Used]
5. Technical Specification Task Force Traveler, TSTF-51, Revision 2, dated July 31, 2003, "Revise Containment Requirements During Handling of Irradiated Fuel and Core Alterations."
6. Nuclear Utilities Management and Resources Council (NUMARC) 93-01, "Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 3.
7. South Texas Project, Units 1 and 2 – Issuance of Amendments on Relocation of Various Technical Specifications (TSs) to the Technical Specification Requirements Manual (TRM), dated December 17, 2002 (TAC NOS. MB3588 and MB3592)
8. South Texas Project, Units 1 and 2 – Issuance of Amendments Re: Allowed Outage Time for Control Room Envelope and the Fuel Handling Building Ventilation Systems, dated September 26, 2000, (TAC NOS. MA3849 and MA3850).
9. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants", USNRC.
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11. Surry Units 1 and 2 - Issuance of Amendments Re: Alternative Source Term (TAC NOS. MA 8649 and MA 8650), dated March 8, 2002.
12. Salem Nuclear Generating Station, Units 1 and 2, Issuance of Amendments Re: Request of Relaxation of Technical Specification Requirements Applicable During Movement of Irradiated Fuel (TAC NOS. MA 5710 and MA 5711), dated September 16, 2004.
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16. NRC Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1, USNRC, November 1982.
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20. "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, Environmental Protection Agency, Washington, D.C., 1988.
21. "Federal Guidance Report No. 12, External Exposure to Radionuclides in Air, Water, and Soil," EPA 420-r-93-081, Environmental Protection Agency, Washington, D.C., 1993.
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Attachment 2

Markup of Technical Specification pages

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CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

   1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or

   2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Specification 3.6.3.

b. All equipment hatches are closed and sealed,

c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,

d. The containment leakage rates are within the limits of Specification 3.6.1 2, and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATIONS

1.9 CORE ALTERATIONS shall be the movement of any fuel, sources, or reactivity control components [excluding rod cluster control assemblies (RCCAs) locked out in the integrated head package] 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

1.9a The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.6. Plant operation within these core operating limits is addressed within the individual Specifications.

DIGITAL CHANNEL OPERATIONAL TEST

1-10 A DIGITAL CHANNEL OPERATIONAL TEST shall consist of injecting simulated process data where available or exercising the digital computer hardware using data base manipulation to verify OPERABILITY of alarm, interlock, and/or trip functions.

DOSE EQUIVALENT I-131

1.11 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same **Committed Effective Dose Equivalent** ~~thyroid~~ dose as the quantity and isotopic mixture of I-131. I-132, I-133, I-134, and I-135 actually present. The **Committed Effective Dose Equivalent** ~~thyroid~~ dose conversion factors used for this calculation shall be those listed in **Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion," 1988; (Table 2.1, Exposure-to-Dose Conversion Factors for Inhalation).** ~~Table E-7 of NRC Regulatory Guide 1.109, Revision 1, October 1977.~~

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. Containment Isolation (Continued)					
b. Containment Ventilation Isolation					
1) Automatic Actuation Logic	2	1	2	1, 2, 3, 4	18
2) Actuation Relays***	3	2	3	1, 2, 3, 4	18
3) Safety Injection ***	See Item 1. above for all Safety Injection initiating functions and requirements.				
4) RCB Purge Radioactivity- High	2	1	2	1, 2, 3, 4, ~~5~~## ~~6~~##	18
5) Containment Spray- Manual Initiation	See Item 2. above for Containment Spray manual initiating functions and requirements.				
6) Phase "A" Isolation- Manual Isolation	See Item 3.a. above for Phase "A" Isolation manual initiating functions and requirements.				
c. Phase "B" Isolation					
1) Automatic Actuation Logic	2	1	2	1, 2, 3, 4	14
2) Actuation Relays	3	2	3	1, 2, 3, 4	14
3) Containment Pressure -- High-3	4	2	3	1, 2, 3	17
4) Containment Spray-- Manual Initiation	See Item 2. above for Containment Spray manual initiating functions and requirements.				
d. RCP Seal Injection Isolation					
1) Automatic Actuation Logic and Actuation Relays	1	1	1	1, 2, 3, 4	16

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
10. Control Room Ventilation					
a. Manual Initiation	3 (1/train)	2 (1/train)	3 (1/train)	~~All~~ 1, 2, 3, 4	27
b. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
c. Automatic Actuation Logic and Actuation Relays	3	2	3	~~All~~ 1, 2, 3, 4	27
d. Control Room Intake Air Radioactivity - High	2	1	2	~~All~~ 1, 2, 3, 4	28
e. Loss of Power	See Item 8. above for all Loss of Power initiating functions and requirements.				
~~11. FHB HVAC~~					
~~a. Manual Initiation~~	~~3 (1/train)~~	~~2 (1/train)~~	~~3 (1/train)~~	~~1, 2, 3, 4 or with irradiated fuel in spent pool~~	~~29, 30~~
~~b. Automatic Actuation Logic and Actuation Relays~~	~~3~~	~~2~~	~~3~~	~~1, 2, 3, 4 or with irradiated fuel in spent pool~~	~~29, 30~~
~~c. Safety Injection~~	~~See Item 1. above for all Safety Injection initiating functions and requirements.~~				
~~d. Spent Fuel Pool Exhaust Radioactivity - High~~	~~2~~	~~1~~	~~2~~	~~With irradiated fuel in spent fuel pool~~	~~30~~

TABLE 3.3-3 (Continued)
TABLE NOTATIONS

***Function is actuated by either actuation train A or actuation train B. Actuation train C is not used for this function.

****Automatic switchover to containment sump is accomplished for each train using the corresponding RWST level transmitter.

Trip function may be blocked in this MODE below the P-11 (Pressurizer Pressure Interlock) Setpoint.

~~## During CORE ALTERATIONS or movement of irradiated fuel within containment.~~

Trip function automatically blocked above P-11 and may be blocked below P-11 when Low Compensated Steamline Pressure Protection is not blocked.

ACTION STATEMENTS

ACTION 14 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 24 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 - (Not Used)

ACTION 16 – With the Charging Header Pressure channel inoperable:

a) Place the Charging Header Pressure channel in the tripped condition within one hour and
b) Restore the Charging Header Pressure channel to operable status within 7 days or be in at least Hot Standby within the next 6 hours and in Cold Shutdown within the following 30 hours.

ACTION 17 - With the number of OPERABLE channels one less than the Total Number of Channels, place the inoperable channel in the bypassed condition within 72 hours, or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours. One additional channel may be bypassed for up to 12 hours for surveillance testing per Specification 4.3.2.1.

ACTION 18 - a) With less than the Minimum Channels OPERABLE requirement for Automatic Actuation Logic or Actuation Relays, operation may continue provided the containment purge supply and exhaust valves are maintained closed.

b) MODE 1, 2, 3, or 4~~, or 5##~~ :

1. With one less than the Minimum Channels OPERABLE requirement for RCB Purge Radioactivity-High, within 30 days restore the inoperable channel or maintain the containment purge supply and exhaust valves closed.

NOTE:
MODE 1, 2, 3, or 4: Supplementary containment purge supply and isolation valves may be open during the allowed outage time for up to 2 hours at a time for required purge operation provided the valves are under administrative control.

~~MODE 5##: Supplementary or Normal containment purge supply and isolation valves may be open during the allowed outage time for up to 6 hours at a time for required purge operation provided the valves are under administrative control.~~

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

2. With two less than the Minimum Channels OPERABLE requirement for RCB Purge Radioactivity-High, operation may continue provided the containment purge supply and exhaust valves are maintained closed.

~~c) MODE 6##: With less than the Minimum Channels OPERABLE requirement for RCB Purge Radioactivity - High, apply the requirements of Technical Specification 3.9.9 for an inoperable Containment Ventilation Isolation System.~~

NOTE:

~~With one less than the Minimum Channels Operable requirement for RCB Purge Radioactivity-High, Supplementary or Normal containment purge supply and isolation valves may be open for up to 6 hours at a time for required purge operation provided the valves are under administrative control.~~

ACTION 19: With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 20: With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. For Functional Units with installed bypass test capability, the inoperable channel may be placed in bypass, and must be placed in the tripped condition within 72 hours.

Note: A channel may be bypassed for up to 12 hours for surveillance testing per Specification 4.3.2.1, provided no more than one channel is in bypass at any time.

b. For Functional Units with no installed bypass test capability,

1. The inoperable channel is placed in the tripped condition within 72 hours, and

2. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels per Specification 4.3.2.1.

~~Note:~~

~~For Unit 1 Train A Loss of Power Instrumentation (Functional Unit 8.a, 8.b, & 8.c) only: In addition to The requirements of ACTION 20.b, the provision below shall apply. This provision will expire 30 days after approval of this amendment.~~

~~With the number of OPERABLE channels more than one less than the Total Number of Channels, within one hour restore all but one channel per bus to OPERABLE status or enter applicable ACTION for the associated standby diesel generator made inoperable by the Loss of Power instrumentation.~~

ACTION 21: With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 22: With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 24 hours, or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 23 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 24 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the ACTION required by Specification 3.7.1.5.

ACTION 25 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 24 hours, or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 26 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, declare the affected Auxiliary Feedwater Pump inoperable and take ACTION required by Specification 3.7.1.2.

ACTION 27 - For an inoperable channel, declare its associated ventilation train inoperable and apply the actions of Specification 3.7.7.

ACTION 28 - a. With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, within 7 days initiate and maintain operation of the Control Room Makeup and Cleanup Filtration System (at 100% capacity) in the recirculation and makeup filtration mode.

b. With the number of OPERABLE channels two less than the Minimum Channels OPERABLE requirement, within 1 hour initiate and maintain operation of the Control Room Makeup and Cleanup Filtration System (at 100% capacity) in the recirculation and makeup filtration mode~~, OR~~

~~immediately suspend CORE ALTERATIONS, movement of irradiated fuel assemblies and crane operations with loads over the spent fuel pool, AND within 12 hours initiate and maintain operation of the Control Room Makeup and Cleanup Filtration System (at 100%capacity) in the recirculation and makeup filtration mode. CORE ALTERATIONS, movement of irradiated fuel assemblies, and crane operations with loads over the spent fuel pool are permitted during operation of the Control Room Makeup and Cleanup Filtration System (at 100% capacity) in the recirculation and makeup filtration mode.~~

c. With required ACTION 28a. or 28b. not met in MODE 1,2,3, or 4, ~~immediately suspend movement of irradiated fuel assemblies and crane operations with loads over *the* spent fuelool. AND~~ be in MODE 3 in 6 hours and in MODE 5 in the following 30 hours.

~~d. With required ACTION 28a. or 28b. not met in MODE 5 or 6, immediately suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and crane operations with loads over the spent fuel pool.~~

~~ACTION 29 - For an inoperable channel, declare its associated ventilation train inoperable and apply the actions of Specification 3.7.8.~~

~~ACTION 30 - With irradiated fuel in the spent fuel pool: With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, fuel movement within the spent fuel pool or crane operation with loads over the spent fuel pool may proceed provided the FHB exhaust air filtration system is in operation and discharging through at least one train of HEPA filters and charcoal adsorbers~~.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT		TRIP SETPOINT	ALLOWABLE VALUE
9.	Engineered Safety Features Actuation System Interlocks		
	a. Pressurizer Pressure, P-11	$\leq$ 1985 psig	$\leq$ 1995 psig
	b. Low-Low T_{AVG}, P-12	$\geq$ 563°F	$\geq$ 560.7°F
	c. Reactor Trip, P-4	N.A.	N.A.
10.	Control Room Ventilation		
	a. Manual Initiation	N.A.	N.A.
	b. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values	
	c. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
	d. Control Room Intake Air Radioactivity – High	$\leq 6.1\times10^{-5}$ µCi/cc	$\leq 7.8\times10^{-5}$ µCi/cc
	e. Loss of Power	See Item 8. above for all Loss of Power Trip Setpoints and Allowable Values	
~~11.~~	~~FHB HVAC~~		
	~~a. Manual Initiation~~	~~N.A.~~	~~N.A.~~

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
~~11. FHB HVAC (Continued)~~		
~~b. Automatic Actuation Logic and Actuation Relays~~	~~N.A.~~	~~N.A.~~
~~c. Safety Injection~~	~~See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values~~	
~~d. Spent Fuel Pool Exhaust Radioactivity – High~~	~~$< 5.0x10^{-4}$~~	~~$< 6.4x10^{-4}$~~

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	DIGITAL OR ANALOG CHANNEL OPERATIONAL TEST (7)	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Containment Isolation (Continued)								
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
4) RCB Purge Radioactivity-High	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4, 5*, 6*
5) Containment Spray - Manual Initiation	See Item 2. above for Containment Spray manual initiation Surveillance Requirements.							
6) Phase "A" Isolation-Manual Initiation	See Item 3. a. above for Phase "A" Isolation manual initiation Surveillance Requirements.							
c. Phase "B" Isolation								
1) Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	Q(1)	N.A.	N.A.	1, 2, 3, 4
2) Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.	Q(6)	Q(8)	1, 2, 3, 4 |
3) Containment Pressure--High-3	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
4) Containment Spray- Manual Initiation	See Item 2. above for Containment Spray manual initiation Surveillance Requirements.							
d. RCP Seal Injection Isolation								
1) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.	Q	Q(8)	1, 2, 3, 4 |
2) Charging Header Pressure - Low	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
Coincident with Phase "A" Isolation	See Item 3.a. above for Phase "A" surveillance requirements.							

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	DIGITAL OR ANALOG CHANNEL OPERATIONAL TEST (7)	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
10. Control Room Ventilation (Continued)								
b. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
c. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	Q(6)	N.A.	N.A.	~~All~~ 1, 2, 3, 4
d. Control Room Intake Air Radioactivity-High	S	R	Q	N.A.	N.A.	N.A.	N.A.	~~All~~ 1, 2, 3, 4
e. Loss of Power	See Item 8. above for all Loss of Power Surveillance Requirements.							
~~11. FHB HVAC~~								
~~a. Manual Initiation~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~R~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~1, 2, 3, 4, or with irradiated fuel in the spent fuel pool~~
~~b. Automatic Actuation Relays~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~Q(6)~~	~~N.A.~~	~~N.A.~~	~~1, 2, 3, 4, or with irradiated fuel in the spent fuel pool.~~

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	DIGITAL OR ANALOG CHANNEL OPERATIONAL TEST (7)	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
~~11. FHB HVAC (Continued)~~								
~~c. Safety Injection~~	~~See Item 1. above for all Safety Injection Surveillance Requirements.~~							
~~d. Spent Fuel Pool Exhaust Radio-activity-High~~	~~S~~	~~R~~	~~Q~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~With irradiated fuel in spent fuel pool.~~

TABLE NOTATION

(1) Each train shall be tested at least every 92 days on a STAGGERED TEST BASIS.

(2) Deleted

(3) Deleted

(4) Deleted

(5) Deleted

(6) Each actuation train shall be tested at least every 92 days on a STAGGERED TEST BASIS. Testing of each actuation train shall include master relay testing of both logic trains. If an ESFAS instrumentation channel is inoperable due to failure of the Actuation Logic Test and/or Master Relay Test, increase the surveillance frequency such that each train is tested at least every 62 days on a STAGGERED TEST BASIS unless the failure can be determined by performance of an engineering evaluation to be a single random failure.

(7) For channels with bypass test instrumentation, input relays are tested on an 18-month (R) frequency.

(8) The test interval is R for Potter & Brumfield MDR Series slave relays.

~~* During CORE ALTERATIONS or movement of irradiated fuel within containment.~~

3/4.7.7 CONTROL ROOM MAKEUP AND CLEANUP FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 Three independent Control Room Makeup and Cleanup Filtration Systems shall be OPERABLE.

