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10 CFR 50.90

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

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. 50-456 and 50-457

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. 50-454 and 50-455

Subject: Request for License Amendment Related to Application of Alternative 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

(3) Technical Specification Task Force (TSTF) Traveler, TSTF-51, "Revise[d] Containment Requirements During Handling [of] Irradiated Fuel and Core Alterations," Revision 2

(4) Letter from Michael P. Gallagher (Exelon/AmerGen), "Exelon/AmerGen 180-Day Response to NRC Generic Letter 2003-01, 'Control Room Habitability'," dated December 9, 2003

In accordance with 10 CFR 50.67, "Accident Source Term," and 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) is requesting an amendment to Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF-72, NPF-77, NPF-37, and NPF-66 for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, respectively. 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, except where alternate

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methods for complying with the specified portions of the NRC's regulations have been used 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 7 of this submittal. Implementation of an AST methodology changes the regulatory assumptions regarding the analytical treatment of the design basis accidents (DBAs). Application of the AST methodology will allow specific changes as noted below.

On December 23, 1999, the NRC issued the Final Rule on "Use of Alternate Source Terms at Operating Reactors." The Final Rule, issued as 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.

In support of a full-scope implementation of the AST methodology, EGC has performed radiological consequence analyses for the following six DBAs that result in control room (CR) and offsite exposure as specified in Reference 1.

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

Proposed changes to the current licensing basis for Byron Station and Braidwood Station justified by the AST analyses include the following items.

- A change in the definition for Dose Equivalent Iodine to reflect the use of exposure-to-dose conversion factors for inhalation from Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1988.
- Addition of a new definition for RECENTLY IRRADIATED FUEL; i.e., fuel that has occupied part of a critical reactor core within the previous 48 hours.
- The Containment Leakage Rate Testing Program maximum allowable containment leakage rate, (i.e., L_a) at design accident pressure (i.e., P_a) is increased from 0.1% to 0.2% of containment air weight per day.
- Emergency Core Cooling System (ECCS) allowable leakage is assumed to be 276,000 cc/hr for the Byron and Braidwood units, rather than 15,249 cc/hr (approximately 4.0 gal/hr) for Braidwood Station and 13,294 cc/hr (approximately 3.5 gal/hr) for Byron Station.
- The Control Room Ventilation (VC) Filtration System makeup filter charcoal adsorber bypass acceptance criterion is increased from 0.05% to 1% and the associated charcoal adsorber penetration acceptance criterion is increased from 0.5% to 2.0%. These changes reflect a

reduction in the assumed filtration efficiency from 99% to 95%.

- TS and associated Bases revisions to change the applicability requirements for the below systems to be applicable only during movement of RECENTLY IRRADIATED FUEL assemblies and to relax the operability requirements of these systems during core alterations, unless required for other reasons.
  - Containment
  - Fuel Handling Building (FHB) Ventilation System

In addition, the change management plan associated with application of the AST methodology will assess potential impact on other issues/programs; e.g., probabilistic risk assessment assumptions, and the severe accident management plan.

The proposed changes related to the applicability requirements during movement of irradiated fuel assemblies are consistent with Technical Specification Task Force Traveler (TSTF)-51, Revision 2 (i.e., Reference 3). TSTF-51, Revision 2, was approved by the NRC on November 1, 1999. TSTF-51 changes 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. Since a portion of this license amendment request is based on TSTF-51, EGC is committing (see Attachment 5, "List of Commitments") 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, as described in TSTF-51. Please note that changes to TS 3.8, "Electrical Power Systems," addressed in TSTF-51, are not included in this submittal.

EGC has been an active participant on the Nuclear Energy Institute (NEI) Control Room Habitability (CRH) Task Force and is aware of the NRC concerns regarding CRH as described in Generic Letter (GL) 2003-01, "Control Room Habitability." EGC has provided a formal response to the Generic Letter in Reference 4. This submittal does not directly address the CRH issue other than to provide an increase in the assumed CR unfiltered inleakage value. The American Society for Testing and Materials (ASTM) E741 tracer gas test of both the Byron Station and Braidwood Station CRs has been performed with satisfactory results. The assumed unfiltered inleakage value in the AST dose analyses is 1000 cfm. The ASTM E741 tracer gas test "as found" unfiltered inleakage was a small percentage of this value for both stations. Note that the pre-AST analysis design leakage limit is 100 cfm.

Approval of this proposed change will provide a source term for Byron Station and Braidwood Station 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, noted above.

This request is subdivided as follows:

- Attachment 1 provides a Description of Proposed Changes, Technical Analysis, and Regulatory Analysis.
- Attachment 2 provides the Markup of Technical Specification pages.

- Attachment 3 provides the Typed Technical Specification pages.
- Attachment 4 provides the Typed Technical Specification Bases pages (for information only).
- Attachment 5 provides the List of Commitments resulting from the proposed changes.
- Attachment 6 contains "Input Parameter Tables," providing the input data for the DBA radiological consequences analyses.
- Attachment 7 contains, "Regulatory Guide Conformance Tables," providing detailed verification that the AST methodology conforms to the guidance in Regulatory Guide (RG) 1.183 (i.e., Reference 1) and RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants."
- Attachment 8 provides a compact disk (CD) containing Byron Station and Braidwood Station meteorological data for the calculation of the atmospheric dispersion factors (χ/Qs). The CD also provides the PAVAN and ARCON96 input parameters.

The proposed changes have been reviewed by the Byron Station and Braidwood Station Plant Operations Review Committees and approved by their respective Nuclear Safety Review Boards in accordance with the requirements of the EGC Quality Assurance Program.

In accordance with 10 CFR 50.91(b), "State consultation," EGC is notifying the State of Illinois of this application for changes to the TS by transmitting a copy of this letter and its attachments to the designated State Official.

Exelon requests approval of the proposed amendment by February 15, 2006. 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 has previously approved implementation of the AST methodology at a number of other nuclear power stations including Surry Power Station in Amendment No. 230, dated March 8, 2002; Kewaunee Nuclear Power Plant in Amendment No. 166, dated March 17, 2003; and H. B. Robinson Steam Electric Plant in Amendment No. 201, dated September 24, 2004.

Should you have any questions or require additional information, please contact Joe Bauer at (630) 657-2801.

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

Executed on February 15, 2005

Keith R. Jury

Keith R. Jury
Director – Licensing and Regulatory Affairs

Attachments:
1. Description of Proposed Changes, Technical Analysis, and Regulatory Analysis
2. Markup of Technical Specification pages
3. Typed Technical Specification pages
4. Typed Technical Specification Bases pages (information only)
5. List of Commitments
6. Input Parameter Tables
7. Regulatory Guide Conformance Tables
8. Byron and Braidwood Meteorological data (information only)

ATTACHMENT 1
Evaluation of Proposed Changes

Braidwood Station
Units 1 and 2

Byron Station
Units 1 and 2

License Amendment Request
"Alternative Source Term Application"

Evaluation of Proposed Changes

ATTACHMENT 1
Evaluation of Proposed Changes

1.0 DESCRIPTION

In accordance with 10 CFR 50.67, "Accident Source Term," and 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) is requesting an amendment to Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF-72, NPF-77, NPF-37, and NPF-66 for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, respectively. 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 Standard Review Plan 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," and Regulatory Guide (RG) 1.183 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 7 of this submittal.

To support a full-scope implementation of the AST methodology, radiological consequence analyses have been performed for the following six bounding Design Basis Accidents (DBAs) that result in control room (CR) and offsite exposure.

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

Proposed changes to the current licensing basis for Byron Station and Braidwood Station justified by the AST analyses include the following:

- A change in the definition for Dose Equivalent Iodine to reflect the use of exposure-to-dose conversion factors for inhalation from Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1988.
- Addition of a new definition for RECENTLY IRRADIATED FUEL; i.e., fuel that has occupied part of a critical reactor core within the previous 48 hours.
- The Containment Leakage Rate Testing Program maximum allowable containment leakage rate, (i.e., L_a) at design accident pressure (i.e., P_a) is increased from 0.1% to 0.2% of containment air weight per day.
- Emergency Core Cooling System (ECCS) allowable leakage is assumed to be 276,000 cc/hr for the Byron and Braidwood units, rather than 15,249 cc/hr (approximately 4.0 gal/hr) for Braidwood Station and 13,294 cc/hr (approximately 3.5 gal/hr) for Byron Station.
- The Control Room Ventilation (VC) Filtration System makeup filter charcoal adsorber bypass acceptance criterion is increased from 0.05% to 1% and the associated charcoal adsorber penetration acceptance criterion is increased from 0.5% to 2.0%. These changes reflect a reduction in the assumed filtration efficiency from 99% to 95%. This penetration acceptance

criteria maintains a safety factor of two consistent with the guidance in NRC GL 99-02, "Laboratory Testing of Nuclear-Grade Activated Charcoal," June 3, 1999.

- TS and associated Bases revisions to change the applicability requirements for the below systems to be applicable only during movement of RECENTLY IRRADIATED FUEL assemblies and to relax the operability requirements of these systems during core alterations, unless required for other reasons.
  - Containment
  - FHB Ventilation System

In addition, the change management plan associated with application of the AST methodology will assess potential impact on other issues/programs; e.g., probabilistic risk assessment assumptions, and the severe accident management plan.

The proposed changes related to the applicability requirements during movement of irradiated fuel assemblies are consistent with Technical Specification Task Force Traveler (TSTF)-51, Revision 2 (i.e., Reference 3). TSTF-51, Revision 2, was approved by the NRC on November 1, 1999. TSTF-51 changes the TS operability requirements for certain engineered safety features such that they are not required to operate after sufficient radioactive decay has occurred to ensure that offsite doses remain within limits. Since a portion of this license amendment request is based on TSTF-51, EGC is committing (see Attachment 5, "List of Commitments") 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, as described in TSTF-51. Please note that changes to TS 3.8, "Electrical Power Systems," addressed in TSTF-51, are not included in this submittal.

EGC has been an active participant on the Nuclear Energy Institute (NEI) Control Room Habitability (CRH) Task Force and is aware of the NRC concerns regarding CRH as described in Generic Letter (GL) 2003-01, "Control Room Habitability." EGC has provided a formal response to the Generic Letter in Reference 4. This submittal does not directly address the CRH issue other than to provide an increase in the assumed CR unfiltered inleakage value. The ASTM E741 tracer gas test of both the Byron Station and Braidwood Station CRs has been performed with satisfactory results. The assumed unfiltered inleakage value in the AST dose analyses is 1000 cfm. The ASTM E741 tracer gas test "as found" unfiltered inleakage was a small percentage of this value for both stations. Note that the pre-AST analysis design leakage limit is 100 cfm.

Approval of this change will provide a source term for Byron and Braidwood 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 below. Upon implementation of AST, containment closure capability will no longer be required for the purpose of moving fuel except during movement of recently irradiated fuel. 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. Other benefits of AST are the cost savings that will be achieved by reducing credited charcoal filter efficiencies in the accident analyses. This additional margin will extend available charcoal filter life by revising the methyl iodine penetration acceptance criteria and will result in less frequent charcoal filter regeneration. HEPA filter efficiency credit reductions provide additional operating margin, but no reduction in test acceptance criteria are proposed. These benefits apply to the CR and FHB Ventilation System.

2.0 PROPOSED CHANGES

The proposed changes related to the applicability requirements during movement of irradiated fuel assemblies are consistent with 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 October 15, 1999. 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.

Since a portion of this license amendment request is based on TSTF-51, EGC 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, as described in TSTF-51 and documented in Attachment 5 to this submittal.

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

TS Section 1.1, "Definitions" (page 1.1-3)

The proposed change revises the definition of DOSE EQUIVALENT I-131 in TS Section 1.1 to remove the word "thyroid" and to add a reference to Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989. [Table 2.1, Exposure-to-Dose Conversion Factors for Inhalation.] This change reflects the application of AST methodology assumptions.

TS Section 1.1, "Definitions" (page 1.1-6)

The proposed change adds the definition of RECENTLY IRRADIATED FUEL. RECENTLY IRRADIATED FUEL is defined as: "Fuel that has occupied part of a critical reactor core within the previous 48 hours. Note that all fuel that has been in a critical reactor core is referred to as irradiated fuel."

TS Section 3.3.7, Required Action E.1 (page 3.3.7-2)

REQUIRED ACTION E.1, "Suspend CORE ALTERATIONS" and its associated completion time are deleted consistent with the guidance in TSTF-51.

TS Section 3.3.8, "FHB Ventilation System Actuation Instrumentation" (page 3.3.8-2)

The words "irradiated fuel" in Required Action B.2.1 and B.2.2 are replaced with the defined term "RECENTLY IRRADIATED FUEL". Required Action B.2.3 is deleted consistent with the guidance in TSTF-51. This revision is supported by the AST analysis.

TS Table 3.3.8-1, "FHB Ventilation System Actuation Instrumentation" (page 3.3.8-4)

In Notes (a) and (b), the term "irradiated fuel" is replaced with the defined term "RECENTLY IRRADIATED FUEL". Note (c) applicable to Function 1, "Fuel Handling Building Radiation" is deleted, as well as the note itself, consistent with the guidance in TSTF-51 and supported by the AST analysis. Deletion of the note eliminates the requirement to have the FHB radiation monitors operable during CORE ALTERATIONS with the equipment hatch not intact.

TS 3.7.10, "VC Filtration System" Condition C (page 3.7.10-2)

REQUIRED ACTION C.2.1 and its completion time regarding suspension of CORE ALTERATIONS are deleted. REQUIRED ACTION C.2.2 and C.2.3 are renumbered to support deletion of C.2.1. This is done in accordance with the guidance in TSTF-51 and supported by the AST analysis.

TS 3.7.10, "VC Filtration System" Condition D (page 3.7.10-3)

REQUIRED ACTION D.1 and its completion time regarding suspension of CORE ALTERATIONS are removed. REQUIRED ACTION D.2 and D.3 are renumbered to support deletion of D.1. This is done in accordance with the guidance in TSTF-51 and supported by the AST analysis.

TS SR 3.7.10.4, "VC Filtration System" (page 3.7.10-4)

The words "the upper cable spreading room at positive pressure of ≥0.02 inches water gauge and" are removed. The word "area" (i.e., in the statement: "...adjacent to the control room *area* during ...") is replaced with the word "envelope". This change is consistent with NUREG 1431, "Standard Technical Specifications, Westinghouse Plants," and is further justified by the requirements of the proposed Byron Station and Braidwood Station Control Room Envelope Integrity Program which verifies that the control room envelope is maintained at a positive pressure.

TS LCO 3.7.11, "VC Temperature Control System" Condition C (page 3.7.11-2)

REQUIRED ACTION C.2.1 and its completion time regarding suspension of CORE ALTERATIONS are removed. Required Action C.2.2 and C.2.3 are renumbered to support deletion of C.2.1. This is done in accordance with the guidance in TSTF-51 and supported by the AST analysis.

TS LCO 3.7.11, "VC Temperature Control System" Condition D (page 3.7.11-3)

REQUIRED ACTION D.1 and its completion time regarding suspension of CORE ALTERATIONS are removed. Required Action D.2 and D.3 are renumbered to support deletion of D.1. This is done in accordance with the guidance in TSTF-51 and supported by the AST analysis.

TS LCO 3.7.13 Applicability Statement (page 3.7.13-1)

The term "irradiated fuel" is replaced with the words "RECENTLY IRRADIATED FUEL" in accordance with the guidance in TSTF-51. The reference to suspension of CORE ALTERATIONS is removed. This change is supported by the AST analysis.

TS LCO 3.7.13 Required Actions for Condition B (continued) (page 3.7.13-2)

The term "irradiated fuel" in REQUIRED ACTION B.2.1 and B.2.2 is replaced with the term "RECENTLY IRRADIATED FUEL". REQUIRED ACTION B.2.3 regarding the suspension of CORE ALTERATIONS is deleted. These changes are in accordance with the guidance in TSTF-51 and supported by the AST analysis.

TS LCO 3.7.13 Required Actions for Condition C (page 3.7.13-3)

The term "irradiated fuel" is replaced with the words "RECENTLY IRRADIATED FUEL" in REQUIRED ACTION C.1 and C.2. REQUIRED ACTION C.3 regarding the suspension of CORE ALTERATIONS is deleted. These changes are in accordance with the guidance in TSTF-51 and supported by the AST analysis.

TS SR 3.7.13.3 (page 3.7.13-4)

In the NOTE, the term "irradiated fuel" is replaced with the term "RECENTLY IRRADIATED FUEL," and the words "or CORE ALTERATIONS," are deleted in accordance with the guidance in TSTF-51. This change is supported by the AST analysis.

TS SR 3.7.13.5 (page 3.7.13-4)

In the NOTE, the term "irradiated fuel" is replaced with the term "RECENTLY IRRADIATED FUEL" in accordance with the guidance in TSTF-51. This change is supported by the AST analysis.

TS LCO 3.9.4 NOTE (page 3.9.4-1)

The reference to "LCO" is changed to "TS". This is an editorial change for clarity.

TS LCO 3.9.4 Applicability Statement (page 3.9.4-1)

The term "irradiated fuel" is replaced with the words "RECENTLY IRRADIATED FUEL." The reference to the suspension of CORE ALTERATIONS is deleted. These changes are in accordance with the guidance in TSTF-51 and supported by the AST analysis.

TS LCO 3.9.4 REQUIRED ACTION A.1 and A.2 (page 3.9.4-2)

In REQUIRED ACTION A.1, the reference to the suspension of CORE ALTERATIONS and its associated COMPLETION TIME are deleted. REQUIRED ACTION A.2 is renumbered and the term "irradiated fuel" is replaced with the term "RECENTLY IRRADIATED FUEL" in accordance with the guidance in TSTF-51 supported by the AST analysis.

TS LCO 3.9.7 Applicability Statement (page 3.9.7-1)

The reference to the suspension of CORE ALTERATIONS is removed; however, the reference to "irradiated fuel assemblies" is retained in this location in accordance with TSTF-51.

TS LCO 3.9.7 REQUIRED ACTION A.2 (page 3.9.7-1)

In REQUIRED ACTION A.1, the reference to the suspension of CORE ALTERATIONS and its associated COMPLETION TIME are deleted in accordance with TSTF-51 supported by the AST analysis. REQUIRED ACTION A.2 is renumbered to A.1.

TS 5.5.11 "Ventilation Filter Testing Program (VFTP)" Items b and c (pages 5.5-17 and 18)

In Item b, the charcoal adsorber bypass acceptance criterion is changed from 0.05% to 1%. In Item c, the charcoal adsorber penetration acceptance criterion is changed from 0.5% to 2.0%. These changes reflect a reduction in assumed filtration efficiency from 99% to 95%, used in the AST analysis, and maintains a safety factor of two for the penetration acceptance criterion.

TS 5.5.16, "Containment Leakage Rate Testing Program" (page 5.5-24)

The maximum allowable containment leakage rate L_a is changed from 0.10% to 0.20% of containment air weight per day. This change is supported by the AST dose consequence analyses.

3.0 BACKGROUND

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.

The fission product release from the reactor core into containment is referred to as the "source term," and it 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" (i.e., Reference 7.1, 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 7.4) 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 LWRs 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.

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. EGC 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, "Radiological Consequence Analyses Using Alternative Source Terms" (i.e., Reference 7.3). Implementation of an AST methodology changes only the regulatory assumptions regarding the analytical treatment of the following DBAs.

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

Implementation of the AST methodology will provide a source term for Byron Station and Braidwood Station that will result in a more accurate assessment of the DBA radiological doses. The improved dose assessment allows cost beneficial relaxation of some current licensing basis requirements as described below.

The NRC has previously approved implementation of the AST methodology at a number of other nuclear power stations including Surry Power Station in Amendment No. 230, dated March 8, 2002; Kewaunee Nuclear Power Plant in Amendment No. 166, dated March 17, 2003; and H. B. Robinson Steam Electric Plant in Amendment No. 201, dated September 24, 2004.

4.0 TECHNICAL ANALYSIS

The AST analyses for Byron Station and Braidwood Station were performed following the guidance in Standard Review Plan 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," and RG 1.183 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 7 of this submittal. In addition, Attachment 7, Table H provides confirmation of conformance to diffuse area source guidance provided in RG 1.194 (i.e., Reference 7.19). The inputs parameters to the radiological analyses are provided in Attachment 6, Tables 1.0 through 8.0.

These analyses have been performed using NRC approved computer codes and have had the appropriate cross-functional reviews.

Implementation of the AST methodology consists of the following major steps.

- Analysis of the atmospheric dispersion for the radiological propagation pathways
- Calculation of offsite Exclusion Area Boundary (EAB) and Low Population Zone (LPZ), CR, and, for LOCA only - Technical Support Center (TSC) personnel Total Effective Dose Equivalent (TEDE) doses
- Identification of the AST based on plant-specific analysis of a bounding core fission product inventory
- Calculation of fission product deposition rates and transport and removal mechanisms
- Calculation of the release fractions for the DBAs that result in the most significant CR and offsite doses (i.e., LOCA, FHA, CREA, LRA, MSLB and SGTR)

This technical analysis section will address the following specific areas/topics.

4.1 Meteorology and Atmospheric Dispersion

4.2 Dose Conversion Factors

4.3 EAB and LPZ Dose Model

4.4 Control Room Dose Model

4.5 Loss of Coolant Accident Radiological Assessment

4.6 Fuel Handling Accident Radiological Assessment

4.7 Control Rod Ejection Radiological Assessment

4.8 Locked Rotor Accident Radiological Assessment

4.9 Main Steam Line Break Radiological Assessment

4.10 Steam Generator Tube Rupture Radiological Assessment

In addition, Attachment 6, Tables 1.0 – 8.0 provide a list of the input parameters used in the radiological consequence analyses; Attachment 7, Tables A – G provide a detailed verification that the AST methodology conforms to the guidance in RG 1.183 and Attachment 7, Table H provides verification that AST methodology conforms to the guidance in RG 1.194 (i.e., Diffuse Area Source Guidance).

4.1 Meteorology and Atmospheric Dispersion

The χ/Q values resulting at the CR intake are calculated using the NRC-sponsored computer codes ARCON96 consistent with the procedures in RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," (i.e., Reference 7.19).

The χ/Q values resulting at the EAB and LPZ are calculated using the NRC-sponsored computer code PAVAN, consistent with the procedures in RG 1.145 (i.e., Reference 7.7).

A five-year (i.e., 1994 -1998) hourly meteorological database was utilized for each station in the ARCON96 and PAVAN modeling of χ/Q.

4.1.1 ARCON96 Modeling Analysis of Control Room χ/Q

Design Input (ARCON96)

NOTE: The ARCON96 computer code utilizes input variables expressed in meters; therefore, a number of the parameters given below will be given both in feet and in meters.

Source/Receptor Scenarios and Configurations

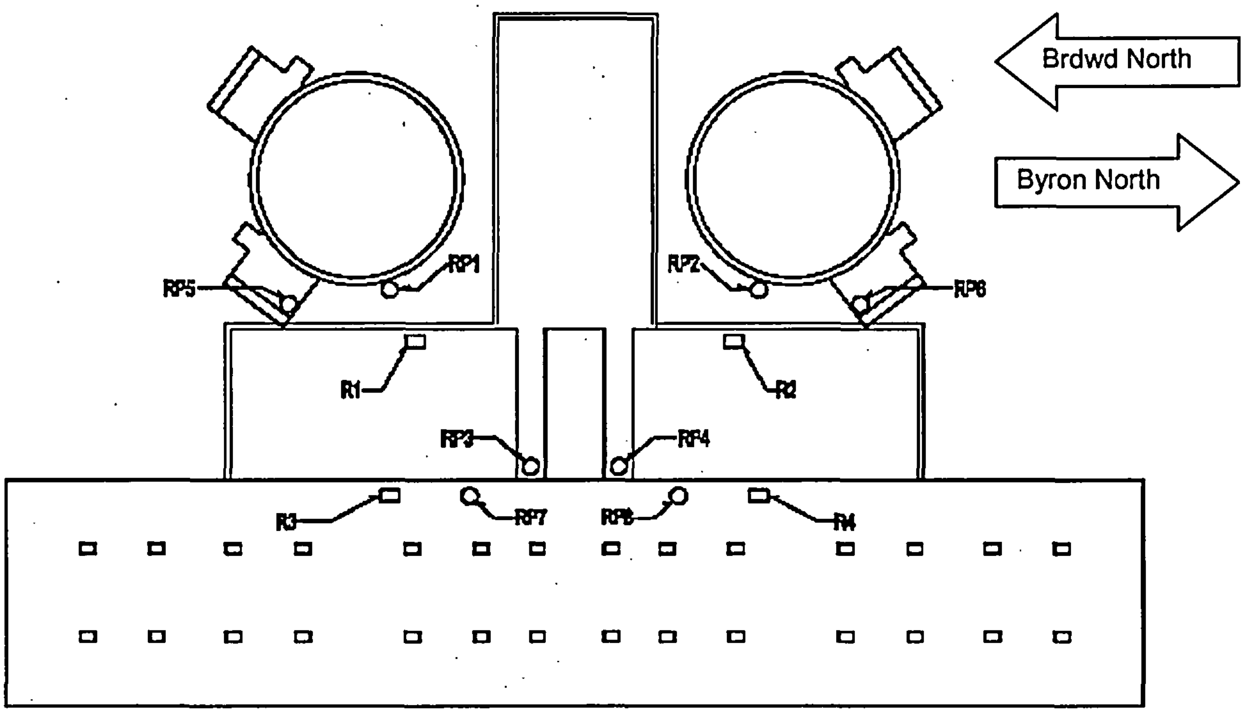
For each unit at Byron Station and Braidwood Station, there are four release points:

- the containment wall;
- the plant vent;
- the SG power operated relief valves (PORVs)/safety valves; and
- through a main steam line break.

The height of these release points are all less than 2.5 times the height of their adjacent buildings and therefore, in accordance with RG 1.194, are modeled as "ground level" releases. The locations of these releases are shown in Figure 4.1.1-1 below for Braidwood and Byron Stations.

Each unit at Byron Station and Braidwood Station has two associated receptors, the CR fresh air intake, with a height of 21.2 m, and the CR Turbine Building Emergency Air Intake, with a height of 20.4 m. The location of these intakes at Byron Station and Braidwood Station are also shown in Figure 4.1.1-1.

Figure 4.1.1-1
Byron/Braidwood Control Room Receptors and Release Points



No.	Relative Elev. (ft)	Receptor
R1	469.4	Unit 1 CR Fresh Air Intake
R2	469.4	Unit 2 CR Fresh Air Intake
R3	466.8	Unit 1 CR Emergency Air Intake
R4	466.8	Unit 2 CR Emergency Air Intake

No.	Relative Elev. (ft)	Release Point
RP1	N/A	Unit 1 Containment Wall
RP2	N/A	Unit 2 Containment Wall
RP3	600	Unit 1 Plant Vent
RP4	600	Unit 2 Plant Vent
RP5	432	Unit 1 SG PORVs/Safety Valves (Worst case of two locations depicted for CR Intake)
RP6	432	Unit 2 SG PORVs/Safety Valves (Worst case of two locations depicted for CR Intake)
RP7	426	Unit 1 Main Steam Line Break
RP8	426	Unit 2 Main Steam Line Break

Notes:
1. MSLB release points RP7 and RP8 in the Turbine Bldg. are closest to the CR air intakes R3 and R4.
2. The MSLB release points closest to the CR fresh air intakes R1 & R2 are from the safety valve rooms (RP5 and RP6)

The ARCON96 computer model is executed for each release point (i.e., source)/receptor combination at Byron Station and Braidwood Station shown below.

1) Unit 1 Containment Wall to Control Room Fresh Air Intake
2) Unit 1 Containment Wall to Control Room Turbine Building (TB) Emergency Air Intake
3) Unit 1 Plant Vent to Control Room Fresh Air Intake
4) Unit 1 Plant Vent to Control Room Turbine Building Emergency Air Intake
5) Unit 1 SG PORVs/Safety Valves to Control Room Fresh Air Intake
6) Unit 1 SG PORVs/Safety Valves to Control Room Turbine Building Emergency Air Intake
7) Unit 1 MSLB to Control Room Fresh Air Intake
8) Unit 1 MSLB to Control Room Turbine Building Emergency Air Intake
9) Unit 2 Containment Wall to Control Room Fresh Air Intake
10) Unit 2 Containment Wall to Control Room Turbine Building Emergency Air Intake
11) Unit 2 Plant Vent to Control Room Fresh Air Intake
12) Unit 2 Plant Vent to Control Room Turbine Building Emergency Air Intake
13) Unit 2 SG PORVs/Safety Valves to Control Room Fresh Air Intake
14) Unit 2 SG PORVs/Safety Valves to Control Room Turbine Building Emergency Air Intake
15) Unit 2 MSLB to Control Room Fresh Air Intake
16) Unit 2 MSLB to Control Room Turbine Building Emergency Air Intake

All scenarios are conservatively assumed to have a "zero" vertical velocity, "zero" exhaust flow and "zero" stack radius.

The containment wall scenarios are modeled as a "diffuse area" source in ARCON96 in accordance with RG 1.194, Section 3.2.4. Attachment 7, Table H contains additional information regarding this issue. All other scenarios are modeled as ground level point sources. The area source representation in ARCON96 requires the building cross-sectional area to be calculated from the maximum building dimensions projected onto a vertical plane perpendicular to the line of sight from the building to the intake. The containment building, with a height of 195 ft (i.e., not including the dome portion above the collar) and width of 161 ft, was utilized for both stations resulting in an area of 31,395 ft^2 (2916.7 m^2). The diffuse area source also requires the release height to be set at the vertical center of the projected area. As specified in RG 1.194, Section 3.2.4.4, the initial diffusion coefficients for both stations are calculated as follows.

$$\sigma_{Y_0} = \frac{Width_{area\ source}}{6} \qquad \sigma_{Z_0} = \frac{Height_{area\ source}}{6}$$

$$\sigma_{Y_0} = \frac{161 ft}{6} = 26.83\ ft = 8.18\ m \qquad \sigma_{Z_0} = \frac{195 ft}{6} = 32.5\ ft = 9.9\ m$$

The remaining three release points at each station (i.e., plant vent, SG PORVs/safety valves, and MSLB) are modeled as a point source. In accordance with RG 1.194 Table A-2, the building area perpendicular to the wind direction should be utilized. For releases from the SG PORVs/safety valves, the containment building area of 2916.7 m^2 was utilized for both stations. There is no change in this building area with a change in wind direction due to its cylindrical shape. The auxiliary building area was utilized for the plant vent and MSLB scenarios. The auxiliary building areas for Byron Station and Braidwood Station were calculated by the following equations.

building area = height x projected building width perpendicular to the wind direction

where:

$y = r\sin(90° - \alpha + FV - \Theta'_N)$; assuming $0° \le (FV - \Theta'_N) \le 90°$

y = projected building width perpendicular to the wind direction (wind direction is conservatively assumed to be directly from source to intake)

r = hypotenuse of the structure, based on length and width

α = angle between the hypotenuse and width of structure

Θ'_N = acute angular deviation in building orientation from north-south (=0° for auxiliary building)

FV = flow vector

ARCON96 requires a horizontal source-receptor distance to be input, which is defined in RG 1.194 Section 3.4 as "the shortest horizontal distance between the release point and the intake." However, for releases in building complexes, a "taut string length" can be utilized if justified. For the "MSLB to Control Room Fresh Air Intake" scenarios at each station, this "taut string length" was utilized to account for the intervening auxiliary building structure. The intake and release height were set equal to each other so as not to also take advantage of the slant distance that ARCON96 calculates. The intake height was set equal to the release height of 7.9 m for both Byron Station and Braidwood Station.

Onsite Meteorological Monitoring Program

The meteorological measurement program at the Byron and Braidwood Stations consists of monitoring wind direction, wind speed, temperature and precipitation. Two methods of determining the appropriate atmospheric stability are:

a. Delta T (i.e., vertical temperature difference), which is the principal method, and

b. Sigma theta (i.e., standard deviation of the horizontal wind direction), for use when delta T is not available.

These data, referenced in ANSI/ANS 2.5 (1984), (i.e., Reference 7.10), are used to determine the meteorological conditions prevailing at the plant site. The meteorological program includes site-specific information on instrumentation and calibration procedures. The meteorological program meets the requirements specified in the Offsite Dose Calculation Manual.

The meteorological tower is equipped with instrumentation that conforms with the system accuracy recommendations in RG 1.23 (Reference 7.11) and ANSI/ANS 2.5 - 1984. The equipment is placed on booms oriented into the generally prevailing wind direction at the site. Equipment signals are transmitted to an instrument building with controlled environmental conditions. The building, at the base of the tower, houses the recording equipment, signal conditioners, etc. This instrumentation is used to process and retransmit the data to the end-point users.

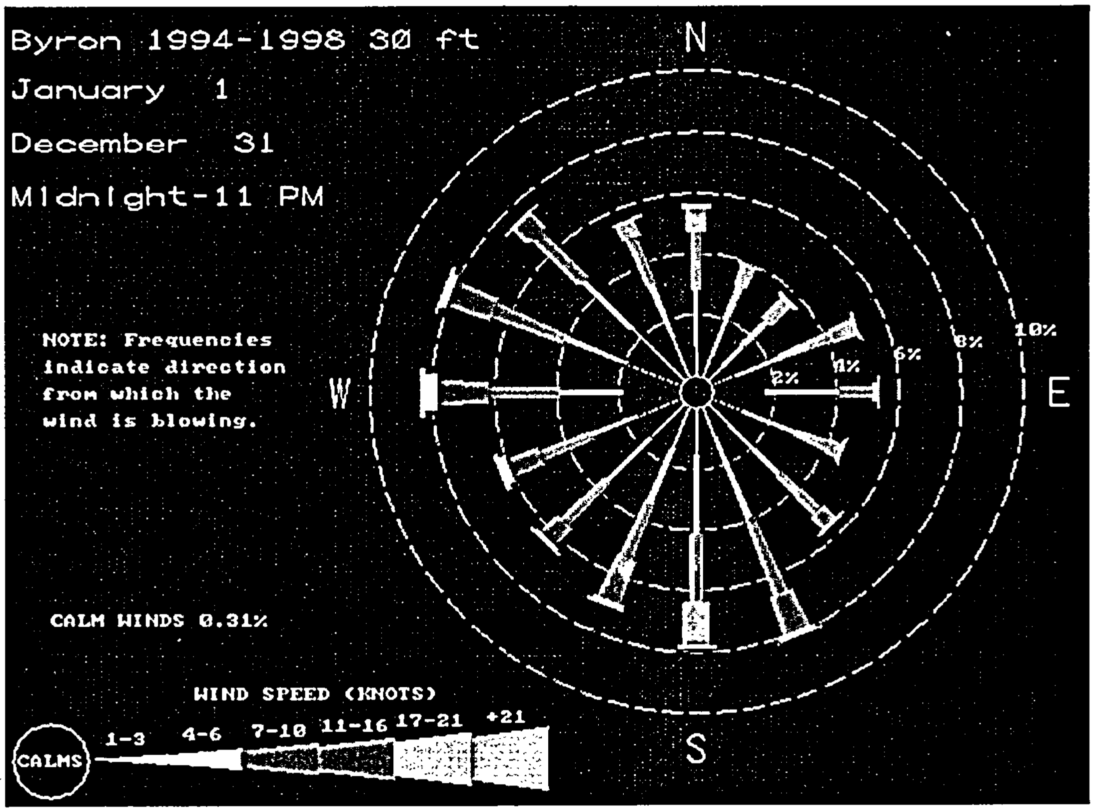
Recorded meteorological data are used to generate wind roses and to provide estimates of airborne concentrations of gaseous effluents and projected offsite radiation dose. Instrument calibrations and data consistency evaluations are performed routinely to ensure maximum data integrity. The data recovery objective is to attain better than 90% from each measuring and recording system. Data storage and records retention are consistent with the requirements of ANSI/ANS 2.5 - 1984).

Meteorological Data

The Byron Station and Braidwood Station meteorological databases for the five-year period, (i.e., 1994 –1998), was applied in the ARCON96 modeling analysis. Wind measurements at Byron Station were taken at 30 ft and 250 ft; and the vertical temperature difference was measured between 250 ft and 30 ft. Braidwood Station wind measurements were taken at 34 ft and 203 ft; and the vertical temperature difference was measured between 199 ft and 30 ft. "Calm" wind speeds at both stations were assigned a value of 0.4 mph (i.e., one-half the threshold value) per UFSAR Section 2.3.4. The minimum wind speed (i.e. wind threshold) was set to the ARCON96 default value of 0.5 m/sec in accordance with RG 1.194, Table A-2.

Figures 4.1.1-2 and 4.1.1-3 below include the five-year wind rose diagrams for Byron Station and Braidwood Station, respectively, based on the lower and upper level data in the meteorological database used for the ARCON96 analysis scenarios. Other ARCON96 parameters are provided in Table 4.1.1-1 (Braidwood Unit 1), Table 4.1.1-2 (Braidwood Unit 2), Table 4.1.1-3 (Byron Unit 1), and Table 4.1.1-4 (Byron Unit 2).

Figure 4.1.1-2
Byron Wind Roses



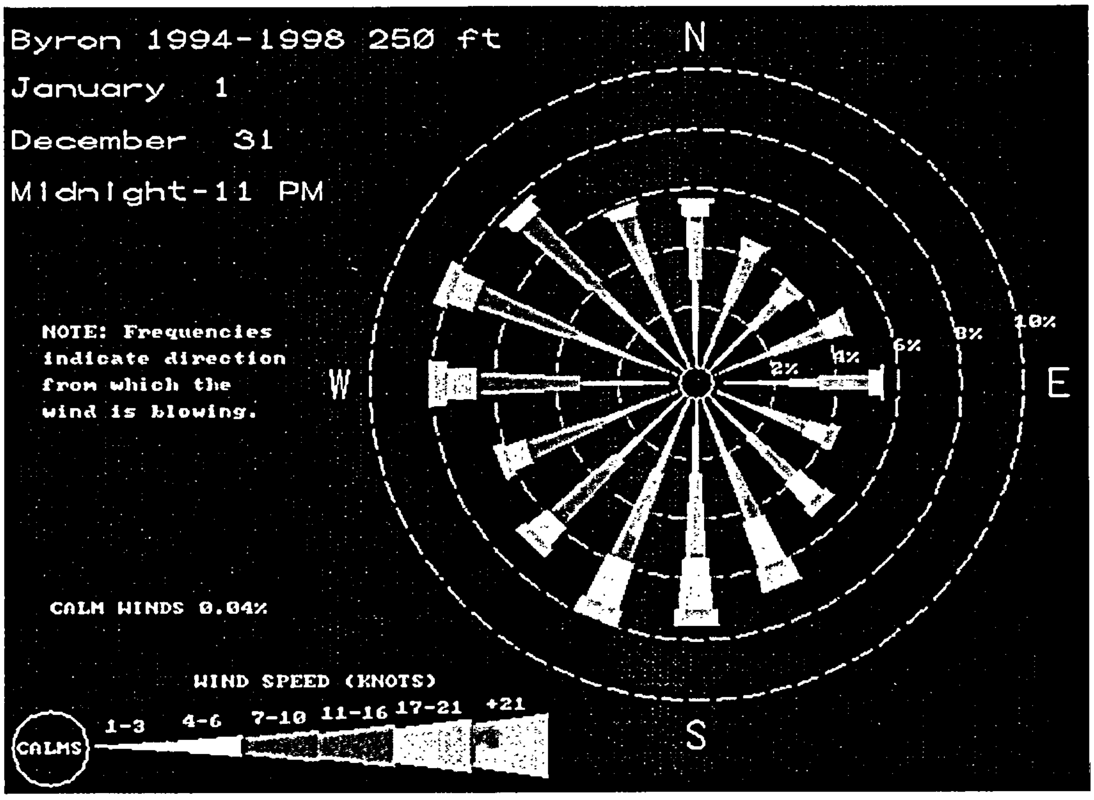
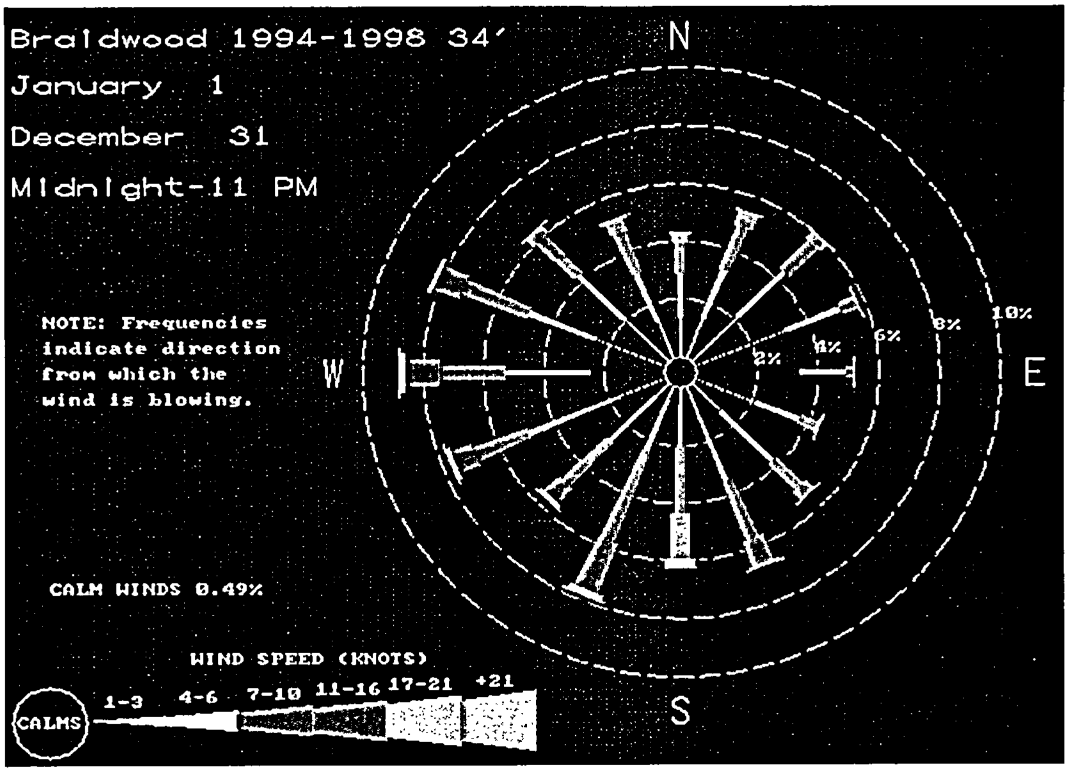


Figure 4.1.1-3
Braidwood Wind Roses



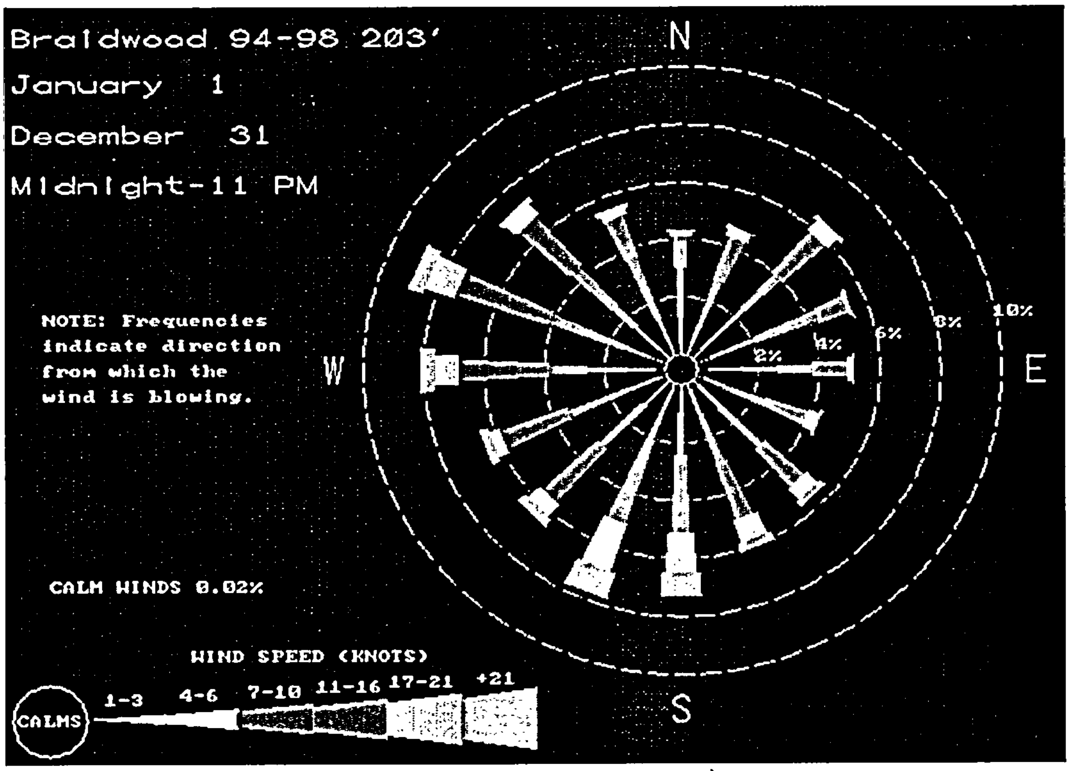


Table 4.1.1-1
Braidwood Unit 1 ARCON96 Input Data Summary

	Containment Wall to Control Room Fresh Air Intake	Containment Wall To Control Room TB Emergency Air Intake	Plant Vent to Control Room Fresh Air Intake	Plant Vent to Control Room TB Emergency Air Intake	PORVs/Safety Valves to Control Room Fresh Air Intake	PORVs/Safety Valves to Control Room TB Emergency Air Intake	Main Steam Line Break to Control Room Fresh Air Intake	Main Steam Line Break to TB Emergency Air Intake
Number of Meteorological Data Files	5	5	5	5	5	5	5	5
Height of lower wind instrument (m)	10.4	10.4	10.4	10.4	10.4	10.4	10.4	10.4
Height of upper wind instrument (m)	61.9	61.9	61.9	61.9	61.9	61.9	61.9	61.9
Release height (m)	29.7	29.7	61	61	9.8	9.8	7.9	7.9
Building Area (m^2)	2916.7	2916.7	2227.6	752.6	2916.7	2916.7	2850.7	752.6
Effluent vertical velocity (m/s)	0	0	0	0	0	0	0	0
Vent or stack flow (m^3/s)	0	0	0	0	0	0	0	0
Vent or stack radius (m)	0	0	0	0	0	0	0	0
Direction .. intake to source (deg)	75	82	217	176	12	51	240	176
Wind direction sector width (deg)	90	90	90	90	90	90	90	90
Wind direction window (deg)	030 - 120	037 - 127	172 - 262	131 - 221	327 - 057	006 - 096	195 - 285	131 - 221
Distance to intake (m)	7.6	30.5	34.1	27.4	22.9	35.1	43.3	13.4
Intake height (m)	21.2	20.4	21.2	20.4	21.2	20.4	7.9	20.4
Terrain elevation difference (m)	0	0	0	0	0	0	0	0
Minimum Wind Speed (m/s)	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
Surface roughness length (m)	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2
Sector averaging constant	4.3	4.3	4.3	4.3	4.3	4.3	4.3	4.3
Initial value of sigma y	8.18	8.18	0	0	0	0	0	0
Initial value of sigma z	9.9	9.9	0	0	0	0	0	0
Total number of hours of data processed	43824	43824	43824	43824	43824	43824	43824	43824
Hours of missing data	34	34	34	34	34	34	34	34
Hours direction in window	9738	9507	12324	11936	9377	9765	13206	11831
Hours elevated plume w/ dir. in window	0	0	0	0	0	0	0	0
Hours of calm winds	220	220	226	226	220	220	220	220
Hours direction not in window or calm	33832	34063	31240	31628	34193	33805	30364	31739

Table 4.1.1-2
Braidwood Unit 2 ARCON96 Input Data Summary

	Containment Wall to Control Room Fresh Air Intake	Containment Wall To Control Room TB Emergency Air Intake	Plant Vent to Control Room Fresh Air Intake	Plant Vent to Control Room TB Emergency Air Intake	PORVs/Safety Valves to Control Room Fresh Air Intake	PORVs/Safety Valves to Control Room TB Emergency Air Intake	Main Steam Line Break to Control Room Fresh Air Intake	Main Steam Line Break to TB Emergency Air Intake
Number of Meteorological Data Files	5	5	5	5	5	5	5	5
Height of lower wind instrument (m)	10.4	10.4	10.4	10.4	10.4	10.4	10.4	10.4
Height of upper wind instrument (m)	61.9	61.9	61.9	61.9	61.9	61.9	61.9	61.9
Release height (m)	29.7	29.7	61	61	9.8	9.8	7.9	7.9
Building Area (m^2)	2916.7	2916.7	2227.6	752.6	2916.7	2916.7	2850.7	752.6
Effluent vertical velocity (m/s)	0	0	0	0	0	0	0	0
Vent or stack flow (m^3/s)	0	0	0	0	0	0	0	0
Vent or stack radius (m)	0	0	0	0	0	0	0	0
Direction .. intake to source (deg)	106	99	323	4	168	129	300	4
Wind direction sector width (deg)	90	90	90	90	90	90	90	90
Wind direction window (deg)	061 - 151	054 - 144	278 - 008	319 - 049	123 - 213	084 - 174	255 - 345	319 - 049
Distance to intake (m)	7.6	30.5	34.1	27.4	22.9	35.1	43.3	13.4
Intake height (m)	21.2	20.4	21.2	20.4	21.2	20.4	7.9	20.4
Terrain elevation difference (m)	0	0	0	0	0	0	0	0
Minimum Wind Speed (m/s)	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
Surface roughness length (m)	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2
Sector averaging constant	4.3	4.3	4.3	4.3	4.3	4.3	4.3	4.3
Initial value of sigma y	8.18	8.18	0	0	0	0	0	0
Initial value of sigma z	9.9	9.9	0	0	0	0	0	0
Total number of hours of data processed	43824	43824	43824	43824	43824	43824	43824	43824
Hours of missing data	34	34	34	34	34	34	34	34
Hours direction in window	9224	9200	12124	9326	11347	9635	13277	9380
Hours elevated plume w/ dir. in window	0	0	0	0	0	0	0	0
Hours of calm winds	220	220	226	226	220	220	220	220
Hours direction not in window or calm	34346	34370	31440	34238	32223	33935	30293	34190

Table 4.1.1-3
Byron Unit 1 ARCON96 Input Data Summary

	Containment Wall to Control Room Fresh Air Intake	Containment Wall To Control Room TB Emergency Air Intake	Plant Vent to Control Room Fresh Air Intake	Plant Vent to Control Room TB Emergency Air Intake	PORVs/Safety Valves to Control Room Fresh Air Intake	PORVs/Safety Valves to Control Room TB Emergency Air Intake	Main Steam Line Break to Control Room Fresh Air Intake	Main Steam Line Break to TB Emergency Air Intake
Number of Meteorological Data Files	5	5	5	5	5	5	5	5
Height of lower wind instrument (m)	9.1	9.1	9.1	9.1	9.1	9.1	9.1	9.1
Height of upper wind instrument (m)	76.2	76.2	76.2	76.2	76.2	76.2	76.2	76.2
Release height (m)	29.7	29.7	61	61	9.8	9.8	7.9	7.9
Building Area (m^2)	2916.7	2916.7	2227.6	752.6	2916.7	2916.7	2850.7	752.6
Effluent vertical velocity (m/s)	0	0	0	0	0	0	0	0
Vent or stack flow (m^3/s)	0	0	0	0	0	0	0	0
Vent or stack radius (m)	0	0	0	0	0	0	0	0
Direction .. intake to source (deg)	255	262	37	356	192	231	60	356
Wind direction sector width (deg)	90	90	90	90	90	90	90	90
Wind direction window (deg)	210 - 300	217 - 307	352 - 082	311 - 041	147 - 237	186 - 276	015 - 105	311 - 041
Distance to intake (m)	7.6	30.5	34.1	27.4	22.9	35.1	43.3	13.4
Intake height (m)	21.2	20.4	21.2	20.4	21.2	20.4	7.9	20.4
Terrain elevation difference (m)	0	0	0	0	0	0	0	0
Minimum Wind Speed (m/s)	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
Surface roughness length (m)	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2
Sector averaging constant	4.3	4.3	4.3	4.3	4.3	4.3	4.3	4.3
Initial value of sigma y	8.18	8.18	0	0	0	0	0	0
Initial value of sigma z	9.9	9.9	0	0	0	0	0	0
Total number of hours of data processed	43824	43824	43824	43824	43824	43824	43824	43824
Hours of missing data	71	71	71	71	71	71	71	71
Hours direction in window	12840	13008	8511	10138	13187	12422	8067	9728
Hours elevated plume w/ dir. in window	0	0	0	0	0	0	0	0
Hours of calm winds	85	85	109	109	85	85	85	85
Hours direction not in window or calm	30828	30660	35133	33506	30481	31246	35601	33940

Table 4.1.1-4
Byron Unit 2 ARCON96 Input Data Summary

	Containment Wall to Control Room Fresh Air Intake	Containment Wall To Control Room TB Emergency Air Intake	Plant Vent to Control Room Fresh Air Intake	Plant Vent to Control Room TB Emergency Air Intake	PORVs/Safety Valves to Control Room Fresh Air Intake	PORVs/Safety Valves to Control Room TB Emergency Air Intake	Main Steam Line Break to Control Room Fresh Air Intake	Main Steam Line Break to TB Emergency Air Intake
Number of Meteorological Data Files	5	5	5	5	5	5	5	5
Height of lower wind instrument (m)	9.1	9.1	9.1	9.1	9.1	9.1	9.1	9.1
Height of upper wind instrument (m)	76.2	76.2	76.2	76.2	76.2	76.2	76.2	76.2
Release height (m)	29.7	29.7	61	61	9.8	9.8	7.9	7.9
Building Area (m^2)	2916.7	2916.7	2227.6	752.6	2916.7	2916.7	2850.7	752.6
Effluent vertical velocity (m/s)	0	0	0	0	0	0	0	0
Vent or stack flow (m^3/s)	0	0	0	0	0	0	0	0
Vent or stack radius (m)	0	0	0	0	0	0	0	0
Direction .. intake to source (deg)	286	279	143	184	348	309	120	184
Wind direction sector width (deg)	90	90	90	90	90	90	90	90
Wind direction window (deg)	241 - 331	234 - 324	098 - 188	139 - 229	303 - 033	264 - 354	075 - 165	139 - 229
Distance to intake (m)	7.6	30.5	34.1	27.4	22.9	35.1	43.3	13.4
Intake height (m)	21.2	20.4	21.2	20.4	21.2	20.4	7.9	20.4
Terrain elevation difference (m)	0	0	0	0	0	0	0	0
Minimum Wind Speed (m/s)	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
Surface roughness length (m)	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2
Sector averaging constant	4.3	4.3	4.3	4.3	4.3	4.3	4.3	4.3
Initial value of sigma y	8.18	8.18	0	0	0	0	0	0
Initial value of sigma z	9.9	9.9	0	0	0	0	0	0
Total number of hours of data processed	43824	43824	43824	43824	43824	43824	43824	43824
Hours of missing data	71	71	71	71	71	71	71	71
Hours direction in window	13487	13313	10160	12830	10453	13144	10017	13145
Hours elevated plume w/ dir. in window	0	0	0	0	0	0	0	0
Hours of calm winds	85	85	109	109	85	85	85	85
Hours direction not in window or calm	30181	30355	33484	30814	33215	30524	33651	30523

Calculations

The χ/Q values resulting from the Byron Station and Braidwood Station ARCON96 modeling analysis of each source/intake scenario for the required time intervals are presented in Table 4.1.3-1.

4.1.2 PAVAN Modeling Analysis of EAB and LPZ χ/Q

Methodology and Acceptance Criteria

The PAVAN model contains certain model options for executing the program. Table 4.1.2-1 below summarizes the options invoked for the EAB and LPZ χ/Q Byron and Braidwood calculations.

Table 4.1.2-1
PAVAN Modeling Options

Option No.	Description	Option Invoked?
1	Calculate σ_y and σ_x based on desert diffusion.	No
2	χ/Q values include evaluation for no building wake.	No
3	ENVLOP calculations printed which describe upper envelope curve.	No
4	Print points used in upper envelope curve and calculation.	Yes
5	Null	---
6	Joint frequency distribution in % frequency format.	No
7	Print χ/Q calculation details	Yes
8	Distribute calm winds observations into first wind speed category.	Yes
9	Use site-specific terrain adjustment factors for the annual average calculations.	Yes*
10	Assume a default terrain adjustment factor for the average annual calculations. Option 10 is applied, which together with application of Option 9 means that site-specific terrain factors will be used.	Yes

* A uniform value of 1.0 is used.

Design Input (PAVAN)

Source/Receptor Scenarios and Configurations

At each station, the outer containment wall and the midpoint between the two reactors are the assumed release points for the EAB and LPZ, respectively. The EAB and LPZ at Byron Station are located at 445 m and 4828 m, respectively, and at 485 m (EAB) and 1810 m (LPZ) at Braidwood Station. These releases do not qualify as elevated releases in accordance with RG 1.145; therefore, they were executed by PAVAN as "ground" type releases requiring an assumption of a 10 m release height. The containment building height of 60.7 m (i.e., including the dome) and the calculated containment building vertical cross-sectional area of 2916.7 m^2 were used for each of the scenarios.

Meteorological Data (PAVAN)

Byron Station and Braidwood Station meteorological data from the five-year period, (i.e., 1994 – 1998), were used in the PAVAN analysis.

The format of PAVAN meteorological input consists of a joint wind direction (based on sixteen 22.5 degree sectors), wind speed (seven intervals), and stability class (seven classes) occurrence frequency distribution.

Each meteorological joint frequency distribution for input to PAVAN was prepared using the Washington Group International (WGI) qualified program ARCONtoPAVANMET Revision 1 to transform the data to a joint wind-stability occurrence frequency distribution. The seven wind speed categories were defined according to RG 1.23 (Reference 7.11) with the first category identified as "calm." For each station, the minimum wind speed (i.e., wind threshold) was set to 0.8 mph and "calm" wind speeds were assigned a value of 0.4 mph (i.e., one-half the threshold value) consistent with UFSAR Section 2.3.4, "Short Term (Accident) Diffusion Estimates." A midpoint was also assumed between each of the RG 1.23 wind speed categories, Nos. 2-6, to be inclusive of all monitored wind speeds. The RG 1.23 wind speed categories have, therefore, been refined in Table 4.1.2-2 as follows.

Table 4.1.2-2
Defined Wind Speed Category Ranges For PAVAN Modeling

Category No.	Regulatory Guide 1.23 Speed Interval (mph)	PAVAN-Assumed Speed Interval (mph)
1 (Calm)	0 to < 1	0 to<0.80
2	1 to 3	≥0.80 to <3.5
3	4 to 7	≥3.5 to <7.5
4	8 to 12	≥7.5 to <12.5
5	13 to 18	≥12.5 to <18.5
6	19 to 24	≥18.5 to <24
7	>24	≥24

Calculations

The Byron Station and Braidwood Station χ/Q values for the EAB and LPZ were calculated by the PAVAN modeling analysis for each time interval in accordance with RG 1.145.

4.1.3 Atmospheric Dispersion Summary and Conclusions

The maximum (i.e., bounding) ARCON96 and PAVAN χ/Q results for all units of Byron and Braidwood are summarized in Table 4.1.3-1 below.

Table 4.1.3-1

Byron-Braidwood Maximum Site-Unit χ/Q Summary

	Release Path		Recommended χ/Q (sec/m3)					
Model	**Release Point**	**Receptor/ Intake**	**0-2 hr**	**2-8 hr(1)**	**8-24 hr**	**1-4 day**	**4-30 day**	**Notes**
ARCON96	Containment Wall	CR Fresh Air	1.73E-03	1.24E-03	5.23E-04	3.55E-04	2.62E-04	Diffuse area source per RG 1.194 Sec. 3.2.4
ARCON96	Containment Wall	CR Turbine Building Emergency Air	1.01E-03	7.25E-04	3.07E-04	2.07E-04	1.46E-04	
ARCON96	Plant Vent	CR Fresh Air	2.22E-03	1.80E-03	7.20E-04	4.75E-04	3.81E-04	Building area perpendicular to wind utilized per RG 1.194 Table A-2
ARCON96	Plant Vent	CR Turbine Building Emergency Air	2.46E-03	1.92E-03	8.14E-04	5.52E-04	4.40E-04	
ARCON96	PORVs/Safety Valves	CR Fresh Air	1.77E-03	1.52E-03	6.98E-04	4.72E-04	3.50E-04	Includes factor of 5 reduction for vertical uncapped release per RG 1.194 Sec. 6
ARCON96	PORVs/Safety Valves	CR Turbine Building Emergency Air	8.14E-04	6.98E-04	3.12E-04	1.95E-04	1.67E-04	
ARCON96	MSLB	CR Fresh Air	3.20E-03	2.73E-03	1.19E-03	8.20E-04	6.60E-04	"Taut String Length" to Fresh Air Intake and building area perpendicular to wind utilized per RG 1.194
ARCON96	MSLB	CR Turbine Building Emergency Air	1.70E-02	1.46E-02	6.68E-03	4.48E-03	3.31E-03	Building area perpendicular to wind utilized per RG 1.194
PAVAN	Outer Containment Wall	EAB	5.36E-04 (N)	2.65E-04 (SE)	1.89E-04 (SE)	9.04E-05 (SE)	3.16E-05 (SE)	In conformance with RG 1.145
PAVAN	Midpoint between two reactors	LPZ	9.32E-05 (ESE)	4.50E-05 (ESE)	3.12E-05 (ESE)	1.41E-05 (ESE)	4.54E-06 (ESE)	

(1) χ/Q values for the PAVAN results are for 0-8 hour time period.

4.2 Dose Conversion Factors

The revised Dose Conversion Factors (DCFs) from the U.S. Federal Guidance Report 11 and 12 (References 7.15 and 7.16) are used for these AST analyses. The RADTRAD Version 3.03 (referred to as RADTRAD) computer code inputs these values directly from its internal database.

4.3 EAB and LPZ Dose Model

Between the Byron and Braidwood Stations, the EAB and LPZ χ/Qs have been determined, and at worst-case, are respectively located 445 m and 1810 m from the postulated release locations.

4.4 Control Room Dose Model

The following CR dose model, with assumed CR volume and filtration system air flow rates, is used to calculate the operator dose for all the re-analyzed accident scenarios.

CR Volume

The air volumes of the CR at the Byron and Braidwood Stations are 230,830 ft^3 and 232,872 ft^3, respectively. These volumes exclude the upper cable spreading rooms (UCSRs) because they do not have return air flow and do not require occupancy during post-accident conditions. For this analysis using the RADTRAD code, these volumes were conservatively modeled as one characteristic 200,000 ft^3 volume.

Using a smaller CR volume, in combination with the high flow rates discussed below, is conservative because it leads to the CR reaching its maximum activity concentration faster than if the actual, substantially larger volume was used. This volume (i.e., the 200,000 ft^3 volume) is also well above the actual 70,275 ft^3 CR shielded volume; therefore, the internal cloud shine dose remains conservatively calculated.

CR Filtration System Flow Rates

Actuation of the CR filtration system places the system in the emergency mode of operation. Actuation of the system to the emergency mode of operation starts the makeup fan, opens the turbine building intake damper, isolates the normal intake from outside dampers, isolates the purge dampers (if open), opens the recirculation charcoal adsorber dampers, and closes the recirculation charcoal adsorber bypass dampers. The operating supply and return fans continue to operate. Outside air from the turbine building is filtered and added to the air being recirculated through the CR.

The Byron and Braidwood Stations have a recirculation flow of 43,500 cfm and make-up airflow of 6000 cfm (± 10%), totaling 49,500 cfm of total combined flow to the CR volume. However, there could be a failure scenario where open inlet and outlet dampers on the unused CR make-up train result in an additional 1500 cfm of filtered intake into the CR during the emergency mode of operation. Therefore, for conservatism, this 1500 cfm was rounded up to 2000 cfm, and is added to the 49,500 cfm total combined flow during the emergency mode of operation. Of this 51,500 cfm, a separate volume designated as the UCSR, receives an un-recirculated intake flow of 1319 cfm at Byron Station, and 2430 cfm at Braidwood Station. Because they are un-recirculated, these flows are subtracted from the 51,500 cfm total combined flow, making the emergency mode of operation adjusted combined flows to the Byron and Braidwood CRs 50,181 cfm and 49,070 cfm, respectively. To calculate the CR volume emergency mode of operation adjusted make-up flow, the ratio of the adjusted combined flow (i.e., 50,181 cfm) to the total combined flow (i.e., 51,500 cfm) is multiplied by the 8000 cfm total make-up flow (i.e., the 6000 cfm make-up air flow plus 2000 cfm from the unused CR make-up train failure). This results in adjusted emergency mode of operation make-up flows of 7795 cfm (±10%) and 7623 cfm (±10%) to the Byron and Braidwood CRs, respectively.

Following CR emergency mode of operation isolation, the credited CR filtration is 99% for the HEPA and 95% for the charcoal filters. The assumed intake flow rate is 8575 cfm, which is the upper 10% bound of the Byron Station 7795 cfm make-up flow noted above. The charcoal filtration efficiency of 95% reflects a reduction from the pre-AST analysis of record (AOR) value of 99%. The maximum intake rate is used because sensitivity analyses have shown that it is conservative to maximize the potentially contaminated emergency mode of operation make-up airflow into the CR.

The AST analysis does not credit the CR filtration system emergency mode of operation during the first 30 minutes of the accident and; therefore, unfiltered make-up air is assumed. This assumption is intended to simulate an allowance for delayed initiation of the CR emergency mode of operation; however, no changes to the Control Room Ventilation (VC) Filtration System Actuation Instrumentation TS (i.e., TS 3.3.7) are being proposed. During this 30-minute period, it is conservative not to consider the potential 2000 cfm of filtered intake entering the CR (i.e., the flow from the unused CR make-up train failure noted above). This assumption is used because filtered flow would act to "clean" the unfiltered air being brought in by the other unfiltered flows (i.e., normal intake and unfiltered inleakage) during these 30 minutes. Sensitivity analyses using RADTRAD confirmed that such additional filtered intake during the first 30 minutes of an accident (i.e., with the CR filtration system in the normal mode of operation) would lower the CR dose consequences. Therefore, during the first 30 minutes of an accident, the adjusted combined flows to the Byron and Braidwood CRs are assumed to be 48,181 cfm (i.e., 43,500 cfm recirculation flow + 6000 cfm make-up flow – 1319 cfm flow to UCSR); and 47,070 cfm (i.e., 43,500 cfm recirculation flow + 6000 cfm make-up flow – 2430 cfm flow to UCSR), respectively, and the subsequent adjusted make-up flows become 5840 cfm (±10%) (i.e., 48,181 cfm / 49,500 cfm x 6000 cfm) and 5705 cfm (±10%) (i.e., 47,070 cfm / 49,500 cfm x 6000) to the Byron and Braidwood CRs, respectively. Therefore, the bounding unfiltered intake flow rate into the CR during this initial 30-minute period is assumed to be 6424 cfm which is the upper bound of the Byron Station 5840 cfm make-up flow noted above.

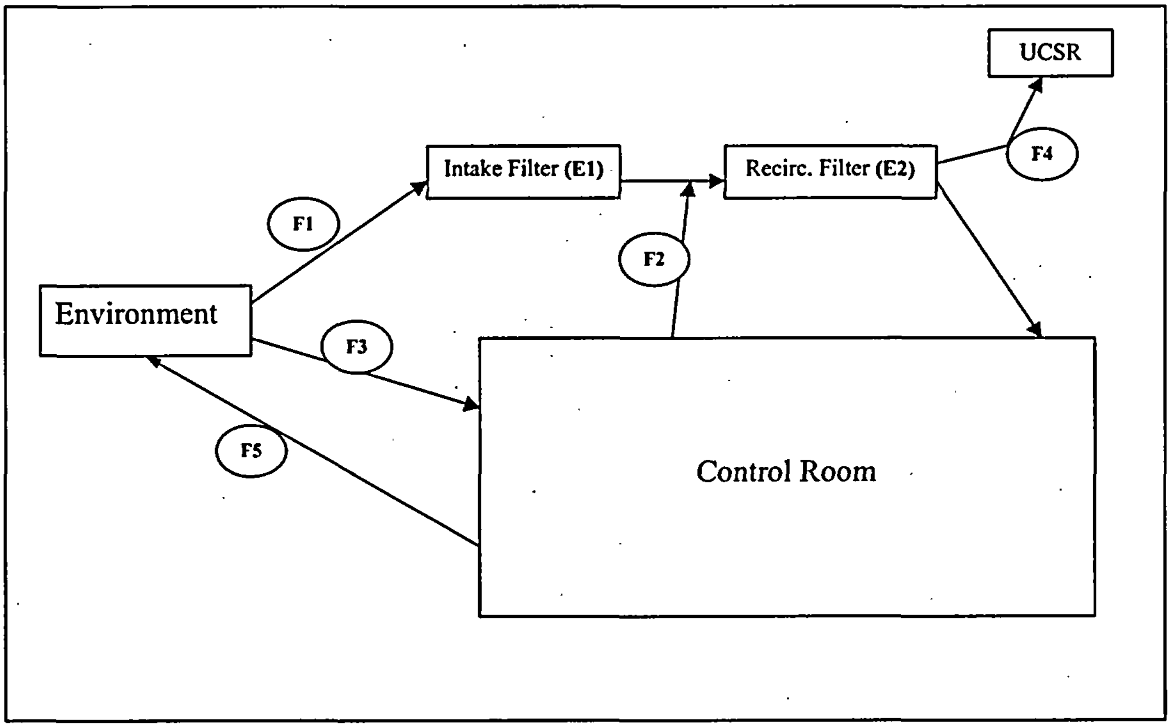
The Byron and Braidwood CRs are modeled using a conservatively reduced (i.e., by the -10% lower bound) recirculation train flow rate of 39,150 cfm (i.e., 0.90 x 43,500 cfm). This recirculation train filtration consists of an 80% aerosol particulate filter (i.e., pre-filter), and a 90% elemental and organic iodine filter (i.e., charcoal filter). No reduction in the efficiency of these filters is proposed.

For additional conservatism, an unfiltered inleakage rate allowance of 1000 cfm is modeled for the accident duration.

Any activity that enters the CR originates from a source that is characterized by a dispersion factor, calculated using ARCON96. The total dose to CR operators over the 720-hour (i.e., 30 day) required mitigation period is the result of the released activities that enter through the air intake, either filtered or unfiltered.

The following Figure 4.4-1 summarizes the CR dose model inputs and assumptions.

Figure 4.4-1
Control Room Ventilation Model, as Analyzed



Time	F1 Intake (cfm)	E1 (%)	F2 Recirc. (cfm)	E2 (%)	F3 Unfiltered Inleakage (cfm)	F4 Flow to UCSR (cfm)	F5 CR Exfiltration (cfm)
0-30 min	6424	0	39,150	0	1000	Byron: 1319 Braidwood: 2430	7424
30 min to 30 days	8575	99 HEPA 95 Charcoal	39,150	80 Pre-Filter 90 Charcoal	1000	Byron: 1319 Braidwood: 2430	9575

F = Flowpath
E = Efficiency
UCSR = Upper Cable Spreading Room

4.5 Loss of Coolant Accident (LOCA)

Input parameters used for the LOCA analysis are given in Attachment 6, Tables 1.0, 2.0 and 3.0. Conformance with RG 1.183 guidance addressing LOCA analysis is provided in Attachment 7, Tables A and B.

4.5.1 Inputs and Assumptions - LOCA

The following primary assumptions from previous LOCA analyses continue to apply:

(a) Two release pathways are considered: containment leakage and ECCS recirculation leakage; and

(b) A single electrical train failure is assumed to remove the following equipment from service: one containment spray train, one ECCS train, and two of the four reactor containment fan coolers (RCFCs).

Primary Containment and ECCS Leakage - LOCA

Primary Containment Leakage

Primary containment leakage is the main contributor to LOCA doses from the Byron and Braidwood Stations. The impacts of the design basis LOCA are mitigated by: (1) a controlling design basis leak rate; and (2) operation of the containment spray system. While most penetrations are into the auxiliary building that has an ESF filtered exhaust, this filtration system is not credited for primary containment leakage.

Primary containment leakage is now modeled as a diffuse area source in conformance with RG 1.194.

Principal differences between pre-AST and post-AST accident analyses are:

- The assumed containment leak rate is increased from 0.1% per day to 0.2% per day. This leak rate continues to be assumed to be reduced to one-half the initial value after 24 hours due to expected reductions in containment pressure.
- 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.
- New dispersion factors, developed in conformance with the latest guidance in RG 1.145 and RG 1.194 are used. Dispersion factors for primary containment leakage for the CR are based on a diffuse area source.

ECCS Leakage

ECCS leakage is a minor contributor to LOCA doses for the Byron and Braidwood units. ECCS leakage is controlled in accordance with TS 5.5.2, "Primary Coolant Sources Outside

Containment." As described in UFSAR Table 15.6-13, "Recirculation Loop Leakage External to Containment," the maximum permitted recirculation loop leakage (i.e., ECCS leakage) is 15,249 cc/hr (i.e., approximately 4.0 gal/hr) for Braidwood Station and 13,294 cc/hr (i.e., approximately 3.5 gal/hr) for Byron Station. These values are assumed in the pre-AST analysis of dose assessment to the CR, with a safety factor of two applied in accordance with Standard Review Plan 15.6.5. The ECCS leakage rate assumed in the AST LOCA analysis is 276,000 cc/hr for the Byron and Braidwood units; and the corresponding SRP 15.6.5 "maximum operational leakage" would be half of this value; i.e., 138,000 cc/hr.

The principal difference between pre-AST and post AST accident analyses is that ECCS maximum operational leakage is assumed to be 138,000 cc/hr for the Byron and Braidwood units, rather than 15,249 cc/hr for Braidwood Station and 13,294 cc/hr for Byron Station. The ECCS leakage flashing fractions assumption continues to be 10% for the duration of the accident.

Fuel Damage and Core Source Term - LOCA

For conservatism, the LOCA core source terms are those associated with a DBA power level of 3658.3 MWth, which includes an additional 2% power over that of the full licensed power to account for uncertainty.

The AST values used in this analysis were derived using guidance outlined in RG 1.183. A list of 60 core isotopic nuclides and their curie per megawatt activities was extracted from the RADTRAD "NIF" files as listed below. The release fractions associated with all of these nuclide groups, as detailed in RG 1.183, were applied to their given groups, and input into the RADTRAD "RTF" files. RADTRAD uses these files combined with the power of 3658.3 MWth to develop the source terms for the DBA LOCA.

Nuclide Inventory (Ci/MWth)	
Co-58	0.2553E+03
Co-60	0.1953E+03
Kr-85	0.2851E+03
Kr-85m	0.8592E+04
Kr-87	0.1696E+05
Kr-88	0.2392E+05
Rb-86	0.6480E+02
Sr-89	0.2907E+05
Sr-90	0.2242E+04
Sr-91	0.3930E+05
Sr-92	0.4136E+05
Y-90	0.2347E+04
Y-91	0.3553E+05
Y-92	0.4150E+05
Y-93	0.4624E+05
Zr-95	0.4560E+05
Zr-97	0.4663E+05
Nb-95	0.4593E+05
Mo-99	0.5058E+05
Tc-99m	0.4429E+05
Ru-103	0.4094E+05

Nuclide Inventory (Ci/MWth)	
Ru-105	0.2798E+05
Ru-106	0.1387E+05
Rh-105	0.2552E+05
Sb-127	0.2848E+04
Sb-129	0.8523E+04
Te-127	0.2812E+04
Te-127m	0.3668E+03
Te-129	0.8389E+04
Te-129m	0.1249E+04
Te-131m	0.3838E+04
Te-132	0.3804E+05
I-131	0.2671E+05
I-132	0.3863E+05
I-133	0.5529E+05
I-134	0.6143E+05
I-135	0.5159E+05
Xe-133	0.5396E+05
Xe-135	0.1532E+05
Cs-134	0.5306E+04
Cs-136	0.1503E+04
Cs-137	0.3077E+04
Ba-139	0.5089E+05
Ba-140	0.4922E+05
La-140	0.5036E+05
La-141	0.4646E+05
La-142	0.4557E+05
Ce-141	0.4498E+05
Ce-143	0.4468E+05
Ce-144	0.3414E+05
Pr-143	0.4350E+05
Nd-147	0.1836E+05
Np-239	0.5178E+06
Pu-238	0.1027E+03
Pu-239	0.7698E+01
Pu-240	0.8971E+01
Pu-241	0.3548E+04
Am-241	0.3921E+01
Cm-242	0.1110E+04
Cm-244	0.1209E+03

Activity Release Fractions - LOCA

Release fractions and timing are taken from RG 1.183, Tables 2 and 4 as shown below.

RG 1.183, Table 2
PWR Core Inventory Fraction Released Into Containment

Group	Gap Release Phase	Early In-vessel Phase	Total
Noble Gases	0.05	0.95	1.0
Halogens	0.05	0.35	0.4
Alkali Metals	0.05	0.25	0.3
Tellurium Metals	0.00	0.05	0.05
Ba, Sr	0.00	0.02	0.02
Noble Metals	0.00	0.0025	0.0025
Cerium Group	0.00	0.0005	0.0005
Lanthanides	0.00	0.0002	0.0002

RG 1.183, Table 4
PWR LOCA Release Phases

Phase	Onset	Duration
Gap Release	30 sec	0.5 hr
Early In-Vessel	0.5 hr	1.3 hr

Airborne Activity Removal Mechanisms in Containment - LOCA

As discussed below, containment spray, natural deposition, decay, and leakage are credited as airborne activity removal mechanisms.

Removal by Containment Spray - LOCA

For Byron and Braidwood Stations, iodine removal by containment spray continues to use Standard Review Plan Chapter 6.5.2, Revision 2 (Reference 7.18) as does the pre-AST design basis. The major impacts of application of AST are that no initial plateout fraction is assumed, and the duration of credited spray is modified. Both changed and unchanged parameters are discussed below.

Sprayed and Unsprayed Volumes, and Air Exchange Rate - LOCA

These subject parameters are consistent with the analysis of record and have not been modified in the AST methodology. The containment volume is 2.85E6 cubic feet with 82.5% of the containment volume sprayed; i.e., the sprayed volume is 2.35125E6 cubic feet and the unsprayed volume is 4.9875E5 cubic feet. These values are rounded to 2.35E6 and 5.0E5 cubic feet, respectively, in the analysis. Initial activity distribution is accordingly, 82.5% in the sprayed region, and 17.5% in the unsprayed region.

Transfer between these two volumes is assumed to be limited to that provided by the reactor containment fan coolers (RCFCs). Even without the RCFCs there would be significant mixing induced by the containment sprays and by the combination of steaming and heat transfer. Two of four RCFCs are credited in the analysis. The assumed flow rate per RCFC is 65,000 cfm for a total of 130,000 cfm.

Spray Removal Coefficients for Aerosols (SRP 6.5.2 particulate removal) - LOCA

From SRP 6.5.2, the first order removal coefficient for particulate (or, effectively, aerosols) may be estimated by:

$$\lambda_p = \frac{3\,(h)\,(F)\,(E)}{2\,(V)\,(D)}$$

where:

λ_p = spray removal constant, hr^{-1}

h = drop fall height = 141 ft

F = volume flow rate of sprays, ft^3/hr
= 2950 gpm (applicable to both injection and recirculation phases)
= (2950 gpm)(60 min/hr) / (7.4805 gal/ft^3) = 23,661.5 ft^3/hr

V = sprayed volume, ft^3
= (0.825)(2.85E6 ft^3)
= 2.35125E6 ft^3

E/D = ratio of a dimensionless collection efficiency "E" to the average spray drop diameter "D"
= 10 m^{-1} for $M_o/M_t \leq 50$
= 1 m^{-1} for $M_o/M_t > 50$
where M_o/M_t is the ratio of the initial aerosol mass to the aerosol mass at time t (note that this ratio also defines the decontamination factor (DF) achieved)

$$\lambda_{P1} = \frac{3\,(141\text{ ft})\,(23{,}661.5\text{ ft}^3/\text{hr})\,(10\text{ m}^{-1})\,(0.3048\text{ m/ft})}{2\,(2.35\text{E6 ft}^3)} = 6.491\text{ hr}^{-1}$$

$$\lambda_{P2} = 0.1 \times 6.491\text{ hr}^{-1} = 0.65\text{ hr}^{-1}$$

(The calculated λ_{P1} and λ_{P2} values above are conservatively reduced to 6.0 hr^{-1} and 0.6 hr^{-1}, respectively).

It is assumed that after the end of the core activity release process the aerosols would continue to be removed at a λ of 6.0 hr^{-1} until an overall DF of 50 is achieved. Tests of RADTRAD aerosol-only runs are used to determined when a DF of 50 is reached with consideration of other spray timing issues discussed below, and with the initial activity assigned a nominal value corresponding to the guidance in RG 1.183.

Spray Removal Coefficients for Elemental Iodine (same methodology as SRP 6.5.2 based elemental iodine removal) - LOCA

The pre-AST SRP 6.5.2 based assessment of elemental iodine removal coefficients during containment spray continue to be used. The original design basis derivation of the spray removal coefficient is 30.3 hr^{-1}. In accordance with SRP 6.5.2, this value is reduced to 20 hr^{-1}. Elemental iodine removal is limited to a DF of 100. Test RADTRAD elemental iodine-only runs were used to determine when a DF of 100 is reached with consideration of other spray timing issues discussed below, and with the initial activity assigned a nominal value corresponding to the guidance in RG 1.183.

Spray Timing - LOCA

The analysis assumes injection spray is initiated at 90 seconds and continues for 20.9 minutes from the time of initiation. There may be a delay of as much as 10 minutes between the termination of containment spray during the ECCS injection phase and the initiation of containment spray during the ECCS recirculation phase.

For aerosol removal, the DF of 50 is reached at 2.21 hours. From that point until eight hours, the removal coefficient of 0.6 hr^{-1} is used. For elemental iodine removal, the DF of 100 is reached at 1.926 hours. After that time, no elemental iodine removal is credited.

Natural Deposition - LOCA

For Byron and Braidwood Stations, the RADTRAD computer program, including the Powers Natural Deposition algorithm based on NUREG/CR-6189 (Reference 7.26), is used for modeling aerosol deposition in containment. No natural deposition is assumed for elemental or organic iodine. The lower bound (i.e., 10%) level of deposition credit is used.

Decay Credited - LOCA

Decay of radioactivity is credited in all compartments prior to release. This is implemented in RADTRAD using the half-lives in the "NIF" files. The RADTRAD decay plus daughter option is used. In reality, daughter products such as xenon from iodines or iodines from tellurium are unlikely to readily escape from the medium in which the parent iodine or tellurium is contained. The RADTRAD feature to include daughter effects is selected for conservatism.

Mechanisms for Environmental Releases - LOCA

The applicable release paths for Byron and Braidwood Stations are leakage of airborne activity from containment, and leakage from the ECCS carrying reactor coolant outside containment into the auxiliary building. These release paths are discussed below.

NUREG 0737 Requirements Resulting From TMI-2 Accident

NUREG 0737, (Reference 7.21), requirements are addressed in the Byron Station and Braidwood Station UFSAR, Appendix E, "Requirements Resulting from TMI-2 Accident." The Byron and Braidwood Stations vital area design features impacted by application of AST and associated changes in plant parameters are specifically addressed in UFSAR, Appendix E, Item E.20, "Plant Shielding," and Item E.75, "Upgrade Emergency Response Facilities."

The radiological impacts evaluated in UFSAR Appendix E, "Requirements Resulting from TMI-2 Accident," are post-accident dose rates in vital areas and post-accident doses for essential post-accident paths due to contained sources. The contained sources are attributed to piping and equipment within the auxiliary building and "shine" from airborne activity within containment. As indicated in Note 3 on Table E.20-2, dose due to the plume (i.e., airborne activity) is not included in the Table. Containment source dose contributions are not significantly impacted by the AST methodology. Airborne containment activity is comparable or slightly smaller using AST due to time dependent release assumptions; therefore, the dose component due to activity in the containment remains bounding. Similarly, auxiliary building contained sources such as ECCS piping are not expected to see an increase in dose. However, airborne activity outside containment may be increased since the allowable containment leak rate has been doubled in the AST analysis. As noted above, the dose contributors to UFSAR Tables E.20-1 and E.20-2 locations and pathways currently do not include airborne contributions due to such factors as containment or ECCS leakage. As indicated in a note in UFSAR Figure E.20-2 (sheet 3 of 12) airborne radiation was accounted for separately and its contribution to the dose rate is added to the Table E.20-1 and E.20-2 doses to obtain total dose.

A review of the impact of AST sources and parameters (i.e., increased containment leak rates of 0.2% per day and 276,000 cc/hr ECCS leak rate, with 10% iodine flashing fraction) on airborne radiation was conducted. Applying the AST methodology results in a dose rate of 3.2 mRem/hr TEDE from ECCS sources, plus a dose rate of 84.6 mRem/hr TEDE from containment leakage sources (i.e., a total of 88 mRem/hr in the largest open volume in the auxiliary building). This dose rate is bounded by the worst one-hour whole body dose rate of 190.3 mRem/hr currently calculated in the pre-AST analysis for the same auxiliary building area. This area was chosen as it represents the most conservative geometry for dose considerations. Based on these calculations, the AST dose results are bounded by the existing pre-AST results. Therefore, the pre-AST conclusion that post-accident vital area doses will not constrain post-accident vital area access or accident response activities remains valid.

The Technical Support Center (TSC) emergency ventilation system, as described in UFSAR Appendix E, Item E.75, is reevaluated below using AST assumptions. Release modeling is identical to that used for the CR, with TSC specific ventilation parameters and χ/Q values used. Some additional margins and relaxation in input variables have been used, as follows.

1. No credit is taken for operation of the emergency filtration system for 30 minutes.

2. The nominal intake rate of 900 cfm is increased by 10% for this 30 minute period to allow for uncertainty. After the 30 minute period it has been determined that a reduction of 10% is the most conservative assumption.

3. An unfiltered inleakage rate of 450 cfm is assumed for the DBA-LOCA duration. This value is 50% of the TSC filtered intake pressurization flow and should be sufficiently conservative to minimize the need for confirmatory testing.

4. The charcoal adsorber credit is reduced to 95% from the pre-AST 99% to provide additional operational margin.

5. Recirculation filtration is based on the nominal flow of 1100 cfm minus 10% to allow for uncertainty and minimize activity removal. No credit is taken for recirculation filtration for the initial 30 minutes.

6. TSC specific χ/Qs are used that have been determined using RG 1.194 methodology.

Control Room Direct Gamma Dose – LOCA

The pre-AST contributors to CR doses due to gamma shine are identified in UFSAR Table 6.4-1, "Expected Dose to Control Room Personnel at Byron (Braidwood) Station Following a Loss of Coolant Accident (LOCA)," as being containment airborne activity, the post-LOCA plume surrounding the CR, and radioactivity accumulated on the CR filter.

The containment activity contribution of 0.023 rem whole body is slightly conservative for AST conditions, and therefore can be treated as a 0.023 rem TEDE dose contribution.

The small 0.003 rem whole-body pre-AST external plume shine dose to the CR is multiplied by a factor of five to yield 0.015 rem TEDE. Generally these doses are noble gas dominated. This conservative multiplier accounts for the factor of two increase in primary containment leak rate assumptions, and other increased contributors such as ECCS iodine. It should be noted that the new χ/Qs based on a diffuse area source are lower than the values historically used in the existing dose analysis.

The fission product filter loading has been reanalyzed using AST assumptions, and compared with those that would be determined using RG 1.4 (Reference 7.12) assumptions. The pre-AST source terms were found to be bounding. Therefore, the existing 0.013 rem whole body dose will continue to be used as a 0.013 rem TEDE dose. The total Direct Gamma Shine Dose is 0.05 rem TEDE.

χ/Q Calculations (Meteorology) - LOCA

Control Room

The CR χ/Q values input to RADTRAD were taken from the ARCON96 results of the Byron and Braidwood Stations design bases analyses. The limiting χ/Q values used are conservatively the worst-case combination of values from all four units at Byron Station and Braidwood Station.

Releases from containment to the environment and subsequently to the CR, utilize the χ/Q value associated with a diffuse area source. The applicability of this model is discussed in Table H, "Conformance with Regulatory Guide 1.194 (Diffuse Area Source Guidance)."

Activity released during the initial 30 minutes of the accident is introduced to the CR via the normal CR fresh air intake. After this period, when the CR outside air intake is assumed to be isolated, the emergency air intake located in the turbine building is used.

The CR, EAB and LPZ χ/Q two-hour values are selected such that they coincide with the release period that caused the highest doses..

High Wind Speed Considerations

Byron Station and Braidwood Station take no credit for any "secondary containment" (i.e., the "auxiliary building" for pressurized water reactors) filtration of primary containment leakage. The auxiliary building filtration is only credited for ECCS leakage. For evaluation of the effect of high wind speeds on the ability of the auxiliary building to maintain a negative pressure, a wind speed of 11.26 meters per second (i.e., 25.2 miles per hour) is used. This is the one-hour average value exceeded only 5% of the time for the total number of hours in the meteorological databases for the five-year period 1994-1998, taken at the highest wind measurement tower elevation of 250 feet for the bounding station (i.e., Byron). Based on the methodology taken from the American Society of Heating, Refrigerating and Air-Conditioning Engineers (ASHRAE) Handbook, 1985 Fundamentals, Table 2, Chapter 14, "Air Flow Around Buildings," a wind speed of greater than 32.2 mph would be required before the TS 3.7.12 minimum negative 0.25 inches water gauge auxiliary building pressure would become positive relative to outside air pressures at any building surface.

Therefore, an adequate vacuum can be obtained up to a wind speed of 32.2 mph, independent of the location of the sensor for outside air pressure used to maintain the negative 0.25 inches water building pressure. Although high wind speeds upwind of building surfaces can result in building inleakage increases, the resulting potential for increases in exhaust flow would be accommodated by changes in damper positioning and would result in better dispersion. The exhaust flow utilized in the RADTRAD modeling accommodates any flow increases without credit for the higher dispersion factors.

4.5.2 Acceptance Criteria – LOCA

Radiological doses resulting from a design basis LOCA to a CR operator and a person located at the EAB or LPZ are to be less than the regulatory dose limits given in Table 4.5.2-1.