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

ACTION:

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

a. With one Control Room Makeup and Cleanup Filtration System inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two Control Room Makeup and Cleanup Filtration Systems inoperable, restore at least two systems to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With three Control Room Makeup and Cleanup Filtration Systems inoperable, ~~suspend all operations involving movement of spent fuel, and crane operation with loads over the spent fuel pool, and~~ restore at least one system to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

~~MODES 5 and 6:~~

~~a. With one Control Room Makeup and Cleanup Filtration System inoperable, restore the inoperable system to OPERABLE status within 7 days or initiate and maintain operation of the remaining OPERABLE Control Room Makeup and Cleanup Filtration Systems in the recirculation and makeup air filtration mode, or suspend all operations involving CORE ALTERATIONS, operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or required boron concentration, movement of spent fuel, and crane operation with loads over the spent fuel pool.~~

~~b. With more than one Control Room Makeup and Cleanup Filtration System inoperable, or with the OPERABLE Control Room Makeup and Cleanup Filtration Systems required to be in the recirculation and makeup air filtration mode by ACTION a. not capable of being powered by an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS, operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or required boron concentration, movement of spent fuel, and crane operations with loads over the spent fuel pool.~~

SURVEILLANCE REQUIREMENTS

4.7.7 Each Control Room Makeup and Cleanup Filtration System shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to 78°F;

b. At least once per 92 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers of the makeup and cleanup air filter units and verifying that the system operates for at least 10 continuous hours with the makeup filter unit heaters operating;

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

1) Verifying that the makeup and cleanup systems satisfy the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% for HEPA filter banks and 0.10% for charcoal adsorber banks and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 6000 cfm ± 10% for the cleanup units and 1000 cfm ± 10% for the makeup units;

2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon," for a methyl iodide penetration of less than 1.0% when tested at a temperature of 30°C and a relative humidity of 70%; and

3) Verifying a system flow rate of 6000 cfm 10% for the cleanup units and 1000 cfm ± 10% for the makeup units during system operation when tested in accordance with ANSI N510-1980.

d. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989 for a methyl iodide penetration of less than 1.0% when tested at a temperature of 30°C and a relative humidity of 70%.

e. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6.1 inches Water Gauge for the makeup units and 6.0 inches Water Gauge for the cleanup units while operating the system at a flow rate of 6000 cfrn + 10% for the cleanup units and 1000 cfm & 10% for the makeup units.

2) Verifying that on a control room emergency ventilation test signal (High Radiation and/or Safety Injection test signal), the system automatically switches into a recirculation and makeup air filtration mode of operation with flow through the HEPA filters and charcoal adsorber banks of the cleanup and makeup units;

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

3) Verifying that the system maintains the control room envelope at a positive pressure of greater than or equal to 1/8 inch Water Gauge at less than or equal to a pressurization flow of 2000 cfm relative to adjacent areas during system operation ~~[1]~~; and

4) Verifying that the makeup filter unit heaters dissipate 4.5 ± 0.45 kW when tested in accordance with ANSI N510-1980.

f. After each complete or partial replacement of a HEPA filter bank, by verifying that the HEPA filter bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 6000 cfm ± 10% for the cleanup units and 1000 cfm ± 10% for the makeup units; and

g. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the charcoal adsorber bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.10% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 6000 cfm ± 10% for the cleanup units and 1000 cfm ± 10% for the makeup units.

~~(1) Measured points at a positive pressure but less than 1/8 inch Water Gauge are acceptable if an evaluation, considering appropriate compensatory action, demonstrates that the condition meets the requirements of GDC-19. The provisions of this note expire at 0800 on September 19, 2005.~~

~~3/4.7.8 FUEL HANDLING BUILDING (FHB) EXHAUST AIR SYSTEM~~ **(This specification is not used)**

~~LIMITING CONDITION FOR OPERATION~~

~~3.7.8 The FHB Exhaust Air System comprised of the following components shall be OPERABLE.~~

~~a. Two independent exhaust air filter trains, and~~

~~b. Three exhaust ventilation trains.~~

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

~~ACTION:~~

~~a. With one FHB exhaust air filter train inoperable, restore the inoperable filter train to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN in the following 30 hours.~~

~~b. With two FHB exhaust air filter trains inoperable, restore at least one inoperable filter train to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN in the following 30 hours.~~

~~c. With one FHB exhaust ventilation train inoperable, restore the inoperable exhaust ventilation train to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN in the following 30 hours.~~

~~d. With more than one FHB exhaust ventilation train inoperable, restore at least two exhaust ventilation trains to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN in the following 30 hours.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.7.8 The Fuel Handling Building Exhaust Air System shall be demonstrated OPERABLE:~~

~~a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating with two of the three exhaust booster fans and two of the three main exhaust fans operating to maintain adequate air flow rate;~~

~~b. At least once per 18 months and (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:~~

~~1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% for HEPA filter banks and 0.10% for charcoal adsorber banks and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 28,000 cfm ± 10%;~~

~~2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52,~~

~~SURVEILLANCE REQUIREMENTS (Continued)~~

~~Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989 "Standard Test Method for Nuclear-Grade Activated Carbon," for a methyl iodide penetration of less than 1.0% when tested at a temperature of 30°C and a relative humidity of 70%; and~~

~~3) Verifying a system flow rate of 29,000 cfm +. 10% during system operation with two of the three exhaust booster fans and two of the three main exhaust fans operating when tested in accordance with ANSI N510-1980. All combinations of two exhaust booster fans and two main exhaust fans shall be tested.~~

~~c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989 for a methyl iodide penetration of less than 1.0% when tested at a temperature of 30°C and a relative I humidity of 70%;~~

~~d. At least once per 18 months by:~~

~~1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 29,000 cfm ± 10%,~~

~~2) Verifying that the system starts on High Radiation and Safety Injection test signals and directs flow through the HEPA filters and charcoal adsorbers,~~

~~3) Verifying that the system maintains the FHB at a negative pressure of greater than or equal to 118 inch Water Gauge relative to the outside atmosphere, and~~

~~4) Verifying that the heaters dissipate 38 f 2.3 kW when tested in accordance with ANSI N510-1980.~~

~~e. After each complete or partial replacement of a HEPA filter bank, by verifying that the HEPA filter bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 29,000 cfm ± 10%; and~~

~~f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the charcoal adsorber bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.10% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 29,000 cfm ± 10%.~~

ELECTRICAL POWER SYSTEMS

A.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the Onsite Class 1E Distribution System, and

b. Two[1] standby diesel generators each with a separate fuel tank containing a minimum volume of 60,500 gallons of fuel.

APPLICABILITY: MODE 5 and MODE 6 with water level in the refueling cavity < 23 ft above the reactor pressure vessel flange.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or required boron concentration, ~~movement of irradiated fuel,~~ **or** operations with a potential for draining the reactor vessel ~~or crane operation with loads over the spent fuel pool~~. Immediately initiate actions to restore the inoperable A.C. electrical power source to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the requirements of Specifications 4.8.1.1, 4.8.1.1.2 (except for Specification 4.8.1.1.2a.3), and 4.8.1.1.3.

4.8.1.2.1 The alternate onsite emergency power source shall be demonstrated functional by:

a. Within 4 hours of taking credit for the onsite emergency power source as a standby diesel generator, verify it starts and achieves steady state voltage ($\pm$10%) and frequency ($\pm$2%) in 5 minutes.

b. Within 4 hours of taking credit for the onsite emergency power source as a standby diesel generator and every 8 hours thereafter, verify the emergency power source is capable of being aligned to the required ESF bus by performing a breaker alignment check.

[1]An alternate onsite emergency power source, capable of supplying power for one train of shutdown cooling may be substituted for one of the required diesels for 14 consecutive days (SR 4.8.1.2.1 is the only requirement applicable).

ELECTRICAL POVER SYSTEMS

A.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.3 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the Onsite Class 1E Distribution System, and

b. One standby diesel generator with a separate fuel tank containing a minimum volume of 60,500 gallons of fuel.

APPLICABILITY: MODE 6 with water level in the refueling cavity $\geq$ 23 ft above the reactor pressure vessel flange.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or required boron concentration, ~~movement of irradiated fuel,~~ **or** operations with a potential for draining the reactor vessel ~~or crane operation with loads over the spent fuel pool~~. Immediately initiate actions to restore the inoperable A.C. electrical power source to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.8.1.3 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the requirements of Specifications 4.8.1.1.1, 4.8.1.1.2 (except for Specification 4.8.1.1.2a.3), and 4.8.1.1.3.

ELECTRICAL POWER SYSTEMS

D.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 DC electrical power subsystem shall be OPERABLE to support the DC electrical power distribution subsystem(s) required by LC0 3.8.3.2, "Onsite Power Distribution - Shutdown."

APPLICABILITY: MODES 5 and 6.

ACTION:

With one or more required DC electrical power subsystems inoperable, immediately declare affected required feature(s) inoperable OR immediately initiate action to suspend operations with a potential for draining the reactor vessel, suspend all operations involving CORE ALTERATIONS, **or** operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or required boron concentration~~, or movement of irradiated fuel~~; initiate corrective action to restore the required DC electrical power subsystems to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENT

4.8.2.2 The required DC sources shall be demonstrated OPERABLE in accordance with Specification 4.8.2.1.

ELECTRICAL POWER SYSTEMS

ONSITE POWER DISTRIBUTION

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 The necessary portion of AC, DC, and AC vital bus electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With one or more required AC, DC, or AC vital bus electrical power distribution subsystems inoperable, immediately declare associated supported required feature(s) inoperable OR immediately initiate action to suspend operations with a potential for draining the reactor vessel, suspend all operations involving CORE ALTERATIONS, operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or required boron concentration, ~~movement of irradiated fuel,~~ and immediately initiate corrective action to restore required AC, DC, and AC vital bus electrical power distribution subsystems to OPERABLE status and declare associated required residual heat removal subsystem(s) inoperable and not in operation.

SURVEILLANCE REQUIREMENT

4.8.3.2 Verify correct breaker alignment and voltage to required AC, DC, and AC vital bus electrical power distribution subsystems at least once per 7 days.

3/4.9.4 (This specification is not used) ~~CONTAINMENT BUILDING PENETRATIONS~~
~~LIMITING CONDITION FOR OPERATION~~

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

~~a. The equipment hatch closed and held in place by a minimum of four bolts~~
~~OR~~
~~1) The Reactor has been subcritical for ≥ 165 hours, AND~~
~~If open, the equipment hatch is capable of being closed.~~

~~b. 1) A minimum of one door in the containment Auxiliary Airlock (AAL) is closed.~~
~~AND~~
~~2) A minimum of one door in the containment Personnel Airlock (PAL) is closed.~~
~~OR~~
~~The water level is ≥ 23 feet above the reactor vessel flange.~~
~~AND~~
~~The Reactor has been subcritical for ≥ 95 hours but <165 hours.~~
~~AND~~
~~An individual is available to close a PAL door when directed (after the initiation of a fuel handling accident inside containment) within~~
~~a. 30 minutes, if the reactor has been subcritical <165 hours.~~
~~OR~~
~~b. As soon as possible but within 2 hours, if the reactor has been subcritical ≥ 165 hours.~~

~~c. All other penetrations providing direct access from the containment atmosphere to the outside atmosphere shall be either:~~

~~1) Closed by an isolation valve, blind flange, or manual valve, or~~

~~2) Be capable of being closed by an OPERABLE automatic containment purge and exhaust isolation valve.~~

~~APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.~~

~~ACTION:~~

~~With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.9.4 Each of the above required containment building penetrations shall be determined to be either in its required condition or capable of being closed as required in specification 3.9.4 within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment building by (as applicable):~~

~~a. Verifying the penetrations are in their required condition~~

~~b. Testing the containment purge and exhaust isolation valves per the applicable portions of Specification 4.6.3.2.~~
~~c. Proper tools are stage and trained personnel are designated to close the equipment hatch if open.~~

REFUELING OPERATIONS

3/4.9.9 (This specification is not used) ~~CONTAINMENT VENTILATION ISOLATION SYSTEM~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.9.9 The Containment Ventilation Isolation System shall be OPERABLE.~~

~~APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.~~

~~ACTION:~~

~~a. With the Containment Ventilation Isolation System inoperable, close each of the purge and exhaust penetrations providing direct access from the containment atmosphere to the outside atmosphere.~~

~~NOTE:~~

~~In accordance with ACTION 18.b and ACTION 18.c of Table 3.3-3, Supplementary or Normal containment purge supply and isolation valves may be open for up to 6 hours at a time for required purge operation provided the valves are under administrative control.~~

~~b. The provisions of Specification 3.0.3 are not applicable.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.9.9 The Containment Ventilation Isolation System shall be demonstrated OPERABLE within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment ventilation isolation occurs on manual initiation and on a High Radiation test signal from each of the RCB purge radiation monitoring instrumentation channels.~~

REFUELING OPERATIONS

3/4.9.12 (This specification is not used) ~~FUEL HANDLING BUILDING EXHAUST AIR SYSTEM~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.9.12 The FHB Exhaust Air System[1] comprised of the following components shall be OPERABLE:~~

~~a. Two exhaust air filter trains,~~

~~b. Two exhaust ventilation trains~~

~~APPLICABILITY: Whenever irradiated fuel is in the spent fuel pool.~~

~~ACTION:~~

~~a. With less than the above FHB Exhaust Air System components OPERABLE but with at least one FHB exhaust air filter train, one FHB exhaust ventilation train, and associated dampers OPERABLE, fuel movement within the spent fuel pool or crane operation with loads over the spent fuel pool may proceed provided the OPERABLE FHB Exhaust Air System components are capable of being powered from an OPERABLE emergency power source and are in operation and discharging through at least one train of HEPA filters and charcoal adsorbers.~~

~~b. With no FHB exhaust air filter train OPERABLE, suspend all operations involving movement fuel within the spent fuel pool or crane operation with loads over the spent fuel pool.~~

~~c. The provisions of Specification 3.0.3 are not applicable.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.9.12 The above required FHB Exhaust Air Systems shall be demonstrated OPERABLE:~~

~~a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating with the operable exhaust booster fans and the operable main exhaust fans operating to maintain adequate air flow rate;~~

~~1 At least one FHB exhaust air filter train, one FHB exhaust booster fan, and one FHB main exhaust fan are capable of being powered from an OPERABLE onsite emergency power source.~~

Pages 3/4 9-15 and 3/4 9-16 have been deleted

~~SURVEILLANCE REQUIREMENTS (Continued)~~

~~b. At least once per 18 months and (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:~~

~~1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% for HEPA filter banks and 0.10% for charcoal adsorber banks and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 29,000 cfm ± 10%;~~

~~2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon," for a methyl iodide penetration of less than 1.0% when tested at a temperature of 30°C and a relative humidity of 70%; and~~

~~3) Verifying a system flow rate of 29,000 cfm k 10% during system operation with two of the three exhaust booster fans and two of the three main exhaust fans operating when tested in accordance with ANSI N510-1980. All combinations of two exhaust booster fans and two main exhaust fans shall be tested.~~

~~c. After every 720 hours of charcoal adsorber operation by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989 for a methyl iodide penetration of less than 1.0% when tested at a temperature of 30°C and a relative humidity of 70%.~~

~~d. At least once per 18 months by:~~

~~1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 29,000 cfm ± 10%.~~

~~2) Verifying that on a High Radiation test signal, the system automatically starts (unless already operating) and directs its exhaust flow through the HEPA filters and charcoal adsorber banks.~~

~~SURVEILLANCE REQUIREMENTS (Continued)~~

~~3) Verifying that the system maintains the spent fuel storage pool area at a negative pressure of greater than or equal to 1/8 inch Water Gauge relative to the outside atmosphere during system operation, and~~

~~4) Verifying that the heaters dissipate 38 ± 2.3 kW when tested in accordance with ANSI N510-1980.*~~

~~e. After each complete or partial replacement of a HEPA filter bank, by verifying that the HEPA filter bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 29,000 cfm ± 10%.~~

~~f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the charcoal adsorber bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.10% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 29,000 cfm ± 10%.~~

~~*During the first six weeks after March 28, 1989, testing will be required for both 50 kW and 38 kW heaters.~~

Attachment 3

Technical Specification Bases pages
(information only)

Page B 3/4 3-2b
Page B 3/4 3-2c
Page B 3/4 3-2c.1 (temp)
Page B 3/4 3-2d
Page B 3/4 3-3
Page B 3/4 7-4
Page B 3/4 7-5
Page B 3/4 7-6
Page B 3/4 7-7
Page B 3/4 8-14 (no changes)
Page B 3/4 8-15
Page B 3/4 8-19
Page B 3/4 8-20 (no changes)
Page B 3/4 9-1b
Page B 3/4 9-2
Page B 3/4 9-3
Page B 3/4 9-3a

INSTRUMENTATION

BASES

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

When control rods are at the top or above the active fuel region (≥ 2 step 259), they are no longer capable of adding positive reactivity to the core, and as such, they are not capable of rod withdrawal as intended by MODE 5*. Therefore, ACTION 10 on Table 3.3-1 is not applicable in this region. This allows the Reactor Trip Breakers to be closed, without meeting the requirements of MODE 5*, while unlocking and stepping the control rods to a position no lower than 259. (CR 97-908-17)

Several ACTIONS in Tables 3.3-1 and 3.3-3 have been revised to change the allowed outage times and bypass test times in accordance with WCAP-10271 and WCAP-14333. Additionally, some ACTIONS have been divided such that only certain requirements apply depending on whether the Functional Units have been modified with installed bypass test capability.