Table 4.5.2-1
Regulatory Dose Limits – LOCA

Dose Type	Control Room (rem)	EAB and LPZ (rem)
TEDE Dose	5[a]	25[a]

[a] 10 CFR 50.67

4.5.3 Summary and Conclusions - LOCA

Table 4.5.3-1 below provides the results from the simulations modeled using the RADTRAD 3.03 code.

Table 4.5.3-1
LOCA Analysis Results

Source	Control Room (rem TEDE)	EAB (rem TEDE)	LPZ (rem TEDE)
Primary Containment Leakage	1.86	11.01 (Maximum 2 hr leakage period from 0.3 to 2.3 hrs)	1.59
ECCS Leakage	2.09	1.19 (Maximum 2 hr leakage period from 1.8 to 3.8 hrs)	1.40
Direct Shine Dose to CR from Containment, External Plume, & CR Filters	0.05	N/A	N/A
Total Dose	**4.00**	**12.2**	**2.99**
Limit	**5.00**	**25.0**	**25.0**

The Technical Support Center evaluations confirm that personnel doses will be within the CR limits in 10 CFR 50.67. The calculated dose is 1.75 rem TEDE for the containment leakage, and 2.06 rem TEDE for ECCS Leakage, for a total of 3.81 rem TEDE. Direct shine dose is assumed negligible because of larger distances from contained sources, and significantly better χ/Qs for filter loading.

4.6 Fuel Handling Accident (FHA)

Input parameters used for the FHA analysis are given in Attachment 6, Tables 1.0, 2.0 and 4.0. Conformance with RG 1.183 guidance addressing FHA analysis is provided in Attachment 7, Tables A and C.

4.6.1 Inputs / Assumptions - FHA

The Byron and Braidwood UFSAR Section 15.7.4, "Fuel Handling Accidents," defines this event as the drop of a spent fuel assembly onto the spent fuel pool floor or the core, resulting in the postulated rupture of the cladding of all the fuel rods in one assembly. Consistent with this UFSAR Section and RG 1.183, two potential accident locations were considered; the FHB and the containment. The dropped fuel assembly is assumed to have been subcritical for ≥48 hrs. The RADTRAD computer program was used to model the associated radiological releases.

The following assumptions and bounding analyzed conditions are utilized in the analysis.

a. The bounding core inventory was based on a DBA power level of 3658.3 MWth, which is 102% of the Rated Thermal Power (RTP) level of 3586.6 MWth, to account for measurement uncertainty.

b. Spent fuel source terms are based on reactor core source terms as previously discussed, with a conservative factor of 2.0 multiplier to account for the gap fractions of fuel exceeding 54 GWD/MTU burnup with a maximum linear heat generation rate exceeding the 6.3 kW/ft peak rod average power limit to address RG 1.183 footnote 11.

c. The damaged fuel is assumed to have operated at a radial peaking factor of 1.7. The damage is assumed to be the rupture of the cladding of all the fuel rods in the one assembly.

d. Movement of fuel will not occur less than 48 hours after the associated reactor shutdown. This establishes a basis for a new TS definition of "RECENTLY IRRADIATED FUEL."

e. A water depth above the damaged fuel of 23 feet is the limiting case, corresponding to the minimum depth of water coverage over the top of irradiated fuel assemblies seated in the spent fuel pool racks within the spent fuel pool.

f. No filtration of the airborne radioactivity released from the pool or automatic isolation of the accident location is assumed. Essentially all of the activity released to the environment is assumed to reach the containment refueling floor or spent fuel pool refueling floor airspace within two hours after the accident.

g. Delayed isolation of the CR via initiation of its emergency mode of operation filtered make-up (i.e., with an assumed charcoal filtration efficiency of 95%) at 30 minutes after the accident. During the first 30 minutes, the maximum unfiltered make-up is assumed at 6424 cfm, i.e., the upper 10% bound of the bounding make-up flow. After 30 minutes, the emergency mode of operation isolation is initiated with an assumed intake flow of 8575 cfm, i.e., the upper 10% bound of the bounding make-up flow which conservatively includes an assumed failure of an open damper on the unused make-up train with 2000 cfm of filtered flow.

h. An amount of unfiltered inleakage into the CR of 1000 cfm, added continuously throughout the accident duration.

An additional analysis was performed for RECENTLY IRRADIATED FUEL, i.e., fuel that has occupied part of a critical reactor core within the previous 48 hours.

Fuel Damage and Core Source Term - FHA

The assumed fuel assembly source term is based on the reactor core source terms previously described. The fraction of the core fuel damaged is based on the current UFSAR design basis of a postulated rupture of all the fuel rods in one assembly. With 193 fuel assemblies in the core operating at full core power of 3658.3 MWth (i.e., 102% of the RTP of 3586.6 MWth), the damaged fuel assembly is assumed to have been operating with a 1.7 peaking factor; therefore, the damaged fuel assembly power = 3658.3 MWth x 1.7/193 = 32.22 MWth.

This analysis assumes that movement of irradiated fuel will not occur less than 48 hours after the associated reactor shutdown, and therefore, a 48-hour delay period is used.

Pool Decontamination Factor (DF) and Fuel Fission Product Gap Inventory - FHA

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 3.7.14, "Spent Fuel Pool Water Level." Twenty-three feet of water is also assumed for an assembly drop in the core. TS 3.9.7, "Refueling Cavity Water Level," requires maintaining at least 23 feet of water above the top of the reactor vessel flange during movement of irradiated fuel assemblies within containment. Since the fuel assemblies are seated below the reactor vessel flange, in actuality, more than 23 feet of water covers the assemblies in the core; therefore, this assumption is conservative for an assembly drop in the core. This assumption is consistent with RG 1.183. As prescribed in Appendix C of RG 1.183, an overall DF of 200 is used as the overall effective iodine DF for this 23-foot water depth, with a DF of 1 for noble gases. Particulate radionuclides are assumed to be retained in the pool water (i.e., a DF of infinity).

RG 1.183, Table 3 allows application of the following gap activity fractions for non-LOCA events. These gap activities apply to fuel whose burnup and power are bounded by those specified in RG 1.183, footnote 11.

- 5% of the noble gases (excluding Kr-85)
- 10% of the Kr-85
- 5% of the iodine inventory (excluding I-131)
- 8% of the I-131
- 12% of the alkali metal inventory

Both Byron and Braidwood Stations may utilize some fuel assemblies with linear heat generation rates in excess of RG 1.183 footnote 11 limits. To account for high burnup assemblies, the above gap activity fractions were doubled. This methodology was previously accepted by the NRC in safety evaluations dated December 5, 2001, (Amendment 201) and March 26, 2002, (Amendment 204) approving the Fort Calhoun Station AST analyses and associated license amendment requests.

Because RADTRAD does not allow for application of isotope specific release fractions, the bounding isotopic core inventory data was used to create the "B/B AST Source Terms for FHA.nif" file based on the curies per megawatt thermal assuming 3586.6 MW reactor thermal power (RTP) for each of the RADTRAD isotopes except to multiply the Kr-85 and I-131 gap activity fractions by a factor of 2.0 and 1.6, respectively, as specified in the RG 1.183 fractions above. The standard RADTRAD PWR library values for Co-58 and Co-60 are used.

The factor of 2.0 applied to all gap activity fractions is combined with the applicable DFs discussed above to provide: 1) a 0.1 release multiplier for noble gases (i.e., 2 x 0.05 x 1.0); 2) a 0.0005 multiplier for iodine (i.e., 2 x 0.05 / 200); and 3) a 0 multiplier for alkali metals (i.e., particulates, retained in the pool water) in the AST FHA.rft file.

Release Model - FHA

Release modeling uses the refueling floor air space (i.e., in the FHB), with the initial air change rate based on the 525,460 cubic feet (86' x 130' x 47' high) volume "exposed to the monitor" as developed for the post-accident radiation monitor response time calculation, divided into the spent fuel storage pool total ventilation exhaust rate of 12,400 cfm. The initial air change rate is therefore 0.0236 per minute, assumed to last for the entire period until initiation of the CR emergency mode of operation. Consistent with RG 1.183, the release from the FHB to the environment is assumed over a two-hour time period. To assure this, the refueling floor exhaust rate is set artificially high at five times this value or 0.118 air changes per minute during the CR emergency mode of operation. The postulated exhaust point is the plant vent, with specific dispersion characteristics to the CR or offsite receiving locations which are defined by unique dispersion factors, or χ/Qs. The alternative release point through a major opening such as a FHB inner or outer trackway roll-up door would have lower χ/Qs and therefore lower calculated dose results.

For the potential accident location in the containment, the corresponding air change rate based on a containment volume of 2,850,000 cubic feet and containment purge ventilation exhaust rate of 40,000 cfm is 0.0140 air changes per minute, considerably lower than the FHB exhaust rates developed above. The purge exhaust point would again be the plant vent, with the same assumed χ/Qs for the CR or offsite receiving locations as for the potential accident location in the FHB. The FHB potential accident location would therefore be bounding. For the alternative release points through the major opening of the personnel/equipment hatch from the containment to the outside, lower χ/Qs apply which would result in lower calculated doses. For an alternative release point through the major opening of the personnel/equipment hatch from the containment to the auxiliary building, the release would be exhausted through the auxiliary building ventilation system to the plant vent, with the same CR χ/Qs as for the FHB potential accident location which would again be bounding.

4.6.2 Acceptance Criteria - FHA

Radiological doses resulting from a design basis FHA for a CR operator and a person located at the EAB or LPZ are to be less than the regulatory dose limits as given below.

Table 4.6.2-1
Regulatory Dose Limits - FHA

Dose Type	Control Room (rem)	EAB and LPZ (rem)
TEDE Dose	5[a]	6.3[b]

[a] 10 CFR 50.67
[b] 10 CFR 50.67 as modified by Regulatory Guide 1.183 (Table 6, Page 1.183-20)

4.6.3 Summary and Conclusions - FHA

Table 4.6.3-1 below provides the results from the limiting case of the unfiltered FHB FHA simulation modeled using the RADTRAD code and the AST assumptions and design input parameters described above.

Table 4.6.3-1

RADTRAD FHA Analysis Results

Fuel Handling Accident in Fuel Handling Building RADTRAD Dose Assessment Results		
Control Room (rem TEDE)	**EAB (rem TEDE)**	**LPZ (rem TEDE)**
4.55	4.24	0.356

Using the design basis assumptions, described methodology, and credited margin relaxation of this analysis, the limiting CR dose is 4.55 rem TEDE (i.e., assuming the dropped fuel assembly has been subcritical for ≥48 hours and the FHB Ventilation System is operating). This limiting dose is below the acceptance criteria, so it is verified that this design basis FHA is sufficiently mitigated at both Byron Station and Braidwood Station.

An additional FHA analysis was performed for RECENTLY IRRADIATED FUEL with containment closure established or with the FHB ventilation system operable. The results of this analysis also met the limits of 10 CFR 50.67 assuming a minimum decay time of six hours. The six-hour minimum decay time is inconsequential as it is physically impossible to remove the reactor head and move fuel within the first six hours after the reactor is subcritical.

4.7 Control Rod Ejection Accident (CREA)

Input parameters used for the CREA analysis are given in Attachment 6, Tables 1.0, 2.0 and 5.0. Conformance with RG 1.183 guidance addressing CREA analysis is provided in Attachment 7, Tables A and D.

4.7.1 Inputs / Assumptions - CREA

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

a. Core inventory was based on a DBA power level of 3658.3 MWth, which is 102% of the RTP level of 3586.6 MWth, to account for measurement uncertainty.

b. 10% of the fuel is damaged during the initiation of this accident, and is assumed to have failed.

c. The damaged fuel is assumed to have operated at a radial peaking factor of 1.7.

d. 10% of the core inventory of noble gases and iodines are released from the fuel gap (RG 1.183, Appendix H). Release fractions of other nuclide groups contained in the fuel gap are detailed in Table 3 of RG 1.183, and to account for gap fraction uncertainty. Due to fuel burnup, fractions from the referenced table are doubled.

e. 2.5% of the damaged fuel rods will experience melting during the CREA.

f. 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 RG 1.183. Fractions of other nuclides released from the melted fuel are used from Table 2 of RG 1.183. Though these are described as LOCA values for fuel melt release, they are conservatively used for the other nuclide groups.

g. 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.

h. All iodine released from the SGs is assumed to be of the elemental species. This is done for RADTRAD simulation considerations, and is consistent with the RG 1.183 specification of 97% elemental and 3% organic, because elemental and organic iodine are identically treated by the computer model.

i. The CR ventilation system at Byron Station and Braidwood Station is assumed to be placed in the emergency mode of operation 30 minutes after the initiation of this design basis accident.

j. For the containment leakage case, all leakage is assumed to be at an increased L_a of 0.2% per day for the first 24 hours and 0.1% per day thereafter.

Radioactive Releases - CREA

Two cases are considered when analyzing the radioactive release due to a CREA.

Case 1: Containment Leakage - CREA

For Case 1, 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 a containment leak rate limit, or L_a, conservatively increased from 0.1% per day to 0.2% per day for this accident analysis. As is historically the case at Byron and Braidwood Stations, and in accordance with RG 1.183 guidance, the leak rate is reduced by 50% after 24 hours, based on the containment pressure decreasing over time.

Case 2: Steam Generator PORV Release - CREA

For Case 2, no breach of the RPV is assumed following the rod ejection. In this case, reactor coolant system (RCS) integrity is maintained and all activity from damaged fuel that has been mixed with the RCS leaks to the secondary side through the steam generator (SG) tubes at a conservative rate of 1.0 gpm total leakage. From here, activity is available for release to the environment by steaming of the SG power operated relief valves (PORVs). An average rate of release is assumed. In addition to the activity released from the primary to secondary coolant, iodine activity in the secondary coolant at the TS limit (i.e., 0.1 μCi/gm Dose Equivalent (DE) I-131) is also assumed to be released.

Fuel Damage and Core Source Term - CREA

For conservatism, the CREA core source terms are those associated with a DBA power level of 3658.3 MWth, which includes an additional 2% power over that of the full licensed power to account for uncertainty.

The sudden rod ejection and localized temperature spike associated with the CREA results in the damage of 10% of the core. Only 2.5% of the damaged core releases melted fuel activity (i.e., 0.250% 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.

The damaged fuel is assumed to have operated at a radial peaking factor of 1.7.

Activity Release Fractions - CREA

Release fractions and transport fractions conform to RG 1.183, Appendix H and Table 3. To conform with this regulatory guidance, 10% of the core inventory of iodine and noble 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. However, to account for fuel burnup in excess of the referenced assumptions, the cesium and rubidium release fraction is doubled. Although analyses have shown that isotopic activity fractions in the fuel-clad gap may in fact decrease when "burning" the fuel longer than the 54 GWd/MTU specified in Regulatory Guide 1.183, this 100% increase in the gap fractions is used as an accepted and conservative means of bounding all extended burnup phenomena. With regard to the fraction released from melted fuel, it is assumed that 90% of the core inventory of iodine and noble gas, and 76% 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). Again, in conformance with RG 1.183, it would be assumed that 100% of the noble gases, 25% of the iodines, and 50% of the cesium and rubidium (i.e., considered particulate/aerosol nuclides) released from the melted fuel would be available for release from containment. However, for this analysis the assumption of 25% of the iodines being available for release was increased to 50%. This was done to prevent a "double counting" of the iodine removal due to plate-out in containment, because this analysis credits Powers' Natural Deposition model for plateout, as opposed to the historically assumed 50% plateout.

These activity release fractions are input to the RADTRAD code through the use of the Release Fractions and Timing (RFT) file.

Airborne Activity Removal Mechanisms in Containment - CREA

As discussed below, natural deposition and decay are credited in the analysis.

Natural Deposition

For Byron Station and Braidwood Station, the RADTRAD computer program, including the Powers' Natural Deposition algorithm based on NUREG/CR-6189, is used for modeling aerosol deposition in containment. No natural deposition is assumed for elemental or organic iodine. The RADTRAD lower bound level (i.e., 10 percent) of deposition credit is used.

Decay Credited

Decay of radioactivity is credited in all compartments, prior to release. This is implemented in RADTRAD using the half-lives in the Nuclide Inventory File (NIF). The RADTRAD decay plus daughter option is used. In reality, daughter products such as xenon from iodines or iodines from tellurium are unlikely to readily escape from the matrix in which the parent iodine or tellurium is contained. Nevertheless, the RADTRAD feature to include daughter effects is selected for conservatism.

Steaming Release Rates and Partition Factors - CREA

Activity that originates in the RCS is released to the secondary coolant by means of the primary-to-secondary coolant leak rate. This assumed leak rate value is a total of 1.0 gpm. For input into RADTRAD this rate is converted from gallons per minute to cubic feet per minute, making it equal to 0.1337 cfm.

For Case 2, the release to the environment is associated with the secondary coolant steaming from the SGs. Because of this release dynamic, RG 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, the partition factor of 0.01 was taken directly from the suggested guidance. However, there is no explicit guidance with regard to other particulate nuclides. 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 cesiums, rubidium, and noble gases. For cesiums and rubidium, a partition factor of 0.0055 is used which bounds the value of 0.00529 shown in ANSI Standard, ANS/ANSI 18.1 - 1999, (Reference 7.14) for Cs-134 which has the largest partition factor of these isotopes. 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 CREA event of Case 2 assumes an average cumulative release rate through the SG PORVs that, for simplicity and conservatism, is reduced in steps. The partition factors discussed above are applied to these release rates, which were derived from the total time increment mass releases. The following table shows the time steps, isotopic partition factors, and associated release rates; conversion of this data from mass to volumetric flow rates was performed based on cooled liquid conditions (i.e., 62.4 lbm/ft^3), as specified by RG 1.183.

Table 4.7.1-1
Partition Factors and Release Rates

Time Interval (hrs)	Total Steam Mass Release (lbm)	Iodine Partition Factor	Cesium Partition Factor	Noble Gas Partition Factor	Steam Release Rate for Iodines (cfm)	Steam Release Rate for Cesiums (cfm)	Steam Release Rate for Noble Gases (cfm)
0 - 0.0556	600,000	0.01	0.0055	1	2.8833E+01	1.5858E+01	2.8833E+03
0.0556 - 1.1111	1,900,000	0.01	0.0055	1	4.8055E+00	2.6430E+00	4.8055E+02
1.1111 - 720	0	0.01	0.0055	1	0	0	0

χ/Q Calculations (Meteorology) - CREA

All releases from the SG PORV/safety valves are considered elevated releases due to the high steaming rates, therefore, in accordance with regulatory guidance, the associated χ/Qs have been reduced by a factor of five.

<u>Justification for Application of Elevated Release Credit</u>

Guidance in RG 1.194 indicates that a factor of five reduction in the ground level χ/Q value calculated with ARCON96 is justified for energetic releases from steam relief valves or atmospheric dump valves, if (1) the release point is uncapped and vertically oriented, and (2) the time-dependent vertical velocity exceeds the 95th-percentile wind speed, at the release point height, by a factor of five.

Byron Station and Braidwood Station meet these criteria as the SG PORVs are uncapped and vertically oriented. The 95th-percentile wind speeds for Byron Station are 7.61 meters per second and 6.89 meters per second at Braidwood Station for the 30 foot tower level. Based on engineering judgment, considering choked sonic flow conditions for saturated steam releases, these steam vertical velocities clearly exceed the 95th-percentile wind speed, at the release point height, by a factor of five.

Therefore, it is concluded that the factor of five reduction is acceptable at Byron Station and Braidwood Station for all steam release accidents involving PORV steam release from the SGs.

4.7.2 Acceptance Criteria - CREA

Radiological doses resulting from a design basis CREA for a CR operator and a person located at the EAB or LPZ are to be less than the regulatory dose limits as given below.

Table 4.7.2-1
Regulatory Dose Limits - CREA

Dose Type	Control Room (rem)	EAB and LPZ (rem)
TEDE Dose	5[a]	6.3[b]

[a] 10 CFR 50.67

[b] 10 CFR 50.67 as modified by Regulatory Guide 1.183

4.7.3 Summary and Conclusions - CREA

Table 4.7.3-1 below provides the results from the simulations modeled using the RADTRAD code.

Table 4.7.3-1

RADTRAD Analysis Results - CREA

Case 1: Containment Leakage CREA RADTRAD Dose Assessment Results		
Control Room (rem TEDE)	EAB (rem TEDE)	LPZ (rem TEDE)
2.549	4.647	1.983
Case 2: Steam Generator PORV Release CREA RADTRAD Dose Assessment Results		
Control Room (rem TEDE)	EAB (rem TEDE)	LPZ (rem TEDE)
0.369	1.480	0.257

For the cases analyzed in this calculation, it is shown that a Case 1 CREA that breaches the RPV, and causes a containment release, would be the bounding CREA scenario. Using the design basis assumptions, described methodology, and credited margin relaxation of this analysis, the limiting CR dose is 2.549 rem TEDE. This limiting dose, and all other doses, are well below their respective acceptance criteria, so it is verified that this design basis Control Rod Ejection Accident is sufficiently mitigated at both Byron and Braidwood Stations.

4.8 Locked Rotor Accident (LRA)

Input parameters used for the LRA analysis are given in Attachment 6, Tables 1.0, 2.0 and 6.0. Conformance with RG 1.183 guidance addressing LRA analysis is provided in Attachment 7, Tables A and E.

4.8.1 Inputs/Assumptions - LRA

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

a. Core inventory is based on a DBA power level of 3658.3 MWth, which is 102% of the RTP level of 3586.6 MWth, to account for measurement uncertainty.

b. Two percent (2%) of the fuel is damaged during the initiation of this accident, and is assumed to have failed.

c. The damaged fuel is assumed to have operated at a radial peaking factor of 1.7.

d. No fuel melts following the postulated LRA.

e. Five percent of the core inventory of noble gases and iodines are released from the fuel gap, excluding I-131 and Kr-85, where 8% and 10% are respectively released. Release fractions of other nuclide groups contained in the fuel gap are detailed in Table 3 of RG 1.183, and to account for gap fraction uncertainty due to expected fuel burnup, these fractions from the referenced table are doubled for conservatism.

f. All iodine released from the SGs is assumed to be of the elemental species. This is done for RADTRAD simulation considerations, and is consistent with the RG 1.183 specification of 97% elemental and 3% organic, because elemental and organic iodine are treated identically by the computer model.

g. The CR ventilation system at Byron Station and Braidwood Station is assumed to be placed in the emergency mode of operation 30 minutes after the initiation of this design basis accident.

h. 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 per RG 1.183.

i. SG PORV releases end at 40 hours, when the RCS has seen a large enough reduction in residual heat to no longer require steaming via the PORVs for temperature reduction.

j. A failure of one SG PORV, in the open position, that takes 20 minutes to isolate, is assumed to conservatively maximize activity release.

Analysis General Description - LRA

This 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 SG PORVs are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism, their pressure reducing effect is ignored and an SG PORV failure, in the open position at the onset of the accident release, is assumed. In addition, the pressure reducing effect of the spray is also ignored for this 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 and leaking SG tubes. 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. Additionally, prior to accident initiation, the reactor is postulated to experience an iodine spike, thereby increasing the RCS iodine activity above that of equilibrium levels.

It is conservatively assumed that, as a result of the postulated LRA, 2% of the fuel rods in the core undergo sufficient clad damage to result in the release of their gap activity. The damaged fuel is assumed to have operated at a radial peaking factor of 1.7.

As a result of this accident, radionuclides carried by the primary coolant to the SGs, via the leaking tubes, are released to the environment via the main steam safety valves or SG PORVs.

This LRA dose assessment is analyzed using two modeled simulations. The first simulation, Case 1, 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 dose equivalent (DE) I-131 spike. Leakage and steaming rates are used to model the transport of activity from the RCS to the environment, through the PORVs of intact SGs, and through the failed PORV of the faulted SG, which is the postulated single-failure for this analysis.

The second simulation, i.e., Case 2, is modeled to calculate the doses due to 0.1 μCi/gm DE I-131 equilibrium activity existing in the secondary coolant prior to accident initiation. This iodine activity is released using the same partitioned steaming rates that are associated with Case 1 SG PORV release to the environment. For the intact SGs, this iodine activity is released at the same partitioned steaming rates that are associated with Case 1. However, for the SG with the failed PORV, referred to as the faulted SG, it is postulated that the activity initially contained in the faulted SG is released to the environment for 20 minutes. The failed SG PORV is isolated by locally closing the associated isolation valve. The operator can identify the failed SG PORV when the faulted SG pressure drops below SG PORV reset value. There is also a positive valve position indication of the open SG PORV on the control board. As a result of the failed SG PORV the operator would enter the appropriate emergency procedure. The dispatch, travel time, and local isolation of this valve by an operator has conservatively been assumed to require 20 minutes, from the time the SG PORV fails open.

Fuel Damage and Core Source Term - LRA

For conservatism, the LRA core source terms are those associated with a DBA power level of 3658.3 MWth, which includes an additional 2% power over that of the full licensed power to account for 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 2% fuel damage. The damaged fuel is assumed to have operated at a radial peaking factor of 1.7. The design basis of this accident assumes that no fuel melt is postulated to occur. Therefore, for Case 1, 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.

The additional source activity modeled in the second case consists simply of the 0.1 µCi/gm DE I-131 equilibrium secondary coolant activity concentration, consistent with the Byron Station and Braidwood Station TS requirements. The total activity available for release from both the intact SGs and the SG with a failed SG PORV are input to the RADTRAD NIF file for Case 2.

Activity Release Fractions - LRA

Release fractions and transport fractions are consistent with RG 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-131and Kr-85, where 8% and 10% are assumed, respectively. Additionally, Table 3 of RG 1.183 shows that 12% of the core cesium and rubidium should be assumed to be in the fuel-clad gap. However, to accommodate the consideration of fuel burnup, in excess of the RG 1.183 assumptions, all RG 1.183 Table 3 "Other Noble Gases", "Other Halogens", and "Alkali Metals" isotopic release fractions are doubled. Although analyses have shown that isotopic activity fractions in the fuel-clad gap may in fact decrease when "burning" the fuel longer than the 54 GWd/MTU specified in RG 1.183, this 100% increase in the gap fractions is used as an accepted and conservative means of bounding all burnup phenomena. All of the gap activity in the damaged fuel is released in its entirety, instantaneously into the RCS and homogeneously mixed.

These activity release fractions are input to the RADTRAD code through the use of the Release Fractions and Timing (RFT) file.

Airborne Activity Removal Mechanisms in Containment - LRA

As discussed below only decay and leakage are credited.

Decay Credited

Decay of radioactivity is credited in all compartments, prior to release. This is implemented in RADTRAD using the half-lives in the Nuclide Inventory File (NIF). The RADTRAD decay plus daughter option is used. In reality, daughter products such as xenon from iodines or iodines from tellurium are unlikely to readily escape from the matrix in which the parent iodine or tellurium is contained. Nevertheless, the RADTRAD feature to include daughter effects is selected for conservatism.

Depletion due to Leakage Credited

For analyses of doses due to release from the RCS volume, the dose results from leakage, and it is reasonable to credit the small amount of depletion from the available RCS activity inventory associated with this leakage. This is calculated inherently by the RADTRAD code.

Release Rates, Steaming Rates, and Partition Factors - LRA

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 0.218 gpm, per intact SG, totaling 0.654 gpm, and 0.5 gpm for the SG with the failed SG PORV. For input into RADTRAD, these rates were converted from gallons per minute to cubic feet per minute, making them 0.02914 cfm, per intact SG, totaling 0.08743 cfm, and 0.06684 cfm for the SG with the failed SG PORV.

Releases to the environment are associated with the secondary coolant steaming from the SGs with intact SG PORVs and releases directly out of a faulted SG. Because of the release dynamic of the activity from the intact SG PORVs, RG 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 0.01 was taken directly from the suggested guidance. However, there is no explicit guidance with regard to other particulate nuclides. 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 cesiums, rubidium, and noble gases. For cesiums and rubidium, a partition factor of 0.0055 is used which bounds the value of 0.00529 shown in ANSI Standard, ANS/ANSI 18.1 - 1999, (Reference 7.14) for Cs-134 which has the largest partition factor of these isotopes. 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 is applicable to both the Case 1 and Case 2 simulations. This methodology assumes an average cumulative release rate through the SG PORVs that, for simplicity and conservatism, is reduced in steps. The partition factors are applied to these release rates, which were derived from the total time increment mass releases. Incremental steam mass releases are in pounds mass. Release rates were derived by dividing these totals by the time increment. Then, this data was converted using the assumption of cooled liquid conditions (i.e., 62.4 lbm/ft^3), as specified by the guidance of RG 1.183. The steaming release and primary-to-secondary coolant leakage is postulated to end at 40 hours, when the RCS and secondary loop have reached equilibrium. Table 4.8.1-1 below shows the time steps, isotopic partition factors, and associated release rates.

Table 4.8.1-1
Partition Factors, and Associated Release Rates

Time Interval (hrs)	Total Steam Mass Release (lbm)	Iodine Partition Factor	Cesium Partition Factor	Noble Gas Partition Factor	Steam Release Rate for Iodines (cfm)	Steam Release Rate for Cesiums (cfm)	Steam Release Rate for Noble Gases (cfm)
0 - 2.0	719,000	0.01	0.0055	1.0	9.5977E-01	5.2788E-01	9.5977E+01
2.0 - 8.0	1,109,000	0.01	0.0055	1.0	4.9346E-01	2.7140E-01	4.9346E+01
8.0 - 40	2,664,000	0.01	0.0055	1.0	2.2226E-01	1.2224E-01	2.2226E+01
40 - 64	0	0.01	0.0055	1.0	0	0	0

The release rate through the failed SG PORV is conservatively assumed to be un-partitioned, and therefore no isotopic partition factors are applied. The rate at which activity is released from this pathway is therefore equal to the primary-to-secondary coolant leak rate discussed above.

In Case 2, for the intact SGs, the iodine activity is released using the same partitioned steaming rates that are associated with Case 1 SG PORV release to the environment. For the faulted SG the release rate is based on a 20-minute failed SG PORV release. The total mass released is 167,000 lbm. Converted to a volume, and divided by the 20-minute release time, this becomes 133.75 cfm volumetric flow rate; entered into RADTRAD.

χ/Q Calculations (Meteorology) - LRA

All releases from the SG PORV/Safety Valves are considered elevated releases due to the high steaming rates, therefore in accordance with regulatory guidance, the associated χ/Qs have been reduced by a factor of five.

<u>Justification for Application of Elevated Release Credit</u>

Justification for application of elevated release criteria for the LRA is identical to that previously presented for the CREA. Based on this discussion, it is concluded that the factor of five reduction is acceptable at Byron Station and Braidwood Station for all steam release accidents involving PORV steam release from the SGs.

4.8.2 Acceptance Criteria - LRA

Radiological doses resulting from a design basis LRA for a CR operator and a person located at the EAB or LPZ are to be less than the regulatory dose limits as given below.

Table 4.8.2-1
Regulatory Dose Limits - LRA

Dose Type	Control Room (rem)	EAB and LPZ (rem)
TEDE Dose	5[a]	2.5[b]

[a] 10 CFR 50.67
[b] 10 CFR 50.67 as modified by Regulatory Guide 1.183 (Table 6, Page 1.183-20)

4.8.3 Summary and Conclusions - LRA

Table 4.8.3-1 below provides the results from the simulations modeled using the RADTRAD code, as well as the summed result.

Table 4.8.3-1
RADTRAD Analysis Results – LRA

Case 1: Doses from Iodine Spike and Fuel Damage RCS Activity RADTRAD Dose Assessment Results		
Control Room (rem TEDE)	EAB (rem TEDE)	LPZ (rem TEDE)
2.466	1.421	0.518
Case 2: Doses from Equilibrium Secondary Coolant Activity RADTRAD Dose Assessment Results		
Control Room (rem TEDE)	EAB (rem TEDE)	LPZ (rem TEDE)
0.063	0.035	0.006
Total Dose from Design Basis Locked Rotor Accident (LRA)		
Control Room (rem TEDE)	EAB (rem TEDE)	LPZ (rem TEDE)
2.529	1.456	0.525

It is shown that for this design basis LRA, assuming a worst-case single failure of one SG PORV, using the design basis assumptions, described methodology, and credited margin relaxation of this analysis, the assessed CR dose is 2.529 rem TEDE. These calculated doses above are well below their respective acceptance criteria, so it is verified that the LRA is sufficiently mitigated at both Byron and Braidwood Stations.

4.9 Main Steam Line Break Accident – MSLB

Input parameters used for the MSLB analysis are given in Attachment 6, Tables 1.0, 2.0 and 7.0. Conformance with RG 1.183 guidance addressing MSLB analysis is provided in Attachment 7, Tables A and F.

4.9.1 Inputs/Assumptions - MSLB

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

a. Core inventory is based on a DBA power level of 3658.3 MWth, which is 102% of the RTP level of 3586.6 MWth, to account for measurement uncertainty.

b. There is no fuel damage as a result of the postulated MSLB accident at Byron and Braidwood Stations.

c. In the case of a postulated iodine activity release rate spike, the spike release is assumed to occur for a period of six hours, when the activity available for release from the fuel has been conservatively depleted.

d. The activity released from the fuel is assumed to be instantaneously mixed with the RCS.

e. All iodine released from the SGs is assumed to be of the elemental species. This is done for RADTRAD simulation considerations, and is consistent with the RG 1.183 specification of 97% elemental and 3% organic, because elemental and organic iodine are treated identically by the computer model.

f. The CR ventilation system at Byron Station and Braidwood Station is assumed to be placed in the emergency mode of operation 30 minutes after the initiation of this design basis accident.

g. The faulted SG is assumed to be in a "dry-out" condition, and does not inhibit activity release from the RCS through that coolant loop.

h. It is conservatively assumed that blowdown of the faulted SG 167,000 lbm fluid takes two minutes to complete.

General Description - MSLB

The 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 RG 1.183, two reactor transients that maximize the radioactivity available for release were modeled.

Case 1: Dose Due to Pre-accident Iodine Spike - MSLB

Case 1 is identified as a reactor pre-accident, transient induced, iodine spike, which raises the primary coolant iodine concentration to the maximum 60 µCi/gm assumed DE I-131 value permitted by the Byron Station and Braidwood Station TS at full power operations, prior to the initiation of the accident. Therefore this case is termed the pre-accident iodine spike case. 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 (i.e., assuming a two minute duration), and the SG PORV release of secondary coolant through the intact SGs (i.e., for a 40 hour duration). This secondary coolant activity, which existed prior to the MSLB accident, is analyzed and the resultant dose is added to the Case 1 results.

Case 2: Dose Due to Accident Initiated Concurrent Iodine Spike - MSLB

Case 2 assumes that the postulated MSLB 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 500 times greater than the release rate corresponding to the 1.0 µCi/gm DE I-131 RCS equilibrium iodine concentration. This 500 times activity release rate spike is assumed to occur for a duration of six hours, as this period has been shown to conservatively deplete the available gap activity in the assumed operating fuel defect. Also, this assumption has been used as the design basis for the pre-AST analysis for this accident at Byron and Braidwood Stations. In RADTRAD, a Nuclide Inventory File (NIF) is designed to input the total isotopic iodine activity that is associated with six hours of activity release at the 500 times rate specified. Then, this NIF is used in conjunction with a modified Release Fraction and Timing (RFT) file, which defines the complete release of this activity over a six hour period.

As described in Case 1 above, the dose due to the equilibrium secondary coolant system iodine activity (0.1 µCi/gm DE I-131) must also be determined and added to this case (i.e., Case 2).

Fuel Damage and Core Source Term - MSLB

The pre-AST design basis at the Byron and Braidwood Stations assumes no fuel damage for the postulated MSLB event. For this MSLB accident, the source terms are defined by the TS activity release rates from a maximum fuel defect assumed during operation, which are characterized by the equilibrium 1.0 µCi/gm DE I-131 activity concentration in the RCS. The noble gas inventory in the RCS is based on operation with a conservative worst-case 1% core fuel defects. Because no fuel damage is assumed due to this accident, only iodine and noble gas isotopes are modeled to contribute to dose. To identify the worst-case MSLB accident, however, two different cases of iodine spiking are analyzed, in accordance with regulatory guidance as previously described.

Activity Removal Mechanisms in Containment - MSLB

The design basis MSLB at the Byron and Braidwood Stations releases activity directly into the RCS, therefore no plateout or other activity deposition is credited.

Decay Credited - MSLB

Decay of radioactivity is credited in all compartments, prior to release. This is implemented in RADTRAD using the half-lives in the NIFs. The RADTRAD decay option is used.

Release Rates, Steaming Rates, and Partition Factors - MSLB

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 0.218 gpm, per intact SG, totaling 0.654 gpm, and 0.5 gpm for the faulted SG with the broken steam line. For input into RADTRAD these rates were converted from gallons per minute to cubic feet per minute, making them 0.02914 cfm, per intact SG, totaling 0.08743 cfm, and 0.06684 cfm for the faulted SG.

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, RG 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, the partition factor of 0.01 was taken directly from the suggested guidance of RG 1.183. 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 that, for simplicity and conservatism, is reduced in steps. The partition factors are applied to these release rates, which were derived from the total time increment mass releases. Incremental steam mass releases are in pounds mass. Release rates were derived by dividing these totals by the time increment. 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 RG 1.183. The steaming release and primary-to-secondary coolant leakage is postulated to end at 40 hours, when the RCS and secondary loop have reached equilibrium. Table 4.9.1-1 below shows the time steps, isotopic partition factors, and associated release rates.

Table 4.9.1-1
Partition Factors, And Associated Release Rates

Time Interval (hrs)	Total Steam Mass Release (lbm)	Iodine Partition Factor	Noble Gas Partition Factor	Steam Release Rate for Iodines (cfm)	Steam Release Rate for Noble Gases (cfm)
0 - 2.0	442,000	0.01	1	5.9001E-01	5.9001E+01
2.0 - 8.0	977,000	0.01	1	4.3472E-01	4.3472E+01
8.0 - 40	2,216,000	0.01	1	1.8488E-01	1.8488E+01

For the loop with the broken steam line, i.e., the faulted SG, it is postulated that the entire release of the secondary coolant of that loop will take two minutes. Therefore, for input into RADTRAD the faulted SG coolant volume of 2675 ft^3 is divided by two minutes to arrive at a design basis value of 1337.5 cfm.

χ/Q Calculations (Meteorology) - MSLB

All releases from the SG PORV/Safety Valves (i.e., from the intact SG) are considered elevated releases due to the high steaming rates, therefore in accordance with regulatory guidance, the associated χ/Qs have been reduced by a factor of five. The χ/Q values for the faulted SG are based on ground level releases.

Justification for Application of Elevated Release Credit

Justification for application of elevated release criteria for the MSLB is identical to that previously presented for the CREA. Based on this discussion, it is concluded that the factor of five reduction is acceptable at Byron Station and Braidwood Station for all steam release accidents involving PORV steam release from the SGs.

Source Term and Dose Calculations – MSLB

Source Term Calculation – MSLB

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 RCS over a six hour period for Case 2.

Dose Calculations – MSLB

The RADTRAD computer code is used to determine the Byron and Braidwood Stations, Units 1 and 2 MSLB accident doses at the EAB, LPZ, and CR, consistent with RG 1.183. For each case, (i.e., Case 1 and Case 2 discussed above), two models were designed for two different release paths, i.e., the intact SG PORVs and the broken steam line.

4.9.2 Acceptance Criteria - MSLB

Radiological doses resulting from a design basis MSLB for a CR operator and a person located at the EAB or LPZ are to be less than the regulatory dose limits as given below.

Table 4.9.2-1
Regulatory Dose Limits - MSLB

	Dose Type	Control Room (rem)	EAB and LPZ (rem)
Case 1	TEDE Dose	5[a]	25[a]
Case 2	TEDE Dose	5[a]	2.5[b]

[a] 10 CFR 50.67
[b] 10 CFR 50.67 as modified by Regulatory Guide 1.183 (Table 6, Page 1.186-20)

4.9.3 Summary and Conclusions - MSLB

Table 4.9.3-1 below provides the results from the Case 1 and Case 2 simulations that were modeled using the RADTRAD code. The total dose for the two cases includes the dose result from the simulations regarding equilibrium secondary coolant activity.

Table 4.9.3-1
RADTRAD Analysis Results - MSLB

Case 1: Pre-Accident 60 µCi/gm DE I-131 Spike RADTRAD Dose Assessment Results		
Control Room (rem TEDE)	**EAB (rem TEDE)**	**LPZ (rem TEDE)**
0.790	0.127	0.073
Case 2: Accident Initiated 500 times Equilibrium Iodine Release Rate Spike RADTRAD Dose Assessment Results		
Control Room (rem TEDE)	**EAB (rem TEDE)**	**LPZ (rem TEDE)**
4.255	0.175	0.406

For the cases analyzed in this calculation, it is shown that a Case 2 MSLB that involves an accident initiated iodine spike, which takes place concurrent with the MSLB releases, would be the bounding MSLB scenario. Using the design basis assumptions, described methodology, and credited margin relaxation of this analysis, the limiting CR dose is 4.255 rem TEDE. This limiting dose, and all other doses, are well below their respective acceptance criteria, so it is verified that this design basis MSLB accident is sufficiently mitigated at both Byron and Braidwood Stations.

4.10 Steam Generator Tube Rupture - SGTR

Input parameters used for the SGTR analysis are given in Attachment 6, Tables 1.0, 2.0 and 8.0. Conformance with RG 1.183 guidance addressing LOCA analysis is provided in Attachment 7, Tables A and G.

4.10.1 Inputs/Assumptions - SGTR

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

a. Core inventory was based on a DBA power level of 3658.3 MWth, which is 102% of the RTP level of 3586.6 MWth, to account for measurement uncertainty.

b. There is no fuel damage as a result of the postulated SG tube rupture accident at Byron and Braidwood Stations.

c. In the case of a postulated iodine activity release rate spike, the spike release is assumed to occur for a period of eight hours, when the activity available for release from the fuel has been conservatively depleted.

d. The activity released from the fuel is assumed to be instantaneously mixed with the RCS.

e. All iodine released from the SGs is assumed to be of the elemental species. This is done for RADTRAD simulation considerations, and is consistent with the RG 1.183 specification of 97% elemental and 3% organic, since elemental and organic iodine are treated identically by the computer model.

f. The CR ventilation system at Byron and Braidwood Stations is assumed to be placed in the emergency mode of operation 30 minutes after the initiation of this design basis accident.

g. The ruptured SG steam release of 9.42E4 lbm was modeled from the time the faulted SG PORV opens (i.e., 465 seconds), which is 20 seconds after reactor trip, until the ruptured SG PORV is isolated (i.e., 1700 seconds).

h. The intact SG steam release of 5.44E5 lbm was modeled from time of RCS cooldown initiation (i.e., 2800 seconds) until the RCS cooldown is terminated and RCS depressurization is initiated (i.e., 3500 seconds).

i. In addition to SG PORV steam releases, the full power steam activity released through the condenser (i.e., 16.04E6 lb/hr) until the time of reactor trip and loss of offsite power was evaluated.

Analysis Description - SGTR

The SGTR accident is postulated as a complete severance of a single SG tube. This is a conservative assumption because tube material is Inconel, 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 Byron Station and Braidwood Station containments. 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.

SGTR accident mitigation can be described in recovery phases. The major phases that this analysis used to model the dose consequences from this event are shown below, along with the time increments that are associated with the sequence of events:

- Start of event until reactor trip; 0 - 445 seconds
- Reactor trip until ruptured SG PORV is opened; 445 - 465 seconds
- Ruptured SG PORV opening until ruptured SG PORV is isolated; 465 - 1700 seconds
- Ruptured SG PORV isolation until RCS cooldown is initiated; 1700 - 2800 seconds
- RCS cooldown initiation until break flow flashing ceases; 2800 - 3000 seconds
- End of break flow flashing until RCS depressurization initiation; 3000 - 3500 seconds
- RCS depressurization initiation until all break flow ceases; 3500 - 3590 seconds

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

Case 1: Dose Due to Pre-accident Iodine Spike - SGTR

Case 1 is identified as a reactor pre-accident, transient induced, iodine spike, which raises the primary coolant iodine concentration to the maximum 60 μCi/gm assumed DE I-131 value at full power operations, prior to the initiation of the accident. Therefore this case is termed the pre-accident iodine spike case. 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 faulted SG is as follows.

Isotope	Intact SGs (Ci)	Ruptured SG (Ci)
I-131	7.10	2.37
I-132	9.37	3.12
I-133	12.92	4.30
I-134	2.33	0.77
I-135	8.06	2.68

These releases of secondary coolant activity, which existed prior to the SGTR accident, are analyzed, and the resultant dose is added to the Case 1 results.

Case 2: Dose Due to Accident Initiated Concurrent Iodine Spike - SGTR

Case 2 assumes that the postulated 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. This 335 times activity release rate spike is assumed to occur for a duration of eight hours, as this period has been shown to conservatively deplete the available gap activity in the assumed operating damaged fuel fraction. Also, this assumption has been used as the existing design basis for this accident at Byron and Braidwood Stations. In RADTRAD, a Nuclide Inventory File (NIF) is designed to input the total isotopic iodine activity that is associated with eight hours of activity release to the RCS at the 335 times rate specified. Then, this NIF is used in conjunction with a modified Release Fraction and Timing (RFT) file, which defines the complete release of this activity over an eight hour period.

As described in Case 1 above, the dose due to the equilibrium secondary coolant system iodine activity must also be determined and added to this case (i.e., Case 2).

Fuel Damage and Core Source Term - SGTR

The pre-AST design basis at the Byron and Braidwood Stations assumes no fuel damage for the postulated SGTR event. For this SGTR accident, the source terms are defined by the TS activity release rates from a maximum fuel defect assumed during operation, which are characterized by the equilibrium 1.0 µCi/gm DE I-131 activity concentration in the RCS. The noble gas inventory in the RCS is based on operation with a conservative worst-case 1% core fuel defects. Because no fuel damage is assumed for this accident, only iodine and noble gas isotopes are modeled to contribute to dose. To identify the worst-case SGTR accident, however, two different cases of iodine spiking are analyzed, per regulatory guidance as previously described.

Activity Removal Mechanisms in Containment - SGTR

The design basis SGTR at Byron Station and Braidwood Station releases activity directly into the RCS, therefore no plateout, or other activity deposition, is credited.

<u>Decay Credited</u>

Decay of radioactivity is credited in all compartments, prior to release. This is implemented in RADTRAD using the half-lives in the NIFs. The RADTRAD decay option is used.

Release Rates and Partition Factors - SGTR

As discussed above, a number of modes of release are indicative of this particular accident scenario. Therefore, the varying releases associated with the timing and sequence of events of this accident have been derived.

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 0.218 gpm, per intact SG, totaling 0.654 gpm. For input into RADTRAD this rate was converted from gallons per minute to cubic feet per minute, making it 0.02914 cfm, per intact SG, totaling 0.08743 cfm.

The methodology used to model steaming of activity through SG 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. Incremental steam mass release rates are given in pounds per second. For the time increments used in this accident scenario, release rates were derived by taking the averages of these rates over each specified time increment. Then these mass flow rates were converted to volumetric flow rates using the assumption of cooled liquid conditions (i.e., 62.4 lbm/ft^3), as specified by the applicable guidance of RG 1.183.

The faulted SG sees two simultaneous release mechanisms. Primary-to-secondary coolant leakage through the ruptured tube of the faulted SG that flashes conservatively goes directly to the environment, without mixing with any secondary coolant. Therefore, with this release mechanism, no partitioning of iodine is expected to occur in this release. However, leakage that does mix with the volume of coolant in the ruptured SG is released by flashing to the environment, and the applicable partition factor is applied, as discussed below.

For all post-accident releases through the SG PORVs, the mechanism for release to the environment is steaming of the coolant in the secondary system. Because of this release dynamic, RG 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, the partition factor of 0.01 was taken directly from the suggested guidance of RG 1.183. Reviewing the specified AST release fractions, it is concluded that the only nuclides to be released from the core source term, other than iodines, are noble gas nuclides. Because of their volatility, 100% of the noble gases are assumed to be released.

In addition to the steam released through the SG PORVs, the steam release through the condenser until the time of reactor trip and loss of offsite power must also be evaluated. The total full power steam flow is 16.04E6 lb/hr; therefore, the total full power steam flow in the three intact SGs is 12.03E6 lb/hr. For steam flow that is released through the condenser from the three intact SGs, an additional 0.01 factor is applied to model partitioning in this pathway.

χ/Q Calculations (Meteorology) - SGTR

All releases from the SG PORV/safety valves are considered elevated releases due to the high steaming rates, therefore in accordance with regulatory guidance, the associated χ/Qs have been reduced by a factor of five.

Justification for Application of Elevated Release Credit

Justification for application of elevated release criteria for the SGTR is identical to that previously presented for the CREA. Based on this discussion, it is concluded that the factor of five reduction is acceptable at Byron Station and Braidwood Station for all steam release accidents involving PORV steam release from the SGs.

Source Term and Dose Calculations – SGTR

Source Term Calculation – SGTR

For the RADTRAD calculation, a list of 60 core isotopic nuclides and their activities were input into the RADTRAD NIF file. However, 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 iodine 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 released over an eight hour period for Case 2.

Dose Calculations - SGTR

The RADTRAD computer code is used to determine Byron Station and Braidwood Station, Units 1 and 2 SGTR accident doses at the EAB, LPZ, and CR, consistent with RG 1.183.

4.10.2 Acceptance Criteria - SGTR

Radiological doses resulting from a design basis SGTR for a CR operator and a person located at the EAB or LPZ are to be less than the regulatory dose limits as given below.

Table 4.10.2-1
Regulatory Dose Limits – SGTR

	Dose Type	Control Room (rem)	EAB and LPZ (rem)
Case 1	TEDE Dose	5[a]	25[a]
Case 2	TEDE Dose	5[a]	2.5[b]

[a] 10 CFR 50.67

[b] 10 CFR 50.67 as modified by Regulatory Guide 1.183 (Table 6, Page 1.183-20)

4.10.3 Summary and Conclusions – SGTR

Table 4.10.3-1 below provides the results from the Case 1 and Case 2 simulations that were modeled using the RADTRAD code. The total dose for each of the two cases includes the dose result due to the equilibrium secondary coolant activity.

Table 4.10.3-1

RADTRAD Analysis Results - SGTR

Case 1: Pre-Accident 60 µCi/gm DE I-131 Spike RADTRAD Dose Assessment Results		
Control Room (rem TEDE)	**EAB (rem TEDE)**	**LPZ (rem TEDE)**
0.760	0.721	0.165
Case 2: Accident Initiated 335 times Equilibrium Iodine Release Rate Spike RADTRAD Dose Assessment Results		
Control Room (rem TEDE)	**EAB (rem TEDE)**	**LPZ (rem TEDE)**
0.196	0.327	0.077

For the cases analyzed in this calculation, it is shown that a Case 1 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. Using the design basis assumptions, described methodology, and credited margin relaxation of this analysis, the limiting CR dose is 0.760 rem TEDE. This limiting dose, and all other doses, are well below their respective acceptance criteria, so it is verified that this design basis SGTR accident is sufficiently mitigated at both Byron and Braidwood Stations.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

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," Exelon Generation Company, LLC (EGC) is requesting an amendment to Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF-72, NPF-77, NPF-37, and NPF-66 for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, respectively. 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, EGC 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 Accidents (FHA)
- Control Rod Ejection Accident (CREA)
- Locked Rotor Accident (LRA)
- Main Steam Line Break (MSLB)
- Steam Generator Tube Rupture (SGTR)

Proposed changes to the current licensing basis for Byron Station and Braidwood Station, justified by the AST analyses, include the following items.

- A change in the definition for Dose Equivalent Iodine to reflect the use of exposure-to-dose conversion factors for inhalation from Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion", 1988.

- Addition of a new definition for RECENTLY IRRADIATED FUEL: "Fuel that has occupied part of a critical reactor core within the previous 48 hours. Note that all fuel that has been in a critical reactor core is referred to as irradiated fuel."
- The Containment Leakage Rate Testing Program maximum allowable containment leakage rate, (i.e., L_a) at design accident pressure (i.e., P_a) is increased from 0.1% to 0.2% of containment air weight per day.
- Emergency Core Cooling System (ECCS) allowable leakage is assumed to be 276,000 cc/hr for the Byron and Braidwood units, rather than 15,249 cc/hr (approximately 4.0 gal/hr) for Braidwood Station and 13,294 cc/hr (approximately 3.5 gal/hr) for Byron Station.
- The Control Room Ventilation (VC) Filtration System makeup filter charcoal adsorber bypass acceptance criterion is increased from 0.05% to 1% and the associated charcoal adsorber penetration acceptance criterion is increased from 0.5% to 2.0%. These changes reflect a reduction in the assumed filtration efficiency from 99% to 95%.
- TS and associated Bases revisions to change the applicability requirements for the below systems to be applicable only during movement of RECENTLY IRRADIATED FUEL assemblies and to relax the operability requirements of these systems during core alterations, unless required for other reasons:
  - Containment
  - Fuel Handling Building (FHB) Ventilation System

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 at the Byron Station and Braidwood Station.

   - 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, it has been demonstrated that, with the requested changes, the dose consequences of these limiting events are within the regulatory guidance provided by the NRC for use with the AST methodology. This guidance is presented in RG 1.183, and Standard Review Plan Section 15.0.1. The AST is an input to calculations used to evaluate the consequences of an accident and does not by itself affect the plant response or the actual pathway of the activity released from the fuel. It does, however, better represent the physical characteristics of the release such that appropriate mitigation techniques may be applied.

The AST methodology follows the guidance provided in RG 1.183 and satisfies the dose limits in 10 CFR 50.67. Even though these limits are not directly comparable to the previously specified whole body and thyroid requirements of 10 CFR 50, Appendix A, General Design Criteria (GDC) 19, "Control room," and 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance," the results of the AST analyses have demonstrated that the 10 CFR 50.67 limits are satisfied. Therefore, it is concluded that AST does not involve a significant increase in the consequences of an accident previously evaluated.

Implementation of AST provides increased operating margins for the control room ventilation system filtration efficiencies. It also relaxes containment integrity requirements while handling irradiated fuel that has decayed for greater than 48 hours and during core alterations. Automatic initiation of the radiation isolation mode for the control room is not credited in the accident analysis which allows relaxation of certain Technical Specification surveillance requirements.

The equipment affected by the proposed changes is mitigative in nature and relied upon after an accident has been initiated. Application of the AST does result in changes to the functions and operation of various filtration systems as described in the Updated Final Safety Analysis Report (UFSAR). These effects have been considered in the evaluations for these proposed changes. While the operation of various systems does change with the implementation of AST, the affected systems are not accident initiators; and application of the AST methodology, itself, is not an initiator of a design basis accident. The proposed changes to the TS revise certain equipment performance requirements but do not require any physical changes to the plant.

As a result, the proposed changes do not affect any of the parameters or conditions that could contribute to the initiation of any accidents. 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.

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 to the plant. Implementation of AST provides increased operating margins for filtration system efficiencies. Application of AST also allows for the relaxation of containment integrity requirements while handling irradiated fuel that has decayed for greater than 48 hours and during core alterations. Automatic initiation of the radiation isolation mode for the control room is no longer credited in the accident analysis.

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.

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.

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, are within the specified regulatory limits.

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 7, "Regulatory Guide Conformance Tables," (i.e., Tables A-G) were developed to show how each section of the RG 1.183 guidance is being addressed. In addition, Table H of Attachment 7 shows conformance with the diffuse area source guidance in RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," December 2001.

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.

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

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," Exelon Generation Company, LLC (EGC) is requesting an amendment to Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF-72, NPF-77, NPF-37, and NPF-66 for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, respectively. 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."

EGC 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." EGC 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.

The following table demonstrates that EGC meets the radiological criteria described in 10 CFR 50.67 for the exclusion area boundary (EAB) and the low population zone (LPZ).

Dose Results (rem TEDE)				
Accident	EAB Doses and Limits		LPZ Doses and Limits	
	Dose	Limit	Dose	Limit
Loss of Coolant Accident	12.2	25	2.99	25
Main Steam Line Break	0.127 (1) 0.175 (2)	25 (1) 2.5 (2)	0.073 (1) 0.406 (2)	25 (1) 2.5 (2)
Control Rod Ejection Accident	4.647	6.3	1.983	6.3
Locked Rotor Accident	1.456	2.5	0.525	2.5
Steam Generator Tube Rupture	0.721 (1) 0.327 (3)	25 (1) 2.5 (3)	0.165 (1) 0.077 (3)	25 (1) 2.5 (3)
Fuel Handling Accident	4.24	6.3	0.356	6.3

Note (1): Pre-accident 60 μCi/gm DEI spike
Note (2): Accident initiated 500 times equilibrium iodine release rate spike
Note (3): Accident initiated 335 times equilibrium iodine release rate spike

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.

The following table demonstrates that EGC meets the radiological criteria described in 10 CFR 50.67 for the control room. Control room exposure to operators is less than the five rem total effective dose equivalent (TEDE) over 30 days for all accidents.

Control Room Dose Results (rem TEDE)		
Accident	**Dose**	**Limit**
Loss of Coolant Accident	4.00	5.0
Main Steam Line Break	0.790 (1) 4.255 (2)	5.0
Control Rod Ejection Accident	2.549	5.0
Locked Rotor Accident	2.529	5.0
Steam Generator Tube Rupture	0.760 (1) 0.196 (3)	5.0
Fuel Handling Accident	4.55	5.0

Note (1): Pre-accident 60 µCi/gm DEI spike
Note (2): Accident initiated 500 times equilibrium iodine release rate spike
Note (3): Accident initiated 335 times equilibrium iodine release rate spike

The implementation of the AST methodology has been evaluated in revisions to the analyses of the limiting design basis accidents at the Byron Station, Units 1 and 2; and Braidwood Station, Units 1 and 2. These accidents include the loss of coolant accident, fuel handling accident, 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

7.1 U. S. Atomic Energy Commission, Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," March 23, 1962

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

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

7.4 NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," February 1995

7.5 A. G. Croff, "A User's Manual for the ORIGEN 2 Computer Code," ORNL/TM-7175, Oak Ridge National Laboratory, July 1980

7.6 S. L. Humphreys et al., "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," NUREG/CR-6604, U. S. Nuclear Regulatory Commission, April 1998 (Note: RADTRAD Version 3.03 used in the analysis)

7.7 U. S. Nuclear Regulatory Commission Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1, November 1982

7.8 T. J. Bander, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations," NUREG-2858, U. S. Nuclear Regulatory Commission, November 1982

7.9 J. V. Ramsdell and C. A. Simonen, "Atmospheric Relative Concentrations in Building Wakes," NUREG-6331, Revision 1, U. S. Nuclear Regulatory Commission, May 1997. (ARCON96)

7.10 ANSI/ANS-2.5-1984, "Standard for Determining Meteorological Information at Nuclear Power Sites", 1984

7.11 U. S. Nuclear Regulatory Commission Regulatory Guide 1.23 (Safety Guide 23), "Onsite Meteorological Programs," February 17, 1972

7.12 U. S. Nuclear Regulatory Commission Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors," Revision 2, June 1974

7.13 U. S. Nuclear Regulatory Commission Standard Review Plan 6.4, "Control Room Habitability Systems," Revision 2, July 1981

7.14 ANSI Standard, ANS/ANSI 18.1, "Radioactive Source Term For Normal Operation for Light Water Reactors," 1999

7.15 Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion", 1988

7.16 Federal Guidance Report No. 12, "External Exposure to Radionuclides in Air, Water, and Soil", 1993.

7.17 Generic Letter 99-02, Laboratory Testing of Nuclear-Grade Activated Charcoal, June 3,1999

7.18 U. S. Nuclear Regulatory Commission Standard Review Plan 6.5.2, "Containment Spray as a Fission Product Cleanup System," Revision 2, December 1988

7.19 Regulatory Guide 1.194; Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants; U.S. Nuclear Regulatory Commission; December 2001.

7.20 Technical Specification Task Force (TSTF) Traveler, TSTF-51, "Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations," Revision 2

7.21 NUREG-0737, "Clarification of TMI Action Plan Requirements", October 1980

7.22 10 CFR 50.67, "Accident source term," December 23, 1999

7.23 10 CFR 50.90, "Application for amendment of license or construction permit", October 4, 1999

7.24 10 CFR 50, Appendix A, General Design Criterion 19, "Control room"

7.25 ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers," 1979

7.26 NUREG/CR-6189, D.A. Powers et al, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments," July 1996

ATTACHMENT 2A

BYRON STATION
UNITS 1 AND 2

Docket Nos. 50-454 and 50-455

License Nos. NPF-37 and NPF-66

License Amendment Request
"Alternative Source Term Implementation"

Markup of Technical Specification Pages

1.1-3
1.1-6
3.3.7-2
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3.7.13-1
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3.7.13-4
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3.9.4-2
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5.5-24

1.1 Definitions

DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same ~~thyroid~~ dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The ~~thyroid~~ dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977, or ICRP 30, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity,"
$\bar{E}$ -AVERAGE DISINTEGRATION ENERGY	$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies (in MeV) per disintegration for non-iodine isotopes, with half lives > 10 minutes, making up at least 95% of the total non-iodine activity in the coolant.
ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

or Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion," 1989; (Table 2.1, Exposure-to-Dose Conversion Factors for Inhalation).

1.1 Definitions

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits including heatup and cooldown rates, and the pressurizer Power Operated Relief Valve (PORV) lift settings for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Unit operation within these limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3586.6 MWt.
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
RECENTLY IRRADIATED FUEL	Fuel that has occupied part of a critical reactor core within the previous 48 hours. (Note that all fuel that has been in a critical reactor core is referred to as irradiated fuel.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met during movement of irradiated fuel assemblies.	D.1 Suspend movement of irradiated fuel assemblies.	Immediately
E. Required Action and associated Completion Time of Condition A or B not met in MODE 5 or 6.	~~E.1 Suspend CORE ALTERATIONS.~~ ~~AND~~ E.~~2~~ 1 Initiate action to restore one VC Filtration System train to OPERABLE status.	~~Immediately~~ Immediately

SURVEILLANCE REQUIREMENTS

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

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

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.7.3	Perform CHANNEL CALIBRATION.	18 months

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Table 3.3.7-1 (page 1 of 1)
VC Filtration System Actuation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1. Control Room Radiation-Gaseous	1,2,3,4,5,6,(a)	2 per train	SR 3.3.7.1 SR 3.3.7.2 SR 3.3.7.3	≤ 2 mR/hr
2. Safety Injection	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 1, for all initiation functions and requirements.			

(a) During movement of irradiated fuel assemblies.

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ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Required Action and associated Completion Time not met. OR Two channels inoperable.	B.1	Place in emergency mode one FHB Ventilation System train capable of being powered by an OPERABLE emergency power source.	Immediately
	OR		
	B.2.1	Suspend movement of ~~irradiated fuel~~ assemblies in the fuel handling building.	Immediately
	AND		
	B.2.2	---------NOTE-------- Only required with equipment hatch not intact. -------------------- Suspend movement of ~~irradiated fuel~~ assemblies in the containment.	Immediately
	~~AND~~		
	~~B.2.3~~	~~---------NOTE---------~~ ~~Only required with equipment hatch not intact.~~ ~~--------------------~~ ~~Suspend CORE ALTERATIONS.~~	~~Immediately~~

RECENTLY IRRADIATED FUEL

Table 3.3.8-1 (page 1 of 1)
FHB Ventilation System Actuation Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1.	Fuel Handling Building Radiation	(a), (b), ~~(c)~~	2	SR 3.3.8.1 SR 3.3.8.2 SR 3.3.8.3	≤ 5 mR/hr
2.	Safety Injection	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 1, for all initiation functions and requirements.			

(a) During movement of ~~irradiated fuel~~ assemblies in the fuel handling building.

(b) During movement of ~~irradiated fuel~~ assemblies in the containment with the equipment hatch not intact.

~~(c) During CORE ALTERATIONS with the equipment hatch not intact.~~

RECENTLY IRRADIATED FUEL

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met in MODE 5 or 6, or during movement of irradiated fuel assemblies.	C.1.1 Place OPERABLE VC Filtration System train in emergency mode.	Immediately
	<u>AND</u>	
	C.1.2 Verify OPERABLE VC Filtration System train is capable of being powered by an OPERABLE emergency power source.	Immediately
	<u>OR</u>	
	~~C.2.1 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~<u>AND</u>~~	
	C.2.~~2~~ 1 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	C.2.~~3~~ 2 Suspend positive reactivity additions.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Two VC Filtration System trains inoperable in MODE 5 or 6, or during movement of irradiated fuel assemblies.	~~D.1~~	~~Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~		
	D.~~2~~ 1	Suspend movement of irradiated fuel assemblies.	Immediately
	AND		
	D.~~3~~ 2	Suspend positive reactivity additions.	Immediately
E. Two VC Filtration System trains inoperable in MODE 1, 2, 3, or 4.	E.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Operate each VC Filtration System train with: a. Flow through the makeup system filters for ≥ 10 continuous hours with the heaters operating; and b. Flow through the recirculation charcoal adsorber for ≥ 15 minutes.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.10.2	Perform required VC Filtration System filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with VFTP
SR 3.7.10.3	Verify each VC Filtration System train actuates on an actual or simulated actuation signal.	18 months
SR 3.7.10.4	Verify one VC Filtration System train can maintain ~~the upper cable spreading room at positive pressure of ≥ 0.02 inches water gauge and~~ the control room at a positive pressure of ≥ 0.125 inches water gauge relative to areas adjacent to the control room envelope ~~area~~ during the emergency mode of operation at a makeup flow rate ≥ 5400 cfm and ≤ 6600 cfm.	18 months on a STAGGERED TEST BASIS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met in MODE 5 or 6, or during movement of irradiated fuel assemblies.	C.1.1 Place OPERABLE VC Temperature Control System train in operation.	Immediately
	<u>AND</u>	
	C.1.2 Verify OPERABLE VC Temperature Control System train is capable of being powered by an OPERABLE emergency power source.	Immediately
	<u>OR</u>	
	~~C.2.1 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	C.2.~~2~~1 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	C.2.~~3~~2 Suspend positive reactivity additions.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Two VC Temperature Control System trains inoperable in MODE 5 or 6, or during movement of irradiated fuel assemblies.	~~D.1~~	~~Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~		
	D.~~2~~ 1	Suspend movement of irradiated fuel assemblies.	Immediately
	AND		
	D.~~3~~ 2	Suspend positive reactivity additions.	Immediately
E. Two VC Temperature Control System trains inoperable in MODE 1, 2, 3, or 4.	E.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.11.1	Verify control room temperature ≤ 90°F.	12 hours
SR 3.7.11.2	Verify each VC Temperature Control System train has the capability to remove the required heat load.	18 months

3.7 PLANT SYSTEMS

3.7.13 Fuel Handling Building Exhaust Filter Plenum (FHB) Ventilation System

LCO 3.7.13 Two FHB Ventilation System trains shall be OPERABLE.

RECENTLY IRRADIATED FUEL

APPLICABILITY: During movement of ~~irradiated fuel~~ assemblies in the fuel building,
During movement of ~~irradiated fuel~~ assemblies in the containment with the equipment hatch not intact,
~~During CORE ALTERATIONS with the equipment hatch not intact.~~

ACTIONS

-----NOTE-----
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One FHB Ventilation System train inoperable.	A.1 Restore FHB Ventilation System train to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1.1 Place OPERABLE FHB Ventilation System train in emergency mode.	Immediately
	<u>AND</u>	
	B.1.2 Verify OPERABLE FHB Ventilation System train is capable of being powered by an OPERABLE emergency power source.	Immediately
	<u>OR</u>	

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2.1 Suspend movement of ~~irradiated fuel~~ assemblies in the fuel handling building.	Immediately
	<u>AND</u>	
RECENTLY IRRADIATED FUEL	B.2.2 --------NOTE--------- Only required with equipment hatch not intact. ---------------------	
	Suspend movement of ~~irradiated fuel~~ assemblies in the containment.	Immediately
	~~<u>AND</u>~~	
	~~B.2.3 --------NOTE---------~~ ~~Only required with equipment hatch not intact.~~ ---------------------	
	~~Suspend CORE ALTERATIONS.~~	~~Immediately~~

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two FHB Ventilation System trains inoperable.	C.1 Suspend movement of ~~irradiated fuel~~ assemblies in the fuel handling building.	Immediately
	<u>AND</u>	
RECENTLY IRRADIATED FUEL	C.2 --------NOTE--------- Only required with equipment hatch not intact. ---------------------	
	Suspend movement of ~~irradiated fuel~~ assemblies in the containment.	Immediately
	~~AND~~	
	~~C.3 NOTE Only required with equipment hatch not intact.~~	
	~~Suspend CORE ALTERATIONS.~~	~~Immediately~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Operate each FHB Ventilation System train for ≥ 15 minutes.	31 days

(continued)

ACTIONS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.13.2	Perform required FHB Ventilation System filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.13.3	-------------------NOTE------------------- Only required during movement of ~~irradiated fuel~~ assemblies ~~or CORE ALTERATIONS~~ with the equipment hatch not intact. -- Verify one FHB Ventilation System train can maintain a pressure ≤ -0.25 inches water gauge relative to atmospheric pressure during the emergency mode of operation.	 7 days on a STAGGERED TEST BASIS
SR 3.7.13.4	Verify each FHB Ventilation System train actuates on an actual or simulated actuation signal.	18 months
SR 3.7.13.5	-------------------NOTE------------------- Only required during movement of ~~irradiated fuel~~ assemblies in the fuel handling building with the equipment hatch intact. -- Verify one FHB Ventilation System train can maintain a pressure ≤ -0.25 inches water gauge relative to atmospheric pressure during the emergency mode of operation at a flow rate ≤ 23,100 cfm.	 18 months on a STAGGERED TEST BASIS

RECENTLY IRRADIATED FUEL

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

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

a. One door in the personnel air lock closed and the equipment hatch held in place by ≥ 4 bolts;

b. One door in the emergency air lock closed; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. Closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. Capable of being closed by an OPERABLE Containment Ventilation Isolation System.

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

TS

Item a. only required when the Fuel Handling Building Exhaust Filter Plenum Ventilation System is not in compliance with ~~LCO~~ 3.7.13, "Fuel Handling Building Exhaust Filter Plenum (FHB) Ventilation System."

APPLICABILITY: ~~During CORE ALTERATIONS,~~
During movement of ~~irradiated fuel~~ RECENTLY IRRADIATED FUEL assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	~~A.1 Suspend CORE ALTERATIONS.~~ ~~AND~~ A.~~2~~ 1 Suspend movement of ~~irradiated fuel~~ RECENTLY IRRADIATED FUEL assemblies within containment.	~~Immediately~~ Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify each required containment penetration is in the required status.	7 days
SR 3.9.4.2	Verify each required containment purge valve actuates to the isolation position on an actual or simulated actuation signal.	18 months
SR 3.9.4.3	Verify the isolation time of each required containment purge valve is within limits.	In accordance with the Inservice Testing Program

3.9 REFUELING OPERATIONS

3.9.7 Refueling Cavity Water Level

LCO 3.9.7 Refueling cavity water level shall be maintained ≥ 23 ft above the top of reactor vessel flange.

APPLICABILITY: ~~During CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts,~~
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling cavity water level not within limit.	~~A.1 Suspend CORE ALTERATIONS.~~ ~~AND~~ A.~~2~~ 1 Suspend movement of irradiated fuel assemblies within containment.	~~Immediately~~ Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.7.1	Verify refueling cavity water level is ≥ 23 ft above the top of reactor vessel flange.	24 hours

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for each of the ESF filter systems that an inplace test of the charcoal adsorber shows a bypass specified below when tested in conformance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980, with any exceptions noted in Appendix A of the UFSAR, at the system flow rate specified below. Verification of the specified flow rates may be accomplished during the performance of SRs 3.7.10.4, 3.7.12.4, and 3.7.13.5, as applicable:

ESF Ventilation System	Flow Rate	Bypass
VC Filtration System (makeup)	≥ 5400 cfm and ≤ 6600 cfm	< ~~0.05%~~ 1%
VC Filtration System (recirculation, charcoal bed after complete or partial replacement)	≥ 44,550 cfm and ≤ 54,450 cfm	< 0.1%
VC Filtration System (recirculation for reasons other than complete or partial charcoal bed replacement)	≥ 44,550 cfm and ≤ 54,450 cfm	< 2%
Nonaccessible Area Exhaust Filter Plenum Ventilation System (after structural maintenance of the charcoal adsorber housings)	≥ 55,669 cfm and ≤ 68,200 cfm per train, and ≥ 18,556 cfm and ≤ 22,733 cfm per bank	< 1%
Nonaccessible Area Exhaust Filter Plenum Ventilation System (for reasons other than structural maintenance of the charcoal adsorber housings)	≥ 55,669 cfm and ≤ 68,200 cfm per train	< 1%
FHB Ventilation System	≥ 18,900 cfm and ≤ 23,100 cfm per train	< 1%

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

c. Demonstrate for each of the ESF filter systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in conformance with Regulatory Guide 1.52, Revision 2, ANSI N510-1980, and ASTM D3803-1989, with any exceptions noted in Appendix A of the UFSAR, at a temperature of 30°C and a Relative Humidity (RH) specified below:

ESF Ventilation System	Penetration	RH
VC Filtration System (makeup)	~~0.5%~~ 2.0%	70%
VC Filtration System (recirculation)	4%	70%
Nonaccessible Area Exhaust Filter Plenum Ventilation System	4.5%	70%
FHB Ventilation System	10%	95%

d. Demonstrate for each of the ESF filter systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is < 6 inches of water gauge when tested in conformance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980, with any exceptions noted in Appendix A of the UFSAR, at the system flow rate specified below. Verification of the specified flow rates may be accomplished during the performance of SRs 3.7.10.4, 3.7.12.4, and 3.7.13.5, as applicable:

ESF Ventilation System	Flow Rate
VC Filtration System (makeup)	≥ 5400 cfm and ≤ 6600 cfm
Nonaccessible Area Exhaust Filter Plenum Ventilation System	≥ 55,669 cfm and ≤ 68,200 cfm per train
FHB Ventilation System	≥ 18,900 cfm and ≤ 23,100 cfm

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.16 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, September 1995 and NEI 94-01, Revision 0.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 42.8 psig for Unit 1 and 38.4 psig for Unit 2

0.20%

The maximum allowable containment leakage rate, L_a, at P_a, shall be ~~0.10%~~ of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and C tests and $< 0.75\ L_a$ for Type A tests; and

ATTACHMENT 2B

BRAIDWOOD STATION
UNITS 1 AND 2

Docket Nos. 50-456 and 50-457

License Nos. NPF-72 and NPF-77

License Amendment Request
"Alternative Source Term Implementation"

Markup of Technical Specification Pages

1.1-3
1.1-6
3.3.7-2
3.3.7-3
3.3.7-4
3.3.8-2
3.3.8-4
3.7.10-2
3.7.10-3
3.7.10-4
3.7.11-2
3.7.11-3
3.7.13-1
3.7.13-2
3.7.13-3
3.7.13-4
3.9.4-1
3.9.4-2
3.9.7-1
5.5-17
5.5-18
5.5-24

1.1 Definitions

DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same ~~thyroid~~ dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The ~~thyroid~~ dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977, or ICRP 30, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity," or Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion," 1989; (Table 2.1, Exposure-to-Dose Conversion Factors for Inhalation).
$\bar{E}$-AVERAGE DISINTEGRATION ENERGY	$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies (in MeV) per disintegration for non-iodine isotopes, with half lives > 10 minutes, making up at least 95% of the total non-iodine activity in the coolant.
ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

1.1 Definitions

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits including heatup and cooldown rates, and the pressurizer Power Operated Relief Valve (PORV) lift settings for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Unit operation within these limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3586.6 MWt.
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
RECENTLY IRRADIATED FUEL	Fuel that has occupied part of a critical reactor core within the previous 48 hours. Note that all fuel that has been in a critical reactor core is referred to as irradiated fuel.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met during movement of irradiated fuel assemblies.	D.1 Suspend movement of irradiated fuel assemblies.	Immediately
E. Required Action and associated Completion Time of Condition A or B not met in MODE 5 or 6.	~~E.1 Suspend CORE ALTERATIONS.~~ ~~AND~~ E.~~2~~1 Initiate action to restore one VC Filtration System train to OPERABLE status.	~~Immediately~~ Immediately

SURVEILLANCE REQUIREMENTS

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

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

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.7.3	Perform CHANNEL CALIBRATION.	18 months

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Table 3.3.7-1 (page 1 of 1)
VC Filtration System Actuation Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1.	Control Room Radiation-Gaseous	1,2,3,4,5,6,(a)	2 per train	SR 3.3.7.1 SR 3.3.7.2 SR 3.3.7.3	≤ 2 mR/hr
2.	Safety Injection	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 1, for all initiation functions and requirements.			

(a) During movement of irradiated fuel assemblies.

Format change only - text rolled forward to previous page. Page 3.3.7-4 is eliminated.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Required Action and associated Completion Time not met. OR Two channels inoperable.	B.1	Place in emergency mode one FHB Ventilation System train capable of being powered by an OPERABLE emergency power source.	Immediately
	OR		
	B.2.1	Suspend movement of ~~irradiated fuel~~ assemblies in the fuel handling building.	Immediately
	AND		
	B.2.2	---------NOTE-------- Only required with equipment hatch not intact. -------------------- Suspend movement of ~~irradiated fuel~~ assemblies in the containment.	Immediately
	~~AND~~		
	~~B.2.3~~	~~---------NOTE---------~~ ~~Only required with equipment hatch not intact.~~ ~~--------------------~~ ~~Suspend CORE ALTERATIONS.~~	~~Immediately~~

RECENTLY IRRADIATED FUEL

Table 3.3.8-1 (page 1 of 1)
FHB Ventilation System Actuation Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1.	Fuel Handling Building Radiation	(a),(b)~~(c)~~	2	SR 3.3.8.1 SR 3.3.8.2 SR 3.3.8.3	≤ 5 mR/hr
2.	Safety Injection	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 1, for all initiation functions and requirements.			

(a) During movement of ~~irradiated fuel~~ assemblies in the fuel handling building.

(b) During movement of ~~irradiated fuel~~ assemblies in the containment with the equipment hatch not intact.

~~(c) During CORE ALTERATIONS with the equipment hatch not intact.~~

RECENTLY IRRADIATED FUEL

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met in MODE 5 or 6, or during movement of irradiated fuel assemblies.	C.1.1	Place OPERABLE VC Filtration System train in emergency mode.	Immediately
	<u>AND</u>		
	C.1.2	Verify OPERABLE VC Filtration System train is capable of being powered by an OPERABLE emergency power source.	Immediately
	<u>OR</u>		
	~~C.2.1~~	~~Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~<u>AND</u>~~		
	C.2.~~2~~1	Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>		
	C.2.~~3~~2	Suspend positive reactivity additions.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Two VC Filtration System trains inoperable in MODE 5 or 6, or during movement of irradiated fuel assemblies.	~~D.1~~	~~Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~		
	D.~~2~~1	Suspend movement of irradiated fuel assemblies.	Immediately
	AND		
	D.~~3~~2	Suspend positive reactivity additions.	Immediately
E. Two VC Filtration System trains inoperable in MODE 1, 2, 3, or 4.	E.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Operate each VC Filtration System train with: a. Flow through the makeup system filters for ≥ 10 continuous hours with the heaters operating; and b. Flow through the recirculation charcoal adsorber for ≥ 15 minutes.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.10.2	Perform required VC Filtration System filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with VFTP
SR 3.7.10.3	Verify each VC Filtration System train actuates on an actual or simulated actuation signal.	18 months
SR 3.7.10.4	Verify one VC Filtration System train can maintain ~~the upper cable spreading room at positive pressure of ≥ 0.02 inches water gauge and~~ the control room at a positive pressure of ≥ 0.125 inches water gauge relative to areas adjacent to the control room ~~area~~ envelope during the emergency mode of operation at a makeup flow rate ≥ 5400 cfm and ≤ 6600 cfm.	18 months on a STAGGERED TEST BASIS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met in MODE 5 or 6, or during movement of irradiated fuel assemblies.	C.1.1 Place OPERABLE VC Temperature Control System train in operation.	Immediately
	AND	
	C.1.2 Verify OPERABLE VC Temperature Control System train is capable of being powered by an OPERABLE emergency power source.	Immediately
	OR	
	~~C.2.1 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	C.2.~~2~~1 Suspend movement of irradiated fuel assemblies.	Immediately
	AND	
	C.2.~~3~~2 Suspend positive reactivity additions.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Two VC Temperature Control System trains inoperable in MODE 5 or 6, or during movement of irradiated fuel assemblies.	~~D.1~~	~~Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~		
	D.~~2~~1	Suspend movement of irradiated fuel assemblies.	Immediately
	AND		
	D.~~3~~2	Suspend positive reactivity additions.	Immediately
E. Two VC Temperature Control System trains inoperable in MODE 1, 2, 3, or 4.	E.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.11.1	Verify control room temperature ≤ 90°F.	12 hours
SR 3.7.11.2	Verify each VC Temperature Control System train has the capability to remove the required heat load.	18 months

3.7 PLANT SYSTEMS

3.7.13 Fuel Handling Building Exhaust Filter Plenum (FHB) Ventilation System

LCO 3.7.13 Two FHB Ventilation System trains shall be OPERABLE.

APPLICABILITY: During movement of ~~irradiated fuel~~ RECENTLY IRRADIATED FUEL assemblies in the fuel building,
During movement of ~~irradiated fuel~~ RECENTLY IRRADIATED FUEL assemblies in the containment with the equipment hatch not intact.
~~During CORE ALTERATIONS with the equipment hatch not intact.~~

ACTIONS

-----------------------------------NOTE-----------------------------------
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One FHB Ventilation System train inoperable.	A.1	Restore FHB Ventilation System train to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1.1	Place OPERABLE FHB Ventilation System train in emergency mode.	Immediately
	<u>AND</u>		
	B.1.2	Verify OPERABLE FHB Ventilation System train is capable of being powered by an OPERABLE emergency power source.	Immediately
	<u>OR</u>		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2.1 Suspend movement of ~~irradiated fuel~~ RECENTLY IRRADIATED FUEL assemblies in the fuel handling building.	Immediately
	<u>AND</u>	
	B.2.2 --------NOTE--------- Only required with equipment hatch not intact. ---------------------	
	Suspend movement of ~~irradiated fuel~~ RECENTLY IRRADIATED FUEL assemblies in the containment.	Immediately
	~~AND~~	
	~~B.2.3~~ ---------~~NOTE~~--------- ~~Only required with equipment hatch not intact.~~ ---------------------	
	~~Suspend CORE ALTERATIONS.~~	~~Immediately~~

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two FHB Ventilation System trains inoperable.	C.1 Suspend movement of ~~irradiated fuel~~ assemblies in the fuel handling building.	Immediately
	AND	
	C.2 --------NOTE--------- Only required with equipment hatch not intact. --------------------- Suspend movement of ~~irradiated fuel~~ assemblies in the containment.	Immediately
	~~AND~~	
	~~C.3 --------NOTE--------- Only required with equipment hatch not intact.~~ --------------------- ~~Suspend CORE ALTERATIONS.~~	~~Immediately~~

RECENTLY IRRADIATED FUEL

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.13.1 Operate each FHB Ventilation System train for ≥ 15 minutes.	31 days

(continued)

ACTIONS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.13.2	Perform required FHB Ventilation System filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.13.3	-------------------NOTE------------------- Only required during movement of ~~irradiated fuel~~ assemblies ~~or CORE ALTERATIONS~~ with the equipment hatch not intact. ---	
	Verify one FHB Ventilation System train can maintain a pressure ≤ -0.25 inches water gauge relative to atmospheric pressure during the emergency mode of operation.	7 days on a STAGGERED TEST BASIS
SR 3.7.13.4	Verify each FHB Ventilation System train actuates on an actual or simulated actuation signal.	18 months
SR 3.7.13.5	-------------------NOTE------------------- Only required during movement of ~~irradiated fuel~~ assemblies in the fuel handling building with the equipment hatch intact. ---	
	Verify one FHB Ventilation System train can maintain a pressure ≤ -0.25 inches water gauge relative to atmospheric pressure during the emergency mode of operation at a flow rate ≤ 23,100 cfm.	18 months on a STAGGERED TEST BASIS

RECENTLY IRRADIATED FUEL

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

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

a. One door in the personnel air lock closed and the equipment hatch held in place by ≥ 4 bolts;

b. One door in the emergency air lock closed; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. Closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. Capable of being closed by an OPERABLE Containment Ventilation Isolation System.

TS

---------------------------NOTE---------------------------
Item a. only required when the Fuel Handling Building Exhaust Filter Plenum Ventilation System is not in compliance with ~~LCO~~ 3.7.13, "Fuel Handling Building Exhaust Filter Plenum (FHB) Ventilation System."

APPLICABILITY: ~~During CORE ALTERATIONS,~~
During movement of ~~irradiated fuel~~ assemblies within containment.

RECENTLY IRRADIATED FUEL

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1	~~Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~		
	~~A.2~~	Suspend movement of ~~irradiated fuel~~ RECENTLY IRRADIATED FUEL assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify each required containment penetration is in the required status.	7 days
SR 3.9.4.2	Verify each required containment purge valve actuates to the isolation position on an actual or simulated actuation signal.	18 months
SR 3.9.4.3	Verify the isolation time of each required containment purge valve is within limits.	In accordance with the Inservice Testing Program

3.9 REFUELING OPERATIONS

3.9.7 Refueling Cavity Water Level

LCO 3.9.7 Refueling cavity water level shall be maintained ≥ 23 ft above the top of reactor vessel flange.

APPLICABILITY: ~~During CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts,~~
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling cavity water level not within limit.	~~A.1 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	A.~~2~~1 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.7.1	Verify refueling cavity water level is ≥ 23 ft above the top of reactor vessel flange.	24 hours

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for each of the ESF filter systems that an inplace test of the charcoal adsorber shows a bypass specified below when tested in conformance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980, with any exceptions noted in Appendix A of the UFSAR, at the system flow rate specified below. Verification of the specified flow rates may be accomplished during the performance of SRs 3.7.10.4, 3.7.12.4, and 3.7.13.5, as applicable:

ESF Ventilation System	Flow Rate	Bypass
VC Filtration System (makeup)	≥ 5400 cfm and ≤ 6600 cfm	< ~~0.05%~~ 1%
VC Filtration System (recirculation, charcoal bed after complete or partial replacement)	≥ 44,550 cfm and ≤ 54,450 cfm	< 0.1%
VC Filtration System (recirculation for reasons other than complete or partial charcoal bed replacement)	≥ 44,550 cfm and ≤ 54,450 cfm	< 2%
Nonaccessible Area Exhaust Filter Plenum Ventilation System (after structural maintenance of the charcoal adsorber housings)	≥ 60,210 cfm and ≤ 73,590 cfm per train, and ≥ 20,070 cfm and ≤ 24,530 cfm per bank	< 1%
Nonaccessible Area Exhaust Filter Plenum Ventilation System (for reasons other than structural maintenance of the charcoal adsorber housings)	≥ 60,210 cfm and ≤ 73,590 cfm per train	< 1%
FHB Ventilation System	≥ 18,900 cfm and ≤ 23,100 cfm per train	< 1%

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

c. Demonstrate for each of the ESF filter systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in conformance with Regulatory Guide 1.52, Revision 2, ANSI N510-1980, and ASTM D3803-1989, with any exceptions noted in Appendix A of the UFSAR, at a temperature of 30°C and a Relative Humidity (RH) specified below:

ESF Ventilation System	Penetration	RH
VC Filtration System (makeup)	~~0.5%~~ 2.0%	70%
VC Filtration System (recirculation)	4%	70%
Nonaccessible Area Exhaust Filter Plenum Ventilation System	4.5%	70%
FHB Ventilation System	10%	95%

d. Demonstrate for each of the ESF filter systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is < 6 inches of water gauge when tested in conformance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980, with any exceptions noted in Appendix A of the UFSAR, at the system flow rate specified below. Verification of the specified flow rates may be accomplished during the performance of SRs 3.7.10.4, 3.7.12.4, and 3.7.13.5, as applicable:

ESF Ventilation System	Flow Rate
VC Filtration System (makeup)	≥ 5400 cfm and ≤ 6600 cfm
Nonaccessible Area Exhaust Filter Plenum Ventilation System	≥ 60,210 cfm and ≤ 73,590 cfm per train
FHB Ventilation System	≥ 18,900 cfm and ≤ 23,100 cfm

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.16 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, September 1995 and NEI 94-01, Revision 0.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 42.8 psig for Unit 1 and 38.4 psig for Unit 2

0.20%

The maximum allowable containment leakage rate, L_a, at P_a, shall be ~~0.10%~~ of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and C tests and $< 0.75\ L_a$ for Type A tests; and

ATTACHMENT 3A

BYRON STATION
UNITS 1 AND 2

Docket Nos. 50-454 and 50-455

License Nos. NPF-37 and NPF-66

License Amendment Request
"Alternative Source Term Implementation"

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3.7.13-4
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1.1 Definitions

DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977, or ICRP 30, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity," or Federal Guidance Report 11 "Limiting Valves of Radionuclide Intake and Air Concentration and Dose Conversion factors for Inhalation, Submersion and Ingestion", 1989; (Table 2.1, Exposure-To-Dose Conversion Factors for Inhalation).
$\bar{E}$ -AVERAGE DISINTEGRATION ENERGY	$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies (in MeV) per disintegration for non-iodine isotopes, with half lives > 10 minutes, making up at least 95% of the total non-iodine activity in the coolant.
ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

1.1 Definitions

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits including heatup and cooldown rates, and the pressurizer Power Operated Relief Valve (PORV) lift settings for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Unit operation within these limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3586.6 MWt.
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
RECENTLY IRRADIATED FUEL	Fuel that has occupied part of a critical reactor core within the previous 48 hours. Note that all fuel that has been in a critical reactor core is referred to as irradiated fuel.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met during movement of irradiated fuel assemblies.	D.1 Suspend movement of irradiated fuel assemblies.	Immediately
E. Required Action and associated Completion Time of Condition A or B not met in MODE 5 or 6.	E.1 Initiate action to restore one VC Filtration System train to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

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

SURVEILLANCE		FREQUENCY
SR 3.3.7.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.7.2	Perform COT.	92 days
SR 3.3.7.3	Perform CHANNEL CALIBRATION.	18 months

Table 3.3.7-1 (page 1 of 1)
VC Filtration System Actuation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1. Control Room Radiation-Gaseous	1,2,3,4,5,6,(a)	2 per train	SR 3.3.7.1 SR 3.3.7.2 SR 3.3.7.3	≤ 2 mR/hr
2. Safety Injection	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 1, for all initiation functions and requirements.			