Regardless of whether the Functional Units have installed bypass test capability, it should be noted that in certain situations, the ACTIONS permit continued operation (for limited periods of time) with less than the minimum number of channels specified in Tables 3.3-1 and 3.3-3. For example, Table 3.3-1 Functional Unit 11 (Pressurizer Pressure - High) requires a minimum of 3 channels operable. However, since continued operation with an inoperable channel is permitted beyond 72 hours, provided the inoperable channel is placed in trip, and since periodic surveillance testing of the other channels must continue to be performed, ACTION 6 permits a channel to be placed in bypass for up to 12 hours to permit testing. Thus, for a limited period of time (12 hours), 2 channels, or one less than the minimum, would be permitted to be inoperable.

Actuation relays consist of slave relays, including the relay contacts for actuating the ESF equipment. If a slave relay becomes inoperable for a particular component(s), then the associated component(s) LCO Required Action should be entered. If an entire train of slave relays for a functional unit becomes inoperable, then the Required Action for the functional unit actuation train should be entered. (CR 00-1 3604-7)

During a plant shutdown for refueling, the Normal Containment Purge System is in operation. The Supplementary Containment Purge System may be used during normal plant operation. Redundant Class 1E radiation monitors (i.e., the Reactor Containment Building [RCB] Purge Isolation) monitor the radiation in these purge lines. Upon either monitor sensing radiation above a preset limit, a signal is sent to the ESFAS logic trains, and the Containment ventilation isolation signal is actuated. In a LOCA, both Normal and Supplementary purge lines are isolated by a Safety Injection (SI) signal. Actuation of the purge isolation by these radiation monitors is not credited in the LOCA accident analyses, and is only a backup function for this event. ~~The subject radiation monitors are credited for purge line isolation for a fuel handling accident.~~

INSTRUMENTATION

BASES

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

ACTION 18.a. applies when the actuation logic for RCB Purge Radioactivity - High is inoperable because it affects both channels. The required action is to maintain the isolation valves closed. Loss of power supply to the output ESF relays of either channel of these monitors will be considered inoperable actuation logic and the isolation valves will be maintained closed in accordance with proposed ACTION 18.a. This is because this failure mode will result in the inability of the other actuation signals to close the purge valves if the initial signal is reset.

In MODE 1, 2, 3, or 4, ~~or 5#,~~ when one of the two required channels of RCB Purge Radioactivity - High is inoperable, ACTION 18.b.1 requires restoration within 30 days. The allowed outage time is a reasonable time for easily accessible non-risk-significant instrumentation. The required action is modified by a note that allows the supplementary purge valves to be opened in MODE 1-4 under administrative control during the 30-day allowed outage time to permit operation of the supplementary purge system for up to 2 hours at a time for the evolutions permitted by the Technical Specifications (containment pressure control, ALARA and respirable air quality needed for personnel entry into containment and for surveillance tests that required the valves to be open). The 2-hour allowance is adequate time for the routine pressure control purge operations during power operation. ~~The note also allows the normal or supplementary purge supply and exhaust valves to be open up to 6 hours at a time in MODE 5# for required purge operations. The 6-hour duration is justified because the design basis event in this MODE would be expected to be a slower developing event and purge operations in support of refueling activities are typically much longer than those done at power.~~ Opening the valves for purge operations is not permitted after the 30-day allowed outage time has expired.

In MODE 1 - 4, the safety analysis credits only the SI signal for actuation of CVI. As a backup, the operable radiation monitoring channel would still be available to actuate containment isolation. ~~In MODE 5 and 6, there is no credible LOCA event and the design basis postulated event is a fuel handling accident in containment.~~

Administrative control during purge evolutions with an inoperable radiation monitoring channel would include the operator ability to manually initiate CVI from the control room handswitch and typically include an assessment of plant conditions for potential actuation precursors, monitoring containment radiation and limiting purge duration.

ACTION 18.b.2 applies in MODE 1, 2, 3, and 4~~, and 5#~~ when both channels of RCB Purge Radioactivity - High are inoperable. The action requires the purge isolation valves to be maintained closed and there is no provision for purge operation under administrative control.

~~ACTION 18.c. applies to the condition where one or both RCB Purge Radioactivity – High channels are inoperable in MODE 6 during movement of irradiated fuel or CORE ALTERATIONS. The ACTION directs the user to apply the requirements of TS 3/4.9.9 for an inoperable Containment Ventilation Isolation System during Refueling. With one inoperable channel of RCB Purge Radioactivity – High inoperable, the action includes a provision that allows purge operations for up to 6 hours at a time. The basis for the 6 hour duration of the purge is the same as described above for MODE 5#.~~

~~INSTRUMENTATION~~

~~BASES~~

~~REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)~~

~~LOP Instrumentation~~

~~Temporary BASES insert for Unit 1 Train A (Functional Unit 8.a, 8.b, 8.c)~~

~~For Unit 1 Train A, a note is added to ACTION 20.b for a condition when more than one loss of voltage or more than one degraded voltage channel per bus are inoperable. The action in the note requires restoring all but one channel per bus to OPERABLE status. The 1 hour Completion Time should allow ample time to repair most failures and takes into account the low probability of an event requiring an LOP start occurring during this interval. If the channels are not restored in the 1 hour completion time, the Conditions specified in TS 3.8.1.1, "AC Sources - Operating," for the DG made inoperable by failure of the LOP DG start instrumentation are required to be entered immediately. The actions of that LCO provides for adequate compensatory actions to assure unit safety. This note supports a one-time change to facilitate corrective maintenance and expires 30 days after approval of the license amendment that added it.~~

INSTRUMENTATION

BASES

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

ACTION 27 for an inoperable channel of control room ventilation requires the associated train of control room ventilation to be declared inoperable and the appropriate action take in accordance with Specification 3.7.7. Each control room ventilation system (train) is actuated by its own instrumentation channel. Consequently an inoperable channel of ventilation actuation instrumentation renders that system/train of ventilation inoperable and Specification 3.7.7 prescribes the appropriate action.

ACTION 28.a. provides 7 days to place the Control Room ventilation in the recirculation and make-up filtration mode of operation at 100% capacity (any two of the three trains of control room makeup and cleanup filtration meet the 100% capacity requirement) when one the two radioactivity high actuation channels is inoperable. This time is acceptable because there is still an operable channel that will function to realign the control room envelope on a high radiation signal unless the failure mode is due to the output power supply. However, in that case, the operator can manually initiate the function. The 7 day allowed outage time is based on the low probability of a Design Basis Accident (DBA) occurring during this time period, and ability of the remaining train to provide the required capability.

ACTION 28.b. applies when both channels of control room ventilation radioactivity-high are inoperable and requires the ventilation system to be placed in recirculation and make-up filtration within 1 hour. ~~The action includes an option to relax the time required to realign the Control Room Makeup and Cleanup Filtration System to the recirculation and makeup mode from 1 hour to 12 hours if the additional requirement to immediately suspend core alterations, movement of irradiated fuel assemblies and crane operations with loads over the spent fuel pool is imposed. The option would apply in any of the four following scenarios: 1) plant operating and no fuel movement or crane operations over the spent fuel pool, 2) plant operating and fuel movement or crane operations being performed, 3) plant shutdown and fuel movement or crane operations being performed, and 4) plant shutdown and no fuel movement or crane operations being performed. (Although Scenario 4 would require no action other than logging the inoperability, it will preclude core alterations, fuel movement and crane movement with loads over the spent fuel pool).~~ The additional restriction provides assurance that potential radiation releases from design basis accidents inside and outside containment have been considered for this configuration. ~~The option permits fuel movement and crane operation with loads over the spent fuel pool if the Control Room makeup and cleanup filtration system is operating at 100% capacity (any two of the three 50% trains that comprise the system).~~

~~The option to allow a 12 hour action time is also consistent with TS 3.7.7 for Control Room HVAC, which allows all three trains of the HVAC to be inoperable for 12 hours.~~

ACTION 28.c. applies for MODEs 1, 2, 3, & 4. It ~~suspends core alterations, movement of irradiated fuel assemblies and crane operations with loads over the spent fuel pool and~~ requires the plant to be placed in a MODE where the Technical Specification does not apply.

INSTRUMENTATION

BASES

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

~~ACTION 28.d. applies in MODE 5 & 6 and requires the suspension of core alterations, movement of irradiated fuel assemblies and crane operations with loads over the spent fuel pool. This effectively precludes the design basis accidents that the control room radioactivity-high actuation system is designed to mitigate.~~

The Engineered Safety Features Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates Turbine trip via P-16, closes main feedwater valves on Tavg below Setpoint, prevents the opening of the main feedwater valves which were closed by a Safety Injection or High Steam Generator Water Level and allows Safety Injection block so that components can be reset or tripped. Reactor tripped with the source range blocked provides a non-protective function that closes the Steam Generator Blowdown isolation valves and allows reopening the valves after the source range block is reset.

Reactor not tripped - prevents manual block of Safety Injection.

P-11 On increasing pressurizer pressure, P-11 automatically reinstates Safety Injection actuation on low pressurizer pressure or low compensated steamline pressure signals, reinstates steamline isolation on low compensated steamline pressure signals, and opens the accumulator discharge isolation valves. On decreasing pressure, P-11 allows the manual block of Safety Injection actuation on low pressurizer pressure or low compensated steamline pressure signals, allows the manual block of steamline isolation on low compensated steamline pressure signals, and enables steam line isolation on high negative steam line pressure rate (when steamline pressure is manually blocked).

P-12 On increasing reactor coolant loop temperature, P-12 automatically provides an arming signal to the Steam Dump System. On decreasing reactor coolant loop temperature, P-l2 automatically removes the arming signal from the Steam Dump System.

P-14 On increasing steam generator water level, P-14 automatically trips the turbine and the main feedwater pumps, and closes all feedwater isolation valves and feedwater control valves.

For Table 4.3-1 Notations 3 and 6, the term "incore" applies to either a PDMS measurement OR a movable incore detector system measurement, because both methods represent a measurement of the reactor core power distribution.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 (NOT USED)

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The limitations on minimum water level and maximum temperature are based on providing a 30-day cooling water supply to safety-related equipment without exceeding its design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants," March 1974.

3/4.7.6 (NOT USED)

3/4.7.7 CONTROL ROOM MAKEUP AND CLEANUP FILTRATION SYSTEM

The Control Room Makeup and Filtration System is comprised of three 50-percent redundant systems (trains) that share a common intake plenum and exhaust plenum. Each system/train is comprised of a makeup fan, a makeup filtration unit, a cleanup filtration unit, a cleanup fan, a control room air handling unit, a supply fan, a return fan, and associated ductwork and dampers. Two of the three 50% design capacity trains are required to be operable during the following modes of operation: shutdown, hot standby, normal operation, postulated accident condition, and loss of offsite power. The toilet kitchen exhaust, heating, and computer room HVAC Subsystem associated with the Control Room Makeup and Filtration System are nonsafety-related and not required for operability.

The OPERABILITY of the Control Room Makeup and Cleanup Filtration System ensures that: (1) the ambient air temperature does not exceed the allowable temperature for continuous-duty rating for the equipment and instrumentation cooled by this system, and (2) the control room will remain habitable for operations personnel during and following all credible accident conditions. Operation of the system with the heaters operating for at least 10 continuous hours in a 92-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem **total effective dose equivalent (TEDE)** ~~s or less whole body, or its equivalent~~. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50. ANSI N510-1980 will be used as a procedural guide for surveillance testing.

~~The ACTIONS specified during modes 5 and 6 with less than the minimum required Control Room Makeup and Cleanup Filtration Systems, or associated power systems, include suspending operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or refueling boron concentration necessary to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum SHUTDOWN MARGIN or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes, including temperature increases when operating with a positive moderator temperature coefficient, must also be evaluated to not result in operation below the required SHUTDOWN MARGIN or refueling boron concentration limits. Control rod withdrawal is not allowed except that it is permissible to unlock the control rods for rapid refueling. To unlock the control rods, they must be withdrawn at least one step. However, since the control rods are above the active fuel when the unlocking process occurs, there is no reactivity addition.~~

The accidents postulated to occur during core alterations, in addition to the fuel handling accident, are: inadvertent criticality (due to a control rod removal error or continuous rod withdrawal error during refueling or boron dilution) and the inadvertent loading of, and subsequent operation with, a fuel assembly in an improper location. These events are not postulated to result in fuel cladding integrity damage. Since the only accident to occur during CORE ALTERATIONS that results in a significant radioactive release is the fuel handling accident and the accident mitigation features of the Control Room Makeup and Cleanup Filtration System are not credited in the accident analysis for a fuel handling accident, there are no OPERABILITY requirements for this system in MODES 5 and 6.

The time limits associated with the ACTIONS to restore an inoperable train to OPERABLE status are consistent with the redundancy and capability of the system and the low probability of a design basis accident while the affected train(s) is out of service. A limited allowed outage time of 12 hours is allowed for all three trains to be out of service simultaneously in recognition of the fact that there are common plenums and some maintenance or testing activities required opening or entry into these common plenums. This time is reasonable to diagnose, plan, and possibly repair problems with the boundary or the ventilation system. This is acceptable based on the low probability of a design basis event in that brief allowed outage time and because administrative controls impose compensatory actions that reduce the already small risk associated with being in the ACTION. The compensatory actions are consistent with the intent of GDC 19 to protect plant personnel from potential hazards such as radioactive contamination, smoke, and temperature, etc. Pre-planned measures should be available to address these concerns for intentional and unintentional entry into the condition. The compensatory actions include:

- Procedures will preclude intentionally removing multiple trains of Control Room Envelope HVAC from service if Containment Spray is not functional or intentionally making a train of Containment Spray unavailable when multiple trains of Control Room Envelope HVAC are out of service. For purposes of this compensatory action, Containment Spray is considered functional if at least one train can be manually or automatically initiated.

- The plant will not make planned simultaneous entries into TS 3.7.7 ACTION c. for MODES 1, 2, 3 and 4 and TRM 3.9.12 ACTION a.

The compensatory action may include placing fans in pull-to-lock as necessary to preclude there being a motive force to transport contaminated air to a clean environment in the event of an accident. These compensatory actions also include administrative controls on opening plenums or other openings such that appropriate communication is established with the control room to assure timely closing of the system if necessary. Since the Control Room Envelope boundary integrity also affects operability of the overall system, entry and exit is administratively controlled. Administrative control of entry and exit through doors is performed by the person(s) entering or exiting the area. Extended opening of the boundary is coordinated with the control room with appropriate plans for closure and communication.

Surveillance Requirement 4.7.7.e.3 verifies the integrity of the control room enclosure, and the assumed inleakage rates of the potentially contaminated air. The control room positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper functioning of the Control Room HVAC. During the emergency mode of operation, the

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Control Room HVAC is designed to pressurize the control room to at least 1/8 inch water gauge (in-wg) positive pressure with respect to adjacent areas in order to prevent unfiltered inleakage. The Control Room HVAC is designed to maintain this positive pressure with two trains at a makeup flow rate of 2000 cfm. The frequency of 18 months is consistent with the guidance provided in NUREG-0800. If the surveillance results are less than 1/8 in-wg and the pressure differential is not positive, the surveillance requirement is considered not met and the appropriate action of TS 3.7.7 must be applied.

~~The footnote for Technical Specification Surveillance Requirement 4.7.7.e.3 has expired and is no longer applicable.~~

3/4.7.8 (Not used) ~~FUEL HANDLING BUILDING EXHAUST AIR SYSTEM~~

~~The FHB exhaust air system is comprised of two independent exhaust air filter trains and three exhaust ventilation trains. Each of the three exhaust ventilation trains has a main exhaust fan, an exhaust booster fan, and associated dampers. The main exhaust fans share a common plenum and the exhaust booster fans share a common plenum. An OPERABLE ventilation exhaust train consists of any OPERABLE main exhaust fan, any OPERABLE exhaust booster fan, and appropriate dampers.~~

~~The OPERABILITY of the Fuel Handling Building Exhaust Air System ensures that radioactive materials leaking from the ECCS equipment within the FHB following a LOCA are filtered prior to reaching the environment. Operation of the system with the heaters operating for the least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the safety analyses. ANSI N510-1980 will be used as a procedural guide for surveillance testing.~~

~~The time limits associated with the ACTIONS to restore an inoperable train to OPERABLE status are consistent with the redundancy and capability of the system and the low probability of a design basis accident while the affected train(s) is out of service. The allowed outage time for one train of FHB exhaust ventilation or one exhaust filtration train being inoperable, or a combination of an inoperable exhaust ventilation train and an inoperable exhaust filtration train is 7 days. With more than one inoperable train of either FHB exhaust filtration or FHB exhaust ventilation, or with combinations involving more than one inoperable train of either the exhaust ventilation or the exhaust filtration, the allowed outage time is 12 hours. A limited allowed outage time of 12 hours is allowed for multiple trains to be out of service simultaneously in recognition of the fact that there are common plenums and some maintenance or testing activities required opening or entry into these common plenums. This time is reasonable to diagnose, plan, and possibly repair problems with the boundary or the ventilation system. This is acceptable based on the low probability of a design basis event in that brief allowed outage time and because administrative controls impose compensatory actions that reduce the already small risk associated with being in the ACTION~~

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~~The compensatory actions are consistent with the intent of GDC 19, GDC 60 and Part 100 to protect plant personnel and the public from potential hazards such as radioactive contamination, smoke, and temperature, etc. Pre-planned measures should be available to address these concerns for intentional and unintentional entry into the condition. The compensatory action may include placing fans in pull-to-lock as necessary to preclude there being a motive force to transport contaminated air to a clean environment in the event of an accident. These compensatory actions include administrative controls on opening plenums or other openings such that appropriate communication is established with the control room to assure timely closing of the system if necessary. Since the Fuel Handling Building boundary integrity also affects operability of the overall system, entry and exit is administratively controlled. Administrative control of entry and exit through doors is performed by the person(s) entering or exiting the area. Extended opening of the boundary is coordinated with the control room with appropriate plans for closure and communication.~~

3/4.7.9 (Not Used)

3/4.7.10 (Not Used)

3/4.7.11 (Not used)

3/4.7.12 (Not used)

3/4.7.13 (Not used)

3/4.7.14 ESSENTIAL CHILLED WATER SYSTEM

The OPERABILITY of the Essential Chilled Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

When a risk-important system or component (for example Essential Chilled Water) is taken out of service, it is important to assure that the impact on plant risk of this and other equipment simultaneously taken out of service is assessed. The Configuration Risk Management Program evaluates the impact on plant risk of equipment out of service. A brief description of the Configuration Risk Management Program is in Section 6.8.3 (administrative controls) of the Technical Specifications.

The extended allowed outage time (EAOT) of 7 days for one inoperable Essential Chilled Water System loop is based on establishing compensatory measures that are consistent with the Configuration Risk Management Program and are controlled by plant procedures to offset the risk impacts of entering the EAOT. Refer to the Bases for 3.8.1.1. Action b for further details.

ELECTRICAL POWER SYSTEMS

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A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION (Continued)

The 10-year Frequency is consistent with the recommendations of Regulatory Guide 1.108, paragraph 2.b, and Regulatory Guide 1.137, paragraph C.2.f.

SR 4.8.1.1.2.9

This SR provided assurance that any accumulation of sediment over time or the normal wear on the system has not degraded the diesels.

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guides 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies," Revision 2, December 1979; 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977; and ASTM D975-81, ASTM D1552-79, ASTM D262282, ASTM D4294-83, and ASTM D2276-78. The standby diesel generators auxiliary systems are designed to circulate warm oil and water through the diesel while the diesel is not running, to preclude cold ambient starts. For the purposes of surveillance testing, ambient conditions are considered to be the hot prelube condition.

3.8.1.3

The OPERABILITY of the minimum AC sources during MODE 6 with ≥23' of water in the cavity is based on the following conditions:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

No Changes on this Page

c. Adequate AC electrical power is provided to mitigate events postulated during shutdown~~, such as a fuel handling accident~~.

In general, when the unit is shutdown, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many Design Basis Accidents (DBAs) that are analyzed in MODES 1,2,3, and 4 have no specific analyses in MODES 5 and 6. Worst case bounding events are deemed not credible in MODES 5 and 6 because the **reduced** energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

During MODES 1, 2, 3, and 4, various deviations from the analysis assumptions and design requirements are allowed within the Required Actions. This allowance is in recognition that certain testing and maintenance activities must be conducted provided an acceptable level of risk is not exceeded. During MODES 5 and 6, performance of a significant number of required testing and maintenance activities is also required. In MODES 5 and 6, the activities are generally planned and administratively controlled. Relaxations from MODES 1, 2, 3, and 4 LCO requirements are acceptable during shutdown modes based on:

a. **Limited** ~~The fact that~~ time in an outage ~~is limited~~. This is a risk prudent goal as well as a utility economic consideration.

b. Requiring appropriate compensatory measures for certain conditions. These may include administrative controls, reliance on systems that do not necessarily meet typical design requirements applied to systems credited in operating MODE analyses, or both.

c. Prudent utility consideration of the risk associated with multiple activities that could affect multiple systems.

d. Maintaining, to the extent practical, the ability to perform required functions (even if not meeting MODE 1,2, 3, and 4 OPERABILITY requirements) with systems assumed to function during an event.

In the event of an accident during shutdown, this LCO ensures the capability to support systems necessary to avoid immediate difficulty, assuming either a loss of all offsite power or a loss of all onsite diesel generator (DG) power.

3.8.2.1

In order to ensure the ability of the batteries to perform their intended function, the batteries are normally maintained in a fully charged state and the environment in which the batteries are located is maintained within the parameters used to determine battery sizing and

ELECTRICAL POWER SYSTEMS

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3.8.3.2

The OPERABILITY of the required DC sources and electrical distribution system during shutdown is based on the following conditions:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate AC electrical power is provided to mitigate events postulated during shutdown~~, such as a fuel handling accident~~.

In general, when the unit is shutdown, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many Design Basis Accidents (DBAs) that are analyzed in MODES 1, 2,3, and 4 have no specific analyses in MODES 5 and 6. Worst case bounding events are deemed not credible in MODES 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

Specifications 3.8.2.2 and 3.8.3.2 require DC power sources and specified electric power distribution for equipment required to be operable during shutdown. If the DC sources or distribution system is inoperable, then the Specifications require the affected components to be declared inoperable or that core alterations and positive reactivity changes be stopped. For a required system or component to be operable, the definition of OPERABLE/OPERABILITY requires the availability of necessary support systems, instrumentation, and electrical power for the required system to meet the design basis requirements. In MODES 5 and 6, the design basis does not include single failure coincident with loss of off-site power. Consequently, where two trains or channels of equipment are required by the Technical Specifications during MODES 5 and 6, only one of the trains or channels is required to be backed by an emergency power source or battery. Inoperability of the battery for one channel or train does not affect components that have an operable battery on the other required channel or train. Required electric power distribution systems must be operable under accident conditions that are applicable during shutdown, including seismic. For components that have only a detection function and no mitigation function during or after the accident, emergency power and safety related normal power are not required (e.g., Source Range instrumentation in Refueling Mode). When the function of those components is lost, the required actions to suspend core alterations or positive reactivity changes preclude the accident the components would be required to detect.

The ACTIONS specified during shutdown with less than the minimum required power sources or distribution systems, include suspending operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or refueling boron concentration necessary to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum SHUTDOWN MARGIN or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes, including temperature increases when operating with a positive moderator temperature coefficient, must also be evaluated to not result in operation below the required SHUTDOWN MARGIN or refueling boron concentration limits. Control rod withdrawal is not allowed except that it is permissible to unlock the control rods for rapid refueling. To unlock the control rods, they must be withdrawn at least one step. However, since the control rods are above the active fuel when the unlocking process occurs, there is no reactivity addition.

3/4.8.4 (Not Used)

No Changes on this Page

BASES

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the Source Range and/or Extended Range Neutron Flux Monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

ACTION a. requires suspending the introduction into the RCS of coolant with boron concentration less than required to meet the refueling boron concentration limit necessary to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes, including temperature increases when operating with a positive moderator temperature coefficient, must also be evaluated to not result in operation below the required refueling boron concentration limit. Control rod withdrawal is not allowed except that it is permissible to unlock the control rods for rapid refueling. To unlock the control rods, they must be withdrawn at least one step. However, since the control rods are above the active fuel when the unlocking process occurs, there is no reactivity addition.

3/4.9.3 (Not Used)

3/4.9.4 (Not Used) ~~CONTAINMENT BUILDING PENETRATIONS~~

~~The requirements on containment building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The containment personnel airlock and auxiliary airlock, which are part of the containment pressure boundary, provide a means for personnel access during MODES 1,2, 3, and 4 operation. The equipment hatch is required to be closed and sealed during MODES 1, 2, 3, and 4. During periods of shutdown, when containment closure is not required, the equipment hatch may be opened to allow passage of material needed to support activities in the containment building. The personnel and auxiliary airlock door interlock mechanisms may be disabled during shutdown, allowing both airlock doors to remain open for extended periods when frequent containment entry is necessary. Both containment personnel airlock doors may be open during CORE ALTERATIONS when specific limitations are satisfied. The specification requires: (1) there is 23 feet of water above the reactor vessel flange, (2) the reactor has been subcritical for 195 hours, (3) one airlock door is OPERABLE and, (4) an individual is available to close one personnel airlock door (if open) following a fuel handling accident inside containment.~~

~~The requirement to have 23 feet of water above the reactor vessel flange is consistent with the fuel handling accident analysis assumptions, Regulatory Guide 1.25, and Technical Specification 3.9.10, Water Level - Refueling Cavity.~~

~~Operability of a containment personnel airlock door requires that the door is capable of being closed, i.e., that the door is unblocked, no cables or hoses run through the personnel airlock, and at least one door seal is capable of being inflated. Containment personnel airlock door closure is required to take place within 30 minutes of initiation of a fuel handling accident inside containment if the reactor has been subcritical for less than 165 hours. Fuel movement is not permitted with personnel airlock doors open, if the reactor has not been subcritical for 295 hours. If the reactor has been subcritical for 165 hours or more, containment personnel airlock door closure is to occur as soon as practicable, but is assumed to occur within 2 hours to be consistent with the accident analysis.~~

3/4.9 REFUELING OPERATIONS

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~~3/4.9.4 CONTAINMENT BUILDING PENETRATIONS (continued)~~

~~The equipment hatch may also be open during CORE ALTERATIONS when specific limitations are satisfied. The specification requires: (1) the reactor has been subcritical for >165 hours and, (2) the equipment hatch (if open) is capable of being closed following a fuel handling accident inside containment. The following administrative requirements will apply whenever the equipment hatch is open during core alterations or the movement of irradiated fuel in containment:~~

~~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~~

~~3. Obstructions (e.g., cables, hoses, and runway) that would prevent closure of the equipment hatch can be quickly removed.~~

~~The containment equipment hatch closure is required to take place upon the occurrence of a fuel handling accident inside containment if the hatch is open. Fuel movement is not permitted with equipment hatch open, if the reactor has not been subcritical for 1165 hours. Equipment hatch closure should occur as soon as practicable, and is normally assumed to occur in 2 hours. Unlike the airlock, the equipment hatch may be blocked by an obstruction (e.g. the removable equipment hatch runway). Fuel movement is not allowed with the runway installed unless the capability to remove all obstructions and close the hatch within the required time is maintained.~~

~~A surveillance requirement verifies that the proper tools are staged at the equipment hatch location and qualified personnel assigned to close the equipment hatch on a seven-day frequency. These requirements assure that the associated doses are limited to within acceptable levels.~~

3/4.9.5 (Not Used)

3/4.9.6 (Not Used)

3/4.9.7 (Not Used)

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

As many as three residual heat removal (RHR) loops may be in operation, but at least one loop must be in operation at all times. One loop in operation ensures that: (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and at least 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

ACTIONS applicable when no RHR loop is in operation require suspending the introduction into the RCS of coolant with boron concentration less than required to meet the refueling boron concentration limit necessary to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes, including temperature increases when operating with a positive moderator temperature coefficient, must be evaluated to not result in operation below the required refueling boron concentration limit.

The 3000 gpm flow rate in 4.9.8.2 refers to total RHR flow through the core, i.e., cold leg injection flow. (CR 97-908-5)

3/4.9.9 (Not used) ~~CONTAINMENT VENTILATION ISOLATION SYSTEM~~

~~The OPERABILITY of this system ensures that the containment purge and exhaust penetrations will be automatically isolated upon detection of high radiation levels in the purge exhaust. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.~~

3/4.9.10 and 3/4.9.11 WATER LEVEL - REFUELING CAVITY AND STORAGE POOLS

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

REFUELING OPERATIONS

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3/4.9.12 (Not used) ~~FUEL HANDLTNG BUILDING EXHAUST AIR SYSTEM~~

~~The FHB exhaust air system is comprised of two independent exhaust air filter trains and three exhaust ventilation trains. Each of the three exhaust ventilation trains has a main exhaust fan, an exhaust booster fan, and associated dampers. The main exhaust fans share a common plenum and the exhaust booster fans share a common plenum. An OPERABLE ventilation exhaust train consists of any OPERABLE main exhaust fan, any OPERABLE exhaust booster fan and appropriate OPERABLE dampers.~~

~~The limitations on the Fuel Handling Building Exhaust Air System ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. Operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses. ANSI N510-1980 will be used as a procedural guide for surveillance testing. This Specification has been modified by a note that states, at least one FHB exhaust air filter train, one FHB exhaust booster fan, and one FHB main exhaust fan are capable of being powered from an Onsite emergency power source. This note ensures that required FKB exhaust train components will have an emergency power source available, even if the limiting conditions for operation can be satisfied.~~

Attachment 4

Planned Changes to the Technical Requirements Manual (information only)

TRM 3/4.9.4 – one page
TRM 3/4.9.12 – two pages
TRM 3/4.9.14 – two pages
TRM 3/4.9.15 – one page
TRM Bases – three pages

REFUELING OPERATIONS

3/4.9.12 CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITIONS FOR OPERATIONS

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

a. The equipment hatch closed and held in place by a minimum of four bolts
OR
1) The Reactor has been subcritical for ≥ 42 hours, AND
If open, the equipment hatch is capable of being closed within 2 hours.

b. A minimum of one door in the containment Auxiliary Airlock (AAL) and a minimum of one door in the containment Personnel Airlock (PAL) are closed.
OR
The water level is ≥ 23 feet above the reactor vessel flange.
AND
The Reactor has been subcritical for ≥ 42 hours
AND
Individuals are available to close a PAL door and AAL door when directed (after the initiation of a fuel handling accident inside containment) as soon as possible but within 2 hours.

c. All other penetrations providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1) Closed by an isolation valve, blind flange, or manual valve, or

2) Be capable of being closed (after the initiation of a fuel handling accident inside containment) as soon as possible but within 2 hours.