(a) During movement of irradiated fuel assemblies.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Required Action and associated Completion Time not met. OR Two channels inoperable.	B.1	Place in emergency mode one FHB Ventilation System train capable of being powered by an OPERABLE emergency power source.	Immediately
	OR		
	B.2.1	Suspend movement of RECENTLY IRRADIATED FUEL assemblies in the fuel handling building.	Immediately
	AND		
	B.2.2	---------NOTE-------- Only required with equipment hatch not intact. -------------------- Suspend movement of RECENTLY IRRADIATED FUEL assemblies in the containment.	Immediately

Table 3.3.8-1 (page 1 of 1)
FHB Ventilation System Actuation Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1.	Fuel Handling Building Radiation	(a),(b)	2	SR 3.3.8.1 SR 3.3.8.2 SR 3.3.8.3	≤ 5 mR/hr
2.	Safety Injection	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 1, for all initiation functions and requirements.			

(a) *During movement of RECENTLY IRRADIATED FUEL assemblies in the fuel handling building.*

(b) During movement of RECENTLY IRRADIATED FUEL assemblies in the containment with the equipment hatch not intact.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met in MODE 5 or 6, or during movement of irradiated fuel assemblies.	C.1.1 Place OPERABLE VC Filtration System train in emergency mode.	Immediately
	<u>AND</u>	
	C.1.2 Verify OPERABLE VC Filtration System train is capable of being powered by an OPERABLE emergency power source.	Immediately
	<u>OR</u>	
	C.2.1 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	C.2.2 Suspend positive reactivity additions.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two VC Filtration System trains inoperable in MODE 5 or 6, or during movement of irradiated fuel assemblies.	D.1 Suspend movement of irradiated fuel assemblies. AND D.2 Suspend positive reactivity additions.	Immediately Immediately
E. Two VC Filtration System trains inoperable in MODE 1, 2, 3, or 4.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.10.1 Operate each VC Filtration System train with: a. Flow through the makeup system filters for ≥ 10 continuous hours with the heaters operating; and b. Flow through the recirculation charcoal adsorber for ≥ 15 minutes.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.10.2	Perform required VC Filtration System filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with VFTP
SR 3.7.10.3	Verify each VC Filtration System train actuates on an actual or simulated actuation signal.	18 months
SR 3.7.10.4	Verify one VC Filtration System train can maintain the control room at a positive pressure of ≥ 0.125 inches water gauge relative to areas adjacent to the control room envelope during the emergency mode of operation at a makeup flow rate ≥ 5400 cfm and ≤ 6600 cfm.	18 months on a STAGGERED TEST BASIS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met in MODE 5 or 6, or during movement of irradiated fuel assemblies.	C.1.1 Place OPERABLE VC Temperature Control System train in operation.	Immediately
	AND	
	C.1.2 Verify OPERABLE VC Temperature Control System train is capable of being powered by an OPERABLE emergency power source.	Immediately
	OR	
	C.2.1 Suspend movement of irradiated fuel assemblies.	Immediately
	AND	
	C.2.2 Suspend positive reactivity additions.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two VC Temperature Control System trains inoperable in MODE 5 or 6, or during movement of irradiated fuel assemblies.	D.1 Suspend movement of irradiated fuel assemblies. AND D.2 Suspend positive reactivity additions.	Immediately Immediately
E. Two VC Temperature Control System trains inoperable in MODE 1, 2, 3, or 4.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.11.1	Verify control room temperature ≤ 90°F.	12 hours
SR 3.7.11.2	Verify each VC Temperature Control System train has the capability to remove the required heat load.	18 months

3.7 PLANT SYSTEMS

3.7.13 Fuel Handling Building Exhaust Filter Plenum (FHB) Ventilation System

LCO 3.7.13 Two FHB Ventilation System trains shall be OPERABLE.

APPLICABILITY: During movement of RECENTLY IRRADIATED FUEL assemblies in the fuel building,
During movement of RECENTLY IRRADIATED FUEL assemblies in the containment with the equipment hatch not intact.

ACTIONS

------------------------------------NOTE------------------------------------
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One FHB Ventilation System train inoperable.	A.1 Restore FHB Ventilation System train to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1.1 Place OPERABLE FHB Ventilation System train in emergency mode. AND B.1.2 Verify OPERABLE FHB Ventilation System train is capable of being powered by an OPERABLE emergency power source. OR	Immediately Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2.1 Suspend movement of RECENTLY IRRADIATED FUEL assemblies in the fuel handling building. AND B.2.2 --------NOTE--------- Only required with equipment hatch not intact. -------------------- Suspend movement of RECENTLY IRRADIATED FUEL assemblies in the containment.	Immediately Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Two FHB Ventilation System trains inoperable.	C.1	Suspend movement of RECENTLY IRRADIATED FUEL assemblies in the fuel handling building.	Immediately
	AND		
	C.2	--------NOTE-------- Only required with equipment hatch not intact. -------------------- Suspend movement of RECENTLY IRRADIATED FUEL assemblies in the containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Operate each FHB Ventilation System train for ≥ 15 minutes.	31 days

(continued)

ACTIONS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.13.2	Perform required FHB Ventilation System filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.13.3	-------------------NOTE------------------- Only required during movement of RECENTLY IRRADIATED FUEL assemblies with the equipment hatch not intact. -- Verify one FHB Ventilation System train can maintain a pressure ≤ -0.25 inches water gauge relative to atmospheric pressure during the emergency mode of operation.	 7 days on a STAGGERED TEST BASIS
SR 3.7.13.4	Verify each FHB Ventilation System train actuates on an actual or simulated actuation signal.	18 months
SR 3.7.13.5	-------------------NOTE------------------- Only required during movement of RECENTLY IRRADIATED FUEL assemblies in the fuel handling building with the equipment hatch intact. -- Verify one FHB Ventilation System train can maintain a pressure ≤ -0.25 inches water gauge relative to atmospheric pressure during the emergency mode of operation at a flow rate ≤ 23,100 cfm.	 18 months on a STAGGERED TEST BASIS

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

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

a. One door in the personnel air lock closed and the equipment hatch held in place by ≥ 4 bolts;

b. One door in the emergency air lock closed; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. Closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. Capable of being closed by an OPERABLE Containment Ventilation Isolation System.

---------------------------NOTE---------------------------
Item a. only required when the Fuel Handling Building Exhaust Filter Plenum Ventilation System is not in compliance with TS 3.7.13, "Fuel Handling Building Exhaust Filter Plenum (FHB) Ventilation System."

APPLICABILITY: During movement of RECENTLY IRRADIATED FUEL assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend movement of RECENTLY IRRADIATED FUEL assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify each required containment penetration is in the required status.	7 days
SR 3.9.4.2	Verify each required containment purge valve actuates to the isolation position on an actual or simulated actuation signal.	18 months
SR 3.9.4.3	Verify the isolation time of each required containment purge valve is within limits.	In accordance with the Inservice Testing Program

3.9 REFUELING OPERATIONS

3.9.7 Refueling Cavity Water Level

LCO 3.9.7 Refueling cavity water level shall be maintained ≥ 23 ft above the top of reactor vessel flange.

APPLICABILITY: During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling cavity water level not within limit.	A.1 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.7.1	Verify refueling cavity water level is ≥ 23 ft above the top of reactor vessel flange.	24 hours

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for each of the ESF filter systems that an inplace test of the charcoal adsorber shows a bypass specified below when tested in conformance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980, with any exceptions noted in Appendix A of the UFSAR, at the system flow rate specified below. Verification of the specified flow rates may be accomplished during the performance of SRs 3.7.10.4, 3.7.12.4, and 3.7.13.5, as applicable:

ESF Ventilation System	Flow Rate	Bypass
VC Filtration System (makeup)	≥ 5400 cfm and ≤ 6600 cfm	< 1%
VC Filtration System (recirculation, charcoal bed after complete or partial replacement)	≥ 44,550 cfm and ≤ 54,450 cfm	< 0.1%
VC Filtration System (recirculation for reasons other than complete or partial charcoal bed replacement)	≥ 44,550 cfm and ≤ 54,450 cfm	< 2%
Nonaccessible Area Exhaust Filter Plenum Ventilation System (after structural maintenance of the charcoal adsorber housings)	≥ 55,669 cfm and ≤ 68,200 cfm per train, and ≥ 18,556 cfm and ≤ 22,733 cfm per bank	< 1%
Nonaccessible Area Exhaust Filter Plenum Ventilation System (for reasons other than structural maintenance of the charcoal adsorber housings)	≥ 55,669 cfm and ≤ 68,200 cfm per train	< 1%
FHB Ventilation System	≥ 18,900 cfm and ≤ 23,100 cfm per train	< 1%

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

c. Demonstrate for each of the ESF filter systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in conformance with Regulatory Guide 1.52, Revision 2, ANSI N510-1980, and ASTM D3803-1989, with any exceptions noted in Appendix A of the UFSAR, at a temperature of 30°C and a Relative Humidity (RH) specified below:

ESF Ventilation System	Penetration	RH
VC Filtration System (makeup)	2.0%	70%
VC Filtration System (recirculation)	4%	70%
Nonaccessible Area Exhaust Filter Plenum Ventilation System	4.5%	70%
FHB Ventilation System	10%	95%

d. Demonstrate for each of the ESF filter systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is < 6 inches of water gauge when tested in conformance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980, with any exceptions noted in Appendix A of the UFSAR, at the system flow rate specified below. Verification of the specified flow rates may be accomplished during the performance of SRs 3.7.10.4, 3.7.12.4, and 3.7.13.5, as applicable:

ESF Ventilation System	Flow Rate
VC Filtration System (makeup)	≥ 5400 cfm and ≤ 6600 cfm
Nonaccessible Area Exhaust Filter Plenum Ventilation System	≥ 55,669 cfm and ≤ 68,200 cfm per train
FHB Ventilation System	≥ 18,900 cfm and ≤ 23,100 cfm

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.16 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, September 1995 and NEI 94-01, Revision 0.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 42.8 psig for Unit 1 and 38.4 psig for Unit 2

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.20% of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and C tests and $< 0.75\ L_a$ for Type A tests; and

ATTACHMENT 3B

BRAIDWOOD STATION
UNITS 1 AND 2

Docket Nos. 50-456 and 50-457

License Nos. NPF-72 and NPF-77

License Amendment Request
"Alternative Source Term Implementation"

Typed Technical Specification Pages

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1.1 Definitions

DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977, or ICRP 30, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity," or Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion," 1989; (Table 2.1, Exposure-To-Dose Conversion Factors for Inhalation).
$\bar{E}$ -AVERAGE DISINTEGRATION ENERGY	$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies (in MeV) per disintegration for non-iodine isotopes, with half lives > 10 minutes, making up at least 95% of the total non-iodine activity in the coolant.
ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

1.1 Definitions

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits including heatup and cooldown rates, and the pressurizer Power Operated Relief Valve (PORV) lift settings for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Unit operation within these limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3586.6 MWt.
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
RECENTLY IRRADIATED FUEL	Fuel that has occupied part of a critical reactor core within the previous 48 hours. Note that all fuel that has been in a critical reactor core is referred to as irradiated fuel.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met during movement of irradiated fuel assemblies.	D.1	Suspend movement of irradiated fuel assemblies.	Immediately
E. Required Action and associated Completion Time of Condition A or B not met in MODE 5 or 6.	E.1	Initiate action to restore one VC Filtration System train to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

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

SURVEILLANCE		FREQUENCY
SR 3.3.7.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.7.2	Perform COT.	92 days
SR 3.3.7.3	Perform CHANNEL CALIBRATION.	18 months

Table 3.3.7-1 (page 1 of 1)
VC Filtration System Actuation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1. Control Room Radiation-Gaseous	1,2,3,4,5,6,(a)	2 per train	SR 3.3.7.1 SR 3.3.7.2 SR 3.3.7.3	≤ 2 mR/hr
2. Safety Injection	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 1, for all initiation functions and requirements.			

(a) During movement of irradiated fuel assemblies.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Required Action and associated Completion Time not met. OR Two channels inoperable.	B.1	Place in emergency mode one FHB Ventilation System train capable of being powered by an OPERABLE emergency power source.	Immediately
	OR		
	B.2.1	Suspend movement of RECENTLY IRRADIATED FUEL assemblies in the fuel handling building.	Immediately
	AND		
	B.2.2	---------NOTE-------- Only required with equipment hatch not intact. -------------------- Suspend movement of RECENTLY IRRADIATED FUEL assemblies in the containment.	Immediately

Table 3.3.8-1 (page 1 of 1)
FHB Ventilation System Actuation Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1.	Fuel Handling Building Radiation	(a),(b)	2	SR 3.3.8.1 SR 3.3.8.2 SR 3.3.8.3	≤ 5 mR/hr
2.	Safety Injection	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 1, for all initiation functions and requirements.			

(a) During movement of RECENTLY IRRADIATED FUEL assemblies in the fuel handling building.

(b) During movement of RECENTLY IRRADIATED FUEL assemblies in the containment with the equipment hatch not intact.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met in MODE 5 or 6, or during movement of irradiated fuel assemblies.	C.1.1	Place OPERABLE VC Filtration System train in emergency mode.	Immediately
	AND		
	C.1.2	Verify OPERABLE VC Filtration System train is capable of being powered by an OPERABLE emergency power source.	Immediately
	OR		
	C.2.1	Suspend movement of irradiated fuel assemblies.	Immediately
	AND		
	C.2.2	Suspend positive reactivity additions.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Two VC Filtration System trains inoperable in MODE 5 or 6, or during movement of irradiated fuel assemblies.	D.1	Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>		
	D.2	Suspend positive reactivity additions.	Immediately
E. Two VC Filtration System trains inoperable in MODE 1, 2, 3, or 4.	E.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Operate each VC Filtration System train with: a. Flow through the makeup system filters for ≥ 10 continuous hours with the heaters operating; and b. Flow through the recirculation charcoal adsorber for ≥ 15 minutes.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.10.2	Perform required VC Filtration System filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with VFTP
SR 3.7.10.3	Verify each VC Filtration System train actuates on an actual or simulated actuation signal.	18 months
SR 3.7.10.4	Verify one VC Filtration System train can maintain the control room at a positive pressure of ≥ 0.125 inches water gauge relative to areas adjacent to the control room envelope during the emergency mode of operation at a makeup flow rate ≥ 5400 cfm and ≤ 6600 cfm.	18 months on a STAGGERED TEST BASIS

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met in MODE 5 or 6, or during movement of irradiated fuel assemblies.	C.1.1	Place OPERABLE VC Temperature Control System train in operation.	Immediately
	<u>AND</u>		
	C.1.2	Verify OPERABLE VC Temperature Control System train is capable of being powered by an OPERABLE emergency power source.	Immediately
	<u>OR</u>		
	C.2.1	Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>		
	C.2.2	Suspend positive reactivity additions.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two VC Temperature Control System trains inoperable in MODE 5 or 6, or during movement of irradiated fuel assemblies.	D.1 Suspend movement of irradiated fuel assemblies. AND D.2 Suspend positive reactivity additions.	Immediately Immediately
E. Two VC Temperature Control System trains inoperable in MODE 1, 2, 3, or 4.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.11.1	Verify control room temperature ≤ 90°F.	12 hours
SR 3.7.11.2	Verify each VC Temperature Control System train has the capability to remove the required heat load.	18 months

3.7 PLANT SYSTEMS

3.7.13 Fuel Handling Building Exhaust Filter Plenum (FHB) Ventilation System

LCO 3.7.13 Two FHB Ventilation System trains shall be OPERABLE.

APPLICABILITY: During movement of RECENTLY IRRADIATED FUEL assemblies in the fuel building,
During movement of RECENTLY IRRADIATED FUEL assemblies in the containment with the equipment hatch not intact.

ACTIONS

-----NOTE-----
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One FHB Ventilation System train inoperable.	A.1 Restore FHB Ventilation System train to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1.1 Place OPERABLE FHB Ventilation System train in emergency mode. AND B.1.2 Verify OPERABLE FHB Ventilation System train is capable of being powered by an OPERABLE emergency power source. OR	Immediately Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2.1 Suspend movement of RECENTLY IRRADIATED FUEL assemblies in the fuel handling building.	Immediately
	AND	
	B.2.2 --------NOTE-------- Only required with equipment hatch not intact. -------------------- Suspend movement of RECENTLY IRRADIATED FUEL assemblies in the containment.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Two FHB Ventilation System trains inoperable.	C.1	Suspend movement of RECENTLY IRRADIATED FUEL assemblies in the fuel handling building.	Immediately
	<u>AND</u>		
	C.2	--------NOTE-------- Only required with equipment hatch not intact. ------------------------- Suspend movement of RECENTLY IRRADIATED FUEL assemblies in the containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Operate each FHB Ventilation System train for ≥ 15 minutes.	31 days

(continued)

ACTIONS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.13.2	Perform required FHB Ventilation System filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.13.3	-------------------NOTE------------------- Only required during movement of RECENTLY IRRADIATED FUEL assemblies with the equipment hatch not intact. -- Verify one FHB Ventilation System train can maintain a pressure ≤ -0.25 inches water gauge relative to atmospheric pressure during the emergency mode of operation.	7 days on a STAGGERED TEST BASIS
SR 3.7.13.4	Verify each FHB Ventilation System train actuates on an actual or simulated actuation signal.	18 months
SR 3.7.13.5	-------------------NOTE------------------- Only required during movement of RECENTLY IRRADIATED FUEL assemblies in the fuel handling building with the equipment hatch intact. -- Verify one FHB Ventilation System train can maintain a pressure ≤ -0.25 inches water gauge relative to atmospheric pressure during the emergency mode of operation at a flow rate ≤ 23,100 cfm.	18 months on a STAGGERED TEST BASIS

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

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

a. One door in the personnel air lock closed and the equipment hatch held in place by ≥ 4 bolts;

b. One door in the emergency air lock closed; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. Closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. Capable of being closed by an OPERABLE Containment Ventilation Isolation System.

---------------------------NOTE---------------------------
Item a. only required when the Fuel Handling Building Exhaust Filter Plenum Ventilation System is not in compliance with TS 3.7.13, "Fuel Handling Building Exhaust Filter Plenum (FHB) Ventilation System."

APPLICABILITY: During movement of RECENTLY IRRADIATED FUEL assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend movement of RECENTLY IRRADIATED FUEL assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify each required containment penetration is in the required status.	7 days
SR 3.9.4.2	Verify each required containment purge valve actuates to the isolation position on an actual or simulated actuation signal.	18 months
SR 3.9.4.3	Verify the isolation time of each required containment purge valve is within limits.	In accordance with the Inservice Testing Program

3.9 REFUELING OPERATIONS

3.9.7 Refueling Cavity Water Level

LCO 3.9.7 Refueling cavity water level shall be maintained ≥ 23 ft above the top of reactor vessel flange.

APPLICABILITY: During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling cavity water level not within limit.	A.1 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.7.1 Verify refueling cavity water level is ≥ 23 ft above the top of reactor vessel flange.	24 hours

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for each of the ESF filter systems that an inplace test of the charcoal adsorber shows a bypass specified below when tested in conformance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980, with any exceptions noted in Appendix A of the UFSAR, at the system flow rate specified below. Verification of the specified flow rates may be accomplished during the performance of SRs 3.7.10.4, 3.7.12.4, and 3.7.13.5, as applicable:

ESF Ventilation System	Flow Rate	Bypass
VC Filtration System (makeup)	≥ 5400 cfm and ≤ 6600 cfm	< 1%
VC Filtration System (recirculation, charcoal bed after complete or partial replacement)	≥ 44,550 cfm and ≤ 54,450 cfm	< 0.1%
VC Filtration System (recirculation for reasons other than complete or partial charcoal bed replacement).	≥ 44,550 cfm and ≤ 54,450 cfm	< 2%
Nonaccessible Area Exhaust Filter Plenum Ventilation System (after structural maintenance of the charcoal adsorber housings)	≥ 60,210 cfm and ≤ 73,590 cfm per train, and ≥ 20,070 cfm and ≤ 24,530 cfm per bank	< 1%
Nonaccessible Area Exhaust Filter Plenum Ventilation System (for reasons other than structural maintenance of the charcoal adsorber housings)	≥ 60,210 cfm and ≤ 73,590 cfm per train	< 1%
FHB Ventilation System	≥ 18,900 cfm and ≤ 23,100 cfm per train	< 1%

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

c. Demonstrate for each of the ESF filter systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in conformance with Regulatory Guide 1.52, Revision 2, ANSI N510-1980, and ASTM D3803-1989, with any exceptions noted in Appendix A of the UFSAR, at a temperature of 30°C and a Relative Humidity (RH) specified below:

ESF Ventilation System	Penetration	RH
VC Filtration System (makeup)	2.0%	70%
VC Filtration System (recirculation)	4%	70%
Nonaccessible Area Exhaust Filter Plenum Ventilation System	4.5%	70%
FHB Ventilation System	10%	95%

d. Demonstrate for each of the ESF filter systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is < 6 inches of water gauge when tested in conformance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980, with any exceptions noted in Appendix A of the UFSAR, at the system flow rate specified below. Verification of the specified flow rates may be accomplished during the performance of SRs 3.7.10.4, 3.7.12.4, and 3.7.13.5, as applicable:

ESF Ventilation System	Flow Rate
VC Filtration System (makeup)	≥ 5400 cfm and ≤ 6600 cfm
Nonaccessible Area Exhaust Filter Plenum Ventilation System	≥ 60,210 cfm and ≤ 73,590 cfm per train
FHB Ventilation System	≥ 18,900 cfm and ≤ 23,100 cfm

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.16 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, September 1995 and NEI 94-01, Revision 0.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 42.8 psig for Unit 1 and 38.4 psig for Unit 2

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.20% of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and C tests and $< 0.75\ L_a$ for Type A tests; and

ATTACHMENT 4A

BYRON STATION
UNITS 1 AND 2

Docket Nos. 50-454 and 50-455
License Nos. NPF-37 and NPF-66

License Amendment Request
"Alternative Source Term Implementation"

Typed Technical Specification Bases Pages
(For information only)

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B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the Reactor Coolant Pressure Boundary (RCPB) design conditions are not to be exceeded during normal operation and Anticipated Operational Occurrences (AOOs). Also, in accordance with GDC 28, "Reactivity Limits" (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.

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

Overpressurization of the RCS could result in a breach of the RCPB reducing the number of protective barriers designed to prevent radioactive releases from exceeding the limits specified in 10 CFR 50.67, "Accident Source Term," (Ref. 4). If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere.

BASES

SAFETY LIMIT VIOLATIONS (continued)

If SL 2.1.2 is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. If the Completion Time is exceeded, actions shall continue in order to reduce pressure to less than the SL. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.
2. ASME, Boiler and Pressure Vessel Code, SECTION III, Article NB-7000.
3. ASME, Boiler and Pressure Vessel Code, SECTION XI.
4. 10 CFR 50.67.
5. UFSAR, Section 5.2.2.
6. UFSAR, Section 7.2.

BASES

LCO

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

The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the SDM value of the LCO. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 50.67, "Accident Source Term," limits (Ref. 7). For the boron dilution accident, if the LCO is violated, the minimum required time assumed for operator action to terminate dilution may no longer be applicable.

APPLICABILITY

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

ACTIONS

A.1

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

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the boric acid storage tank or the refueling water storage tank. The operator should borate with the best source available for the unit conditions.

BASES

SURVEILLANCE REQUIREMENTS (continued)

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

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. UFSAR, Section 15.1.5.
3. UFSAR, Section 15.4.6.
4. UFSAR, Section 15.4.1.
5. UFSAR, Section 15.4.2.
6. UFSAR, Section 15.4.8.
7. 10 CFR 50.67.

B 3.3 INSTRUMENTATION

B 3.3.1 Reactor Trip System (RTS) Instrumentation

BASES

BACKGROUND

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

The protection and monitoring systems have been designed to assure safe operation of the reactor. This is achieved by specifying Limiting Safety System Settings (LSSS) in terms of parameters directly monitored by the RTS, as well as specifying LCOs on other reactor system parameters and equipment performance.

The LSSS, defined in this specification as the Allowable Values, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits during Design Basis Accidents (DBAs).

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

1. The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent Departure from Nucleate Boiling (DNB);
2. Fuel centerline melt shall not occur; and
3. The RCS pressure SL of 2735 psig shall not be exceeded.

Operation within the SLs of Specification 2.0, "Safety Limits (SLs)," also maintains the above values and assures that dose will be within the 10 CFR 50.67 and 10 CFR 100 limits during AOOs.

BASES

BACKGROUND (continued)

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

The RTS instrumentation is segmented into four distinct but interconnected modules as identified below. The RTS process is illustrated in UFSAR, Chapter 7 (Ref. 1):

1. Field transmitters or process sensors: provide a measurable electronic signal based upon the physical characteristics of the parameter being measured;

2. Signal Process Control and Protection System, including Analog Protection System, Nuclear Instrumentation System (NIS), field contacts, and protection channel sets: provide signal conditioning, bistable setpoint comparison, process algorithm actuation, compatible electrical signal output to protection system devices, and control board/control room/miscellaneous indications;

3. Solid State Protection System (SSPS), including input, logic, and output bays: initiates proper unit shutdown and/or ESF actuation in accordance with the defined logic, which is based on the bistable outputs from the signal process control and protection system; and

4. Reactor trip switchgear, including Reactor Trip Breakers (RTBs) and bypass breakers: provides the means to interrupt power to the Control Rod Drive Mechanisms (CRDMs) and allows the Rod Cluster Control Assemblies (RCCAs), or "rods," to fall into the core and shut down the reactor. The bypass breakers allow testing of the RTBs at power.

BASES

APPLICABLE SAFETY ANALYSES

The safety analyses assume that the containment remains intact with penetrations unnecessary for core cooling isolated early in the event (i.e., within approximately 60 seconds). The isolation of the purge valves has not been analyzed mechanistically in the dose calculations, although its rapid isolation is assumed. The containment ventilation isolation radiation monitors act as backup to the SI signal to ensure closing of the purge valves. They are also the primary means for automatically isolating containment in the event of a fuel handling accident. Containment isolation in turn ensures meeting the containment leakage rate assumptions of the safety analyses, and ensures that the calculated accidental radiological doses are below 10 CFR 50.67, "Accident Source Term," (Ref. 1) limits.

The containment ventilation isolation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requirements ensure that the instrumentation necessary to initiate Containment Ventilation Isolation, listed in Table 3.3.6-1, is OPERABLE.

1. <u>Manual Initiation - Phase A</u>

   Refer to LCO 3.3.2, Function 3.a.1, for all initiating Functions and requirements.

2. <u>Manual Initiation - Phase B</u>

   Refer to LCO 3.3.2, Function 3.b.1, for all initiating Functions and requirements.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.6

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

REFERENCES

1. 10 CFR 50.67.
2. NUREG-1366, December 1992.
3. WCAP-13877, Revision 2-P, "Reliability Assessment of Westinghouse Type AR Relays Used as SSPS Slave Relays," October 1999.
4. WCAP-13878-P, Revision 2, "Reliability Assessment of Potter & Brumfield MDR Series Relays," October 1999.
5. WCAP-13900, Revision 0, "Extension of Slave Relay Surveillance Test Intervals," April 1994.

B 3.3 INSTRUMENTATION

B 3.3.7 Control Room Ventilation (VC) Filtration System Actuation Instrumentation

BASES

BACKGROUND

The VC Filtration System provides an enclosed control room environment from which the unit can be operated following an uncontrolled release of radioactivity. During normal operation, the VC Filtration System provides control room ventilation. Upon receipt of an actuation signal, the VC Filtration System initiates filtered ventilation and pressurization of the control room. This system is described in the Bases for LCO 3.7.10, "Control Room Ventilation (VC) Filtration System."

The actuation instrumentation consists of two high radiation channels in each of the outside air intakes. A high radiation (gaseous) signal from one of two channels will initiate its associated train of the VC Filtration System and trip the Control Room Offices HVAC (VV) System. The VC Filtration System is also actuated by a Safety Injection (SI) signal. The SI Function is discussed in LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation."

The radiological dose assessments performed for the applicable Design Basis Accidents (DBAs) assume initiation of the VC Filtration System within 30 minutes.

APPLICABLE SAFETY ANALYSES

The control room must be kept habitable for the operators stationed there during accident recovery and post accident operations.

The VC Filtration System acts to terminate the supply of unfiltered outside air to the control room, initiate filtration, and pressurize the control room. These actions are necessary to ensure the control room is kept habitable for the operators stationed there during accident recovery and post accident operations by minimizing the radiation exposure of control room personnel.

In MODES 1, 2, 3, and 4, the radiation monitor actuation of the VC Filtration System provides a protected environment from which operators can control the unit following a DBA.

BASES

ACTIONS (continued)

C.1 and C.2

Condition C applies when the Required Action and associated Completion Time of Condition A or B have not been met and the unit is in MODE 1, 2, 3, or 4. The unit must be brought to a MODE in which the likelihood of an event requiring the VC Filtration System is minimized. To achieve this status, the unit must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1

Condition D applies when the Required Action and associated Completion Time of Condition A or B have not been met when irradiated fuel assemblies are being moved. Movement of irradiated fuel assemblies must be suspended immediately to reduce the risk of accidents that would require VC Filtration System actuation.

E.1

Condition E applies when the Required Action and associated Completion Time of Condition A or B have not been met in MODE 5 or 6. Actions must be initiated immediately to restore the inoperable train(s) to OPERABLE status to provide protection from significant radioactivity releases.

B 3.3 INSTRUMENTATION

B 3.3.8 Fuel Handling Building Exhaust Filter Plenum (FHB) Ventilation System Actuation Instrumentation

BASES

BACKGROUND

The FHB Ventilation System ensures that radioactive materials in the fuel handling building atmosphere following a fuel handling accident involving RECENTLY IRRADIATED FUEL are filtered and adsorbed prior to exhausting to the environment. The system is described in the Bases for LCO 3.7.13, "Fuel Handling Building Exhaust Filter Plenum (FHB) Ventilation System." The system initiates filtered ventilation of the fuel handling building automatically following receipt of a high radiation signal or safety injection signal.

Two radiation monitoring channels (ORE-AR055 and ORE-AR056) provide input to the FHB Ventilation System isolation. A high radiation signal from ORE-AR055 initiates Train A FHB Ventilation System isolation. A high radiation signal from ORE-AR056 initiates Train B FHB Ventilation System isolation. High radiation detected by any monitor initiates fuel handling building isolation and starts the FHB Ventilation System. These actions function to prevent exfiltration of contaminated air by initiating filtered ventilation, which imposes a negative pressure on the fuel handling building.

For a fuel handling accident, involving irradiated fuel assemblies other than RECENTLY IRRADIATED FUEL assemblies, the radiological dose assessments did not credit an OPERABLE FHB Ventilation System. Given the fuel handling accidents described above, the dose limits of 10 CFR 50.67 are not exceeded.

APPLICABLE SAFETY ANALYSES

The FHB Ventilation System ensures that radioactive materials in the fuel handling building atmosphere following a fuel handling accident involving RECENTLY IRRADIATED FUEL are filtered and adsorbed prior to being exhausted to the environment. This action reduces the radioactive content in the fuel handling building exhaust following a fuel handling accident so that doses remain within the limits specified in 10 CFR 50.67 (Ref. 1).

The FHB Ventilation System actuation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The LCO requires two channels to ensure that the radiation monitoring instrumentation necessary to initiate the FHB Ventilation System remains OPERABLE.

APPLICABILITY

High radiation initiation of the FHB Ventilation System must be OPERABLE during movement of RECENTLY IRRADIATED FUEL assemblies in the fuel handling building to ensure automatic initiation of the FHB Ventilation System when the potential for a fuel handling accident exists. Due to radioactive decay, the FHB Ventilation System instrumentation is only required to be OPERABLE during fuel handling involving handling RECENTLY IRRADIATED FUEL.

During movement of RECENTLY IRRADIATED FUEL assemblies with the containment equipment hatch not intact, the FHB Ventilation System actuation instrumentation is required to be OPERABLE to alleviate the consequences of an accident inside containment. The containment equipment hatch "not intact" refers to the requirement to have one door in the personnel air lock closed and the equipment hatch closed and held in place by a minimum of four bolts as described in the Bases for LCO 3.9.4, "Containment Penetrations."

While in MODES 1, 2, 3, 4, 5, and 6 without fuel handling involving handling RECENTLY IRRADIATED FUEL in progress, the FHB Ventilation System instrumentation need not be OPERABLE since a fuel handling accident involving RECENTLY IRRADIATED FUEL cannot occur.

ACTIONS

The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by plant specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a COT, when the process instrumentation is set up for adjustment to bring it within specification. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

BASES

ACTIONS (continued)

A Note has been added to the ACTIONS to clarify the application of LCO 3.0.3. LCO 3.0.3 is not applicable while in MODE 5 or 6. If moving RECENTLY IRRADIATED FUEL assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving RECENTLY IRRADIATED FUEL assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of RECENTLY IRRADIATED FUEL assemblies would not be sufficient reason to require a reactor shutdown.

A.1

Condition A applies to the failure of a single radiation monitor channel. If one channel is inoperable, a period of 7 days is allowed to restore it to OPERABLE status. The 7 day Completion Time is the same as is allowed if one train of the mechanical portion of the system is inoperable. The basis for this time is the same as that provided in LCO 3.7.13.

B.1, B.2.1, and B.2.2

Condition B applies if the Required Action or associated Completion Time of Condition A is not met or the failure of two radiation monitors. If the train cannot be restored to OPERABLE status, one FHB Ventilation System train must be immediately placed in the emergency mode. The FHB Ventilation System train placed in operation must be capable of being powered by an OPERABLE emergency power source. This accomplishes the actuation instrumentation function and places the unit in a conservative mode of operation.

BASES

ACTIONS (continued)

Alternative actions may be taken if the FHB Ventilation System train is not placed in emergency mode or does not have an associated OPERABLE diesel generator. Required Action B.2.1 requires the suspension of fuel movement of RECENTLY IRRADIATED FUEL assemblies in the Fuel Handling Building, precluding a fuel handling accident. Required Action B.2.2 requires suspending movement of RECENTLY IRRADIATED FUEL assemblies inside containment, precluding an accident that would require FHB Ventilation System actuation when the equipment hatch is not intact. These actions do not preclude the movement of fuel assemblies to a safe position.

Required Action B.2.2 is modified by a Note which indicates that this Required Action is only required if the equipment hatch is not intact. If the hatch is intact, only Required Action B.2.1 is required.

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.8-1 determines which SRs apply to which Fuel Handling Building (FHB) Radiation Actuation Functions.

SR 3.3.8.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Agreement criteria are determined, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.8.2

A COT is performed once every 92 days on each required channel to ensure the entire channel will perform the intended function. This test verifies the capability of the instrumentation to provide the FHB Ventilation System actuation. The setpoints shall be left consistent with the plant specific calibration procedure tolerance. The Frequency of 92 days is based on the known reliability of the monitoring equipment and has been shown to be acceptable through operating experience.

SR 3.3.8.3

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

REFERENCES

1. 10 CFR 50.67.

BACKGROUND (continued)

This LCO deals with protection of the Reactor Coolant Pressure Boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a Loss Of Coolant Accident (LOCA). However, the ability to monitor leakage provides advance warning to permit unit shutdown before a LOCA occurs. This advantage has been shown by "leak before break" studies.

APPLICABLE SAFETY ANALYSIS

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

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a Main Steam Line Break (MSLB). The leakage contaminates the secondary fluid. Other accidents or transients involving secondary steam release to the atmosphere are the Steam Generator Tube Rupture (SGTR), Control Rod Ejection, and the Locked Rotor event (note that the Locked Rotor analysis assumes a concurrent Steam Generator (SG) Power Operated Relief Valve (PORV) failure). The MSLB is more limiting than the SGTR, Control Rod Ejection and Locked Rotor event for main control room radiation dose.

The UFSAR (Ref. 3) analysis for 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.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The dose consequences resulting from the MSLB, SGTR, Control Rod Ejection and Locked Rotor accidents are within the limits defined in 10 CFR 50.67 (Ref. 5).

To support the use of sleeving techniques for SG tube repair, the Unit 1 primary to secondary leakage limits are conservatively reduced from 500 gpd for any single SG and 1 gpm total to 150 gpd for any single SG and 600 gpd total (Ref. 4).

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

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

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

BASES

LCO (continued)

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump discharge flow monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled Reactor Coolant Pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through All SGs

Total primary to secondary LEAKAGE amounting to 600 gallons per day through all SGs not isolated from the RCS produces acceptable doses for all applicable Design Basis Accidents (DBAs). Violation of this LCO could result in exceeding the dose limits specified in 10 CFR 50.67 for the applicable DBAs . Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.13.2

This SR provides the means necessary to determine SG OPERABILITY. The requirement to demonstrate SG tube integrity in accordance with the Steam Generator Tube Surveillance Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
2. Regulatory Guide 1.45, May 1973.
3. UFSAR, Chapter 15.
4. Safety Evaluation Report, dated May 7, 1994.
5. 10 CFR 50.67.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND

For the bounding accidents specified in Regulatory Guide (RG) 1.183 (Ref. 7), the maximum Total Effective Dose Equivalent (TEDE) that an individual at the exclusion area boundary, at the outer boundary of the low population zone and in the control room can receive is specified in 10 CFR 50.67 (Ref. 6). The limits on specific activity ensure that the doses are held to a fraction of the 10 CFR 50.67 limits.

For other non-bounding transients and accidents analyzed in the Updated Final Safety Analysis Report (UFSAR), the maximum dose to the whole body and the thyroid that an individual at the site boundary can receive for 2 hours during an accident is specified in 10 CFR 100 (Ref. 1). Any future modification to the facility design bases for these events will use source term assumptions and radiological criteria in the affected analyses that are established in RG 1.183 and 10 CFR 50.67.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize 10 CFR 50.67 control room and offsite radioactivity dose consequences in the event of accidents such as a Main Steam Line Break (MSLB), Steam Generator Tube Rupture (SGTR), Control Rod Ejection or a Locked Rotor.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The allowable levels are intended to limit dose as specified by 10 CFR 50.67 and RG 1.183.

BASES

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensures that individual doses for all applicable design basis accidents will be within 10 CFR 50.67 limits and RG 1.183 guidance. These accident analyses assume the specific activity of the reactor coolant at the LCO limit and an existing reactor coolant Steam Generator (SG) tube leakage rate of 1 gpm, which conservatively bounds the limit specified in LCO 3.4.13.d. The safety analysis assumes the specific activity of the secondary coolant at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.3, "Secondary Specific Activity."

The bounding design basis accident analysis that establishes the acceptability of the limits for RCS specific activity is the MSLB Control Room Operator dose limit as specified in 10 CFR 50.67. Reference to this analysis is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The analysis is for two cases of reactor coolant specific activity. One case assumes specific activity at 1.0 μCi/gm DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the I-131 iodine release rate from the fuel to the coolant to a value 500 times greater than the release rate corresponding to the initial primary system iodine concentration. The second case assumes the initial reactor coolant iodine activity at 60.0 μCi/gm DOSE EQUIVALENT I-131 due to a pre-accident iodine spike caused by an RCS transient. In both cases, the noble gas activity in the reactor coolant assumes 1% failed fuel, which closely equals the LCO limit of $100/\bar{E}$ μCi/gm for gross specific activity.

The radiological consequence of an MSLB event is described in UFSAR Section 15.1.5.3. The safety analysis shows the radiological consequences of a MSLB accident are within the Reference 6 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed the limits shown in Figure 3.4.16-1, in the applicable specification, for more than 48 hours. The safety analysis has concurrent and pre-accident iodine spiking levels up to 60.0 μCi/gm DOSE EQUIVALENT I-131.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

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

LCO

The specific iodine activity is limited to 1.0 μCi/gm DOSE EQUIVALENT I-131. The gross specific activity in the reactor coolant is limited to the number of μCi/gm equal to 100 divided by $\bar{E}$ (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on DOSE EQUIVALENT I-131 and the limit on gross specific activity ensure that individual doses for all applicable design basis accidents will be within 10 CFR 50.67 limits and RG 1.183 guidance.

Violation of the Technical Specification may result in reactor coolant radioactivity levels that could, in the event of an MSLB, SGTR, Control Rod Ejection or Locked Rotor accident, lead to doses that exceed the 10 CFR 50.67 dose limits.

BASES

APPLICABILITY

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

For operation in MODE 3 with RCS average temperature < 500°F, and in MODES 4 and 5, the release of radioactivity in the event of a MSLB, SGTR, Control Rod Ejection or Locked Rotor accident is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves.

ACTIONS

A.1 and A.2

With the DOSE EQUIVALENT I-131 specific activity greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the limits of Figure 3.4.16-1 are not exceeded. The Completion Time of 4 hours provides sufficient time to obtain and analyze a sample. Sampling is done to continue to provide a trend.

The DOSE EQUIVALENT I-131 specific activity must be restored to within limits within 48 hours. The Completion Time of 48 hours is required, if the limit violation resulted from normal iodine spiking.

A Note to the Required Actions excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require unit shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the unit remains at, or proceeds to power operation.

BASES

ACTIONS (continued)

B.1

If the Required Action and associated Completion Time of Condition A is not met or if the DOSE EQUIVALENT I-131 specific activity is in the unacceptable region of Figure 3.4.16-1, the reactor must be brought to MODE 3 with RCS average temperature < 500°F within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

C.1

With the gross specific activity in excess of the allowed limit, the unit must be placed in MODE 3 with RCS average temperature < 500°F. This action lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves and prevents venting the SG to the environment in an SGTR, Control Rod Ejection or Locked Rotor event. MSLB releases are similarly limited with the exception of initial blowdown and leakage through the affected SG. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

SR 3.4.16.1 requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant at least once every 7 days. A gross radioactivity analysis consists of the quantitative measurement of the total specific activity of the reactor coolant except for radionuclides with half lives < 10 minutes and all radioiodines. The total specific activity is the sum of the degassed beta-gamma activity and the total of all identified gaseous activities in the sample within 2 hours after the sample was taken. Determination of the contributors to the gross specific activity are based upon those energy peaks identifiable with a 95% confidence level. The latest available data may be used for pure beta emitting radionuclides. This Surveillance provides an indication of any increase in gross specific activity.

BASES

REFERENCES

1. 10 CFR 100.11, 1973.
2. UFSAR, Section 15.6.3.
3. Safety Evaluation Report, dated May 7, 1994.
4. Safety Evaluation Report, dated August 18, 1994.
5. Safety Evaluation Report, dated November 9, 1995.
6. 10 CFR 50.67.
7. Regulatory Guide 1.183, dated July 2000.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.4.2

Heat traced portions of the RWST vent path should be verified every 24 hours to be within the temperature limit needed to prevent ice blockage and subsequent vacuum formation in the tank during rapid level decreases caused by accident conditions. This Frequency is sufficient to identify a temperature change that would approach the lower limit and has been shown to be acceptable through operating experience.

The SR is modified by a Note that eliminates the requirement to perform this Surveillance when the ambient air temperature is ≥ 35°F. With ambient air temperature above this limit, the RWST vent path will be free of ice blockage.

SR 3.5.4.3

The RWST water volume should be verified every 7 days to be above the required minimum level of 89% (useable volume of > 395,000 gallons) in order to ensure that a sufficient initial supply is available for injection and to support continued ECCS and Containment Spray System pump operation on recirculation. Since the RWST volume is normally stable and protected by a low level alarm, a 7 day Frequency is appropriate and has been shown to be acceptable through operating experience.

SR 3.5.4.4

The boron concentration of the RWST should be verified every 7 days to be within the required limits. This SR ensures that the reactor will remain subcritical following a LOCA and will limit the power level increase and subsequently returns the reactor to subcritical immediately following an MSLB. Further, it assures that the resulting sump pH will be maintained in an acceptable range so that boron precipitation in the core will not occur, sufficient iodine will be retained to limit doses, stress corrosion cracking of equipment will be minimized, and hydrogen production will be minimized. Since the RWST volume is normally stable, a 7 day sampling Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a LOCA and a steam line break (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA, secondary system pipe break, or fuel handling accident (Ref. 3). In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment leakage rate, used to evaluate doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting design basis LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.20% per day in the safety analysis at P_a = 42.8 psig for Unit 1 and P_a = 38.4 psig for Unit 2 (Ref. 3).

The radiological dose assessments performed for the design basis LOCA assume a maximum allowable containment leakage rate of 0.20% per day. In this case, the dose limits of 10 CFR 50.67 (Ref. 7) are not exceeded.

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

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

LCO

Containment OPERABILITY is maintained by limiting leakage to ≤ 1.0 L_a, except prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test. At this time, applicable leakage limits must be met.

Compliance with this LCO will ensure a containment configuration, including the equipment hatch, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. Failure to meet air lock and purge valve leakage limits specified in LCO 3.6.2 and LCO 3.6.3 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required leakage test is required to be < 0.6 L_a for combined Type B and C leakage and < 0.75 L_a for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of ≤ 1.0 L_a. At ≤ 1.0 L_a the dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

SR 3.6.1.2

This SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and Frequency are consistent with the requirements of 10 CFR50.55a(b)(2)(vi) "Effective Edition and Addenda of Subsection IWE and Subsection IWL, SECTION XI" (Ref. 4), and Section 10 CFR50.55a(b)(2)(ix), "Examination of Concrete Containments" (Ref. 5). Predicted tendon lift off forces will be determined consistent with the recommendations of Regulatory Guide 1.35.1, (Ref. 6).