APPLICABILITY: During movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving movement of irradiated fuel in the containment building.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment building penetrations shall be determined to be either in its required condition or capable of being closed as required in specification 3.9.4 within 100 hours prior to the start of and at least once per 7 days during movement of irradiated fuel in the containment building by (as applicable):

a. Verifying the penetrations are in their required condition or capable of being placed in their required condition.

b. Proper tools are staged and trained personnel are designated to close the equipment hatch if open.

REFUELING OPERATIONS

3/4.9.12 FUEL HANDLING BUILDING EXHAUST AIR SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 One FHB Exhaust Air Train[1] shall be OPERABLE
OR
If not OPERABLE, capable of being restored to an OPERABLE status within two hours

APPLICABILITY: During the movement of fuel within the spent fuel pool or when conducting crane operation with loads over the spent fuel pool.

ACTION: With no FHB exhaust air train OPERABLE or capable of being restored to an OPERABLE status within two hours, suspend all operations involving movement of fuel within the spent fuel pool or crane operation with loads over the spent fuel pool.

SURVEILLANCE REQUIREMENTS

4.9.12 The FHB Exhaust Air System shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating with the operable exhaust booster fans and the operable main exhaust fans operating to maintain adequate air flow rate;

b. At least once per 18 months and (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

 1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% for HEPA filter banks and 0.10% for charcoal adsorber banks and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 29,000 cfm $\pm$ 10%;

 2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon," for a methyl iodide penetration of less than 1.0% when tested at a temperature of 30°C and a relative humidity of 70%; and

[1] At least one FHB exhaust air filter train, one FHB exhaust booster fan, and one FHB main exhaust fan are capable of being powered from an OPERABLE onsite emergency power source.

3/4.9.12 FUEL HANDLING BUILDING EXHAUST AIR SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

3) Verifying a system flow rate of 29,000 cfm ± 10% during system operation with two of the three exhaust booster fans and two of the three main exhaust fans operating when tested in accordance with ANSI N510-1980. All combinations of two exhaust booster fans and two main exhaust fans shall be tested.

c. After every 720 hours of charcoal adsorber operation by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989 for a methyl iodide penetration of less than 1.0% when tested at a temperature of 30°C and a relative humidity of 70%.

d. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 29,000 cfm ± 10%,

2) Verifying that the system maintains the spent fuel storage pool area at a negative pressure of greater than or equal to 1/8 inch Water Gauge relative to the outside atmosphere during system operation, and

3) Verifying that the heaters dissipate 38 ± 2.3 kW when tested in accordance with ANSI N510-1980.*

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the HEPA filter bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 29,000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the charcoal adsorber bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.10% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 29,000 cfm ± 10%.

REFUELING OPERATIONS

3/4.9.14 CONTROL ROOM MAKEUP AND CLEANUP FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.14 One Control Room Makeup and Cleanup Filtration System shall be OPERABLE
OR
If not OPERABLE, capable of being restored to an OPERABLE status within two hours.

APPLICABILITY: During the movement of irradiated fuel or when conducting crane operation with loads over the spent fuel pool.

ACTION: With no Control Room Makeup and Cleanup Filtration Systems OPERABLE or capable of being restored to an OPERABLE status within two hours, suspend all operations involving movement of irradiated fuel and crane operation with loads over the spent fuel pool.

SURVEILLANCE REQUIREMENTS

4.9.14 Each Control Room Makeup and Cleanup Filtration System shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to 78°F;

b. At least once per 92 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers of the makeup and cleanup air filter units and verifying that the system operates for at least 10 continuous hours with the makeup filter unit heaters operating;

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

1) Verifying that the makeup and cleanup systems satisfy the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% for HEPA filter banks and 0.10% for charcoal adsorber banks and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 6000 cfm $\pm$ 10% for the cleanup units and 1000 cfm $\pm$ 10% for the makeup units;

2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon," for a methyl iodide penetration of less than 1.0% when tested at a temperature of 30°C and a relative humidity of 70%; and

3) Verifying a system flow rate of 6000 cfm 10% for the cleanup units and 1000 cfm $\pm$ 10% for the makeup units during system operation when tested in accordance with ANSI N510-1980.

REFUELING OPERATIONS

3/4.9.14 CONTROL ROOM MAKEUP AND CLEANUP FILTRATION SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

d. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989 for a methyl iodide penetration of less than 1.0% when tested at a temperature of 30°C and a relative humidity of 70%.

e. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6.1 inches Water Gauge for the makeup units and 6.0 inches Water Gauge for the cleanup units while operating the system at a flow rate of 6000 cfm + 10% for the cleanup units and 1000 cfm & 10% for the makeup units.

2) Verifying that the system maintains the control room envelope at a positive pressure of greater than or equal to 1/8 inch Water Gauge at less than or equal to a pressurization flow of 2000 cfm relative to adjacent areas during system operation; and

3) Verifying that the makeup filter unit heaters dissipate 4.5 ± 0.45 kW when tested in accordance with ANSI N510-1980.

f. After each complete or partial replacement of a HEPA filter bank, by verifying that the HEPA filter bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0,05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 6000 cfm ± 10% for the cleanup units and 1000 cfm ± 10% for the makeup unlts; and

g. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the charcoal adsorber bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.10% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 6000 cfm ± 10% for the cleanup units and 1000 cfm ± 10% for the makeup units.

REFUELING OPERATIONS

3/4.9.15 RADIOACTIVE MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.15 The Unit Vent Gas Activity Monitor RT-8010B shall be functional.

APPLICABILITY: During the movement of irradiated fuel or when conducting crane operation with loads over the spent fuel pool.

ACTION: With the requirements of the above specification not met, ensure that a method is available for monitoring a radioactive release in the event of a fuel handling accident OR suspend operations involving the movement of irradiated fuel or loads over the spent fuel pool.

SURVEILLANCE REQUIREMENTS

4.9.15 Surveillance testing for the Unit Vent Gas Activity Monitor RT-8010B will be in accordance with the methodology in the ODCM.

REFUELING OPERATIONS

BASES

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

Containment building penetration closure is not credited or required as a mitigation function to meet the accident analyses. The OPERABILITY requirements provide the restoration of a monitored release path as a defense-in-depth measure to mitigate the consequences of a postulated FHA. This is consistent with NUMARC 93-01, Revision 3, Section 11.2.6, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants", subheading "Containment – Primary (PWR)/Secondary (BWR)." The closure requirement is a regulatory commitment for licensing the Alternative Source Term. (REF: Licensing Amendments No. xx and No. xx for Units 1 and 2 respectively)

The containment personnel airlock and auxiliary airlock, which are part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 operation. The equipment hatch is required to be closed and sealed during MODES 1, 2, 3, and 4. During periods of shutdown, when containment closure is not required, the equipment hatch may be opened to allow passage of material needed to support activities in the containment building. The personnel and auxiliary airlock door interlock mechanisms may be disabled during shutdown, allowing both airlock doors to remain open for extended periods when frequent containment entry is necessary. Both containment personnel airlock doors and/or auxiliary airlock doors may be open during the movement of irradiated fuel when specific limitations are satisfied. The specification requires: (1) there is 23 feet of water above the reactor vessel flange, (2) the reactor has been subcritical for 42 hours, (3) one airlock door in each containment entry point is OPERABLE and, (4) an individual is available to close one door in each entry point (if open) following a fuel handling accident inside containment.

The requirement to have 23 feet of water above the reactor vessel flange is consistent with the fuel handling accident analysis assumptions, Regulatory Guide 1.183, and Technical Specification 3.9.10, Water Level - Refueling Cavity.

Operability of each airlock entry door requires that the doors are capable of being closed, i.e., that the door is unblocked, no cables or hoses run through the airlock, and at least one door seal is capable of being inflated. Containment airlock door closure should occur as soon as practicable, but within at least 2 hours.

The equipment hatch may also be open during the movement of irradiated fuel when specific limitations are satisfied. The specification requires: (1) the reactor has been subcritical for 42 hours and, (2) the equipment hatch (if open) is capable of being closed following a fuel handling accident inside containment. The following administrative requirements will apply whenever the equipment hatch is open during the movement of irradiated fuel in containment:

1. Appropriate personnel are aware of the open status of the containment during movement of irradiated fuel.

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

3. Obstructions (e.g., cables, hoses, and runway) that would prevent closure of the equipment hatch can be quickly removed.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS (continued)

The containment equipment hatch closure is required to take place upon the occurrence of a fuel handling accident inside containment if the hatch is open. Fuel movement is not permitted with the equipment hatch open if the reactor has not been subcritical for 42 hours. Equipment hatch closure should occur as soon as practicable, and is normally assumed to occur in 2 hours. Unlike the airlock, the equipment hatch may be blocked by an obstruction (e.g. the removable equipment hatch runway). Fuel movement is not allowed with the runway installed unless the capability to remove all obstructions and close the hatch within the required time is maintained.

A surveillance requirement verifies that the proper tools are staged at the equipment hatch location and qualified personnel assigned to close the equipment hatch on a seven-day frequency.

3/4.9.12 FHB EXHAUST AIR SYSTEM

The FHB exhaust air system is comprised of two independent exhaust air filter trains and three exhaust ventilation trains. Each of the three exhaust ventilation trains has a main exhaust fan, an exhaust booster fan, and associated dampers. The main exhaust fans share a common plenum and the exhaust booster fans share a common plenum. An OPERABLE FHB Exhaust Air Train consists of any OPERABLE exhaust filter train, any OPERABLE main exhaust fan, any OPERABLE exhaust booster fan and appropriate OPERABLE dampers.

The Fuel Handling Building Exhaust Air System is not credited or required as a mitigation function to meet the accident analyses. The OPERABILITY requirements provide the restoration of a filtered release path as a defense-in-depth measure to mitigate the consequences of a postulated FHA. This is consistent with NUMARC 93-01, Revision 3, Section 11.2.6, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants", subheading "Containment – Primary (PWR)/Secondary (BWR)." The OPERABILITY requirement is a regulatory commitment for licensing the Alternative Source Term. (REF: Licensing Amendments No. xx and No. xx for Units 1 and 2 respectively)

Operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The iodine removal capacity of the system is consistent with NRC RG 1.52. ANSI N510-1980 will be used as a procedural guide for surveillance testing. This Specification has been modified by a note that states, at least one FHB exhaust air filter train, one FHB exhaust booster fan, and one FHB main exhaust fan are capable of being powered from an Onsite emergency power source. This note ensures that required FHB exhaust train components will have an emergency power source available.

3/4.9.14 CONTROL ROOM MAKEUP AND CLEANUP FILTRATION SYSTEM

The Control Room Makeup and Cleanup Filtration System is comprised of three 50-percent redundant systems (trains) that share a common intake plenum and exhaust plenum. Each system/train is comprised of a makeup fan, a makeup filtration unit, a cleanup filtration unit, a

3/4.9.14 CONTROL ROOM MAKEUP AND CLEANUP FILTRATION SYSTEM (continued)

cleanup fan, a control room air handling unit, a supply fan, a return fan, and associated ductwork and dampers.

The Control Room Makeup and Cleanup Filtration System is not credited or required as a mitigation function to meet the accident analyses for a fuel handling accident. The OPERABILITY requirement during the movement of irradiated fuel or when irradiated fuel is in the spent fuel pool provides for the restoration of a filtered path as a defense-in-depth measure to further lower the consequences to the control room operator from a postulated FHA. This is consistent with NUMARC 93-01, Revision 3, Section 11.2.6, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants", subheading "Containment – Primary (PWR)/Secondary (BWR)" The OPERABILITY requirement is a regulatory commitment for licensing the Alternative Source Term. (REF: Licensing Amendments No. xx and No. xx for Units 1 and 2 respectively)

3/4.9.15 RADIOACTIVE MONITORING INSTRUMENTATION

The Radioactive Monitoring Instrumentation provides the capability of monitoring a release from a fuel handling accident in either the Reactor Containment Building or the Fuel Handling Building. This is consistent with NUMARC 93-01, Revision 3, Section 11.2.6, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants", subheading "Containment – Primary (PWR)/Secondary (BWR)". Maintaining this instrumentation functional when moving irradiated fuel or performing operations involving crane operation with loads over the spent fuel pool is a regulatory commitment for licensing the Alternative Source Term. (REF: Licensing Amendments No. xx and No. xx for Units 1 and 2 respectively)

Attachment 5

List of Commitments

List of Commitments

The following table identifies those actions committed to by STPNOC in this document. Any statements in this submittal with the exception of those in the table below are provided for information purposes and are not considered commitments. Please direct questions regarding these commitments to Ken Taplett at (361) 972-8416.

Using insights from TSTF-51, and consistent with the guidance in NUMARC 93-01, Revision 3, Section 11.3.6.5, "Safety Assessment for Removal of Equipment from Service During Shutdown Conditions," subheading "Containment - Primary (PWR)/Secondary (BWR)," STPNOC makes the following commitments to mitigate the consequences of a potential fuel handling accident.

NOTE: The purpose of these commitments are to maintain the Fuel Handling Building (FHB) Ventilation System and associated radiation monitoring availability to reduce doses even further below that provided by the natural decay and to avoid unmonitored releases; and to enable the FHB Ventilation System to draw the release from a postulated fuel handling accident in the FHB in the proper direction such that it can be treated and monitored.

In addition, the purpose of these commitments is to isolate the Reactor Containment Building (RCB) for postulated fuel handling accident in the RCB and further reduce dose by natural decay and to enable the Ventilation System to draw the release from a postulated fuel handling accident in the containment in the proper direction such that it can be monitored (not treated).

Commitment	Continuing Compliance	Scheduled Completion Date
1. Whenever fuel is being moved in the spent fuel pool or when conducting crane operation with loads over the spent fuel pool, at least one train of FHB Exhaust Air shall be OPERABLE or capable of being restored to an OPERABLE status within two hours[1].	X	Upon Implementation
2. Whenever irradiated fuel is being moved or when conducting crane operation with loads over the spent fuel pool, at least one Control Room Makeup and Cleanup Filtration System shall be OPERABLE or capable of being restored to an OPERABLE status within two hours[1].	X	Upon Implementation

Commitment	Continuing Compliance	Scheduled Completion Date
3. Whenever irradiated fuel is being moved within the Reactor Containment Building, the following will be closed or capable of being closed within two hours[1]. a. The equipment hatch b. At least one door in the Auxiliary Airlock and one door in the Personnel Airlock. c. All other penetrations providing direct access from the containment atmosphere to the outside atmosphere.	X	Upon Implementation
4. Within two hours[1] of a fuel handling accident in the FHB, at least one train of FHB Exhaust Air will be placed in operation.	X	Upon Implementation
5. Within two hours[1] of a fuel handling accident, at least one Control Room Makeup and Cleanup Filtration System will be placed in operation.	X	Upon Implementation
6. Within two hours[1] of a fuel handling accident in the Reactor Containment Building, the following actions will be taken: a. Close the equipment hatch, b. Close at least one of the Auxiliary Airlock doors and one of the Personnel Airlock doors, and c. Close all other penetrations providing direct access from the containment atmosphere to the outside atmosphere.	X	Upon Implementation
7. Whenever irradiated fuel is being moved or when conducting crane operation with loads over the spent fuel pool, radiation monitoring instrumentation will remain functional to ensure that a release following a fuel handling accident is monitored.	X	Upon Implementation

[1] The two hours to restore the FHB Exhaust Air System and the Control Room Makeup and Cleanup Filtration System to OPERABLE status and to close containment penetrations or openings in the event of a fuel handling accident is reasonable because these systems are not required to mitigate the accident. These systems are not credited in the accident analyses. Dose limits are within requirements assuming an instantaneous release from the FHA. These additional administrative actions are taken to further filter and monitor the release as a defense-in-depth measure.