BASES

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
2. UFSAR, Chapter 15.
3. UFSAR, Section 6.2.
4. 10 CFR50.55a(b)(2)(vi) "Effective Edition and Addenda of Subsection IWE and Subsection IWL, SECTION XI."
5. 10 CFR50.55a(b)(2)(ix), "Examination of Concrete Containments."
6. Regulatory Guide 1.35.1, July 1990.
7. 10 CFR 50.67.

BASES

APPLICABLE SAFETY ANALYSES

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a Loss of Coolant Accident (LOCA) and a steam line break. In addition, release of significant fission product radioactivity within containment can occur from a LOCA, secondary system pipe break, or fuel handling accident. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment leakage rate, used to evaluate doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting design basis LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.20% per day in the safety analysis at P_a = 42.8 psig for Unit 1 and P_a = 38.4 psig for Unit 2 (Ref. 2).

The radiological dose assessments performed for the design basis LOCA assume a maximum allowable containment leakage rate of 0.20% per day. In this case, the dose limits of 10 CFR 50.67 (Ref. 3) are not exceeded.

The containment air locks satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Each containment air lock forms part of the containment pressure boundary. As part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into or exit from containment.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is not normally challenged when the containment air lock door is used for entry and exit (procedures require strict adherence to single door opening), this test is only required to be performed every 24 months. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, and the potential for loss of containment OPERABILITY if the Surveillance were performed with the reactor at power. The 24 month Frequency for the interlock is justified based on generic operating experience. The 24 month Frequency is based on engineering judgment and is considered adequate given that the interlock is not challenged during use of the air lock.

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
2. UFSAR, Section 6.2.
3. 10 CFR 50.67.

BASES

APPLICABLE SAFETY ANALYSES

The containment isolation valve LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during major accidents. As part of the containment boundary, containment isolation valve OPERABILITY supports leak tightness of the containment. Therefore, the safety analyses of any event requiring isolation of containment is applicable to this LCO.

In MODES 1 through 4, the DBAs that result in a release of radioactive material within containment are a Loss Of Coolant Accident (LOCA), secondary system pipe break, and rod ejection accident (Ref. 1). In the analyses for each of these accidents, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through containment isolation valves (including containment purge valves) are minimized. The safety analyses assume that the 48 inch purge valves are closed at event initiation.

In the calculation of control room and offsite doses following a LOCA, the accident analyses are performed in conformance with the requirements of Regulatory Guide 1.183 (Ref. 2).

The containment isolation valves ensure that the containment design leakage rate remains within L_a by automatically isolating penetrations that do not serve post accident functions and providing isolation capability for penetrations associated with safe shutdown functions. The maximum isolation time for automatic containment isolation valves is 60 seconds (Ref. 1). This isolation time is based on engineering judgement since the control room and offsite dose calculations are performed assuming that the leakage from containment begins immediately following the accident.

BASES

REFERENCES

1. UFSAR, Section 6.2.
2. Regulatory Guide 1.183, July 2000.
3. Standard Review Plan 6.2.4.
4. Generic Issue B-20, "Containment Leakage Due to Seal Deterioration."

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Containment Spray and Cooling Systems

BASES

BACKGROUND

The Containment Spray and Containment Cooling Systems provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine and aerosol removal capability of the spray reduces the release of fission product radioactivity from containment to the environment, in the event of a Design Basis Accident (DBA), to within limits. The Containment Spray and Containment Cooling Systems are designed to meet the requirements of 10 CFR 50, Appendix A, GDC 38, "Containment Heat Removal," GDC 39, "Inspection of Containment Heat Removal Systems," GDC 40, "Testing of Containment Heat Removal Systems," GDC 41, "Containment Atmosphere Cleanup," GDC 42, "Inspection of Containment Atmosphere Cleanup Systems," and GDC 43, "Testing of Containment Atmosphere Cleanup Systems" (Ref. 1).

The Containment Cooling System and Containment Spray System are Engineered Safety Feature (ESF) Systems and are discussed in UFSAR, Sections 9.4.8 and 6.5.2, respectively (Refs. 2 and 3). They are designed to ensure that the heat removal capability required during the post accident period can be attained. The Containment Spray System in conjunction with the Containment Cooling System limit and maintain post accident conditions to less than the containment design values. In addition, the Containment Spray System and Containment Cooling System provide an alternate hydrogen control function to the hydrogen recombiners, hydrogen mixing, during post Loss Of Coolant Accident (LOCA) conditions.

BASES

BACKGROUND (continued)

Containment Spray System

The Containment Spray System consists of two separate 100% capacity trains, each capable of meeting the design bases. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate ESF bus. The Refueling Water Storage Tank (RWST) supplies borated water to the Containment Spray System during the injection phase of operation. In the recirculation mode of operation, containment spray pump suction is transferred from the RWST to the containment sump(s).

The Containment Spray System provides a spray of cold borated water mixed with sodium hydroxide (NaOH) from the spray additive tank into the upper regions of containment to reduce the containment pressure and temperature and to reduce fission products from the containment atmosphere during a DBA. The RWST solution temperature is an important factor in determining the heat removal capability of the Containment Spray System during the injection phase. In the recirculation mode of operation, heat is removed from the containment sump water by the residual heat removal heat exchangers. Each train of the Containment Spray System provides adequate spray coverage to meet the system design requirements for containment heat removal.

The Spray Additive System injects an NaOH solution into the spray. The resulting alkaline pH of ≥ 8.0 for the spray ensures that radioiodines removed from the containment atmosphere by spray or natural deposition will remain in solution and not re-evolve from water in the containment sump. The upper pH limit minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the fluid. The chemical aspects of iodine removal capability are addressed in LCO 3.6.7, "Spray Additive System."

B 3.6 CONTAINMENT SYSTEMS

B 3.6.7 Spray Additive System

BASES

BACKGROUND

The Spray Additive System is a subsystem of the Containment Spray System. It provides an available inventory of sodium hydroxide (NaOH) that is added to the containment spray and is accumulated in the containment sump. The resulting pH of the sump water will be ≥ 8.0, thus ensuring that iodine in the sump water will not re-evolve and be available for release from the containment to the environment. The Spray Additive System is described in UFSAR Section 6.5.2 (Ref. 1).

Airborne radioiodine in its various forms is the fission product of primary concern in the evaluation of a design basis Loss of Coolant Accident (LOCA). It is absorbed by the spray from the containment atmosphere. To enhance the iodine absorption capacity of the spray, the spray solution is adjusted to an alkaline pH that promotes iodine hydrolysis, in which iodine is converted to nonvolatile forms. Because of its stability when exposed to radiation and elevated temperature, NaOH is the preferred spray additive. The NaOH added to the spray also ensures an equilibrium sump pH value of ≥ 8.0 and ≤ 11.0 of the solution recirculated from the containment sump. This pH band minimizes the evolution of iodine, while minimizing the occurrence of chloride and caustic stress corrosion on mechanical systems and components.

The Spray Additive System consists of one spray additive tank that is shared by the two trains of spray additive equipment. Each train of equipment provides a flow path from the spray additive tank to a containment spray pump and consists of an eductor for each containment spray pump, valves, instrumentation, and connecting piping. Each eductor draws the NaOH spray solution from the common tank using a portion of the borated water discharged by the containment spray pump as the motive flow. The eductor mixes the NaOH solution and the borated water and discharges the mixture into the spray pump suction line. The eductors are designed to ensure that the pH of the spray mixture is between 8.5 and 12.8.

BASES

BACKGROUND (continued)

An automatic or manual Containment Spray System actuation signal opens the valves from the spray additive tank to the eductor (CS019A/B), the discharge valve to the eductor from the CS pump discharge (CS010A/B), if not already open, and the isolation valve into containment (CS007A/B); in addition to starting the CS pumps. The 30% to 36% NaOH solution is drawn into the spray pump suctions. The spray additive tank capacity provides for the addition of NaOH solution to water sprayed into the containment from either the Refueling Water Storage Tank (RWST) during the injection phase, or recirculated from the containment sump during the recirculation phase. The inventory of NaOH in the spray additive tank assures a long term containment sump pH of ≥ 8.0. The percent solution and volume of solution sprayed into containment ensures a long term containment sump pH of ≥ 8.0 and ≤ 11.0. This ensures the continued iodine retention effectiveness of the sump water during the recirculation phase of spray operation and also minimizes the occurrence of chloride induced stress corrosion cracking of the stainless steel recirculation piping.

APPLICABLE SAFETY ANALYSES

The Spray Additive System is essential to prevent re-evolution of iodine collected in the sump following a DBA.

Following the assumed release of radioactive materials into containment, the containment is assumed to leak at its design value volume following the accident. The analysis assumes that 82.5% of containment is covered by the spray (Ref. 2).

The accident analysis assumes that the Spray Additive System is initiated after start of the DBA. The accident analysis does not credit iodine removal due to NaOH in the spray droplets, but only credits NaOH in the containment sump for pH control thus ensuring that iodine in the sump water will not re-evolve and be available for release from the containment to the environment.

The DBA analyses assume that one train of the Containment Spray System/Spray Additive System is inoperable and that the useable spray additive tank volume is added to the remaining Containment Spray System flow path.

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

BASES

LCO	The Spray Additive System is necessary to prevent re-evolution of radioactive iodine collected in the sump into the containment atmosphere in the event of a design basis LOCA. To be considered OPERABLE, the volume and concentration of the spray additive solution must be sufficient to provide NaOH injection into the spray flow in either the injection or the recirculation phase to increase the sump water pH to a value ≥ 8.0. This minimum pH assures retention of iodine collected in the containment sump. An upper bound on pH of 11.0 prevents conditions that may induce caustic stress corrosion cracking of mechanical system components. In addition, it is essential that valves in the Spray Additive System flow paths are properly positioned and that automatic valves are capable of activating to their correct positions.
APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment requiring the operation of the Spray Additive System. The Spray Additive System assists in reducing the iodine fission product inventory prior to release to the environment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Thus, the Spray Additive System is not required to be OPERABLE in MODE 5 or 6.
ACTIONS	<u>A.1</u> If the Spray Additive System is inoperable, it must be restored to OPERABLE within 7 days. The pH adjustment of the Containment Spray System flow for corrosion protection and iodine removal enhancement is reduced in this condition. The Containment Spray System would still be available and would remove some iodine from the containment atmosphere in the event of a DBA. The 7 day Completion Time takes into account the redundant flow path capabilities and the low probability of the worst case DBA occurring during this period.

BASES

ACTIONS (continued)

B.1 and B.2

If the Spray Additive System cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 84 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows additional time for attempting restoration of the Spray Additive System and is reasonable when considering the driving force for a release of radioiodine from the Reactor Coolant System is reduced in MODE 3.

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1

Verifying the correct alignment of Spray Additive System manual and automatic valves in the spray additive flow path provides assurance that the system is able to provide additive to the Containment Spray System in the event of a DBA. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment and capable of potentially being mispositioned are in the correct position.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.7.2

To provide effective iodine retention in the containment sump, the containment spray must be an alkaline solution. Since the RWST contents are normally acidic, the volume of the spray additive tank must provide a sufficient volume of spray additive to adjust pH for all water injected. This SR is performed to verify the availability of sufficient NaOH solution in the Spray Additive System. The 184 day Frequency was developed based on the low probability of an undetected change in tank volume occurring during the SR interval (the tank is isolated during normal unit operations). Tank level is also indicated and alarmed in the control room, so that there is high confidence that a substantial change in level would be detected.

SR 3.6.7.3

This SR provides verification of the NaOH concentration in the spray additive tank and is sufficient to ensure that the spray solution being injected into containment is at the correct pH level. The 184 day Frequency is sufficient to ensure that the concentration level of NaOH in the spray additive tank remains within the established limits. This is based on the low likelihood of an uncontrolled change in concentration (the tank is normally isolated) and the probability that any substantial variance in tank volume will be detected.

SR 3.6.7.4

This SR provides verification that each automatic valve in the Spray Additive System flow path actuates to its correct position. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

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

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

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

LCO

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

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in exposures comparable to the 10 CFR 50.67 (Ref. 4) limits or the NRC staff approved licensing basis.

BASES

REFERENCES

1. UFSAR, Section 10.3.
2. UFSAR, Section 15.1.5.
3. UFSAR, Section 6.2.
4. 10 CFR 50.67.
5. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.7 PLANT SYSTEMS

B 3.7.3 Secondary Specific Activity

BASES

BACKGROUND

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

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

This limit is lower than the activity value that might be expected from a 1 gpm tube leak (LCO 3.4.13, "RCS Operational LEAKAGE") of primary coolant at the limit of 1.0 μCi/gm (LCO 3.4.16, "RCS Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives, (i.e., < 20 hours). I-131, with a half life of 8.04 days, concentrates faster than it decays, but does not reach equilibrium because of blowdown and other losses.

With the specified activity limit, the resultant dose would be within 10 CFR 50.67 (Ref. 1) limits assuming a secondary coolant release following a trip from full power.

BASES

APPLICABLE SAFETY ANALYSES

The accident analysis of the Main Steam Line Break (MSLB), as discussed in the UFSAR, Chapter 15 (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.1 μCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed the 10 CFR 50.67 limits (Ref. 1).

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

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

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

BASES

LCO	As indicated in the Applicable Safety Analyses, the specific activity of the secondary coolant is required to be $\leq 0.1\ \mu Ci/gm$ DOSE EQUIVALENT I-131 to limit the radiological consequences of a Design Basis Accident (DBA) to the required limits (Ref. 1). Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.
APPLICABILITY	In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere. In MODES 5 and 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.
ACTIONS	<u>A.1 and A.2</u> DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS and contributes to increased post accident doses. If the secondary specific activity is not within limits, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS	SR 3.7.3.1 This SR verifies that the secondary specific activity is within the limits of the accident analysis. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.
REFERENCES	1. 10 CFR 50.67. 2. UFSAR, Chapter 15.

BASES

APPLICABLE SAFETY ANALYSES

The SG PORV lines provide an alternate method for cooling the unit to RHR entry condition whenever the preferred heat sink via the Steam Dump System to the condenser is unavailable due to a loss of offsite power. Prior to operator actions to cool down the unit, the SG PORVs and main steam safety valves (MSSVs) are assumed to operate automatically to relieve steam and maintain the SG pressure below the design value. For the recovery from a steam generator tube rupture (SGTR) event (Ref. 2), the operator is also required to perform a limited cooldown to establish adequate subcooling as a necessary step to terminate the primary to secondary break flow into the ruptured SG. The time required to terminate the primary to secondary break flow for an SGTR is more critical than the time required to cool down to RHR conditions for this event and also for other accidents. After primary to secondary break flow termination, it is assumed that SG PORVs on two intact SGs are used to cool the RCS down to 350°F. Thus, the SGTR is the limiting event for the SG PORVs. The number of SG PORVs required to be OPERABLE to satisfy the SGTR accident analysis requirements depends upon the number of reactor coolant system loops and consideration of a single failure assumption regarding the failure of one SG PORV to open on demand.

In addition, the SGTR analysis considers SG overfill. The limiting single failure with respect to SG overfill is the failure of one SG PORV on an intact SG to open when required for cooldown of the RCS. The analysis assumes four SG PORVs are OPERABLE at the start of the SGTR event. One SG PORV is on the ruptured SG, another SG PORV is assumed to fail to open and the remaining SG PORVs are used to perform the RCS cooldown. The analysis shows that cooldown using two SG PORVs results in no SG overfill.

The SG PORVs are equipped with manual block valves in the event a SG PORV fails to close during use. The SG PORVs are also credited as CIVs (refer to LCO 3.6.3).

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.4.2

The function of the block valve is to isolate a failed open SG PORV. Cycling the block valve both closed and open demonstrates its capability to perform this function. Performance of inservice testing or use of the block valve during unit cooldown may satisfy this requirement. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. The Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 10.3.
2. UFSAR, Section 15.6.3.

BASES

BACKGROUND (continued)

Actuation of the VC Filtration System places the system in the emergency mode of operation. Actuation of the system to the emergency mode of operation; starts the makeup fan, opens the turbine building intake damper, isolates the normal intake from outside dampers, isolates the purge dampers (if open), opens the recirculation charcoal adsorber dampers, and closes the recirculation charcoal adsorber bypass dampers. The operating supply and return fans continue to operate. Interlocks are provided such that the makeup fan will not start unless the associated supply fan is in operation. Outside air is filtered and then mixed with the air being recirculated through the control room. Pressurization of the control room minimizes infiltration of unfiltered air from the areas adjacent to the control room envelope.

The air entering the control room is continuously monitored by radiation detectors. One outside air intake detector output above the alarm setpoint will cause actuation of the emergency mode of operation and trip the Control Room Offices HVAC (VV) System.

The VC Filtration System will not automatically realign to the Turbine Building makeup air intake upon receipt of a high radiation or Safety Injection (SI) signal when a VC Filtration System Emergency Makeup Filter unit is in operation and aligned to the outside air intake.

One VC Filtration System train can pressurize the control room to ≥ 0.125 inches water gauge, relative to areas adjacent to the control room envelope.

The control room and the control room envelope are defined in UFSAR Section 6.4 (Ref. 1). The control room is contained within the control room envelope. The areas within the control room envelope, external to the control room, are maintained at a positive pressure as described in the Control Room Envelope Integrity Program.

Redundant filter trains are provided such that if an excessive pressure drop develops across one filter train, the other train is available to provide the required filtration.

BASES

BACKGROUND (continued)

The normally open intake isolation dampers are arranged in a series so that the failure of one damper to shut will not result in a breach of isolation. The VC Filtration System is designed in accordance with Seismic Category I requirements.

The VC Filtration System is designed to maintain the control room environment for 30 days after a Design Basis Accident (DBA) without exceeding the dose limits of 10 CFR 50.67 (Ref. 7), (i.e., 5 rem TEDE).

APPLICABLE SAFETY ANALYSES

The VC System components are arranged in redundant, safety related ventilation trains. The location of components and ducting within the control room envelope ensures an adequate supply of filtered air to all areas requiring access. The VC Filtration System provides airborne radiological protection for the control room operators, as demonstrated by the control room accident dose analyses for the most limiting design basis Loss of Coolant Accident, fission product release presented in the UFSAR, Chapter 15 (Ref. 3). The safety analyses assume a 95% filter efficiency for the makeup charcoal adsorber and a 90% filter efficiency for the recirculation charcoal adsorber. For design basis accident radiological dose assessments, the VC Filtration System is assumed to be initiated within 30 minutes.

As described in UFSAR Section 6.4 (Ref. 1) and Section 2.2 (Ref. 4), there are no potential toxic releases that could pose a risk to the control room operators.

The worst case single active failure of a component of the VC Filtration System, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

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

BASES

LCO

Two independent and redundant VC Filtration System trains are required to be OPERABLE to ensure that at least one is available assuming a single failure disables the other train. Total system failure could result in exceeding a TEDE dose of 5 rem to the control room operator in the event of a large radioactive release.

The VC Filtration System is considered OPERABLE when the individual components necessary to limit operator exposure are OPERABLE in both trains. A VC Filtration System train is OPERABLE when the associated:

a. Makeup air fan is OPERABLE;

b. Supply fan is OPERABLE;

c. Return air fan is OPERABLE;

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

e. Makeup filter unit heater, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

The control room boundary must be maintained within the assumptions of the design analysis and in accordance with the Control Room Envelope Integrity Program.

APPLICABILITY

In MODES 1, 2, 3, 4, 5, and 6, and at all times during movement of irradiated fuel assemblies in the fuel handling building or containment, the VC Filtration System must be OPERABLE to control operator exposure during and following a DBA, including the release from a fuel handling accident.

In MODE 5 or 6, the VC Filtration System provides protection from significant radioactive releases.

BASES

ACTIONS (continued)

C.1.1, C.1.2, C.2.1, and C.2.2

In MODE 5 or 6, or during movement of irradiated fuel assemblies, if the inoperable VC Filtration System train cannot be restored to OPERABLE status within the required Completion Time, action must be taken to immediately place the OPERABLE VC Filtration System train in the emergency mode. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure would be readily detected. Action C.1.2 requires the VC Filtration System train placed in operation be capable of being powered by an OPERABLE emergency power source. This action assures availability of electric power in the unlikely event of a loss of offsite power. This power source can be either from Unit 1 or Unit 2, via OPERABLE crosstie breakers.

An alternative to Required Action C.1.1 and C.1.2 is to immediately suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

D.1 and D.2

In MODE 5 or 6, or during movement of irradiated fuel assemblies, with two VC Filtration System trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might enter the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

E.1

If both VC Filtration System trains are inoperable in MODE 1, 2, 3, or 4, the VC Filtration System may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.10.4

This SR verifies the capability of the VC Filtration System to pressurize the control room. The control room positive pressure, with respect to potentially contaminated areas adjacent to the control room envelope, is periodically tested to verify the function of the VC Filtration System. During the emergency mode of operation, the VC Filtration System is designed to pressurize the control room to ≥ 0.125 inches water gauge, relative to areas adjacent to the control room envelope in order to minimize unfiltered inleakage. The VC Filtration System is designed to maintain this positive pressure with one train at a makeup flow rate ≥ 5400 cfm and ≤ 6600 cfm. The Frequency of 18 months on a STAGGERED TEST BASIS is consistent with the guidance provided in NUREG-0800 (Ref. 6).

REFERENCES

1. UFSAR, Section 6.4.
2. UFSAR, Section 9.4.
3. UFSAR, Chapter 15.
4. UFSAR, Section 2.2.
5. Regulatory Guide 1.52, Rev. 2.
6. NUREG-0800, Section 6.4, Rev. 2, July 1981.
7. 10 CFR 50.67

BASES

ACTIONS (continued)

C.1.1, C.1.2, C.2.1, and C.2.2

In MODE 5 or 6, or during movement of irradiated fuel, if the inoperable VC Temperature Control System train cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE VC Temperature Control System train must be placed in operation immediately. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that active failures will be readily detected. Action C.1.2 requires the VC Temperature Control System train placed in operation be capable of being powered by an OPERABLE emergency power source. This action assures availability of electric power in the unlikely event of a loss of offsite power. This power source can be either from Unit 1 or Unit 2, via OPERABLE crosstie breakers.

An alternative to Required Action C.1.1 and C.1.2 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

D.1 and D.2

In MODE 5 or 6, or during movement of irradiated fuel assemblies, with two VC Temperature Control System trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

E.1

If both VC Temperature Control System trains are inoperable in MODE 1, 2, 3, or 4, the control room VC Temperature Control System may not be capable of performing its intended function. Therefore, LCO 3.0.3 must be entered immediately.

BASES

APPLICABLE SAFETY ANALYSES

The design basis of the Nonaccessible Area Exhaust Filter Plenum Ventilation System is established by the large break LOCA. The system evaluation assumes a passive failure of the ECCS outside containment, such as an SI pump seal failure, during the recirculation mode. In such a case, the system limits radioactive release to within the 10 CFR 50.67 (Ref. 4) limits, or the NRC staff approved licensing basis (e.g., a specified fraction of Reference 5 limits). While the system is automatically initiated on an SI signal, manual actuation/alignment of the system is acceptable. The system is not required until initiation of the ECCS recirculation mode. The analysis of the effects and consequences of a large break LOCA is presented in Reference 3. The Nonaccessible Area Exhaust Filter Plenum Ventilation System also actuates following a small break LOCA, in those cases where the ECCS goes into the recirculation mode of long term cooling, to clean up releases of smaller leaks, such as from valve stem packing. The Nonaccessible Area Exhaust Filter Plenum Ventilation System is also credited in the control room habitability analysis (Ref. 5). The safety analyses assume a 90% filter efficiency.

Two types of system failures are considered in the accident analysis: complete loss of function, and excessive LEAKAGE. Either type of failure may result in a lower efficiency of removal for any gaseous and particulate activity released to the ECCS pump rooms following a LOCA.

The Nonaccessible Area Exhaust Filter Plenum Ventilation System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The Nonaccessible Area Exhaust Filter Plenum Ventilation System is required to be OPERABLE to ensure that atmospheric releases from the ECCS pump rooms do not exceed those assumed in the safety analysis. Total system failure could result in the atmospheric release, 1) exceeding 10 CFR 50.67 limits in the event of a Design Basis Accident (DBA), and 2) exceeding the limits for control room habitability. The Nonaccessible Area Exhaust Filter Plenum Ventilation System is considered OPERABLE when the individual components, necessary to maintain ECCS pump rooms and equipment rooms filtration are OPERABLE.

In order for the Nonaccessible Area Exhaust Filter Plenum Ventilation System to perform its function, filtration and motive flow must be provided by two of the three trains, the bypass path(s) to the normal auxiliary building exhaust system must be isolated, and the third train's inlet damper must be closed. The closure of the third train's inlet damper, prevents starting of a third fan and also ensures filtration of the exhaust from the ECCS pump rooms, by eliminating potential bypass flow paths.

Three trains of the Nonaccessible Area Exhaust Filter Plenum Ventilation System are required to be OPERABLE to ensure that at least two are available, assuming a single failure coincident with loss of offsite power on the affected unit and an orderly shutdown on the other unit. In addition due to design considerations, two of the trains must be aligned for operation and one train must be aligned in standby (i.e., the inlet damper closed).

To accommodate the single failure and loss of offsite power assumptions, the required fans in each of the Nonaccessible Area Exhaust Filter Plenum Ventilation System trains must be independent of the credited fans in the other trains.

BASES

SURVEILLANCE REQUIREMENTS (continued)

If a particular pump room is isolated such that there is no potential for post accident fluids to pass through the room, or that room's ECCS equipment is not required, that room can be excluded from meeting the acceptance criteria of the SR. Performance of this SR with a room excluded, represents a change in the ECCS pump room area volume that the system is maintaining at a negative pressure. Prior to the room being put back in service, this SR would have to be performed with the new volume, to assure that the system can maintain the entire volume at the required negative pressure.

The 18 month Frequency on a STAGGERED TEST BASIS is consistent with that specified in Reference 6.

REFERENCES

1. UFSAR, Section 6.5.1.
2. UFSAR, Section 9.4.5.
3. UFSAR, Section 15.6.5.
4. 10 CFR 50.67.
5. UFSAR, Section 6.4.
6. Regulatory Guide 1.52 (Rev. 2).
7. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.

B 3.7 PLANT SYSTEMS

B 3.7.13 Fuel Handling Building Exhaust Filter Plenum (FHB) Ventilation System

BASES

BACKGROUND

The FHB Ventilation System is available to filter airborne radioactive particulates from the area of the fuel pool following a fuel handling accident. The FHB Ventilation System, in conjunction with other normally operating systems, also provides environmental control of temperature in the fuel pool area.

The FHB Ventilation System is a subsystem of the common auxiliary building heating, ventilation, and air conditioning system (VA). Each unit has two VA supply and two VA exhaust fans. The VA supply and exhaust fans are not required for FHB Ventilation System OPERABILITY.

The FHB Ventilation System consists of two independent and redundant trains. Each train 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 follows the adsorber section to collect carbon fines and provide backup in case the main HEPA filter bank fails. The downstream HEPA filter is not credited in the analysis, but serves to collect charcoal fines, and to back up the upstream HEPA filter should it develop a leak. The system initiates filtered ventilation of the fuel handling building following receipt of a high radiation signal or a Safety Injection (SI) on either unit.

BASES

BACKGROUND (continued)

The FHB Ventilation System is a standby system. During normal operation flow from the fuel handling building is routed through the FHB Ventilation System prefilters and HEPA filters and then through the VA exhaust plenum via the VA exhaust fans. Upon FHB Ventilation System actuation (emergency mode of operation), the bypass dampers close, and the FHB Ventilation System fans start, drawing air through the FHB Ventilation System charcoal filters. The prefilters remove any large particles in the air to prevent excessive loading of the HEPA filters and charcoal adsorbers. The FHB Ventilation System may also be initiated manually.

The FHB Ventilation System is discussed in the UFSAR, Sections 6.5.1, 9.4.5, and 15.7.4 (Refs. 1, 2, and 3, respectively).

APPLICABLE SAFETY ANALYSES

The FHB Ventilation System design basis is established by the consequences of the limiting Design Basis Accident (DBA), which is a fuel handling accident involving handling RECENTLY IRRADIATED FUEL. The analysis of the fuel handling accident, given in Reference 3, assumes that all fuel rods in an assembly are damaged. The DBA analysis of the fuel handling accident assumes that only one train of the FHB Ventilation System is functional due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the one remaining train of this filtration system. The amount of fission products available for release from the fuel handling building is determined for a fuel handling accident. The accident analyses assume a 90% filter efficiency for elemental iodine and a 70% filter efficiency for methyl iodine. Due to radioactive decay, the FHB Ventilation System is only required to isolate during fuel handling accidents involving handling RECENTLY IRRADIATED FUEL. These assumptions and analyses follow the guidance provided in Regulatory Guide 1.183 (Ref. 4).

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

BASES

LCO

Two independent and redundant trains of the FHB Ventilation System are required to be OPERABLE to ensure that at least one train is available, assuming a single failure that disables the other train, coincident with a loss of offsite power. Total system failure could result in the atmospheric release from the fuel handling building exceeding the 10 CFR 50.67 (Ref. 5) limits in the event of a fuel handling accident involving handling RECENTLY IRRADIATED FUEL.

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

a. Fan is OPERABLE;

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

c. Ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

APPLICABILITY

During movement of RECENTLY IRRADIATED FUEL in the fuel handling building, the FHB Ventilation System is required to be OPERABLE to alleviate the consequences of a fuel handling accident.

During movement of RECENTLY IRRADIATED FUEL in the containment with the containment equipment hatch not intact, the FHB Ventilation System is required to be OPERABLE to mitigate the consequences of an accident inside containment. The equipment hatch is considered not intact if both personnel air lock doors associated with the equipment hatch are open or the hatch is not held in place with at least four bolts.

BASES

ACTIONS

The Actions Table is modified by a Note indicating that LCO 3.0.3 does not apply. If moving RECENTLY IRRADIATED FUEL assemblies in the fuel handling building while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving RECENTLY IRRADIATED FUEL assemblies while in MODES 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of RECENTLY IRRADIATED FUEL assemblies is not sufficient to require a reactor shutdown.

A.1

With one FHB Ventilation System train inoperable, action must be taken to restore OPERABLE status within 7 days. During this period, the remaining OPERABLE train is adequate to perform the FHB Ventilation System function. The 7 day Completion Time is based on the risk from an event occurring requiring the inoperable FHB Ventilation System train, and the remaining FHB Ventilation System train providing the required protection.

B.1.1, B.1.2, B.2.1, and B.2.2

When Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE FHB Ventilation System train must be placed in the emergency mode or RECENTLY IRRADIATED FUEL movement suspended. This action ensures that the remaining train is OPERABLE, that no undetected failures preventing system operation will occur, and that any active failure will be readily detected. Required Action B.1.2 requires the FHB Ventilation System train placed in operation be capable of being powered by an OPERABLE emergency power source. This action assures availability of electric power in the unlikely event of a loss of offsite power. This power source can be from Unit 1 or Unit 2, via OPERABLE crosstie breakers.

If the system is not placed in the emergency mode, Action B.2.1 requires suspension of RECENTLY IRRADIATED FUEL movement in the fuel handling building, which precludes a fuel handling accident involving handling RECENTLY IRRADIATED FUEL in the fuel handling building. This does not preclude the movement of fuel assemblies to a safe position.

BASES

ACTIONS (continued)

Required Action B.2.2 requires suspension of movement of RECENTLY IRRADIATED FUEL assemblies inside containment, precluding an accident that might require actuation of the FHB Ventilation System (when the equipment hatch is not intact).

Required Action B.2.2 is modified by a Note which indicates that this Required Action is only required if the equipment hatch is not intact. If the hatch is intact, only Required Action B.2.1 is required.

C.1 and C.2

When two trains of the FHB Ventilation System are inoperable action must be taken to place the unit in a condition in which the LCO does not apply. Action must be taken immediately to suspend movement of RECENTLY IRRADIATED FUEL assemblies in the fuel handling building. This does not preclude the movement of fuel to a safe position.

Required Action C.2 requires suspension of movement of RECENTLY IRRADIATED FUEL assemblies inside containment, precluding an accident that might require actuation of the FHB Ventilation System (when the equipment hatch is not intact).

Required Action C.2 is modified by a Note which indicates that this Required Action is only required if the equipment hatch is not intact. If the hatch is intact, only Required Action C.1 is required.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.13.3

This SR verifies the integrity of the fuel handling building and containment enclosure. The ability of the fuel handling building and containment to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the FHB Ventilation System and enclosure integrity. During the emergency mode of operation the FHB Ventilation System is designed to maintain a slight negative pressure in the fuel handling building to prevent unfiltered leakage. The FHB Ventilation System is designed to maintain a ≤ -0.25 inches water gauge with respect to atmospheric pressure. The Frequency of 7 days on a STAGGERED TEST BASIS, is based on the increased containment activity that occurs when the equipment hatch is not intact, that could affect containment integrity.

This SR is modified by a Note that requires this SR only during movement of RECENTLY IRRADIATED FUEL assemblies (in the fuel building or in the containment) when the equipment hatch is not intact.

SR 3.7.13.4

This SR verifies that each FHB Ventilation System train aligns, starts, and operates on an actual or simulated actuation signal. The 18 month Frequency is consistent with Reference 6.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.13.5

This SR verifies the integrity of the fuel handling building enclosure. The ability of the fuel handling building to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the FHB Ventilation System. During the emergency mode of operation the FHB Ventilation System is designed to maintain a slight negative pressure in the fuel handling building, to prevent unfiltered leakage. The FHB Ventilation System is designed to maintain a ≤ -0.25 inches water gauge with respect to atmospheric pressure at a flow rate ≤ 23,100 cfm to the fuel handling building. The Frequency of 18 months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 7).

An 18 month Frequency (on a STAGGERED TEST BASIS) is consistent with Reference 6.

This SR is modified by a Note that requires this SR only during movement of RECENTLY IRRADIATED FUEL assemblies in the fuel handling building when the equipment hatch is intact.

REFERENCES

1. UFSAR, Section 6.5.1.
2. UFSAR, Section 9.4.5.
3. UFSAR, Section 15.7.4.
4. Regulatory Guide 1.183, July 2000.
5. 10 CFR 50.67.
6. Regulatory Guide 1.52 (Rev. 2).
7. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.

B 3.7 PLANT SYSTEMS

B 3.7.14 Spent Fuel Pool Water Level

BASES

BACKGROUND	The minimum water level in the spent fuel pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel. A general description of the spent fuel pool design is given in the UFSAR, Section 9.1.2 (Ref. 1). A description of the Spent Fuel Pool Cooling and Cleanup System is given in the UFSAR, Section 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in the UFSAR, Section 15.7.4 (Ref. 3).
APPLICABLE SAFETY ANALYSES	The minimum water level in the spent fuel pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.183 (Ref. 4). The resultant Total Effective Dose Equivalent (TEDE) dose is within 10 CFR 50.67 limits (Ref. 5). According to Reference 4, there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool water surface during a fuel handling accident. With 23 ft of water, the assumptions of Reference 4 can be used directly. In practice, this LCO preserves the assumption for the bulk of the fuel in the storage racks. In the case of a single bundle dropped and lying horizontally on top of the spent fuel racks, however, there may be < 23 ft of water above the width of the bundle. To offset this small nonconservatism, the analysis assumes that all fuel rods fail, although analysis shows that only the first few rows fail from a hypothetical maximum drop. The spent fuel pool water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

This SR verifies sufficient spent fuel pool water is available in the event of a fuel handling accident. The water level in the spent fuel pool must be checked periodically. The 7 day Frequency is appropriate because the volume in the pool is normally stable. Water level changes are controlled by plant procedures and are acceptable based on operating experience.

During refueling operations, the level in the spent fuel pool is in equilibrium with the refueling cavity when they are hydraulically coupled, and the level in the refueling cavity is checked daily in accordance with SR 3.9.7.1.

REFERENCES

1. UFSAR, Section 9.1.2.
2. UFSAR, Section 9.1.3.
3. UFSAR, Section 15.7.4.
4. Regulatory Guide 1.183, July 2000.
5. 10 CFR 50.67.

B 3.9 REFUELING OPERATIONS

B 3.9.4 Containment Penetrations

BASES

BACKGROUND

During movement of RECENTLY IRRADIATED FUEL assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODES 5 and 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are filtered, closed, or capable of being closed. Since there is no significant potential for containment pressurization, the 10 CFR 50, Appendix J, leakage criteria and tests are not required.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR 50.67 (Ref 5). In addition, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

BASES

BACKGROUND (continued)

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During movement of RECENTLY IRRADIATED FUEL assemblies within containment with the equipment hatch installed, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts be approximately equally spaced. During movement of RECENTLY IRRADIATED FUEL assemblies within containment and the equipment hatch not intact, the OPERABILITY requirements of the Fuel Handling Building Exhaust Filter Plenum (FHB) Ventilation System must be met. The OPERABILITY requirements of the FHB Ventilation System are provided in TS 3.7.13, "Fuel Handling Building Exhaust Filter Plenum (FHB) Ventilation System."

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 in accordance with LCO 3.6.2, "Containment Air Locks." The two air locks are the personnel air lock and the emergency air lock. Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During movement of RECENTLY IRRADIATED FUEL assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain closed. An exception, however, is provided for the personnel air lock. It is acceptable to have both doors of the personnel air lock opened simultaneously provided the FHB Ventilation System is in compliance with TS 3.7.13.

The closure restrictions are sufficient to restrict unfiltered fission product radioactivity releases from containment to the environment due to a fuel handling accident involving RECENTLY IRRADIATED FUEL during refueling.

BASES

BACKGROUND (continued)

The Containment Ventilation Isolation System consists of the normal purge subsystem, the mini purge subsystem, and the post Loss Of Coolant Accident purge subsystem. These three subsystems contain penetrations which provide direct access from the containment to the outside atmosphere. In MODE 6, the minipurge subsystem is normally used to exchange large volumes of containment air to support refueling operations. Each penetration contains inside and outside containment isolation valves which close automatically on an actuation signal. During movement of RECENTLY IRRADIATED FUEL within containment, all required valves within a subsystem must be capable of being closed by a containment ventilation isolation signal whenever the associated subsystem is in operation. A list of the instrumentation which functions to isolate the valves in these penetrations is provided in LCO 3.3.6, "Containment Ventilation Isolation Instrumentation."

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by a closed automatic isolation valve, a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods allowed under the provisions of 10 CFR 50.59 may include use of a material that can provide a temporary atmospheric pressure ventilation barrier during movement of RECENTLY IRRADIATED FUEL within the containment.

BASES

APPLICABLE SAFETY ANALYSES	During movement of RECENTLY IRRADIATED FUEL assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 1). Fuel handling accidents, analyzed in Reference 2, include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.7, "Refueling Cavity Water Level," ensure that the release of fission product radioactivity, subsequent to a fuel handling accident in containment, results in doses that are within the 10 CFR 50.67 (Ref. 5) limits. The radiological dose assessments for the Design Basis Fuel Handling Accident in containment were performed in accordance with the guidance of Regulatory Guide 1.183 (Ref 4). When moving RECENTLY IRRADIATED FUEL in containment, the requirements of the LCO must be met with the exception that LCO 3.9.4.a need not be met if at least one train of the FHB Ventilation System is OPERABLE as specified in the Note. This exception permits movement of RECENTLY IRRADIATED FUEL with both personnel air lock doors open or the equipment hatch not intact. When moving fuel in the containment that is not RECENTLY IRRADIATED FUEL, the LCO is not applicable. Due to radioactive decay, neither containment closure nor an OPERABLE FHB Ventilation System train are required to meet the dose limits of 10 CFR 50.67 during a fuel handling accident. Another consideration, which may result in a limiting decay time prior to fuel handling, is the impact of decay heat on the spent fuel pool cooling requirements described in Reference 3. Containment penetrations satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	This LCO limits the consequences of a fuel handling accident involving handling RECENTLY IRRADIATED FUEL in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge (supply and exhaust) penetrations. For the OPERABLE containment purge

BASES

LCO (continued)

penetrations, this LCO ensures that unisolated penetrations are isolable by the Containment Ventilation Isolation System. The OPERABILITY requirements for this LCO ensure the automatic purge valve closure times specified in the UFSAR can be achieved and, therefore, meet the assumptions used in the safety analysis to ensure that releases through the valves are terminated, such that radiological doses are within the acceptance limits.

The LCO is modified by a Note which allows both personnel air lock doors to be open or the equipment hatch not intact when the FHB Ventilation System is in compliance with TS 3.7.13. When the equipment hatch is installed it serves to contain fission product radioactivity that may be released following a fuel handling accident in the containment. When the equipment hatch is not intact, or when both doors of the personnel air lock are simultaneously opened, the internal containment pressure is essentially equal to the internal pressure of the fuel handling building. In the event of a fuel handling accident in the containment, realigning of the fuel handling building ventilation system creates a negative pressure in the containment and fuel handling building relative to the auxiliary building and outside atmosphere. The negative pressure ensures that any radioactivity released to the containment atmosphere will either remain in the containment or be filtered through a FHB Ventilation System train. As such, with the equipment hatch not intact, or with both personnel air lock doors open, the consequences of a fuel handling accident involving RECENTLY IRRADIATED FUEL in containment would not exceed those calculated for a fuel handling accident involving RECENTLY IRRADIATED FUEL in the fuel handling building.

In addition, a commitment has been made to implement compensatory measures during movement of irradiated fuel as described in UFSAR Section 15.7.4, "Fuel Handling Accidents." These compensatory measures support the Alternate Source Term methodology and reduce doses even further below that provided by natural decay and avoid unmonitored releases in the event of a postulated fuel handling accident.

BASES

APPLICABILITY

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

ACTIONS

A.1

If the containment equipment hatch, air lock doors, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, the unit must be placed in a condition where containment closure is not needed. This is accomplished by immediately suspending movement of RECENTLY IRRADIATED FUEL assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that each of the containment penetrations required to be isolated is isolated. This Surveillance for the open purge valves demonstrates that the valves are not blocked from closing. Also the Surveillance will demonstrate that each valve operator has motive power which will ensure that each valve is capable of being closed by an OPERABLE automatic Containment Ventilation Isolation signal.

The Surveillance is performed every 7 days during movement of RECENTLY IRRADIATED FUEL assemblies within containment. As such, this Surveillance ensures that a postulated fuel handling accident involving RECENTLY IRRADIATED FUEL that releases fission product radioactivity within the containment will not result in a release of fission product radioactivity to the environment.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.9.4.2

This Surveillance demonstrates that each required containment purge valve actuates to its isolation position on an actual or simulated high radiation signal. In TS 3.3.6, the Containment Ventilation Isolation instrumentation requires a CHANNEL CHECK every 12 hours and a COT every 92 days to ensure the channel OPERABILITY during refueling operations. Every 18 months a CHANNEL CALIBRATION is performed. SR 3.9.4.3 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. These Surveillances will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

SR 3.9.4.3

This Surveillance demonstrates that the isolation time of each required containment purge valve providing direct access from the containment atmosphere to the outside atmosphere is in accordance with the Inservice Testing Program requirements. This SR, along with SR 3.9.4.2, ensures the containment purge valves in penetrations which provide direct access from the containment atmosphere to the outside atmosphere are capable of closing after a postulated fuel handling accident to limit the release of fission product radioactivity from the containment.

REFERENCES

1. UFSAR, Section 15.7.4.
2. NUREG-0800, Section 15.0.1, Rev. 0, July 2000.
3. NUREG-0800, Section 9.1.3.
4. Regulatory Guide 1.183, July 2000.
5. 10 CFR 50.67.

B 3.9 REFUELING OPERATIONS

B 3.9.7 Refueling Cavity Water Level

BASES

BACKGROUND

The movement of irradiated fuel assemblies within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange. This requirement ensures a sufficient level of water is maintained in the refueling cavity to retain iodine fission product activity resulting from a fuel handling accident in containment (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to within 10 CFR 50.67 limits (Ref. 3).

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies, the water level in the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment, as postulated by Regulatory Guide 1.183 (Ref. 1).

The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and doses are maintained within allowable limits (Refs. 2 and 3).

The minimum decay time prior to allowing fuel handling is addressed in the fuel handling accident dose analysis. However, another consideration, which may result in a limiting decay time prior to fuel handling, is the impact of decay heat on the spent fuel pool cooling requirements described in Reference 4.

Refueling cavity water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO	A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits.
APPLICABILITY	LCO 3.9.7 is applicable when moving irradiated fuel assemblies within containment. The LCO ensures a sufficient level of water is present in the refueling cavity to minimize the radiological consequences of a fuel handling accident in containment. If irradiated fuel assemblies are not present in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.14, "Spent Fuel Pool Water Level."
ACTIONS	<u>A.1</u> With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving movement of irradiated fuel assemblies within the containment shall be suspended immediately to ensure that a fuel handling accident cannot occur. The suspension of irradiated fuel movement shall not preclude completion of movement of a fuel assembly to a safe position.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1

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

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

REFERENCES

1. Regulatory Guide 1.183, July 2000.
2. UFSAR, Section 15.7.4.
3. 10 CFR 50.67.
4. NUREG-0800, Section 9.1.3.

ATTACHMENT 4B

BRAIDWOOD STATION
UNITS 1 AND 2

Docket Nos. 50-456 and 50-457
License Nos. NPF-72 and NPF-77

License Amendment Request
"Alternative Source Term Implementation"

Typed Technical Specification Bases Pages
(For information only)

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B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the Reactor Coolant Pressure Boundary (RCPB) design conditions are not to be exceeded during normal operation and Anticipated Operational Occurrences (AOOs). Also, in accordance with GDC 28, "Reactivity Limits" (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.

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

Overpressurization of the RCS could result in a breach of the RCPB reducing the number of protective barriers designed to prevent radioactive releases from exceeding the limits specified in 10 CFR 50.67, “Accident Source Term” (Ref. 4). If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere.

BASES

SAFETY LIMIT VIOLATIONS (continued)

If SL 2.1.2 is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. If the Completion Time is exceeded, actions shall continue in order to reduce pressure to less than the SL. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.
2. ASME, Boiler and Pressure Vessel Code, SECTION III, Article NB-7000.
3. ASME, Boiler and Pressure Vessel Code, SECTION XI.
4. 10 CFR 50.67.
5. UFSAR, Section 5.2.2.
6. UFSAR, Section 7.2.

BASES

LCO

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

The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the SDM value of the LCO. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 50.67, "Accident Source Term," limits (Ref. 7). For the boron dilution accident, if the LCO is violated, the minimum required time assumed for operator action to terminate dilution may no longer be applicable.

APPLICABILITY

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

ACTIONS

A.1

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

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the boric acid storage tank or the refueling water storage tank. The operator should borate with the best source available for the unit conditions.

BASES

SURVEILLANCE REQUIREMENTS (continued)

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

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. UFSAR, Section 15.1.5.
3. UFSAR, Section 15.4.6.
4. UFSAR, Section 15.4.1.
5. UFSAR, Section 15.4.2.
6. UFSAR, Section 15.4.8.
7. 10 CFR 50.67.

B 3.3 INSTRUMENTATION

B 3.3.1 Reactor Trip System (RTS) Instrumentation

BASES

BACKGROUND

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

The protection and monitoring systems have been designed to assure safe operation of the reactor. This is achieved by specifying Limiting Safety System Settings (LSSS) in terms of parameters directly monitored by the RTS, as well as specifying LCOs on other reactor system parameters and equipment performance.

The LSSS, defined in this specification as the Allowable Values, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits during Design Basis Accidents (DBAs).

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

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

2. Fuel centerline melt shall not occur; and

3. The RCS pressure SL of 2735 psig shall not be exceeded.

Operation within the SLs of Specification 2.0, "Safety Limits (SLs)," also maintains the above values and assures that dose will be within the 10 CFR 50.67 and 10 CFR 100 limits during AOOs.

BASES

BACKGROUND (continued)

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

The RTS instrumentation is segmented into four distinct but interconnected modules as identified below. The RTS process is illustrated in UFSAR, Chapter 7 (Ref. 1):

1. Field transmitters or process sensors: provide a measurable electronic signal based upon the physical characteristics of the parameter being measured;

2. Signal Process Control and Protection System, including Analog Protection System, Nuclear Instrumentation System (NIS), field contacts, and protection channel sets: provide signal conditioning, bistable setpoint comparison, process algorithm actuation, compatible electrical signal output to protection system devices, and control board/control room/miscellaneous indications;

3. Solid State Protection System (SSPS), including input, logic, and output bays: initiates proper unit shutdown and/or ESF actuation in accordance with the defined logic, which is based on the bistable outputs from the signal process control and protection system; and

4. Reactor trip switchgear, including Reactor Trip Breakers (RTBs) and bypass breakers: provides the means to interrupt power to the Control Rod Drive Mechanisms (CRDMs) and allows the Rod Cluster Control Assemblies (RCCAs), or "rods," to fall into the core and shut down the reactor. The bypass breakers allow testing of the RTBs at power.

BASES

APPLICABLE SAFETY ANALYSES

The safety analyses assume that the containment remains intact with penetrations unnecessary for core cooling isolated early in the event (i.e., within approximately 60 seconds). The isolation of the purge valves has not been analyzed mechanistically in the dose calculations, although its rapid isolation is assumed. The containment ventilation isolation radiation monitors act as backup to the SI signal to ensure closing of the purge valves. They are also the primary means for automatically isolating containment in the event of a fuel handling accident. Containment isolation in turn ensures meeting the containment leakage rate assumptions of the safety analyses, and ensures that the calculated accidental radiological doses are below 10 CFR 50.67, "Accident Source Term" (Ref. 1) limits.

The containment ventilation isolation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requirements ensure that the instrumentation necessary to initiate Containment Ventilation Isolation, listed in Table 3.3.6-1, is OPERABLE.

1. <u>Manual Initiation - Phase A</u>

   Refer to LCO 3.3.2, Function 3.a.1, for all initiating Functions and requirements.

2. <u>Manual Initiation - Phase B</u>

   Refer to LCO 3.3.2, Function 3.b.1, for all initiating Functions and requirements.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.6

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

REFERENCES

1. 10 CFR 50.67.
2. NUREG-1366, December 1992.
3. WCAP-13877, Revision 2-P, "Reliability Assessment of Westinghouse Type AR Relays Used as SSPS Slave Relays," October 1999.
4. WCAP-13878-P, Revision 2, "Reliability Assessment of Potter & Brumfield MDR Series Relays," October 1999.
5. WCAP-13900, Revision 0, "Extension of Slave Relay Surveillance Test Intervals," April 1994.

B 3.3 INSTRUMENTATION

B 3.3.7 Control Room Ventilation (VC) Filtration System Actuation Instrumentation

BASES

BACKGROUND

The VC Filtration System provides an enclosed control room environment from which the unit can be operated following an uncontrolled release of radioactivity. During normal operation, the VC Filtration System provides control room ventilation. Upon receipt of an actuation signal, the VC Filtration System initiates filtered ventilation and pressurization of the control room. This system is described in the Bases for LCO 3.7.10, "Control Room Ventilation (VC) Filtration System."

The actuation instrumentation consists of two high radiation channels in each of the outside air intakes. A high radiation (gaseous) signal from one of two channels will initiate its associated train of the VC Filtration System and trip the Control Room Offices HVAC (VV), Laboratory HVAC (VL), and Radwaste Building Ventilation (VW) Systems. The VC Filtration System is also actuated by a Safety Injection (SI) signal. The SI Function is discussed in LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation."

The radiological dose assessments performed for the applicable Design Basis Accidents (DBAs) assume initiation of the VC Filtration System within 30 minutes.

APPLICABLE SAFETY ANALYSES

The control room must be kept habitable for the operators stationed there during accident recovery and post accident operations.

The VC Filtration System acts to terminate the supply of unfiltered outside air to the control room, initiate filtration, and pressurize the control room. These actions are necessary to ensure the control room is kept habitable for the operators stationed there during accident recovery and post accident operations by minimizing the radiation exposure of control room personnel.

In MODES 1, 2, 3, and 4, the radiation monitor actuation of the VC Filtration System provides a protected environment from which operators can control the unit following a DBA.

BASES

ACTIONS (continued)

C.1 and C.2

Condition C applies when the Required Action and associated Completion Time of Condition A or B have not been met and the unit is in MODE 1, 2, 3, or 4. The unit must be brought to a MODE in which the likelihood of an event requiring the VC Filtration System is minimized. To achieve this status, the unit must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1

Condition D applies when the Required Action and associated Completion Time of Condition A or B have not been met when irradiated fuel assemblies are being moved. Movement of irradiated fuel assemblies must be suspended immediately to reduce the risk of accidents that would require VC Filtration System actuation.

E.1

Condition E applies when the Required Action and associated Completion Time of Condition A or B have not been met in MODE 5 or 6. Actions must be initiated immediately to restore the inoperable train(s) to OPERABLE status to provide protection from significant radioactivity releases.

B 3.3 INSTRUMENTATION

B 3.3.8 Fuel Handling Building Exhaust Filter Plenum (FHB) Ventilation System Actuation Instrumentation

BASES

BACKGROUND

The FHB Ventilation System ensures that radioactive materials in the fuel handling building atmosphere following a fuel handling accident involving RECENTLY IRRADIATED FUEL are filtered and adsorbed prior to exhausting to the environment. The system is described in the Bases for LCO 3.7.13, "Fuel Handling Building Exhaust Filter Plenum (FHB) Ventilation System." The system initiates filtered ventilation of the fuel handling building automatically following receipt of a high radiation signal or safety injection signal.

Two radiation monitoring channels (0RE-AR055 and 0RE-AR056) provide input to the FHB Ventilation System isolation. A high radiation signal from 0RE-AR055 initiates Train A FHB Ventilation System isolation. A high radiation signal from 0RE-AR056 initiates Train B FHB Ventilation System isolation. High radiation detected by any monitor initiates fuel handling building isolation and starts the FHB Ventilation System. These actions function to prevent exfiltration of contaminated air by initiating filtered ventilation, which imposes a negative pressure on the fuel handling building.

For a fuel handling accident, involving irradiated fuel assemblies other than RECENTLY IRRADIATED FUEL assemblies, the radiological dose assessments did not credit an OPERABLE FHB Ventilation System. Given the fuel handling accidents described above, the dose limits of 10 CFR 50.67 are not exceeded.

APPLICABLE SAFETY ANALYSES

The FHB Ventilation System ensures that radioactive materials in the fuel handling building atmosphere following a fuel handling accident involving RECENTLY IRRADIATED FUEL are filtered and adsorbed prior to being exhausted to the environment. This action reduces the radioactive content in the fuel handling building exhaust following a fuel handling accident so that doses remain within the limits specified in 10 CFR 50.67 (Ref. 1).

The FHB Ventilation System actuation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO	The LCO requires two channels to ensure that the radiation monitoring instrumentation necessary to initiate the FHB Ventilation System remains OPERABLE.
APPLICABILITY	High radiation initiation of the FHB Ventilation System must be OPERABLE during movement of RECENTLY IRRADIATED FUEL assemblies in the fuel handling building to ensure automatic initiation of the FHB Ventilation System when the potential for a fuel handling accident exists. Due to radioactive decay, the FHB Ventilation System instrumentation is only required to be OPERABLE during fuel handling involving handling RECENTLY IRRADIATED FUEL. During movement of RECENTLY IRRADIATED FUEL assemblies with the containment equipment hatch not intact, the FHB Ventilation System actuation instrumentation is required to be OPERABLE to alleviate the consequences of an accident inside containment. The containment equipment hatch "not intact" refers to the requirement to have one door in the personnel air lock closed and the equipment hatch closed and held in place by a minimum of four bolts as described in the Bases for LCO 3.9.4, "Containment Penetrations." While in MODES 1, 2, 3, 4, 5, and 6 without fuel handling involving handling RECENTLY IRRADIATED FUEL in progress, the FHB Ventilation System instrumentation need not be OPERABLE since a fuel handling accident involving RECENTLY IRRADIATED FUEL cannot occur.
ACTIONS	The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by plant specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a COT, when the process instrumentation is set up for adjustment to bring it within specification. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

BASES

ACTIONS (continued)

A Note has been added to the ACTIONS to clarify the application of LCO 3.0.3. LCO 3.0.3 is not applicable while in MODE 5 or 6. If moving RECENTLY IRRADIATED FUEL assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving RECENTLY IRRADIATED FUEL assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of RECENTLY IRRADIATED FUEL assemblies would not be sufficient reason to require a reactor shutdown.

A.1

Condition A applies to the failure of a single radiation monitor channel. If one channel is inoperable, a period of 7 days is allowed to restore it to OPERABLE status. The 7 day Completion Time is the same as is allowed if one train of the mechanical portion of the system is inoperable. The basis for this time is the same as that provided in LCO 3.7.13.

B.1, B.2.1, and B.2.2

Condition B applies if the Required Action or associated Completion Time of Condition A is not met or the failure of two radiation monitors. If the train cannot be restored to OPERABLE status, one FHB Ventilation System train must be immediately placed in the emergency mode. The FHB Ventilation System train placed in operation must be capable of being powered by an OPERABLE emergency power source. This accomplishes the actuation instrumentation function and places the unit in a conservative mode of operation.

BASES

ACTIONS (continued)

Alternative actions may be taken if the FHB Ventilation System train is not placed in emergency mode or does not have an associated OPERABLE diesel generator. Required Action B.2.1 requires the suspension of fuel movement of RECENTLY IRRADIATED FUEL assemblies in the Fuel Handling Building, precluding a fuel handling accident. Required Action B.2.2 requires suspending movement of RECENTLY IRRADIATED FUEL assemblies inside containment, precluding an accident that would require FHB Ventilation System actuation when the equipment hatch is not intact. These actions do not preclude the movement of fuel assemblies to a safe position.

Required Action B.2.2 is modified by a Note which indicates that this Required Action is only required if the equipment hatch is not intact. If the hatch is intact, only Required Action B.2.1 is required.

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.8-1 determines which SRs apply to which Fuel Handling Building (FHB) Radiation Actuation Functions.

SR 3.3.8.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Agreement criteria are determined, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.8.2

A COT is performed once every 92 days on each required channel to ensure the entire channel will perform the intended function. This test verifies the capability of the instrumentation to provide the FHB Ventilation System actuation. The setpoints shall be left consistent with the plant specific calibration procedure tolerance. The Frequency of 92 days is based on the known reliability of the monitoring equipment and has been shown to be acceptable through operating experience.

SR 3.3.8.3

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

REFERENCES

1. 10 CFR 50.67.

BASES

BACKGROUND (continued)

This LCO deals with protection of the Reactor Coolant Pressure Boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a Loss Of Coolant Accident (LOCA). However, the ability to monitor leakage provides advance warning to permit unit shutdown before a LOCA occurs. This advantage has been shown by "leak before break" studies.

APPLICABLE SAFETY ANALYSIS

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

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a Main Steam Line Break (MSLB). The leakage contaminates the secondary fluid. Other accidents or transients involving secondary steam release to the atmosphere are the Steam Generator Tube Rupture (SGTR), Control Rod Ejection, and the Locked Rotor event (note that the Locked Rotor analysis assumes a concurrent Steam Generator (SG) Power Operated Relief Valve (PORV) failure). The MSLB is more limiting than the SGTR, Control Rod Ejection and Locked Rotor event for main control room radiation dose.

The UFSAR (Ref. 3) analysis for 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.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The dose consequences resulting from the MSLB, SGTR, Control Rod Ejection and Locked Rotor accidents are within the limits defined in 10 CFR 50.67 (Ref. 5).

To support the use of sleeving techniques for SG tube repair, the Unit 1 primary to secondary leakage limits are conservatively reduced from 500 gpd for any single SG and 1 gpm total to 150 gpd for any single SG and 600 gpd total (Ref. 4).

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

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

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

BASES

LCO (continued)

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump discharge flow monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled Reactor Coolant Pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through All SGs

Total primary to secondary LEAKAGE amounting to 600 gallons per day through all SGs not isolated from the RCS produces acceptable doses for all applicable Design Basis Accidents (DBAs). Violation of this LCO could result in exceeding the dose limits specified in 10 CFR 50.67 for the applicable DBAs . Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.13.2

This SR provides the means necessary to determine SG OPERABILITY. The requirement to demonstrate SG tube integrity in accordance with the Steam Generator Tube Surveillance Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
2. Regulatory Guide 1.45, May 1973.
3. UFSAR, Chapter 15.
4. Safety Evaluation Report, dated May 7, 1994.
5. 10 CFR 50.67.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND

For the bounding accidents specified in Regulatory Guide (RG) 1.183 (Ref. 7), the maximum Total Effective Dose Equivalent (TEDE) that an individual at the exclusion area boundary, at the outer boundary of the low population zone and in the control room can receive is specified in 10 CFR 50.67 (Ref. 6). The limits on specific activity ensure that the doses are held to a fraction of the 10 CFR 50.67 limits.

For other non-bounding transients and accidents analyzed in the Updated Final Safety Analysis Report (UFSAR), the maximum dose to the whole body and the thyroid that an individual at the site boundary can receive for 2 hours during an accident is specified in 10 CFR 100 (Ref. 1). Any future modification to the facility design bases for these events will use source term assumptions and radiological criteria in the affected analyses that are established in RG 1.183 and 10 CFR 50.67.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize 10 CFR 50.67 control room and offsite radioactivity dose consequences in the event of accidents such as a Main Steam Line Break (MSLB), Steam Generator Tube Rupture (SGTR), Control Rod Ejection or a Locked Rotor.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The allowable levels are intended to limit dose as specified by 10 CFR 50.67 and RG 1.183.

BASES

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensures that individual doses for all applicable design basis accidents will be within 10 CFR 50.67 limits and RG 1.183 guidance. These accident analyses assume the specific activity of the reactor coolant at the LCO limit and an existing reactor coolant Steam Generator (SG) tube leakage rate of 1 gpm, which conservatively bounds the limit specified in LCO 3.4.13.d. The safety analysis assumes the specific activity of the secondary coolant at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.3, "Secondary Specific Activity."

The bounding design basis accident analysis that establishes the acceptability of the limits for RCS specific activity is the MSLB Control Room Operator dose limit as specified in 10 CFR 50.67. Reference to this analysis is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The analysis is for two cases of reactor coolant specific activity. One case assumes specific activity at 1.0 μCi/gm DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the I-131 iodine release rate from the fuel to the coolant to a value 500 times greater than the release rate corresponding to the initial primary system iodine concentration. The second case assumes the initial reactor coolant iodine activity at 60.0 μCi/gm DOSE EQUIVALENT I-131 due to a pre-accident iodine spike caused by an RCS transient. In both cases, the noble gas activity in the reactor coolant assumes 1% failed fuel, which closely equals the LCO limit of $100/\bar{E}$ μCi/gm for gross specific activity.

The radiological consequence of an MSLB event is described in UFSAR Section 15.1.5.3. The safety analysis shows the radiological consequences of a MSLB accident are within the Reference 6 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed the limits shown in Figure 3.4.16-1, in the applicable specification, for more than 48 hours. The safety analysis has concurrent and pre-accident iodine spiking levels up to 60.0 μCi/gm DOSE EQUIVALENT I-131.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

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

LCO

The specific iodine activity is limited to 1.0 μCi/gm DOSE EQUIVALENT I-131. The gross specific activity in the reactor coolant is limited to the number of μCi/gm equal to 100 divided by $\bar{E}$ (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on DOSE EQUIVALENT I-131 and the limit on gross specific activity ensure that individual doses for all design basis accidents will be within 10 CFR 50.67 limits and RG 1.183 guidance.

Violation of the Technical Specification may result in reactor coolant radioactivity levels that could, in the event of an MSLB, SGTR, Control Rod Ejection or Locked Rotor accident, lead to doses that exceed the 10 CFR 50.67 dose limits.

BASES

APPLICABILITY

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

For operation in MODE 3 with RCS average temperature < 500°F, and in MODES 4 and 5, the release of radioactivity in the event of a MSLB, SGTR, Control Rod Ejection or Locked Rotor accident is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves.

ACTIONS

A.1 and A.2

With the DOSE EQUIVALENT I-131 specific activity greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the limits of Figure 3.4.16-1 are not exceeded. The Completion Time of 4 hours provides sufficient time to obtain and analyze a sample. Sampling is done to continue to provide a trend.

The DOSE EQUIVALENT I-131 specific activity must be restored to within limits within 48 hours. The Completion Time of 48 hours is required, if the limit violation resulted from normal iodine spiking.

A Note to the Required Actions excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require unit shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the unit remains at, or proceeds to power operation.

BASES

ACTIONS (continued)

B.1

If the Required Action and associated Completion Time of Condition A is not met or if the DOSE EQUIVALENT I-131 specific activity is in the unacceptable region of Figure 3.4.16-1, the reactor must be brought to MODE 3 with RCS average temperature < 500°F within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

C.1

With the gross specific activity in excess of the allowed limit, the unit must be placed in MODE 3 with RCS average temperature < 500°F. This action lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves and prevents venting the SG to the environment in an SGTR, Control Rod Ejection or Locked Rotor event. MSLB releases are similarly limited with the exception of initial blowdown and leakage through the affected SG. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

SR 3.4.16.1 requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant at least once every 7 days. A gross radioactivity analysis consists of the quantitative measurement of the total specific activity of the reactor coolant except for radionuclides with half lives < 10 minutes and all radioiodines. The total specific activity is the sum of the degassed beta-gamma activity and the total of all identified gaseous activities in the sample within 2 hours after the sample was taken. Determination of the contributors to the gross specific activity are based upon those energy peaks identifiable with a 95% confidence level. The latest available data may be used for pure beta emitting radionuclides. This Surveillance provides an indication of any increase in gross specific activity.

BASES

REFERENCES

1. 10 CFR 100.11, 1973.
2. UFSAR, Section 15.6.3.
3. Safety Evaluation Report, dated May 7, 1994.
4. Safety Evaluation Report, dated August 18, 1994.
5. Safety Evaluation Report, dated November 9, 1995.
6. 10 CFR 50.67.
7. Regulatory Guide 1.183, dated July 2000.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.4.2

Heat traced portions of the RWST vent path should be verified every 24 hours to be within the temperature limit needed to prevent ice blockage and subsequent vacuum formation in the tank during rapid level decreases caused by accident conditions. This Frequency is sufficient to identify a temperature change that would approach the lower limit and has been shown to be acceptable through operating experience.

The SR is modified by a Note that eliminates the requirement to perform this Surveillance when the ambient air temperature is ≥ 35°F. With ambient air temperature above this limit, the RWST vent path will be free of ice blockage.

SR 3.5.4.3

The RWST water volume should be verified every 7 days to be above the required minimum level of 89% (useable volume of > 395,000 gallons) in order to ensure that a sufficient initial supply is available for injection and to support continued ECCS and Containment Spray System pump operation on recirculation. Since the RWST volume is normally stable and protected by a low level alarm, a 7 day Frequency is appropriate and has been shown to be acceptable through operating experience.

SR 3.5.4.4

The boron concentration of the RWST should be verified every 7 days to be within the required limits. This SR ensures that the reactor will remain subcritical following a LOCA and will limit the power level increase and subsequently returns the reactor to subcritical immediately following an MSLB. Further, it assures that the resulting sump pH will be maintained in an acceptable range so that boron precipitation in the core will not occur, sufficient iodine will be retained to limit doses, stress corrosion cracking of equipment will be minimized, and hydrogen production will be minimized. Since the RWST volume is normally stable, a 7 day sampling Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a LOCA and a steam line break (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA, secondary system pipe break, or fuel handling accident (Ref. 3). In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment leakage rate, used to evaluate doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting design basis LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.20% per day in the safety analysis at P_a = 42.8 psig for Unit 1 and P_a = 38.4 psig for Unit 2 (Ref. 3).

The radiological dose assessments performed for the design basis LOCA assume a maximum allowable containment leakage rate of 0.20% per day. In this case, the dose limits of 10 CFR 50.67 (Ref. 7) are not exceeded.

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

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

LCO

Containment OPERABILITY is maintained by limiting leakage to ≤ 1.0 L_a, except prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test. At this time, applicable leakage limits must be met.

Compliance with this LCO will ensure a containment configuration, including the equipment hatch, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. Failure to meet air lock and purge valve leakage limits specified in LCO 3.6.2 and LCO 3.6.3 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required leakage test is required to be $< 0.6\ L_a$ for combined Type B and C leakage and $< 0.75\ L_a$ for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0\ L_a$. At $\leq 1.0\ L_a$ the dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

SR 3.6.1.2

This SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and Frequency are consistent with the requirements of 10 CFR50.55a(b)(2)(vi) "Effective Edition and Addenda of Subsection IWE and Subsection IWL, SECTION XI" (Ref. 4), and Section 10 CFR50.55a(b)(2)(ix), "Examination of Concrete Containments" (Ref. 5). Predicted tendon lift off forces will be determined consistent with the recommendations of Regulatory Guide 1.35.1, (Ref. 6).

BASES

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
2. UFSAR, Chapter 15.
3. UFSAR, Section 6.2.
4. 10 CFR50.55a(b)(2)(vi) "Effective Edition and Addenda of Subsection IWE and Subsection IWL, SECTION XI."
5. 10 CFR50.55a(b)(2)(ix), "Examination of Concrete Containments."
6. Regulatory Guide 1.35.1, July 1990.
7. 10 CFR 50.67.

APPLICABLE SAFETY ANALYSES

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a Loss of Coolant Accident (LOCA) and a steam line break. In addition, release of significant fission product radioactivity within containment can occur from a LOCA, secondary system pipe break, or fuel handling accident. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment leakage rate, used to evaluate doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting design basis LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.20% per day in the safety analysis at P_a = 42.8 psig for Unit 1 and P_a = 38.4 psig for Unit 2 (Ref. 2).

The radiological dose assessments performed for the Design Basis LOCA assume a maximum allowable containment leakage rate of 0.20% per day. In this case, the dose limits of 10 CFR 50.67 (Ref. 3) are not exceeded.

The containment air locks satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Each containment air lock forms part of the containment pressure boundary. As part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into or exit from containment.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is not normally challenged when the containment air lock door is used for entry and exit (procedures require strict adherence to single door opening), this test is only required to be performed every 24 months. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, and the potential for loss of containment OPERABILITY if the Surveillance were performed with the reactor at power. The 24 month Frequency for the interlock is justified based on generic operating experience. The 24 month Frequency is based on engineering judgment and is considered adequate given that the interlock is not challenged during use of the air lock.

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
2. UFSAR, Section 6.2.
3. 10 CFR 50.67.

BASES

APPLICABLE SAFETY ANALYSES

The containment isolation valve LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during major accidents. As part of the containment boundary, containment isolation valve OPERABILITY supports leak tightness of the containment. Therefore, the safety analyses of any event requiring isolation of containment is applicable to this LCO.

In Modes 1 through 4, the DBAs that result in a release of radioactive material within containment are a Loss Of Coolant Accident (LOCA), secondary system pipe break, and rod ejection accident (Ref. 1). In the analyses for each of these accidents, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through containment isolation valves (including containment purge valves) are minimized. The safety analyses assume that the 48 inch purge valves are closed at event initiation.

In the calculation of control room and offsite doses following a LOCA, the accident analyses are performed in conformance with the requirements of Regulatory Guide 1.183 (Ref. 2)

The containment isolation valves ensure that the containment design leakage rate remains within L_a by automatically isolating penetrations that do not serve post accident functions and providing isolation capability for penetrations associated with safe shutdown functions. The maximum isolation time for automatic containment isolation valves is 60 seconds (Ref. 1). This isolation time is based on engineering judgement since the control room and offsite dose calculations are performed assuming that the leakage from containment begins immediately following the accident.

REFERENCES

1. UFSAR, Section 6.2.
2. Regulatory Guide 1.183, July 2000.
3. Standard Review Plan 6.2.4.
4. Generic Issue B-20, "Containment Leakage Due to Seal Deterioration."

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Containment Spray and Cooling Systems

BASES

BACKGROUND

The Containment Spray and Containment Cooling Systems provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine and aerosol removal capability of the spray reduces the release of fission product radioactivity from containment to the environment, in the event of a Design Basis Accident (DBA), to within limits. The Containment Spray and Containment Cooling Systems are designed to meet the requirements of 10 CFR 50, Appendix A, GDC 38, "Containment Heat Removal," GDC 39, "Inspection of Containment Heat Removal Systems," GDC 40, "Testing of Containment Heat Removal Systems," GDC 41, "Containment Atmosphere Cleanup," GDC 42, "Inspection of Containment Atmosphere Cleanup Systems," and GDC 43, "Testing of Containment Atmosphere Cleanup Systems" (Ref. 1).

The Containment Cooling System and Containment Spray System are Engineered Safety Feature (ESF) Systems and are discussed in UFSAR, Sections 9.4.8 and 6.5.2, respectively (Refs. 2 and 3). They are designed to ensure that the heat removal capability required during the post accident period can be attained. The Containment Spray System in conjunction with the Containment Cooling System limit and maintain post accident conditions to less than the containment design values. In addition, the Containment Spray System and Containment Cooling System provide an alternate hydrogen control function to the hydrogen recombiners, hydrogen mixing, during post Loss Of Coolant Accident (LOCA) conditions.

BACKGROUND (continued)

Containment Spray System

The Containment Spray System consists of two separate 100% capacity trains, each capable of meeting the design bases. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate ESF bus. The Refueling Water Storage Tank (RWST) supplies borated water to the Containment Spray System during the injection phase of operation. In the recirculation mode of operation, containment spray pump suction is transferred from the RWST to the containment sump(s).

The Containment Spray System provides a spray of cold borated water mixed with sodium hydroxide (NaOH) from the spray additive tank into the upper regions of containment to reduce the containment pressure and temperature and to reduce fission products from the containment atmosphere during a DBA. The RWST solution temperature is an important factor in determining the heat removal capability of the Containment Spray System during the injection phase. In the recirculation mode of operation, heat is removed from the containment sump water by the residual heat removal heat exchangers. Each train of the Containment Spray System provides adequate spray coverage to meet the system design requirements for containment heat removal.

The Spray Additive System injects an NaOH solution into the spray. The resulting alkaline pH of $\geq$ 8.0 for the spray ensures that radioiodines removed from the containment atmosphere by spray or natural deposition will remain in solution and not re-evolve from water in the containment sump. The upper pH limit minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the fluid. The chemical aspects of iodine removal capability are addressed in LCO 3.6.7, "Spray Additive System."

B 3.6 CONTAINMENT SYSTEMS

B 3.6.7 Spray Additive System

BASES

BACKGROUND

The Spray Additive System is a subsystem of the Containment Spray System. It provides an available inventory of sodium hydroxide (NaOH) that is added to the containment spray and is accumulated in the containment sump. The resulting pH of the sump water will be ≥ 8.0, thus ensuring that iodine in the sump water will not re-evolve and be available for release from the containment to the environment. The Spray Additive System is described in UFSAR Section 6.5.2 (Ref. 1).

Airborne radioiodine in its various forms is the fission product of primary concern in the evaluation of a design basis Loss of Coolant Accident (LOCA). It is absorbed by the spray from the containment atmosphere. To enhance the iodine absorption capacity of the spray, the spray solution is adjusted to an alkaline pH that promotes iodine hydrolysis, in which iodine is converted to nonvolatile forms. Because of its stability when exposed to radiation and elevated temperature, NaOH is the preferred spray additive. The NaOH added to the spray also ensures an equilibrium sump pH value of ≥ 8.0 and ≤ 11.0 of the solution recirculated from the containment sump. This pH band minimizes the evolution of iodine, while minimizing the occurrence of chloride and caustic stress corrosion on mechanical systems and components.

The Spray Additive System consists of one spray additive tank that is shared by the two trains of spray additive equipment. Each train of equipment provides a flow path from the spray additive tank to a containment spray pump and consists of an eductor for each containment spray pump, valves, instrumentation, and connecting piping. Each eductor draws the NaOH spray solution from the common tank using a portion of the borated water discharged by the containment spray pump as the motive flow. The eductor mixes the NaOH solution and the borated water and discharges the mixture into the spray pump suction line. The eductors are designed to ensure that the pH of the spray mixture is between 8.5 and 12.8.

BASES

BACKGROUND (continued)

An automatic or manual Containment Spray System actuation signal opens the valves from the spray additive tank to the eductor (CS019A/B), the discharge valve to the eductor from the CS pump discharge (CS010A/B), if not already open, and the isolation valve into containment (CS007A/B); in addition to starting the CS pumps. The 30% to 36% NaOH solution is drawn into the spray pump suctions. The spray additive tank capacity provides for the addition of NaOH solution to water sprayed into the containment from either the Refueling Water Storage Tank (RWST) during the injection phase, or recirculated from the containment sump during the recirculation phase. The inventory of NaOH in the spray additive tank assures a long term containment sump pH of ≥ 8.0. The percent solution and volume of solution sprayed into containment ensures a long term containment sump pH of ≥ 8.0 and ≤ 11.0. This ensures the continued iodine retention effectiveness of the sump water during the recirculation phase of spray operation and also minimizes the occurrence of chloride induced stress corrosion cracking of the stainless steel recirculation piping.

APPLICABLE SAFETY ANALYSES

The Spray Additive System is essential to prevent re-evolution of iodine collected in the sump following a DBA.

Following the assumed release of radioactive materials into containment, the containment is assumed to leak at its design value volume following the accident. The analysis assumes that 82.5% of containment is covered by the spray (Ref. 2).

The accident analysis assumes that the Spray Additive System is initiated after start of the DBA. The accident analysis does not credit iodine removal due to NaOH in the spray droplets, but only credits NaOH in the containment sump for pH control thus ensuring that iodine in the sump water will not re-evolve and be available for release from the containment to the environment.

The DBA analyses assume that one train of the Containment Spray System/Spray Additive System is inoperable and that the useable spray additive tank volume is added to the remaining Containment Spray System flow path.

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

BASES

LCO

The Spray Additive System is necessary to prevent re-evolution of radioactive iodine collected in the sump into the containment atmosphere in the event of a design basis LOCA. To be considered OPERABLE, the volume and concentration of the spray additive solution must be sufficient to provide NaOH injection into the spray flow in either the injection or the recirculation phase to increase the sump water pH to a value ≥ 8.0. This minimum pH assures retention of iodine collected in the containment sump. An upper bound on pH of 11.0 prevents conditions that may induce caustic stress corrosion cracking of mechanical system components. In addition, it is essential that valves in the Spray Additive System flow paths are properly positioned and that automatic valves are capable of activating to their correct positions.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment requiring the operation of the Spray Additive System. The Spray Additive System assists in reducing the iodine fission product inventory prior to release to the environment.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Thus, the Spray Additive System is not required to be OPERABLE in MODE 5 or 6.

ACTIONS

<u>A.1</u>

If the Spray Additive System is inoperable, it must be restored to OPERABLE within 7 days. The pH adjustment of the Containment Spray System flow for corrosion protection and iodine removal enhancement is reduced in this condition. The Containment Spray System would still be available and would remove some iodine from the containment atmosphere in the event of a DBA. The 7 day Completion Time takes into account the redundant flow path capabilities and the low probability of the worst case DBA occurring during this period.

BASES

ACTIONS (continued)

B.1 and B.2

If the Spray Additive System cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 84 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows additional time for attempting restoration of the Spray Additive System and is reasonable when considering the driving force for a release of radioiodine from the Reactor Coolant System is reduced in MODE 3.

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1

Verifying the correct alignment of Spray Additive System manual and automatic valves in the spray additive flow path provides assurance that the system is able to provide additive to the Containment Spray System in the event of a DBA. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment and capable of potentially being mispositioned are in the correct position.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.7.2

To provide effective iodine retention in the containment sump, the containment spray must be an alkaline solution. Since the RWST contents are normally acidic, the volume of the spray additive tank must provide a sufficient volume of spray additive to adjust pH for all water injected. This SR is performed to verify the availability of sufficient NaOH solution in the Spray Additive System. The 184 day Frequency was developed based on the low probability of an undetected change in tank volume occurring during the SR interval (the tank is isolated during normal unit operations). Tank level is also indicated and alarmed in the control room, so that there is high confidence that a substantial change in level would be detected.

SR 3.6.7.3

This SR provides verification of the NaOH concentration in the spray additive tank and is sufficient to ensure that the spray solution being injected into containment is at the correct pH level. The 184 day Frequency is sufficient to ensure that the concentration level of NaOH in the spray additive tank remains within the established limits. This is based on the low likelihood of an uncontrolled change in concentration (the tank is normally isolated) and the probability that any substantial variance in tank volume will be detected.

SR 3.6.7.4

This SR provides verification that each automatic valve in the Spray Additive System flow path actuates to its correct position. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

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

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

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

LCO

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

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in exposures comparable to the 10 CFR 50.67 (Ref. 4) limits or the NRC staff approved licensing basis.

BASES

REFERENCES

1. UFSAR, Section 10.3.
2. UFSAR, Section 15.1.5.
3. UFSAR, Section 6.2.
4. 10 CFR 50.67.
5. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.7 PLANT SYSTEMS

B 3.7.3 Secondary Specific Activity

BASES

BACKGROUND

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

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

This limit is lower than the activity value that might be expected from a 1 gpm tube leak (LCO 3.4.13, "RCS Operational LEAKAGE") of primary coolant at the limit of 1.0 μCi/gm (LCO 3.4.16, "RCS Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives, (i.e., < 20 hours). I-131, with a half life of 8.04 days, concentrates faster than it decays, but does not reach equilibrium because of blowdown and other losses.

With the specified activity limit, the resultant doses would be within 10 CFR 50.67 (Ref. 1) limits assuming a secondary coolant release following a trip from full power.

BASES

APPLICABLE SAFETY ANALYSES

The accident analysis of the Main Steam Line Break (MSLB), as discussed in the UFSAR, Chapter 15 (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.1 μCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed the 10 CRF 50.67 limits (Ref. 1).

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

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

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

BASES

LCO	As indicated in the Applicable Safety Analyses, the specific activity of the secondary coolant is required to be $\leq$ 0.1 μCi/gm DOSE EQUIVALENT I-131 to limit the radiological consequences of a Design Basis Accident (DBA) to the required limits (Ref. 1). Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.
APPLICABILITY	In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere. In MODES 5 and 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.
ACTIONS	A.1 and A.2 DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS and contributes to increased post accident doses. If the secondary specific activity is not within limits, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

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

REFERENCES

1. 10 CFR 50.67.

2. UFSAR, Chapter 15.

BASES

APPLICABLE SAFETY ANALYSES

The SG PORV lines provide an alternate method for cooling the unit to RHR entry condition whenever the preferred heat sink via the Steam Dump System to the condenser is unavailable due to a loss of offsite power. Prior to operator actions to cool down the unit, the SG PORVs and main steam safety valves (MSSVs) are assumed to operate automatically to relieve steam and maintain the SG pressure below the design value. For the recovery from a steam generator tube rupture (SGTR) event (Ref. 2), the operator is also required to perform a limited cooldown to establish adequate subcooling as a necessary step to terminate the primary to secondary break flow into the ruptured SG. The time required to terminate the primary to secondary break flow for an SGTR is more critical than the time required to cool down to RHR conditions for this event and also for other accidents. After primary to secondary break flow termination, it is assumed that SG PORVs on two intact SGs are used to cool the RCS down to 350°F. Thus, the SGTR is the limiting event for the SG PORVs. The number of SG PORVs required to be OPERABLE to satisfy the SGTR accident analysis requirements depends upon the number of reactor coolant system loops and consideration of a single failure assumption regarding the failure of one SG PORV to open on demand.

In addition, the SGTR analysis considers SG overfill. The limiting single failure with respect to SG overfill is the failure of one SG PORV on an intact SG to open when required for cooldown of the RCS. The analysis assumes four SG PORVs are OPERABLE at the start of the SGTR event. One SG PORV is on the ruptured SG, another SG PORV is assumed to fail to open and the remaining SG PORVs are used to perform the RCS cooldown. The analysis shows that cooldown using two SG PORVs results in no SG overfill.

The SG PORVs are equipped with manual block valves in the event a SG PORV fails to close during use. The SG PORVs are also credited as CIVs (refer to LCO 3.6.3).

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.4.2

The function of the block valve is to isolate a failed open SG PORV. Cycling the block valve both closed and open demonstrates its capability to perform this function. Performance of inservice testing or use of the block valve during unit cooldown may satisfy this requirement. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. The Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 10.3.
2. UFSAR, Section 15.6.3.

BASES

BACKGROUND (continued)

Actuation of the VC Filtration System places the system in the emergency mode of operation. Actuation of the system to the emergency mode of operation; starts the makeup fan, opens the turbine building intake damper, isolates the normal intake from outside dampers, isolates the purge dampers (if open), opens the recirculation charcoal adsorber dampers, and closes the recirculation charcoal adsorber bypass dampers. The operating supply and return fans continue to operate. Interlocks are provided such that the makeup fan will not start unless the associated supply fan is in operation. Outside air is filtered and then mixed with the air being recirculated through the control room. Pressurization of the control room minimizes infiltration of unfiltered air from the areas adjacent to the control room envelope.

The air entering the control room is continuously monitored by radiation detectors. One outside air intake detector output above the alarm setpoint will cause actuation of the emergency mode of operation and trip the Control Room Offices HVAC (VV), Laboratory HVAC (VL), and Radwaste Building Ventilation (VW) Systems.

The VC Filtration System will not automatically realign to the Turbine Building makeup air intake upon receipt of a high radiation or Safety Injection (SI) signal when a VC Filtration System Emergency Makeup Filter unit is in operation and aligned to the outside air intake.

One VC Filtration System train can pressurize the control room to $\geq$ 0.125 inches water gauge, relative to areas adjacent to the control room envelope.

The control room and the control room envelope are defined in UFSAR Section 6.4 (Ref. 1). The control room is contained within the control room envelope. The areas within the control room envelope, external to the control room, are maintained at a positive pressure as described in the Control Room Envelope Integrity Program.

Redundant filter trains are provided such that if an excessive pressure drop develops across one filter train, the other train is available to provide the required filtration.

BASES

BACKGROUND (continued)

The normally open intake isolation dampers are arranged in a series so that the failure of one damper to shut will not result in a breach of isolation. The VC Filtration System is designed in accordance with Seismic Category I requirements.

The VC Filtration System is designed to maintain the control room environment for 30 days after a Design Basis Accident (DBA) without exceeding the dose limits of 10 CFR 50.67 (Ref. 7), (i.e., 5 rem TEDE).

APPLICABLE SAFETY ANALYSES

The VC System components are arranged in redundant, safety related ventilation trains. The location of components and ducting within the control room envelope ensures an adequate supply of filtered air to all areas requiring access. The VC Filtration System provides airborne radiological protection for the control room operators, as demonstrated by the control room accident dose analyses for the most limiting design basis Loss of Coolant Accident, fission product release presented in the UFSAR, Chapter 15 (Ref. 3). The safety analyses assume a 95% filter efficiency for the makeup charcoal adsorber and a 90% filter efficiency for the recirculation charcoal adsorber. For design basis accident radiological dose assessments, the VC Filtration System is assumed to be initiated within 30 minutes.

As described in UFSAR Section 6.4 (Ref. 1) and Section 2.2 (Ref. 4), the only potential toxic releases that could pose a risk to the control room operators, are from offsite sources. The probability of such a release is low and there would be sufficient time upon notification to initiate the VC isolation mode of operation.

The worst case single active failure of a component of the VC Filtration System, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

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

BASES

LCO

Two independent and redundant VC Filtration System trains are required to be OPERABLE to ensure that at least one is available assuming a single failure disables the other train. Total system failure could result in exceeding a TEDE dose of 5 rem to the control room operator in the event of a large radioactive release.

The VC Filtration System is considered OPERABLE when the individual components necessary to limit operator exposure are OPERABLE in both trains. A VC Filtration System train is OPERABLE when the associated:

a. Makeup air fan is OPERABLE;

b. Supply fan is OPERABLE;

c. Return air fan is OPERABLE;

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

e. Makeup filter unit heater, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

The control room boundary must be maintained within the assumptions of the design analysis and in accordance with the Control Room Envelope Integrity Program.

APPLICABILITY

In MODES 1, 2, 3, 4, 5, and 6, and at all times during movement of irradiated fuel assemblies in the fuel handling building or containment, the VC Filtration System must be OPERABLE to control operator exposure during and following a DBA, including the release from a fuel handling accident.

In MODE 5 or 6, the VC Filtration System provides protection from significant radioactive releases.

BASES

ACTIONS (continued)

C.1.1, C.1.2, C.2.1, and C.2.2

In MODE 5 or 6, or during movement of irradiated fuel assemblies, if the inoperable VC Filtration System train cannot be restored to OPERABLE status within the required Completion Time, action must be taken to immediately place the OPERABLE VC Filtration System train in the emergency mode. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure would be readily detected. Action C.1.2 requires the VC Filtration System train placed in operation be capable of being powered by an OPERABLE emergency power source. This action assures availability of electric power in the unlikely event of a loss of offsite power. This power source can be either from Unit 1 or Unit 2, via OPERABLE crosstie breakers.

An alternative to Required Action C.1.1 and C.1.2 is to immediately suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

D.1 and D.2

In MODE 5 or 6, or during movement of irradiated fuel assemblies, with two VC Filtration System trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might enter the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

E.1

If both VC Filtration System trains are inoperable in MODE 1, 2, 3, or 4, the VC Filtration System may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.10.4

This SR verifies the capability of the VC Filtration System to pressurize the control room. The control room positive pressure, with respect to potentially contaminated areas adjacent to the control room envelope, is periodically tested to verify the function of the VC Filtration System. During the emergency mode of operation, the VC Filtration System is designed to pressurize the control room to ≥ 0.125 inches water gauge, relative to areas adjacent to the control room envelope in order to minimize unfiltered inleakage. The VC Filtration System is designed to maintain this positive pressure with one train at a makeup flow rate ≥ 5400 cfm and ≤ 6600 cfm. The Frequency of 18 months on a STAGGERED TEST BASIS is consistent with the guidance provided in NUREG-0800 (Ref. 6).

REFERENCES

1. UFSAR, Section 6.4.
2. UFSAR, Section 9.4.
3. UFSAR, Chapter 15.
4. UFSAR, Section 2.2.
5. Regulatory Guide 1.52, Rev. 2.
6. NUREG-0800, Section 6.4, Rev. 2, July 1981.
7. 10 CFR 50.67.

BASES

ACTIONS (continued)

C.1.1, C.1.2, C.2.1, and C.2.2

In MODE 5 or 6, or during movement of irradiated fuel, if the inoperable VC Temperature Control System train cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE VC Temperature Control System train must be placed in operation immediately. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that active failures will be readily detected. Action C.1.2 requires the VC Temperature Control System train placed in operation be capable of being powered by an OPERABLE emergency power source. This action assures availability of electric power in the unlikely event of a loss of offsite power. This power source can be either from Unit 1 or Unit 2, via OPERABLE crosstie breakers.