For License Amendments 139/128, Units 1 and 2 respectively, two hours to close the equipment hatch was acceptable to meet the dose guidelines of 10 CFR 100 and General Design Criterion 19. For this proposed licensing amendment request, the dose guidelines of 10 CFR 100 and General Design Criterion 19 are met without restoring the systems described above. Therefore, two hours is a reasonable time to put these defense-in-depth measures in place.

Reference: South Texas Project, Units 1 and 2 – Issuance of Amendments 139/128, Units 1 and 2 respectively, dated July 18, 2003 on Equipment Hatch Open During Refuel Operations (TAC NOS. MB3587 and MB3591)

Other Commitment		
Commitment	**Continuing Compliance**	**Scheduled Completion Date**
8. Until a plant modification is completed for supporting the limiting single failure assumptions in the steam generator tube rupture (SGTR) analysis, STP will maintain an administrative limit for reactor coolant system dose equivalent iodine so that the radiological dose limits for the SGTR analysis remain bounding.	X	Upon Implementation

Attachment 6

Regulatory Guide 1.183 Conformance Tables

REGULATORY GUIDE 1.183 CONFORMANCE TABLE

Notes: [a] Any reference to Tables or Sections in this column refers to Attachment 1, Section 4.0, "Technical Analysis" of the Licensee Evaluation

Table A: Conformance with Regulatory Guide 1.183 Main Sections

RG Section	Regulatory Position	Analysis	Comments [a]
3.1 – AST Fission Product Inventory	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. The uncertainty factor used in determining the core inventory should be that value provided in Appendix K to 10 CFR Part 50, typically 1.02.	Conforms	The core power level assumed for the DBA analyses is 4100 MWt. This is greater than the TS rated thermal power of the core of 3853 MWt with an uncertainty factor of 0.6%. The peak burnup assumed is 60030 MWD/MTU. The inventory of fission products is based on the current licensed values for fuel enrichment. (Section 4.2.4.2)
	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.		The assumed period of irradiation was sufficient (three-region equilibrium cycle core at end of life with the three regions having operated at 39.31 MW/MTU for 509, 1018, and 1527 EFPD, respectively) to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. (Section 4.2.4.2)
	The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 or ORIGEN-ARP.		The ORIGEN 2.1 code was used to calculate plant-specific fission product inventories. (Section 4.2.4.2)

Table A: Conformance with Regulatory Guide 1.183 Main Sections			
RG Section	**Regulatory Position**	**Analysis**	**Comments [a]**
	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. 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.		For the DBA LOCA, all fuel assemblies in the core are assumed to be affected and the core average inventory is used. (Section 4.3.5) The fission product inventory for the FHA is provided in Table 4.4-1. The core inventory for each fission product in Table 4.4-1 is multiplied by the peaking factor of 1.7, which bounds values in the Core Operating Limits Report, and by the fraction of the fuel in the core that is damaged (314 pins out of 50952). (Section 4.4.3) For the remaining accidents, the fuel damage is specified in the accident discussion in Section 4. All accident analyses were performed assumed 4100 Mwt. Rated full power for the STP units is 3853 Mwt. Each accident analyses in Section 4.0. A radioactive decay time of 42 hours while the facility is shutdown in modeled in the FHA. (Section 4.4.3)
3.2 – Release fractions	The core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage phases for DBA LOCAs are listed in Table 2 for PWRs. These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1. For non-LOCA events, the fractions of the core	Conforms	The DBA LOCA release fractions given as release rates over the given duration are found in Table 4.3-10. For non-LOCA events, the fractions of the

Table A: Conformance with Regulatory Guide 1.183 Main Sections			
RG Section	**Regulatory Position**	**Analysis**	**Comments [a]**
	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.		core inventory assumed to be in the gap for the various radio-nuclides are based on Table 3. See sections 4.4.3 (FHA), 4.7.3.1.1 (CREA), and 4.8.3 (LRA). The MSLB and SGTR accidents do not assume clad damage.
3.3 – Timing of Release Phases	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. 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. 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.	Conforms	The DBA LOCA release fractions given as release rates over the given duration are found in Table 4.3-10. The LOCA activity released from the core is modeled in a linear fashion over the duration of the release phases. This is true, except that the elemental and particulate iodines released from the fuel for an accident-induced spike during a SGTR are released into the RCS over time. (4.6.3.1.1) Leak before break is not credited in the AST analyses.
3.4 – Radionuclide Composition	Table 5 lists the elements in each radionuclide group that should be considered in design basis analyses.	Conforms	The fission product inventory for the LOCA is listed in Table 4.3-1. This table does not list Br, Se, Pd, Co, Eu, Pm & Sm – elements listed in Table 5 of RG 1.183 In addition to the radionuclides appearing in

Table A: Conformance with Regulatory Guide 1.183 Main Sections			
RG Section	Regulatory Position	Analysis	Comments [a]
			the RADTRAD list, Kr83m, Xe131m, Xe133m, and Xe135m were added for dose analysis purposes based on their inclusion in TID-14844. Xe138 was also added. Co58 and Co60 were deleted from the list because only 63 radionuclides can be used. A study performed for another licensee indicated that omitting Co58 and Co 60 decreased the control room dose by about 0.01 percent while adding the noble gas isotopes increased the control room dose by about 0.1 percent. (Section 4.3.2)
3.5 – Chemical Form	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	95 percent of the iodine released from the reactor coolant system to the containment is 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. Fission products are assumed to be in particulate form, with the exception of elemental and organic iodine and noble gases. (Section 4.3.5) 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. The accident-specific appendices to this regulatory guide provide additional details.
3.6 – Fuel	The amount of fuel damage caused by non-LOCA	Conforms	The non-LOCA design bases analyses used

Table A: Conformance with Regulatory Guide 1.183 Main Sections			
RG Section	**Regulatory Position**	**Analysis**	**Comments [a]**
Damage in Non-LOCA DBAs	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.		DNBR as a fuel damage criterion. (Section 4.2.4.2)
4.1 – Offsite Dose Consequences The following assumptions should be used in determining the TEDE for persons located at or beyond the boundary of the exclusion area (EAB)			
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.	Conforms	TEDE is calculated, with significant progeny included. The DBA radiological consequences are listed in a "Summary and Conclusions" table at the end of the discussion of each DBA in Section 4.
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	Conforms	Table 2.1 of Federal Guidance Report 11 were used. (Section 4.2.4.1)

Table A: Conformance with Regulatory Guide 1.183 Main Sections			
RG Section	**Regulatory Position**	**Analysis**	**Comments [a]**
	factors acceptable to the NRC staff.		
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 standard breathing rates specified in RG 1.183. (Table 4.2-1)
4.1.4	The DDE should be calculated assuming submergence in semi-infinite cloud assumptions with appropriate credit for attenuation by body tissue. EDE may be used in lieu of DDE in determining the contribution of external dose to the TEDE.	Conforms	The analyses were performed using the NRC RADTRAD computer code. (Section 4.2.1)
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. 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.	Conforms	The TEDE was determined for the most limiting person at the EAB and the maximum two-hour dose has been reported. (Section 4.2.1)
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 person at the LPZ. (Section 4.2.1)
4.1.7	No correction should be made for depletion of the		No plume depletion has been credited.

Table A: Conformance with Regulatory Guide 1.183 Main Sections			
RG Section	**Regulatory Position**	**Analysis**	**Comments** [a]
	effluent plume by deposition on the ground.		(Section 4.2.1)
4.2 – Control Room Dose Consequences			
4.2.1	The TEDE analysis should consider all sources of radiation that will cause exposure to control room personnel.	Conforms	The DBA LOCA radiation dose to personnel in the CR and TSC includes the gamma shine from the primary containment airborne activity (CR and TSC), from airborne activity in the electrical penetration area (CR only), from activity in the radioactive cloud surrounding the plant structures (CR and TSC), and from trapped activity on filters (CR and TSC). (Section 4.3.4.2)
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 non-conservative results for the control room.	Conforms	Gamma shine dose contribution to the control room is discussed in Section 4.3.4.2. Source to receptor models are discussed in Section 4.1.3.
4.2.3	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, should be structured to provide suitably conservative estimates of the exposure to control room personnel.	Conforms	Gamma shine dose contribution to the control room is discussed in Section 4.3.4.2.
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-	Conforms	For the DBA LOCA, no credit is taken for any filtration other than for the recirculation filters for the CR. The recirculation filter features are qualified and acceptable per the referenced guidance. (Section 4.2.2) For the FHA, no credit is taken for any filtration (make-up or recirculation clean-up)

Table A: Conformance with Regulatory Guide 1.183 Main Sections			
RG Section	**Regulatory Position**	**Analysis**	**Comments [a]**
	Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" (Ref. 25), for guidance.		for either the CR or the TSC. (Section 4.2.2 and 4.2.3)
4.2.5	Credit should generally not be taken for the use of personal protective equipment or prophylactic drugs.	Conforms	No credit is taken for the use of personal protective equipment or prophylactic drugs. (Section 4.2.2 and 4.2.3)
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. 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 standard breathing rates specified in RG 1.183 and the standard CR and TSC occupancy factors specified in RG 1.183 have been used. (Table 4.2-3)
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.	Conforms	See response to Position 4.1.2.
4.3 – Other Dose Consequences	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. 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.		Technical Support Center (TSC) doses were calculated for the analyzed accidents and the results are found in the "Summary and Conclusions" section of each accident discussion in Section 4. The radiation doses used for the current licensing basis environmental qualification analyses were calculated using source terms determined by TID-14844 methodology.

Table A: Conformance with Regulatory Guide 1.183 Main Sections			
RG Section	**Regulatory Position**	**Analysis**	**Comments** [a]
			AST impact on these doses has been considered. (Section 1.0)
4.4 – Acceptance Criteria	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 19 (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 an AST, the applicable criteria should be updated for consistency with the TEDE criterion in 10 CFR 50.67(b)(2)(iii).		All accident analyses consequences meet 10 CFR 50.67 and the SRP. See the "Summary and Conclusions" section of each accident discussion in Section 4. The accident analyses consequences are well within the criteria tabulated in Table 6. See the "Summary and Conclusions" section of each accident discussion in Section 4. For post-accident vital area access, the results of the assessment of dose impact of containment shine demonstrates that the current calculated doses (based on TID-14844 source terms) bound the corresponding doses that would be calculated based on the AST. (Section 4.9) The post-accident containment high range radiation monitors are determined not to be impacted by the AST. (Section 4.9) The CR radiological dose impact of AST is specifically calculated for the six Design Basis Accidents. See the "Summary and Conclusions" section of each accident

Table A: Conformance with Regulatory Guide 1.183 Main Sections			
RG Section	Regulatory Position	Analysis	Comments [a]
			discussion in Section 4.
5.1 – General Considerations for Analysis Assumptions and Methodology			
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. These design basis analyses were structured to provide a conservative set of assumptions to test the performance of one or more aspects of the facility design. Licensees should exercise caution in proposing deviations based upon data from a specific accident sequence since the DBAs were never intended to represent any specific accident sequence -- the proposed deviation may not be conservative for other accident sequences.	Conforms	Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50 was applied. (Section 4.0) Care has been taken to ensure that dose analyses have not been "tuned" to a specific set of accident progression assumptions. (Section 4.0)
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		For the DBA LOCA, credit is taken for the CR recirculation filters. These are safety-related and are required to be operable by TS 3.7.7. They are powered by emergency power sources and are automatically actuated by SI signal. TS 3.3.2 applies. (Section 4.2.2) Without credit being taken for the FHB filters or for the CR make-up filters (and the associated heaters to control intake humidity),

Table A: Conformance with Regulatory Guide 1.183 Main Sections			
RG Section	Regulatory Position	Analysis	Comments [a]
	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.		the single-failure assessment becomes much simpler for application of the AST than that of the CLB. For the AST DBA LOCA, an electrical division electrical failure is assumed as a single failure to minimize containment mixing via the containment fan coolers. This assumption maximizes dose. Only two out of three trains of containment ventilation are assumed to operate, and one fan-cooler on one of the operating trains is assumed to be out of service, as well. The spray removal lambdas used are also consistent with the loss of one spray train, as are the assumptions regarding CR ventilation and filtration. (Section 4.3.4.1) The LOCA, MSLB, SGTR, CREA and LRA analyses assume a loss of offsite power concurrent with the accident.
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 non-conservative in another portion of the same analysis.	Conforms	Conservative parameters were used when calculating components in the dose analyses. (Section 4.0)
5.1.4	In order to issue a license amendment authorizing the use of an AST and the TEDE dose criteria, the NRC staff must make a current finding of compliance with regulations applicable to the amendment. The	Conforms	The analysis assumptions and methods are compatible with the ASTs and the TEDE criteria.

Table A: Conformance with Regulatory Guide 1.183 Main Sections			
RG Section	**Regulatory Position**	**Analysis**	**Comments [a]**
	characteristics of the ASTs and the revised dose calculational methodology may be incompatible with many of the analysis assumptions and methods currently reflected in the facility's design basis analyses. Licensees should ensure that analysis assumptions and methods are compatible with the ASTs and the TEDE criteria.		
5.2 – Accident-Specific Assumptions for Analysis Assumptions and Methodology	Licensees should analyze the DBAs that are affected by the specific proposed applications of an AST. The NRC staff has determined that the analysis assumptions in the appendices to this guide provide an integrated approach to performing the individual analyses and generally expects licensees to address each assumption or propose acceptable alternatives. The NRC will consider licensee proposals for changes in analysis assumptions based upon risk insights. The staff will not approve proposals that would reduce the defense in depth deemed necessary to provide adequate protection for public health and safety.	Conforms	The postulated accident radiological consequence analyses specified in this RG are updated for AST implementation impact. Each assumption is addressed with conformance with the RG accident analyses assumptions provided by this table. No changes were made to analysis assumptions based upon risk insights. (Section 4.0) Defense-in-depth has not been compromised by the changes proposed in this application. (Section 5.2)
5.3 – Meteorology Assumptions	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. References 22 and 28 of this RG should be used if the FSAR χ/Q values are to be revised or if values are to be	Conforms	All χ/Q values have been recalculated for the AST application. (Section 4.1) Reference 28 of RG 1.183 has been used to calculate offsite χ/Q values (for the EAB, the

Table A: Conformance with Regulatory Guide 1.183 Main Sections			
RG Section	Regulatory Position	Analysis	Comments [a]
	determined for new release points or receptor distances.		LPZ). RG 1.194 guidance has been used, as well. Reference 22 of RG 1.183 has not been used. (Sections 4.1.2 and 4.1.3)
	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.		Fumigation has not been included since no credit is taken for an elevated release.
	The NRC computer code PAVAN implements Regulatory Guide 1.145 and its use is acceptable to the NRC staff.		The PAVAN code was used for determining of offsite χ/Q values. (Section 4.1.2)
	The methodology of the NRC computer code ARCON96 is generally acceptable to the NRC staff for use in determining control room χ/Q values.		ARCON96 was used for determining χ/Q values for onsite receptors near building structures. (Section 4.1.3)
	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.		Recently acquired meteorological data (five years from 2000 to 2004) is used to calculate onsite and offsite atmospheric dispersion. (Section 4.1)
	All changes in χ/Q analysis methodology should be reviewed by the NRC staff.		χ/Q values for radiological dose calculations are found in Tables 4.1-24 and 4.1-37.
5.6 – Assumptions for Evaluating the Radiation Doses for Equipment	The assumptions in Appendix I to this guide are acceptable to the NRC staff for performing radiological assessments associated with equipment qualification. The assumptions in Appendix I will supersede Regulatory Positions 2.c(1) and 2.c(2) and Appendix D of Revision 1 of Regulatory	Conforms	The radiation doses used for the CLB environmental qualification analyses were calculated using source terms determined by TID-14844 methodology. (Section 1.0)

Table A: Conformance with Regulatory Guide 1.183 Main Sections			
RG Section	**Regulatory Position**	**Analysis**	**Comments [a]**
Qualification	Guide 1.89, for operating reactors that have amended their licensing basis to use an alternative source term. Except as stated in Appendix I, all other assumptions, methods, and provisions of Revision 1 of Regulatory Guide 1.89 remain effective. 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. However, no plant modifications are required to address the impact of the difference in source term characteristics (i.e., AST vs TID14844) on EQ doses pending the outcome of the evaluation of the generic issue.		

Table B – Comparison with Regulatory Guide 1.183 Appendix A (PWR Loss-of-Coolant Accident)			
RG Section	**Regulatory Position**	**Analysis**	**Comments [a]**
1 – Source Term Assumptions	Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.	Conforms	The core inventory and release of radio-nuclides in the AST analysis were derived using the guidance outlined in this RG. ORIGIN 2.1 code was used to calculate plant-specific fission product inventories. (Section 4.3.2) The release fractions are provided in Table 4.3-10.
2 – Source Term Assumptions	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	Sump pH is less than 7. The plant-	A calculation was performed to evaluate containment sump pH in the event of a DBA LOCA. The objective of the analysis was to

Table B – Comparison with Regulatory Guide 1.183 Appendix A (PWR Loss-of-Coolant Accident)			
RG Section	**Regulatory Position**	**Analysis**	**Comments [a]**
	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.	specific calculation will be provided.	determine the transient containment sump pH so that the removal of elemental and particulate iodine (cesium iodide - CsI) from the containment atmosphere in the course of the DBA LOCA would not be overstated. The analysis credits the pH buffering effect of trisodium phosphate (TSP) stored in the containment sump. The pH decreases slightly below 7.0 over the 30-day duration of the radiological consequence analysis for the DBA-LOCA, and the impact of that decrease has been reflected in the CR, TSC, and offsite doses. Because the pH of the containment sump falls below 7.0 after one day, a fractional iodine release for ESF leakage greater than 10% was considered per RG 1.183. (Sections 4.3.3.1.1)
3 – Assumptions on Transport in Primary Containment			
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 as it is released. This distribution should be adjusted if there are internal compartments that have limited ventilation exchange.	Conforms	The radioactivity release from the fuel is assumed to mix instantaneously and homogeneously throughout the containment air space as it is released. (Section 4.3.5)
3.2	Reduction in airborne radioactivity in the containment by natural deposition within the containment may be credited. The prior practice of deterministically assuming that a 50% plate-out 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	Conforms	The natural removal rate for elemental iodine is 4.5 per hour (Section 4.3.5).

Table B – Comparison with Regulatory Guide 1.183 Appendix A (PWR Loss-of-Coolant Accident)			
RG Section	**Regulatory Position**	**Analysis**	**Comments [a]**
	source terms.		
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 may be credited. 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 un-sprayed 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.	Conforms	The containment spray systems are designed and maintained in accordance with Chapter 6.5.2 of the SRP. Values of reduction in airborne radioactivity in the containment by containment spray systems is given in Table 4.3-13 and discussed in Section 4.3.5. Forced mixing crediting containment fan-cooler units is used. The AST spray removal parameters are given in Table 4.3-13. The table demonstrates that the particulate iodine removal rate is reduced from 6.9 to 0.7 when a DF of 50 is reached.
3.7	The primary containment should be assumed to leak at	Conforms	The volumetric leak rate from containment is

Table B – Comparison with Regulatory Guide 1.183 Appendix A (PWR Loss-of-Coolant Accident)			
RG Section	**Regulatory Position**	**Analysis**	**Comments [a]**
	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.		assumed to be 0.3%/day for the first 24 hours and 0.15%/day for the remainder of the accident – this is consistent with the CLB. (Section 4.3.3.1) Maximum post-LOCA containment temperature and pressure were assumed. (Section 4.3-11)
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.	Bounding	Primary containment is routinely purged during power operations. Releases via the purge system prior to containment isolation are analyzed and the resulting doses are summed with the postulated doses from other release paths. The reactor coolant concentrations are based on 1% failed fuel that is greater than the TS limit of 1.0 μCi/gm. Containment leakage is assumed via open purge lines for the first 23 seconds of the accident. This leakage is released to the environment via the plant vent. (Section 4.3.3.1.1)
4 – Assumptions of Dual Containment		Not Applicable	
5 – Assumptions on ESF System	ESF systems that re-circulate sump water outside of the primary containment are assumed to leak during their intended operation. The radiological consequences from the postulated leakage should be analyzed and combined	Conforms	The radiological consequences from the postulated ESF systems leakage is analyzed and combined with consequences postulated for other fission product release paths.

Table B – Comparison with Regulatory Guide 1.183 Appendix A (PWR Loss-of-Coolant Accident)			
RG Section	**Regulatory Position**	**Analysis**	**Comments [a]**
Leakage	with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA.		(Section 4.3.3.2)
5.1	With the exception of noble gases, all the fission products released from the fuel to the containment should be assumed to instantaneously and homogeneously mix in the primary containment sump water 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.	Conforms	For determination of the dose contribution from ESF leakage, all radionuclides assumed to be released from the core (except noble gases) are assumed to be instantaneously and homogeneously mixed in the containment sump. (Section 4.3.3.2)
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 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 mini-flow return to the refueling water storage tank.	Conforms	ESF leak rate is 4140 cc/hr and analyzed as 8280 cc/hr per the CLB. ESF leakage is assumed to begin at time = 0. Leakage past valves and into tanks vented to the atmosphere has been evaluated to have no impact on the dose consequences because of the distances involved and the time of travel. (Section 4.3.3.2)
5.3	With the exception of iodine, all radioactive materials in the re-circulating liquid should be assumed to be retained in the liquid phase.	Conforms	Complies. (Section 4.3.3.2)
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	Higher amount assumed	The fractional release of iodine from ESF sump water leakage uses a value of 10% only for the first 24 hours. Beyond that time, the

Table B – Comparison with Regulatory Guide 1.183 Appendix A (PWR Loss-of-Coolant Accident)			
RG Section	**Regulatory Position**	**Analysis**	**Comments** [a]
	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.	because sump pH goes below 7.0	release fraction is increased to 16% (24-480 hours) and then to 25% (480 hours to 720 hours) based on the calculated volatility of the iodine for the pH values over those intervals relative to the volatility for a pH of 7.0. (Section 4.3.3.2)
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.	Conforms	The radioiodine that is postulated to be available for release to the environment is assumed to be 97% elemental and 3% organic. The ESF leakage is assumed to release directly into the FHB from the leaked reactor coolant. It is then assumed to be released instantaneously into the environment without benefit of filtration via the plant vent. (Table 4.3-11 & Section 4.3.3.2)
6 – Assumption on Main Steam Isolation Valve Leakage in BWRs		Not Applicable	
7 – Assumption on Containment Purging	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.	Not Applicable	Hydrogen control by purge is not part of the licensing basis. (Ref: Amendment 165 for Unit 1 and 155 for Unit 2 – TAC Nos. MC4229 and MC4290)

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

RG Section	Regulatory Position	Analysis	Comments [a]
1 – Source Term	Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.	Conforms	The assumptions regarding core inventory and the release of radionuclides from the fuel are consistent with Regulatory Position 3 of this guide. (Tables 4.4-1 and 4.4-2)
1.1	The number of fuel rods damaged during the accident should be based on a conservative analysis that considers the most limiting case.	Conforms	The mechanical part of the analysis remains unchanged from the STPEGS CLB; the total number of failed fuel rods is 314 (out of 50952 for an entire core). (Section 4.4.5)
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 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. Since alkali metal releases (as particulates) are assumed to experience an infinite DF due to the water submergence (per RG 1.183), no alkali metals (e.g., Cs and Rb) are included. (Sections 4.4.4 and 4.4.5)
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.	Conforms	This position is not explicitly used. Given the 23 feet of water depth and the effective DF of 200, the iodine released to the atmosphere is 57% elemental and 43% organic, as discussed in Position 2 of Appendix B of RG 1.183. (Section 4.4.4)
2 – Water Depth	If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental	Conforms	The depth of water over the damaged fuel is not less than 23 feet.

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

RG Section	Regulatory Position	Analysis	Comments [a]
	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.		Due to the submergence of the damaged fuel, the iodine release is assumed to experience a DF of 200. The assumed iodine chemical form after decontamination by the water pool is 43% organic and 57% elemental. (Section 4.4.4)
3 – Noble Gases	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	No DF is applied to the noble gas. The DF for particulate is assumed to be infinite. (Section 4.4.4)
4 – FHA Within the Fuel Building			
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.	Assumption more conservative	Following accident initiation at t = 42 hours after shutdown, the radionuclide inventory from the damaged fuel pins is assumed to leak out to the environment instantaneously. (Sections 4.4.1 and 4.4.2)
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.	Not Applicable	No credit is taken for filtration by the FHB filters or for hold-up in the FHB. (Section 4.4.2)
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.	Not Applicable	Following accident initiation at t = 42 hours after shutdown, the radionuclide inventory from the damaged fuel pins is assumed to leak out to the environment instantaneously.

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

RG Section	Regulatory Position	Analysis	Comments [a]
			(Sections 4.4.1 and 4.4.2)
5 – FHA Within Containment			
5.3	If the containment is open during fuel handling operations (e.g., personnel air lock or equipment hatch is open), the radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period.	Assumption more conservative	Following accident initiation at t = 42 hours after shutdown, the radionuclide inventory from the damaged fuel pins is assumed to leak out to the environment instantaneously. (Sections 4.4.1 and 4.4.2)
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.	Not Applicable	Following accident initiation at t = 42 hours after shutdown, the radionuclide inventory from the damaged fuel pins is assumed to leak out to the environment instantaneously. There is no credit taken for activity removal other than by scrubbing by the water in the refueling cavity. (Sections 4.4.1 and 4.4.2)
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.	Not Applicable	Following accident initiation at t = 42 hours after shutdown, the radionuclide inventory from the damaged fuel pins is assumed to leak out to the environment instantaneously. (Section 4.4.2)

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

RG Section	Regulatory Position	Analysis	Comments [a]
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	Conforms	Assumptions regarding core inventory and the release of radionuclides from the fuel are consistent with Regulatory Position 3 of this regulatory guide. No fuel damage is postulated to occur for the Main Steam Line

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

RG Section	Regulatory Position	Analysis	Comments [a]
	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.		Break. (Section 4.5.3)
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. Footnote 2: The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum technical specification values, whichever maximizes the radiological consequences. In determining dose equivalent I-131 (DE I-131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.	Conforms	The activity assumed in the analysis is based on the activity associated with the maximum technical specification values. In determining dose equivalent I-131, only the radioiodine associated with normal operations or iodine spikes is included. (Section 4.5.5)
2.1	<u>Case 1</u> 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	This analyzed case involves a 60 μCi/gm pre-accident iodine spike, consistent with the Technical Specification operational Reactor Coolant System activity concentration limit for assumed spike. All of the spike activity is homogeneously mixed in the primary coolant, prior to accident initiation. (Section 4.5.1)
2.2	<u>Case 2</u> 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	Conforms	This case involves an accident initiated iodine spike that occurs concurrently with the release of the fluid from the primary and secondary coolant systems. This transient is associated with an iodine spike which assumes that the

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

RG Section	Regulatory Position	Analysis	Comments [a]
	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.		iodine release rate from the fuel rods to the primary coolant increases to a value 500 times greater than the release rate corresponding to the 1.0 μCi/gm DE I-131 RCS equilibrium iodine. The elemental and particulate iodines release rate spike is assumed to occur for eight hours. (Section 4.5.1)
3	The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant.	Conforms	The activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant. (Section 4.5.5)
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.	Bounds	STP has taken a more conservative approach and assumes 4.2% elemental iodine, 13.1% organic iodine and 82.7% particulate released from the steam generators to the environment. (Section 4.5.4)
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.	Conforms	The primary-to-secondary leak rate in the steam generators is based on one gallon/minute (gpm)/1440 gallon/day (gpd) total leakage. The leak rate is apportioned between the faulted steam generator as 0.35

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

RG Section	Regulatory Position	Analysis	Comments [a]
	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.		gpm/504 gpd and 0.65 gpm/936 gpd from the intact steam generators. This is the leakage assumption in the current accident analysis. This assumption is conservative when compared with the Technical Specification limit of 0.1 gpm/150 gpd limit. (Sections 4.5.4 and 4.5.5)
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	The density is assumed to be 8.33 lbs/gal. (Section 4.5.5)
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 steaming release and primary-to-secondary coolant leakage is postulated to end at 8 hours, when the primary and secondary loops have reached equilibrium. This consistent with the current licensing basis. (Section 4.5.4)
5.4	All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.	Conforms	Noble gases are released without reduction or mitigation. (Section 4.5.4)
5.5	The transport model described in Appendix E of RG	Conforms	The transport model described is utilized for

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

RG Section	Regulatory Position	Analysis	Comments [a]
	1.183 should be utilized for iodine and particulate releases from the steam generators. This model is shown in Figure E-1 of RG 1.183.		iodine and particulate releases from the steam generators. (Section 4.5.1.1)
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	Primary-to-secondary coolant leakage through the faulted steam generator conservatively goes directly to the environment without mixing with the secondary coolant. Therefore, under the assumed dry-out conditions, no partitioning of any nuclides is postulated to occur in this release pathway. (Section 4.5.4) For all post-accident releases via the intact steam generator loops, the mechanism for release to the environment is steaming of the secondary coolant. Because of this release dynamic, a reduction is taken in the amount of activity released to the environment based on partitioning of nuclides between the liquid and gas states of water. For iodine, the partitioning factor of 100 is used, per RG 1.183. (Sections 4.5.4 and 4.5.5)
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" , during periods of total submergence of the tubes.	Conforms	A partition factor of 100 is used for elemental and particulate iodines from primary-to-secondary leakage in the intact steam generators. Noble gases and organic iodines are released with no partitioning. (Sections 4.5.4 and 4.5.5)

Table D: Conformance with Regulatory Guide 1.183 Appendix E (PWR Main Steam Line Break)			
RG Section	**Regulatory Position**	**Analysis**	**Comments [a]**
5.5.3	The leakage that does not immediately flash is assumed to mix with the bulk water.	Conforms	See comments for Section 5.5.1 above.
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 for iodine is assumed in the analysis. (Section 4.5.5)
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. 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.	Conforms	Tube uncovery does not occur in the intact steam generators. (Section 4.5.5)

Table E: Conformance with Regulatory Guide 1.183 Appendix F (PWR Steam Generator Tube Rupture)			
RG Section	**Regulatory Position**	**Analysis**	**Comments [a]**
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.	Conforms	Assumptions regarding core inventory and the release of radio-nuclides from the fuel are consistent with Regulatory Position 3 of this guide. No fuel damage is assumed due to the SGTR accident. (Section 4.6.3)
2	If no or minimal[2] fuel damage is postulated for the	Conforms	For this analysis, only the iodine and noble

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

RG Section	Regulatory Position	Analysis	Comments [a]
	limiting event, the activity released should be the maximum coolant activity allowed by technical specification. Two cases of iodine spiking should be assumed. Footnote 2. The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum technical specification values, whichever maximizes the radiological consequences. In determining dose equivalent I-131 (DE I-131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.		gas activities, which are conservatively characterized by operation with 1% core fuel defects and the equilibrium and spiked release rates from that fuel, define the source terms. The AST release fractions associated with iodines and noble gases are assumed to be 100%, and are released to the reactor coolant. No additional fuel damage is assumed due to this accident. Two different cases of iodine spiking are analyzed, in accordance with regulatory guidance as previously described. (Section 4.6.3)
2.1	Case 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	This analyzed case involves a 60 μCi/gm pre-accident iodine spike, consistent with the Technical Specification operational Reactor Coolant System activity concentration limit for assumed spike. All of the spike activity is homogeneously mixed in the primary coolant, prior to accident initiation. (Section 4.6.1)
2.2	Case 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)	Conforms	This case involves an accident initiated iodine spike that occurs concurrently with the release of the fluid from the primary and secondary coolant systems. This spike results in a release rate that is 335 times greater than the release rate corresponding to the 1.0 μCi/gm DE I-131 RCS equilibrium iodine concentration, and lasts for 8 hours. (Section 4.6.1)

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

RG Section	Regulatory Position	Analysis	Comments [a]
	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.		
3	The activity released from the fuel, if any, should be assumed to be released instantaneously and homogeneously through the primary coolant.	Conforms	Mixing in the primary coolant is assumed to be released instantaneously and homogeneously. (Section 4.6.5)
4	Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic.	Bounds	STP has taken a more conservative approach and assumes 4.2% elemental iodine, 13.1% organic iodine and 82.7% particulate. (Section 4.6.4)
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 primary-to-secondary leak rate in the steam generators is based on one gallon/minute (gpm)/1440 gallon/day (gpd) total leakage. The leak rate is apportioned between the ruptured steam generator as 0.35 gpm/504 gpd and 0.65 gpm/936 gpd from the intact steam generators. This is the leakage assumption in the current accident analysis. This assumption is conservative when compared with the Technical Specification limit of 0.1 gpm/150 gpd limit. (Sections 4.6.4 and 4.6.5)
5.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be	Conforms	The density is assumed to be 8.33 lbs/gal. (Section 4.6.5)

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

RG Section	Regulatory Position	Analysis	Comments [a]
	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).		
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	Release of activity terminates after 8 hours when shutdown cooling has been established. (Section 4.6.5)
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 coincident loss of offsite power is assumed. (Section 4.6.5)
5.5	All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.	Conforms	Noble gases are released without reduction or mitigation. (Section 4.6.4)
5.6	The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E of RG 1.183 should be utilized for iodine and particulates.	Conforms	The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E of RG 1.183 is utilized for iodine and particulates. (Section 4.6.1.1)

Table F: Conformance with Regulatory Guide 1.183 Appendix H (PWR Control Rod Ejection Accident)

RG Section	Regulatory Position	Analysis	Comments [a]
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	The core inventory in Regulatory Position 3 of RG 1.183 is assumed. The CREA results in damage to 10% of the core. 10% of the core inventory of the noble gases and iodines is in the fuel gap and available for release. One quarter percent of the core experiences fuel melting. 100% of the noble gases, 25% of the iodines, and 50% of the cesium and rubidium contained in the fraction of melted fuel are available for release from containment. For the secondary system release pathway, 100% of the noble gases and 50% of the iodines in the fraction of melted fuel are released to the reactor coolant. (Sections 4.7.2 and 4.7.3)
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	Since fuel damage is postulated, a radiological consequence analysis is 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	For the first case, the activity for leakage from the containment assumes that the activity in the fuel pellet-clad gap and the activity released due to fuel melting is instantaneously mixed in the containment. For the second case, 100% of the noble gases and iodines released by fuel failed during the

Table F: Conformance with Regulatory Guide 1.183 Appendix H (PWR Control Rod Ejection Accident)

RG Section	Regulatory Position	Analysis	Comments [a]
			accident is available for release to the secondary system. (Section 4.7.2)
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	Iodines released to the containment (from the fuel and RCS) are assumed to be 95% particulate, 4.85% elemental, and 0.15% organic. (Section 4.7.5)
5	Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic.	Bounds	STP has taken a more conservative approach and assumes 4.2% elemental iodine, 13.1% organic iodine and 82.7% particulate. (Section 4.7.5)
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.	Conforms	No credit is assumed for removal of iodine in the containment due to containment sprays. (Section 4.7.4)

Table F: Conformance with Regulatory Guide 1.183 Appendix H (PWR Control Rod Ejection Accident)

RG Section	Regulatory Position	Analysis	Comments [a]
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 containment leaks for the first 24 hours at its design leak rate of 0.3 percent per day (i.e., the Technical Specification limit). Thereafter, the containment leak rate is 0.15 percent per day. (Section 4.7.5)
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	For the case of the secondary release pathway, the assumed primary-to-secondary leak rate is 1 gpm (1440 gpd) which is more conservative than the Technical Specification limit of 0.1 gpm per steam generator. Shutdown cooling is assumed to be in operation within 8 hours after accident initiation. Leakage via the MSIV above seat drain orifices is assumed to continue for 36 hours. (Section 4.7.4.2) No releases from the secondary side are postulated for the case of the containment release pathway. (Section 4.7.4.1)
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	Conforms	The density is assumed to be 8.33 lbm/gal. (Section 4.7.5)

Table F: Conformance with Regulatory Guide 1.183 Appendix H (PWR Control Rod Ejection Accident)			
RG Section	**Regulatory Position**	**Analysis**	**Comments [a]**
	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).		
7.3	All noble gas radionuclides released to the secondary system are assumed to be released to the environment without reduction or mitigation.	Conforms	100% of the noble gases released via the secondary system are assumed to be released to the environment without reduction or mitigation.. (Section 4.7.4.2)
7.4	The transport model described in assumptions 5.5 and 5.6 of Appendix E of RG 1.183 should be utilized for iodine and particulates.	Conforms	(Section 4.7.1.1)

Table G: Conformance with Regulatory Guide 1.183 Appendix G (PWR Locked Rotor Accident)			
RG Section	**Regulatory Position**	**Analysis**	**Comments [a]**
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 are those in Regulatory Position 3 of RG 1.183. The release from the breached fuel is based on Regulatory Position 3.2 of RG 1.183. 10% of the fuel rods are assumed to fail. (Section 4.8.3)
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	10% fuel failure is assumed. A radiological calculation analysis was performed.
3	The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant.	Conforms	(Section 4.8.5)

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

RG Section	Regulatory Position	Analysis	Comments [a]
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.	Bounds	The chemical form of the iodine in the RCS is 95% CsI, 4.85% elemental, and 0.15% organic. The chemical form of the iodine released from the secondary side to the environment is 4.2% elemental, 13.1% organic, and 82.7% particulate. (Section 4.8.3.4)
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 assumed primary-to-secondary leak rate is 1 gpm (1440 gpd) which is more conservative than the Technical Specification limit of 0.1 gpm per steam generator. A 0.35 gpm (504 gpd) leak rate is assumed for one steam generator that experiences tube uncovery during the accident due to a postulated single failure in the feedwater system. A 0.65 gpm (936 gpd) leak rate is assumed for the remaining steam generators. (Section 4.8.4 and 4.8.5)
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	The density is assumed to be 8.33 lbm/gal. (Section 4.8.5)
5.3	The primary-to-secondary leakage should be assumed to	Conforms	The primary-to-secondary leakage of 1 gpm is

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

RG Section	Regulatory Position	Analysis	Comments [a]
	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.		assumed to continue for 8 hours following the accident. Eight hours after the accident, the Residual Heat Removal System starts operation to cool down the plant. No further steam or activity is released to the environment. (Section 4.8.5)
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	The release of fission products from the secondary system is evaluated with the assumption of a coincident loss of offsite power. The secondary system condensers are not available for dumping steam. (Section 4.8.5)
5.5	All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.	Conforms	Noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation. (Section 4.8.4)
5.6	The transport model described in assumptions 5.5 and 5.6 of Appendix E of RG 1.183 should be utilized for iodine and particulates.	Conforms	(Section 4.8.1.1)

[a] The Section or Table number indicated in the parentheses, in this column, refers to the Section or Table in Attachment 1, "Licensee's Evaluation" of this licensing amendment request, where the regulatory position is addressed.

Attachment 7

NRC
Regulatory Issue Summary 2006-04
Table

NRC Regulatory Issue Summary 2006-04 Table

The purpose of RIS 2006-04 is to discuss the more frequent and significant issues encountered by the NRC staff during its review of AST submittals and to provide information for licensees to consider when developing submittals for implementation of an AST. This table provides comments describing how STPNOC addressed each RIS issue in the application request for implementing AST.

RIS Issue	Licensee Comments [a]
1. Level of Detail Contained in LARs	
(1) The AST amendment request should provide justification for each individual proposed change to the technical specifications (TS)	Provided in Section 2.0 of the Licensee's Evaluation for each individual proposed change to the TS.
(2) The AST amendment request should identify and justify each change to the licensing basis accident analyses	Section 2.0 of the Licensee's Evaluation provides an overview justification. Section 4.0 and 5.0 provide the detailed justification.
(3) The AST amendment request should 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.	Sufficient detail in tabular format is provided in Section 4.0 of the Licensee's Evaluation to allow the NRC staff to confirm the dose analyses results in independent calculations. In addition, the dose consequent calculations will be submitted under separate cover letter.
Licensees should identify the most current analyses, assumptions, and TS changes in their submittal and supplements to the submittal.	The most current analyses, assumptions, and TS changes are identified throughout Attachment 1 of the Licensing Amendment Request.
2. Main Steam Isolation Valve (MSIV) Leakage and Fission Product Deposition in Piping	Not applicable to PWRs

RIS Issue	Licensee Comments [a]
3. Control Room Habitability	
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.	No credit is taken for use of non-ESF ventilation systems during a DBA. (Section 4.2.2)
Generic Letter (GL) 2003-01, "Control Room Habitability" requested licensees to confirm the ability of their facility's control room to meet applicable habitability regulatory requirements. 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 CR make-up flow is increased from 2000 cfm to 2200 cfm for conservatism. The CR make-up filtration is conservatively ignored. 100 cfm of unfiltered inleakage is assumed in addition to the 2200 cfm of make-up flow that is assumed to experience no filtration. (Section 4.2.2) Unfiltered inleakage testing performed in Unit 1 using the tracer gas method in response to GL 2003-01 measured 9.4 scfm for the limiting case. Unfiltered inleakage testing performed in Unit 2 using the tracer gas method in response to GL 2003-01 measured 62 scfm for the limiting case. (Section 4.2.2)
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.	Control room and other ventilation systems which affect areas adjacent to the CRE and are the same as the operation and performance described in the response to the GL. No credit is taken for the control room ventilation system makeup filters in the AST application. (Section 4.2.2)
4. Atmospheric Dispersion	
Licensees 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".	Updated CR χ/Q values for releases from the containment, from the plant vent, and from the PORV nearest the CR intake were calculated using the computer code ARCON96 using the methods of Regulatory Guide 1.194. The revision to the atmospheric dispersion analyses is provided in Section 4.1 of the Licensee's Evaluation. MET data and inputs to ARCON 96 plus marked-up updates to Chapter 2 of the UFSAR will be provided as part of the submittal.

RIS Issue	Licensee Comments [a]
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".	The χ/Q values for offsite locations were evaluated using the methods of Regulatory Guide 1.145. (Section 4.1) MET data and inputs to PAVAN plus marked-up updates to Chapter 2 of the UFSAR will be provided as part of the submittal.
The submittal should include 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).	Figure 4.1-13 provides a simplified plot plan with release points and receptors.
The submittal should include a justification for using control room intake χ/Q values for modeling the unfiltered inleakage, if applicable.	Section 4.2.2.1 of Attachment 1 provides justification.
The submittal should include 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).	The revised χ/Q values used for the AST application have been developed using more recent meteorological data than that used for the CLB. These more recent data were obtained for the years 2000 to 2004 (five years worth of data) and are documented in ABS Consulting Report R-1459208-01, September 2005. Submittal will include a copy of the meteorological data inputs and program outputs. The MET data was collected per station procedures. (Section 4.1)
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.	No credit is taken for an elevated release. (Section 4.1.3)
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°.	All releases are assumed to be at ground level, and therefore, only 10 m (lower) elevation wind speed data is relevant. (Section 4.1.2) Valid wind direction data provided from 1° to 360°. (Tables 4.1-1 and Figures 4.1-2 through 4.1-7)

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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.	Buoyancy or mechanical jets of high energy releases are not credited. (Section 4.1.3)
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.	No credit is taken for an elevated release. (Sections 4.1.2 and 4.1.3)
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).	No credit is taken for an elevated release. (Section 4.1.2)
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	An adequate number of categories at lower wind speed were used. Seven wind speed groups for STP are used which are as follows: 0.5 mph, 3.5 mph, 7.5 mph, 12.5 mph, 18.5 mph, 24.5 mph, and 36.0 mph (i.e., 0.22 m/s, 1.56 m/s, 3.35 m/s, 5.59 m/s, 8.27 m/s, 10.95 m/s, and 16.09 m/s). These are judged to be adequate for determining the offsite X/Q values using PAVAN. (Section 4.1.2)
5. Modeling of ESF Leakage	
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 postulated [ESF] leakage is 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. (Section 4.3.3.2)
Licensees should account for ESF leakage at accident conditions in their dose analyses so as not to underestimate the release rate.	ESF leakage was accounted for at accident conditions. (Section 4.3.3.2)

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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 RG 1.183 recommended value of 10% is used only for the first 24 hours. Beyond that time, the release fraction is increased to 16% (24-480 hours) and then to 25% (480 hours to 720 hours) based on the calculated volatility of the iodine for the pH values over those intervals relative to the volatility for a pH of 7.0. (Section 4.3.3.2)
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.	The calculation methodology for containment sump pH control was based on the approach outlined in NUREG-1465 and NUREG/CR-5950. The assessment included I-127 and I-129. (Section 4.3.3.1.2.1)
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. Releases from personnel air locks and equipment hatches exposed to the environment and containment purge releases prior to containment isolation need to be addressed.	The AST application re-analyzes the design basis dose calculations for an open containment during refueling. (Section 4.4.2) The current dose analysis considered the open personnel air lock and equipment hatch as release paths. (Section 4.4.2) For the LOCA analysis, the duration of flow through open purge valves is assumed to be 23 seconds – same as the current licensing basis. (Section 4.3.5)

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Licensees are responsible for identifying all release pathways and for considering these pathways in their AST analyses, consistent with any proposed modification.	Revised control room, exclusion area boundary, and low population zone atmospheric dispersion factors (χ/Q) for the containment leakage, plant vent, and steam generator (SG) secondary side power-operated relief valve (PORV) release paths were calculated. (Section 4.1) A new release path consisting of the containment electrical penetration area volume, the leak rate per penetration, the number of penetrations, and the ventilation exhaust rate for the electrical penetration area are analyzed to determine the post-accident transient activity airborne in the electrical penetration area to support the calculation of the gamma shine contribution to the CR. (Section 4.2.2.1 and Table 4.3-11)
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.	All analyses assume a water density of 1 gram/cubic centimeter (i.e. 8.33 lbm/gallon). (Sections 4.5.5, 4.6.5, 4.7.5 and 4.8.5)
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, fespectively, 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.	The depth of water over the damaged fuel is not less than 23 feet. Due to the submergence of the damaged fuel, the iodine release is assumed to experience a DF of 200 per RG 1.183. (Section 4.4.4)

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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.	Noble gas and cesium isotopes were included in the dose assessment.
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 ^{131}I. 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.	The definition of DE I-131 is modified to reflect that the dose conversion factors are those listed in Federal Guidance Report 11. (Section 2.0 and 4.2.4.1)
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.	Accident not included with this submittal.
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.	The volumetric flow rate between sprayed and unsprayed regions of containment is found in Table 4.3-11 in the Licensee's Evaluation. Less than 10% of this total will re-circulate in the unsprayed region. The dose is insensitive to mixing flow bypass of this magnitude.

[a] The Section or Table number indicated in the parentheses, in this column, refers to the Section or Table in Attachment 1, "Licensee's Evaluation" of this licensing amendment request.