An alternative to Required Action C.1.1 and C.1.2 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

D.1 and D.2

In MODE 5 or 6, or during movement of irradiated fuel assemblies, with two VC Temperature Control System trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

E.1

If both VC Temperature Control System trains are inoperable in MODE 1, 2, 3, or 4, the control room VC Temperature Control System may not be capable of performing its intended function. Therefore, LCO 3.0.3 must be entered immediately.

BASES

APPLICABLE SAFETY ANALYSES

The design basis of the Nonaccessible Area Exhaust Filter Plenum Ventilation System is established by the large break LOCA. The system evaluation assumes a passive failure of the ECCS outside containment, such as an SI pump seal failure, during the recirculation mode. In such a case, the system limits radioactive release to within the 10 CFR 50.67 (Ref. 4) limits, or the NRC staff approved licensing basis (e.g., a specified fraction of Reference 5 limits). While the system is automatically initiated on an SI signal, manual actuation/alignment of the system is acceptable. The system is not required until initiation of the ECCS recirculation mode. The analysis of the effects and consequences of a large break LOCA is presented in Reference 3. The Nonaccessible Area Exhaust Filter Plenum Ventilation System also actuates following a small break LOCA, in those cases where the ECCS goes into the recirculation mode of long term cooling, to clean up releases of smaller leaks, such as from valve stem packing. The Nonaccessible Area Exhaust Filter Plenum Ventilation System is also credited in the control room habitability analysis (Ref. 5). The safety analyses assume a 90% filter efficiency.

Two types of system failures are considered in the accident analysis: complete loss of function, and excessive LEAKAGE. Either type of failure may result in a lower efficiency of removal for any gaseous and particulate activity released to the ECCS pump rooms following a LOCA.

The Nonaccessible Area Exhaust Filter Plenum Ventilation System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The Nonaccessible Area Exhaust Filter Plenum Ventilation System is required to be OPERABLE to ensure that atmospheric releases from the ECCS pump rooms do not exceed those assumed in the safety analysis. Total system failure could result in the atmospheric release, 1) exceeding 10 CFR 50.67 limits in the event of a Design Basis Accident (DBA), and 2) exceeding the limits for control room habitability. The Nonaccessible Area Exhaust Filter Plenum Ventilation System is considered OPERABLE when the individual components, necessary to maintain ECCS pump rooms and equipment rooms filtration are OPERABLE.

In order for the Nonaccessible Area Exhaust Filter Plenum Ventilation System to perform its function, filtration and motive flow must be provided by two of the three trains, the bypass path(s) to the normal auxiliary building exhaust system must be isolated, and the third train's inlet damper must be closed. The closure of the third train's inlet damper, prevents starting of a third fan and also ensures filtration of the exhaust from the ECCS pump rooms, by eliminating potential bypass flow paths.

Three trains of the Nonaccessible Area Exhaust Filter Plenum Ventilation System are required to be OPERABLE to ensure that at least two are available, assuming a single failure coincident with loss of offsite power on the affected unit and an orderly shutdown on the other unit. In addition due to design considerations, two of the trains must be aligned for operation and one train must be aligned in standby (i.e., the inlet damper closed).

To accommodate the single failure and loss of offsite power assumptions, the required fans in each of the Nonaccessible Area Exhaust Filter Plenum Ventilation System trains must be independent of the credited fans in the other trains.

BASES

SURVEILLANCE REQUIREMENTS (continued)

If a particular pump room is isolated such that there is no potential for post accident fluids to pass through the room, or that room's ECCS equipment is not required, that room can be excluded from meeting the acceptance criteria of the SR. Performance of this SR with a room excluded, represents a change in the ECCS pump room area volume that the system is maintaining at a negative pressure. Prior to the room being put back in service, this SR would have to be performed with the new volume, to assure that the system can maintain the entire volume at the required negative pressure.

The 18 month Frequency on a STAGGERED TEST BASIS is consistent with that specified in Reference 6.

REFERENCES

1. UFSAR, Section 6.5.1.
2. UFSAR, Section 9.4.5.
3. UFSAR, Section 15.6.5.
4. 10 CFR 50.67.
5. UFSAR, Section 6.4.
6. Regulatory Guide 1.52 (Rev. 2).
7. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.

B 3.7 PLANT SYSTEMS

B 3.7.13 Fuel Handling Building Exhaust Filter Plenum (FHB) Ventilation System

BASES

BACKGROUND

The FHB Ventilation System is available to filter airborne radioactive particulates from the area of the fuel pool following a fuel handling accident. The FHB Ventilation System, in conjunction with other normally operating systems, also provides environmental control of temperature in the fuel pool area.

The FHB Ventilation System is a subsystem of the common auxiliary building heating, ventilation, and air conditioning system (VA). Each unit has two VA supply and two VA exhaust fans. The VA supply and exhaust fans are not required for FHB Ventilation System OPERABILITY.

The FHB Ventilation System consists of two independent and redundant trains. Each train 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 follows the adsorber section to collect carbon fines and provide backup in case the main HEPA filter bank fails. The downstream HEPA filter is not credited in the analysis, but serves to collect charcoal fines, and to back up the upstream HEPA filter should it develop a leak. The system initiates filtered ventilation of the fuel handling building following receipt of a high radiation signal or a Safety Injection (SI) on either unit.

BASES

BACKGROUND (continued)

The FHB Ventilation System is a standby system. During normal operation flow from the fuel handling building is routed through the FHB Ventilation System prefilters and HEPA filters and then through the VA exhaust plenum via the VA exhaust fans. Upon FHB Ventilation System actuation (emergency mode of operation), the bypass dampers close, and the FHB Ventilation System fans start, drawing air through the FHB Ventilation System charcoal filters. The prefilters remove any large particles in the air to prevent excessive loading of the HEPA filters and charcoal adsorbers. The FHB Ventilation System may also be initiated manually.

The FHB Ventilation System is discussed in the UFSAR, Sections 6.5.1, 9.4.5, and 15.7.4 (Refs. 1, 2, and 3, respectively).

APPLICABLE SAFETY ANALYSES

The FHB Ventilation System design basis is established by the consequences of the limiting Design Basis Accident (DBA), which is a fuel handling accident involving handling RECENTLY IRRADIATED FUEL. The analysis of the fuel handling accident, given in Reference 3, assumes that all fuel rods in an assembly are damaged. The DBA analysis of the fuel handling accident assumes that only one train of the FHB Ventilation System is functional due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the one remaining train of this filtration system. The amount of fission products available for release from the fuel handling building is determined for a fuel handling accident. The accident analyses assume a 90% filter efficiency for elemental iodine and a 70% filter efficiency for methyl iodine. Due to radioactive decay, the FHB Ventilation System is only required to isolate during fuel handling accidents involving handling RECENTLY IRRADIATED FUEL. These assumptions and analyses follow the guidance provided in Regulatory Guide 1.183 (Ref. 4).

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

BASES

LCO

Two independent and redundant trains of the FHB Ventilation System are required to be OPERABLE to ensure that at least one train is available, assuming a single failure that disables the other train, coincident with a loss of offsite power. Total system failure could result in the atmospheric release from the fuel handling building exceeding the 10 CFR 50.67 (Ref. 5) limits in the event of a fuel handling accident involving handling RECENTLY IRRADIATED FUEL.

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

a. Fan is OPERABLE;

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

c. Ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

APPLICABILITY

During movement of RECENTLY IRRADIATED FUEL in the fuel handling building, the FHB Ventilation System is required to be OPERABLE to alleviate the consequences of a fuel handling accident.

During movement of RECENTLY IRRADIATED FUEL in the containment with the containment equipment hatch not intact, the FHB Ventilation System is required to be OPERABLE to mitigate the consequences of an accident inside containment. The equipment hatch is considered not intact if both personnel air lock doors associated with the equipment hatch are open or the hatch is not held in place with at least four bolts.

BASES

ACTIONS

The Actions Table is modified by a Note indicating that LCO 3.0.3 does not apply. If moving RECENTLY IRRADIATED FUEL assemblies in the fuel handling building while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving RECENTLY IRRADIATED FUEL assemblies while in MODES 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of RECENTLY IRRADIATED FUEL assemblies is not sufficient to require a reactor shutdown.

A.1

With one FHB Ventilation System train inoperable, action must be taken to restore OPERABLE status within 7 days. During this period, the remaining OPERABLE train is adequate to perform the FHB Ventilation System function. The 7 day Completion Time is based on the risk from an event occurring requiring the inoperable FHB Ventilation System train, and the remaining FHB Ventilation System train providing the required protection.

B.1.1, B.1.2, B.2.1, and B.2.2

When Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE FHB Ventilation System train must be placed in the emergency mode or RECENTLY IRRADIATED FUEL movement suspended. This action ensures that the remaining train is OPERABLE, that no undetected failures preventing system operation will occur, and that any active failure will be readily detected. Required Action B.1.2 requires the FHB Ventilation System train placed in operation be capable of being powered by an OPERABLE emergency power source. This action assures availability of electric power in the unlikely event of a loss of offsite power. This power source can be from Unit 1 or Unit 2, via OPERABLE crosstie breakers.

If the system is not placed in the emergency mode, Action B.2.1 requires suspension of RECENTLY IRRADIATED FUEL movement in the fuel handling building, which precludes a fuel handling accident involving handling RECENTLY IRRADIATED FUEL in the fuel handling building. This does not preclude the movement of fuel assemblies to a safe position.

BASES

ACTIONS (continued)

Required Action B.2.2 requires suspension of movement of RECENTLY IRRADIATED FUEL assemblies inside containment, precluding an accident that might require actuation of the FHB Ventilation System (when the equipment hatch is not intact).

Required Action B.2.2 is modified by a Note which indicates that this Required Action is only required if the equipment hatch is not intact. If the hatch is intact, only Required Action B.2.1 is required.

C.1 and C.2

When two trains of the FHB Ventilation System are inoperable action must be taken to place the unit in a condition in which the LCO does not apply. Action must be taken immediately to suspend movement of RECENTLY IRRADIATED FUEL assemblies in the fuel handling building. This does not preclude the movement of fuel to a safe position.

Required Action C.2 requires suspension of movement of RECENTLY IRRADIATED FUEL assemblies inside containment, precluding an accident that might require actuation of the FHB Ventilation System (when the equipment hatch is not intact).

Required Action C.2 is modified by a Note which indicates that this Required Action is only required if the equipment hatch is not intact. If the hatch is intact, only Required Action C.1 is required.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.13.3

This SR verifies the integrity of the fuel handling building and containment enclosure. The ability of the fuel handling building and containment to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the FHB Ventilation System and enclosure integrity. During the emergency mode of operation the FHB Ventilation System is designed to maintain a slight negative pressure in the fuel handling building to prevent unfiltered leakage. The FHB Ventilation System is designed to maintain a ≤ -0.25 inches water gauge with respect to atmospheric pressure. The Frequency of 7 days on a STAGGERED TEST BASIS, is based on the increased containment activity that occurs when the equipment hatch is not intact, that could affect containment integrity.

This SR is modified by a Note that requires this SR only during movement of RECENTLY IRRADIATED FUEL assemblies (in the fuel building or in the containment) when the equipment hatch is not intact.

SR 3.7.13.4

This SR verifies that each FHB Ventilation System train aligns, starts, and operates on an actual or simulated actuation signal. The 18 month Frequency is consistent with Reference 6.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.13.5

This SR verifies the integrity of the fuel handling building enclosure. The ability of the fuel handling building to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the FHB Ventilation System. During the emergency mode of operation the FHB Ventilation System is designed to maintain a slight negative pressure in the fuel handling building, to prevent unfiltered leakage. The FHB Ventilation System is designed to maintain a ≤ -0.25 inches water gauge with respect to atmospheric pressure at a flow rate ≤ 23,100 cfm to the fuel handling building. The Frequency of 18 months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 7).

An 18 month Frequency (on a STAGGERED TEST BASIS) is consistent with Reference 6.

This SR is modified by a Note that requires this SR only during movement of RECENTLY IRRADIATED FUEL assemblies in the fuel handling building when the equipment hatch is intact.

REFERENCES

1. UFSAR, Section 6.5.1.
2. UFSAR, Section 9.4.5.
3. UFSAR, Section 15.7.4.
4. Regulatory Guide 1.183, July 2000.
5. 10 CFR 50.67.
6. Regulatory Guide 1.52 (Rev. 2).
7. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.

B 3.7 PLANT SYSTEMS

B 3.7.14 Spent Fuel Pool Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

A general description of the spent fuel pool design is given in the UFSAR, Section 9.1.2 (Ref. 1). A description of the Spent Fuel Pool Cooling and Cleanup System is given in the UFSAR, Section 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in the UFSAR, Section 15.7.4 (Ref. 3).

APPLICABLE SAFETY ANALYSES

The minimum water level in the spent fuel pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.183 (Ref. 4). The resultant Total Effective Dose Equivalent (TEDE) dose is within 10 CFR 50.67 limits (Ref. 5).

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

The spent fuel pool water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

This SR verifies sufficient spent fuel pool water is available in the event of a fuel handling accident. The water level in the spent fuel pool must be checked periodically. The 7 day Frequency is appropriate because the volume in the pool is normally stable. Water level changes are controlled by plant procedures and are acceptable based on operating experience.

During refueling operations, the level in the spent fuel pool is in equilibrium with the refueling cavity when they are hydraulically coupled, and the level in the refueling cavity is checked daily in accordance with SR 3.9.7.1.

REFERENCES

1. UFSAR, Section 9.1.2.
2. UFSAR, Section 9.1.3.
3. UFSAR, Section 15.7.4.
4. Regulatory Guide 1.183, July 2000.
5. 10 CFR 50.67.

B 3.9 REFUELING OPERATIONS

B 3.9.4 Containment Penetrations

BASES

BACKGROUND

During movement of RECENTLY IRRADIATED FUEL assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODES 5 and 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are filtered, closed, or capable of being closed. Since there is no significant potential for containment pressurization, the 10 CFR 50, Appendix J, leakage criteria and tests are not required.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR 50.67 (Ref. 5). In addition, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

BASES

BACKGROUND (continued)

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During movement of RECENTLY IRRADIATED FUEL assemblies within containment with the equipment hatch installed, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts be approximately equally spaced. During movement of RECENTLY IRRADIATED FUEL assemblies within containment and the equipment hatch not intact, the OPERABILITY requirements of the Fuel Handling Building Exhaust Filter Plenum (FHB) Ventilation System must be met. The OPERABILITY requirements of the FHB Ventilation System are provided in TS 3.7.13, "Fuel Handling Building Exhaust Filter Plenum (FHB) Ventilation System."

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 in accordance with LCO 3.6.2, "Containment Air Locks." The two air locks are the personnel air lock and the emergency air lock. Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During movement of RECENTLY IRRADIATED FUEL assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain closed. An exception, however, is provided for the personnel air lock. It is acceptable to have both doors of the personnel air lock opened simultaneously provided the FHB Ventilation System is in compliance with TS 3.7.13.

The closure restrictions are sufficient to restrict unfiltered fission product radioactivity releases from containment to the environment due to a fuel handling accident involving RECENTLY IRRADIATED FUEL during refueling.

BASES

BACKGROUND (continued)

The Containment Ventilation Isolation System consists of the normal purge subsystem, the mini purge subsystem, and the post Loss Of Coolant Accident purge subsystem. These three subsystems contain penetrations which provide direct access from the containment to the outside atmosphere. In MODE 6, the minipurge subsystem is normally used to exchange large volumes of containment air to support refueling operations. Each penetration contains inside and outside containment isolation valves which close automatically on an actuation signal. During movement of RECENTLY IRRADIATED FUEL within containment, all required valves within a subsystem must be capable of being closed by a containment ventilation isolation signal whenever the associated subsystem is in operation. A list of the instrumentation which functions to isolate the valves in these penetrations is provided in LCO 3.3.6, "Containment Ventilation Isolation Instrumentation."

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by a closed automatic isolation valve, a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods allowed under the provisions of 10 CFR 50.59 may include use of a material that can provide a temporary atmospheric pressure ventilation barrier during movement of RECENTLY IRRADIATED FUEL within the containment.

BASES

APPLICABLE SAFETY ANALYSES	During movement of RECENTLY IRRADIATED FUEL assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 1). Fuel handling accidents, analyzed in Reference 2, include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.7, "Refueling Cavity Water Level," ensure that the release of fission product radioactivity, subsequent to a fuel handling accident in containment, results in doses that are within the 10 CFR 50.67 (Ref. 5) limits. The radiological dose assessments for the Design Basis Fuel Handling Accident in containment were performed in accordance with the guidance of Regulatory Guide 1.183 (Ref 4). When moving RECENTLY IRRADIATED FUEL in containment, the requirements of the LCO must be met with the exception that LCO 3.9.4.a need not be met if at least one train of the FHB Ventilation System is OPERABLE as specified in the Note. This exception permits movement of RECENTLY IRRADIATED FUEL with both personnel air lock doors open or the equipment hatch not intact. When moving fuel in the containment that is not RECENTLY IRRADIATED FUEL, the LCO is not applicable. Due to radioactive decay, neither containment closure nor an OPERABLE FHB Ventilation System train are required to meet the dose limits of 10 CFR 50.67 during a fuel handling accident. Another consideration, which may result in a limiting decay time prior to fuel handling, is the impact of decay heat on the spent fuel pool cooling requirements described in Reference 3. Containment penetrations satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	This LCO limits the consequences of a fuel handling accident involving handling RECENTLY IRRADIATED FUEL in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge (supply and exhaust) penetrations. For the OPERABLE containment purge

BASES

LCO (continued)

penetrations, this LCO ensures that unisolated penetrations are isolable by the Containment Ventilation Isolation System. The OPERABILITY requirements for this LCO ensure the automatic purge valve closure times specified in the UFSAR can be achieved and, therefore, meet the assumptions used in the safety analysis to ensure that releases through the valves are terminated, such that radiological doses are within the acceptance limits.

The LCO is modified by a Note which allows both personnel air lock doors to be open or the equipment hatch not intact when the FHB Ventilation System is in compliance with TS 3.7.13. When the equipment hatch is installed it serves to contain fission product radioactivity that may be released following a fuel handling accident in the containment. When the equipment hatch is not intact, or when both doors of the personnel air lock are simultaneously opened, the internal containment pressure is essentially equal to the internal pressure of the fuel handling building. In the event of a fuel handling accident in the containment, realigning of the fuel handling building ventilation system creates a negative pressure in the containment and fuel handling building relative to the auxiliary building and outside atmosphere. The negative pressure ensures that any radioactivity released to the containment atmosphere will either remain in the containment or be filtered through a FHB Ventilation System train. As such, with the equipment hatch not intact, or with both personnel air lock doors open, the consequences of a fuel handling accident involving RECENTLY IRRADIATED FUEL in containment would not exceed those calculated for a fuel handling accident involving RECENTLY IRRADIATED FUEL in the fuel handling building.

In addition, a commitment has been made to implement compensatory measures during movement of irradiated fuel as described in UFSAR Section 15.7.4, "Fuel Handling Accidents." These compensatory measures support the Alternate Source Term methodology and reduce doses even further below that provided by natural decay and avoid unmonitored releases in the event of a postulated fuel handling accident.

BASES

APPLICABILITY

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

ACTIONS

A.1

If the containment equipment hatch, air lock doors, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, the unit must be placed in a condition where containment closure is not needed. This is accomplished by immediately suspending movement of RECENTLY IRRADIATED FUEL assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that each of the containment penetrations required to be isolated is isolated. This Surveillance for the open purge valves demonstrates that the valves are not blocked from closing. Also the Surveillance will demonstrate that each valve operator has motive power which will ensure that each valve is capable of being closed by an OPERABLE automatic Containment Ventilation Isolation signal.

The Surveillance is performed every 7 days during movement of RECENTLY IRRADIATED FUEL assemblies within containment. As such, this Surveillance ensures that a postulated fuel handling accident involving RECENTLY IRRADIATED FUEL that releases fission product radioactivity within the containment will not result in a release of fission product radioactivity to the environment.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.9.4.2

This Surveillance demonstrates that each required containment purge valve actuates to its isolation position on an actual or simulated high radiation signal. In TS 3.3.6, the Containment Ventilation Isolation instrumentation requires a CHANNEL CHECK every 12 hours and a COT every 92 days to ensure the channel OPERABILITY during refueling operations. Every 18 months a CHANNEL CALIBRATION is performed. SR 3.9.4.3 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. These Surveillances will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

SR 3.9.4.3

This Surveillance demonstrates that the isolation time of each required containment purge valve providing direct access from the containment atmosphere to the outside atmosphere is in accordance with the Inservice Testing Program requirements. This SR, along with SR 3.9.4.2, ensures the containment purge valves in penetrations which provide direct access from the containment atmosphere to the outside atmosphere are capable of closing after a postulated fuel handling accident to limit the release of fission product radioactivity from the containment.

REFERENCES

1. UFSAR, Section 15.7.4.
2. NUREG-0800, Section 15.0.1, Revision 0, July 2000.
3. NUREG-0800, Section 9.1.3.
4. Regulatory Guide 1.183, July 2000.
5. 10 CFR 50.67.

B 3.9 REFUELING OPERATIONS

B 3.9.7 Refueling Cavity Water Level

BASES

BACKGROUND	The movement of irradiated fuel assemblies within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange. This requirement ensures a sufficient level of water is maintained in the refueling cavity to retain iodine fission product activity resulting from a fuel handling accident in containment (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to within 10 CFR 50.67 limits (Ref. 3).
APPLICABLE SAFETY ANALYSES	During movement of irradiated fuel assemblies, the water level in the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment, as postulated by Regulatory Guide 1.183 (Ref. 1). The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and doses are maintained within allowable limits (Refs. 2 and 3). The minimum decay time prior to allowing fuel handling is addressed in the fuel handling accident dose analysis. However, another consideration, which may result in a limiting decay time prior to fuel handling, is the impact of decay heat on the spent fuel pool cooling requirements described in Reference 4. Refueling cavity water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO	A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits.
APPLICABILITY	LCO 3.9.7 is applicable when moving irradiated fuel assemblies within containment. The LCO ensures a sufficient level of water is present in the refueling cavity to minimize the radiological consequences of a fuel handling accident in containment. If irradiated fuel assemblies are not present in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.14, "Spent Fuel Pool Water Level."
ACTIONS	<u>A.1</u> With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving movement of irradiated fuel assemblies within the containment shall be suspended immediately to ensure that a fuel handling accident cannot occur. The suspension of irradiated fuel movement shall not preclude completion of movement of a fuel assembly to a safe position.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1

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

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

REFERENCES

1. Regulatory Guide 1.183, July 2000.
2. UFSAR, Section 15.7.4.
3. 10 CFR 50.67.
4. NUREG-0800, Section 9.1.3.

ATTACHMENT 5

BYRON STATION
UNITS 1 AND 2

Docket Nos. 50-454 and 50-455
License Nos. NPF-37 and NPF-66

and

BRAIDWOOD STATION
UNITS 1 AND 2

Docket Nos. 50-456 and 50-457
License Nos. NPF-72 and NPF-77

License Amendment Request
"Alternative Source Term Implementation"

List of Commitments

List of Commitments

The following table identifies those actions committed to by Exelon in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

Commitment	Continuing Compliance	Scheduled Completion Date
In accordance with 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)," Byron and Braidwood Stations make 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 proper direction such that it can be treated and monitored. NOTE: It is unnecessary to maintain a contingency to close the containment equipment hatch or the personnel air lock doors during a postulated fuel handling accident as these containment penetrations open directly into the FHB where air flow in the proper direction will be maintained as noted below. 1. At least one train of the FHB Ventilation System will be available when moving irradiated fuel in the containment or FHB. 2. At least one FHB trackway door will be maintained closed when moving irradiated fuel in the containment or FHB. 3. When moving irradiated fuel in the FHB, within one hour of a fuel handling accident, air flow in the proper direction will be established in the FHB. 4. When moving irradiated fuel in the containment with the equipment hatch off; or with the equipment hatch installed and both personnel air lock doors open, within one hour of a fuel handling accident: a. air flow in the proper direction will be established in the containment/FHB, and b. all other penetrations in the containment to the outside air will be closed.	X	Upon Implementation.

Commitment	Continuing Compliance	Scheduled Completion Date
5. When moving irradiated fuel in the containment with the equipment hatch installed and at least one of the personnel air lock doors closed, within one hour of a fuel handling accident, all other penetrations in the containment to the outside air will be closed. 6. The containment and FHB effluent will be monitored consistent with TS 5.5.4, "Radioactive Effluent Controls Program."		

ATTACHMENT 6
Evaluation of Proposed Changes

Input Parameter Tables

Table 1.0 Input Parameters Common to All AST Radiological Analyses

Item	Parameter	Unit	Value	Reference	Notes
1.1	Core power	MWt	3658.3	UFSAR Tables 15.1-3, 15.3-3, 15.3-4, 15.6-7, 15.6-12	Per 10CFR50 Appendix K, the LOCA source term requires consideration of a 2% calorimetric uncertainty; this is applied to the core source term also for all events involving fuel damage. (1.02)(3586.6 MWt) = 3658.3 MWt
1.2	Off-site Breathing Rates 0-8 Hours 8-24 Hours 1-30 Days	m^3/sec	 3.50E-4 1.80E-4 2.30E-4	NRC Regulatory Guide (RG) 1.183	
1.3	Control Room Breathing Rate	m^3/sec	3.50E-4	RG 1.183	
1.4	CR Occupancy Factor 0-24 Hours 1-4 Days 4-30 Days	-	 1.0 0.6 0.4	RG 1.183	
1.5	Control Room Dose Acceptance Criteria TEDE	rem	5	10CFR 50.67	

Table 1.0 Input Parameters Common to All AST Radiological Analyses					
Item	**Parameter**	**Unit**	**Value**	**Reference**	**Notes**
1.6	Control Room Volume served by Recirculation Filtration to the Control Room Ventilation System	ft^3	200,000 ft^3	See Notes	The calculated values are 230,830 ft^3 (Byron) and 232,872 ft^3 (Braidwood). These volumes exclude the Upper Cable Spreading Room and other "supply only" rooms, which are not served by recirculation filtration. The smaller volume of 200,000 ft^3 is demonstrated by calculation to be conservative with respect to Control Room doses. This smaller "analyzed volume" of 200,000 ft^3 is used to account for VC supply and return duct configurations and other uncertainties.
1.7	CR HVAC Mode 1 Operation, Normal CR Unfiltered Makeup Air Flow Rate CR Unfiltered Recirc. Air Flow Rate	 cfm cfm	 6000 ±10% 43,500 ±10%	Drawing M-96 P&ID, Tech. Spec. 3.7.10	A portion of the flow is directed to non-recirculated volumes such as the Upper cable Spreading Rooms that do not require occupancy during post-accident conditions. Accordingly, the intake into the "analyzed occupied volume" can be adjusted to reflect this.

Table 1.0 Input Parameters Common to All AST Radiological Analyses					
Item	Parameter	Unit	Value	Reference	Notes
1.8	CR HVAC Emergency Mode of Operation, Post Accident CR filtered Makeup Air Flow Rate CR filtered Recirc. Air Flow Rate CR unfiltered inleakage	 cfm cfm cfm	 6000 ±10% 43,500±10% 1000	 Tech. Spec. 3.7.10 See Notes Drawing M-96 P&ID See Notes	 Bounding Values of Tech Spec LCO 3.7.10. Proposed value of 1,000 cfm is to be used for AST.
1.9	Control Room Makeup Filter Efficiency Elemental Iodine Organic Iodine Aerosol/Particulate	 % 	 95 95 99	UFSAR Table 6.4-1a UFSAR Section 6.5.1 See Notes	These values listed represent reductions in filter efficiencies from current UFSAR values to be used in the AST analyses.
1.10	Control Room Recirculation Filter Efficiency Elemental Iodine Organic Iodine Aerosol/Particulate	 % 	 90 90 80	UFSAR Table 6.4-1a	

Table 1.0 Input Parameters Common to All AST Radiological Analyses					
Item	**Parameter**	**Unit**	**Value**	**Reference**	**Notes**
1.11	Delay to Switch CR HVAC from Mode 1 (Normal) to Mode 2 (Emergency) after receiving an isolation signal (manual initiation)	minutes	30	See Notes	Switchover is currently an automatic function. The potential for manual initiation by 30 minutes is analyzed as an AST benefit for analysis purposes. TS changes to reflect manual initiation may be made in the future. Potential manual initiation at 30 minutes is an increase from current accident analysis values.
1.12	Technical Support Center Parameters				
	Volume	ft^3	86,300	Existing value	
	Flow Rates				
	Makeup	cfm	900 ±10%	Drawing M-94 Sheet 2 Rev. P	
	Recirc		1100 ±10%	UFSAR Section E.75.3.2 UFSAR Table E.75-1	
	Makeup and Recirculation Filter Efficiency				
	Elemental Iodine	%	99		Determination of allowable inleakage and reductions in filter efficiencies from current UFSAR Table E.75-1 HEPA values are investigated as an AST benefit.
	Organic Iodine		99		
	Aerosol/Particulate		99		

Table 2.0 Input Parameters Common to All Radiological Analyses Modeling Coolant Activity Release

Item	Parameter	Unit	Value	Reference	Notes
2.1	Primary Coolant Activity Equilibrium Operation Associated Iodine concentrations I-131 I-132 I-133 I-134 I-135	μCi /gram dose equivalent I-131	 1.0 0.742 0.979 1.350 0.243 0.842	 Tech Spec LCO 3.4.16 RG 1.183 Existing values	For full power operation.
2.2	Secondary Coolant Activity	μCi /gram dose equivalent I-131	0.1	Tech Spec LCO 3.7.3	
2.3	Reactor Coolant System Operational Leakage	gpd gpd	600 For 4 steam generators 150 For one steam generator	Tech Spec LCO 3.4.13	The TS leakage is 600 gpd for 4 steam generators, 150 gpd for one steam generator. The total leakage to be evenly divided for the four steam generators is 1 gpm. For events involving a faulted steam generator, 0.5 gpm leak rate shall be used for faulted generator and 0.218 gpm shall be used for each of the intact generators. Assumed to be based on water at cold conditions. These assumptions are chosen to optimize accident induced leak rate limit and bound the values specified in Tech Spec LCO 3.4.13.

Table 2.0 Input Parameters Common to All Radiological Analyses Modeling Coolant Activity Release					
Item	**Parameter**	**Unit**	**Value**	**Reference**	**Notes**
2.4	Reactor Coolant System Operational Unidentified and Identified Leakage	gpm	12	Tech Spec Section 3.4.13	Based on leakage limits of TS 3.4.13 of 1.0 gpm unidentified leakage, 10 gpm of identified leakage, and 600 gpd of P/S leakage (rounded up to 12 gpm). This leakage, combined with the letdown flow cleanup, is used to determine the iodine appearance rate. Leakage is based on cold conditions.
2.5	Assumed Fuel Damage MSLB SGTR	-	 0 0	 UFSAR Section 15.1.5 UFSAR Section 15.6.3	No fuel damage is assumed. Use RG-1.183 guidance.
2.6	Iodine spike pre-accident (normal) appearance rate I-131 I-132 I-133 I-134 I-135	curies/ minute	 0.416 1.754 0.924 0.926 0.826	Existing values	These values are consistent with the historic Power Uprate calcs. These values were derived from the "500 times values" specified in the calc.

3.0 Input Parameters for the LOCA Radiological Consequence Analysis					
Item	**Parameter**	**Unit**	**Value**	**Reference**	**Notes**
3.1	Core Release Fraction Iodine Noble Gases	-	Per Table 2 in RG 1.183	RG 1.183	
3.2	Iodine Plate-out Fraction on Containment Surfaces	-	Time-dependent	RG 1.183	Powers natural deposition model in RADTRAD
3.3	Initial Iodine Species in Containment Aerosol Elemental Organic	 %	 95 4.85 0.15	RG 1.183	
3.4	Iodine released from ECCS to the environment Elemental Organic	 %	 97 3	RG 1.183	
3.5	Containment Leak Rate 0-24 Hours > 24 Hours	 weight %/day	 0.20 0.10	UFSAR Table 15.6-7 New AST analyzed values	The reduction in the design basis containment leak rate by 50% at 24 hours is consistent with guidance of RG 1.183. These leak rate values are increased in the analysis by a factor of 2 from those in the UFSAR Table 15.6-7.
3.6	Containment Spray Flow Actuation Time Including Delay	seconds	90	UFSAR Table 15.6-7	Biased high for conservatism based on 88.1 or 53.1+X* where X* denotes the time at which the containment pressure setpoint is reached (High-3 for CS).

3.0 *Input Parameters for the LOCA Radiological Consequence Analysis*					
Item	**Parameter**	**Unit**	**Value**	**Reference**	**Notes**
3.7	Spray Injection Flow Duration	Minutes	20.9	UFSAR Table 15.6-7	This value is based on containment spray switchover at lo-3 RWST level, maximum SI, and one containment spray pump running (because the analysis assumes the failure of one containment spray pump). The duration is from spray initiation.
3.8	Containment Spray Parameters: Containment Spray Flow Injection Recirculation	 gpm	 2950 2950	 UFSAR Section A6.5.4 UFSAR Table A6.5-1	The minimum amount of delivered flow based on 219 spray nozzles and a minimum delivered flow of 15 gpm per nozzle is 3285 gpm. For conservatism, a value of approximately 90 % is used.
	Fraction of Containment Sprayed	-	0.825	UFSAR Section 6.5.2.3 UFSAR Table A6.5-1	The value is based on the most conservative approach of minimum net sprayed containment volume and maximum net containment volume, as per the referenced UFSAR Table. Since the same flow rate is used during injection and recirculation phases, this value is also valid for both phases.
	Containment Volume Sprayed Unsprayed	ft^3 ft^3 ft^3	2.85E6 2.35E6 5.0E5	UFSAR Table 15.6-7 UFSAR Table 15.6-7 UFSAR Table 15.6-7	This volume is used to determine fraction of containment sprayed. See Note above.
	Containment Spray Fall Height	ft	141	UFSAR Table A6.5-1	This value bounds the minimum height between the spray ring header (elevation 567 feet, and the operating floor (elevation 426 feet. This value is unchanged from the historical Power Uprate calculations
	Containment Spray				

3.0 Input Parameters for the LOCA Radiological Consequence Analysis					
Item	**Parameter**	**Unit**	**Value**	**Reference**	**Notes**
	Temperature Nominal (norm. op.)	°F	120	See Notes	Unchanged. Conservatively greater than maximum RWST temperature in Tech Spec section 3.5.4. Consistent with containment analysis input per the historical Power Uprate calculations.
	Containment Temperature Maximum during LOCA	°F	260	UFSAR Table A6.5-1	Unchanged. Consistent with the historical Power Uprate calculations.
3.9	Number of Deck Fans Operating	#	2 of 4	UFSAR Table 15.6-7	
3.10	Deck Fan Flow Rate, per fan	cfm	65,000	UFSAR Table 15.6-7	Based on minimum heat removal and service water temperature of 100 °F.
3.11	Time from Spray Cessation Before Spray Recirculation Starts	min	10	UFSAR Table 15.6-7 See Notes	The instructions for switchover from injection to recirculation are provided in station procedures. The instructions call for continuous spray flow. However, if the RWST level drops to the empty level prior to the completion of the switchover, the operators are directed to secure the CS pump prior to performing the switchover. A conservative delay of 10 minutes is assumed.

3.0 Input Parameters for the LOCA Radiological Consequence Analysis					
Item	**Parameter**	**Unit**	**Value**	**Reference**	**Notes**
3.12	ECCS Recirculation Leakage	cc/hr	276,000	Engineering Evaluation New AST analytical allowance See Note	UFSAR Table 15.6-13 identifies the leakage as 3910 cc/hr. Per NRC guidance, this is doubled (SRP Section 15.6.5, Appendix B). Tech Spec Section 5.5.2 requires a primary coolant sources outside containment program. Procedures implement this program. The acceptance criterion specified for ECCS leakage is 3860 cc/hr. The proposed value bounds the procedure requirements. ECCS leakage at the proposed value will not interfere with ECCS operation over the accident duration.
3.13	Recirculation Loop Water Volume at 11.6 minutes	ft^3	38,979	UFSAR Table 15.6-12	This value is considered to be the minimum containment sump volume at ECCS switchover. The proposed value assumes the RCS volume for unit 2, which bounds unit 1 and takes into account maximum steam generator tube plugging level. At CS switchover, an additional 19,527 ft^3 should be added, for a total of 58,506 ft^3.
3.14	Fraction of core iodine in sump solution	-	0.4	RG 1.183	
3.15	Iodine partition coefficient for recirculation leakage	-	0.1	RG 1.183	
3.16	Sump Water Temperature	°F	260	UFSAR Table 15.6-12 See Note	Consistent with the historical Power Uprate calculations.

3.0 Input Parameters for the LOCA Radiological Consequence Analysis					
Item	**Parameter**	**Unit**	**Value**	**Reference**	**Notes**
3.17	Auxiliary Building Release Path Iodine Removal Efficiency Elemental iodine Organic iodine Particulates	%	 90 90 99	UFSAR Section 6.5.1.1.2 See Notes	 Flashed fluid Iodine is assumed to contain no aerosol.
3.18	Passive failure release from the ECCS recirculation path	-	N/A	RG 1.183	Since credit is taken for safety grade filtration system in the Auxiliary Building for ECCS leakage, the passive failure at 24 hours is not applicable.
3.19	Offsite Dose Acceptance Criteria TEDE Analysis Release Duration	 rem days	 25 30	RG 1.183	The limits apply to the doses due to the combined release paths.
3.20	CR Dose Acceptance Criteria TEDE	rem	5	10CFR 50.67	

Table 4.0 Input Parameters for the FHA Outside Containment Radiological Consequence Analysis

Item	Parameter	Unit	Value	Reference	Notes
4.1	Power Peaking Factor for damaged fuel	-	1.7	UFSAR Section 4.3.2.2.6 (p. 4.3-20) UFSAR Section 15.7.4.2	
4.2	Time From Reactor Shutdown to Beginning of Fuel Handling Operations	hr	48	UFSAR Section 15.7.4.2	Bounds TRM 3.9.a
4.3	Gap Activity I-131 Kr-85 Other Noble Gases Other Halogens Alkali Metals	 %	2 times the following: 0.08 0.10 0.05 0.05 0.12	R.G. 1.183	As a significant number of fuel assemblies not qualifying for AST due to their containing fuel rods with maximum linear heat generation rates exceeding 6.3 kilowatts per foot peak rod average power for burnups exceeding 54 GWD/MTU, the fuel will be treated as having gap fractions a factor of 2 greater than the RG 1.183 values, as accepted by NRC for Fort Calhoun AST applications.
4.4	Number of Assemblies Damaged	#	1	UFSAR Section 15.7.4.2.1 See Notes	264 rods are assumed to be damaged consistent with current analysis (UFSAR Section 15.7.4.2.1). This is one full assembly.
4.5	Pool Scrubbing Factor – Overall Elemental Organic (and Noble Gases)	 -	 200 500 1	RG 1.183	Per RG 1.183 based on 23 feet of water coverage.
4.6	Release Path Filter Efficiency	%	0	See Note	Filtration is not credited for AST analysis.
4.7	Duration of release	hr	2	RG 1.183	

Table 4.0 Input Parameters for the FHA Outside Containment Radiological Consequence Analysis

Item	Parameter	Unit	Value	Reference	Notes
4.8	Chemical form of radioiodine released from the fuel to the spent fuel pool Cesium iodide Elemental iodine Organic iodine	% 	95 4.85 0.15	RG 1.183	
4.9	Chemical form of radioiodine released from the pool to the building Elemental iodine Organic iodine	%	57 43	RG 1.183	
4.10	Depth of Water above the Top of Reactor Vessel Flange and Fuel Assemblies in Spent Fuel Pool	ft	23	Tech Spec LCO 3.9.7 Tech Spec 3.7.14	
4.11	Offsite Dose Acceptance Criteria TEDE	rem	6.3	RG 1.183	
4.12	CR Offsite Dose Acceptance Criteria TEDE	rem	5	10CFR 50.67	

Table 5.0 Input Parameters for the Rod Ejection Radiological Consequence Analysis

Item	Parameter	Unit	Value	Reference	Notes
5.1	Portion of Core Experiencing Cladding Damage	%	10	UFSAR Table 15.4-4	Bounds value predicted by safety analysis. The acceptance criteria for fuel melting is 10%.
5.2	Melted fuel	% of core	0.250	New AST analyzed value	Based on 50% of the rods that violate the DNB limit having melting in the inner 10% over 50% of the axial length: 0.10 x 0.5 x 0.1 x 0.5 = 0.00250 = 0.250%
5.3	Gap Activity Alkali Metals	 %	 0.24	RG 1.183 RG 1.183 Section 3.2, and Appendix H, Paragraph 1	As a significant number of fuel assemblies not qualifying for AST due to their containing fuel rods with maximum linear heat generation rates exceeding 6.3 kilowatts per foot peak rod average power for burnups exceeding 54 GWD/MTU, the fuel will be treated as having gap fractions a factor of 2 greater than the RG 1.183 values, as accepted by NRC for Fort Calhoun AST applications.
5.4	Fraction of activity released to containment From gap inventory Iodine Noble Gases From melted fuel Iodine Noble gas Iodine plateout onto containment surfaces	 - - -	 1.0 1.0 0.5 1.0 0.5	RG 1.183	

Table 5.0 Input Parameters for the Rod Ejection Radiological Consequence Analysis					
Item	**Parameter**	**Unit**	**Value**	**Reference**	**Notes**
5.5	Iodine Chemical Species in Containment Aerosol Elemental Organic Iodine Chemical Species in Release from SG to Environment Elemental Organic	% %	95 4.85 0.15 97 3	RG 1.183	.
5.6	Iodine Removal Coefficients in Containment	N/A	See Notes	RG 1.183	Typically, no credit is taken for continuing iodine removal in the containment for the rod ejection accident, however under provisions allowed by the AST governing RG 1.183, Power's model for particulate deposition removal may be credited, if 50% plateout is not credited.
5.7	Containment Leak Rate 0-24 Hours > 24 Hours	weight %/day	0.20 0.10	UFSAR Table 15.4-4 See Notes New AST analyzed values	The reduction in the design basis containment leak rate by 50% at 24 hours is consistent with guidance of RG 1.183. These values are increased in the analysis by a factor of 2 from those in the UFSAR Table 15.6-7.

Table 5.0 Input Parameters for the Rod Ejection Radiological Consequence Analysis

Item	Parameter	Unit	Value	Reference	Notes
5.8	Fraction of activity released to primary coolant (for primary to secondary leakage pathway) From gap inventory Iodine Noble Gases From melted fuel Iodine Noble gas	 - -	 1.0 1.0 0.5 1.0	RG 1.183	
5.9	Iodine Chemical Species in Primary Coolant Elemental Organic Particulate	%	100 0 0	Existing values	
5.10	Steam Released to Environment 0 - 200 sec 200 - 4000 sec	lb/sec	3000 500	UFSAR Table 15.4-4 Existing values	Bounds release predicted by small break LOCA analysis.
5.11	Chemical form of radioiodine released to the containment atmosphere Cesium iodine Elemental iodine Organic iodide	%	95 4.85 0.15	RG 1.183	
5.12	Offsite Dose Acceptance Criteria TEDE	rem	6.3	RG 1.183	
5.13	CR Offsite Dose Acceptance Criteria (TEDE)	rem	5	10CFR 50.67	

Table 6.0 Input Parameters for the Locked Rotor Radiological Consequence Analysis

Item	Parameter	Unit	Value	Reference	Notes
6.1	Fraction of Core Experiencing Cladding Damage with Failed-Open PORV	%	2.0	UFSAR Table 15.3-4	The value of 2% is used, the reload limit for safety analysis, bounding the 0.1% value in the existing design basis calculation. The current alternative analysis for no PORV failure and 5% fuel failure is overly conservative.
6.2	Gap Fractions I-131 Kr-85 Other Noble Gases Other Halogens Alkali Metals	-	(2 times the following): 0.08 0.10 0.05 0.05 0.12	RG 1.183	As a significant number of fuel assemblies not qualifying for AST due to their containing fuel rods with maximum linear heat generation rates exceeding 6.3 kilowatts per foot peak rod average power for burnups exceeding 54 GWD/MTU, the fuel will be treated as having gap fractions a factor of 2 greater than the RG 1.183 values, as accepted by NRC for Fort Calhoun AST applications.
6.3	SG Iodine Partition Factor SG Aerosol Carryover	-	0.01 0.001	RG 1.183 ANSI/ANS-18.1	Credited only for non-faulted SG, with PORV failure assumed in faulted SG
6.4	Steam Released to Environment 0 - 2 hrs 2 - 8 hrs 8 - 40 hrs 40 - 64 hrs	lbm	 719,000 1,109,000 2,664,000 0	Existing values, UFSAR Table 15.3-3 UFSAR Table 15.3-4	As per RG 1.183, release duration is until cold shutdown is established and releases from the steam generators have been terminated.
6.5	RHR Cut-in Time	hours	40	UFSAR Table 15.3-3 UFSAR Table 15.3-3	Time for termination of release due to PORV steaming.
6.6	Chemical form of radioiodine released from the fuel Cesium iodide Elemental iodide Organic iodide	%	 95 4.85 0.15	RG 1.183	

Table 6.0 Input Parameters for the Locked Rotor Radiological Consequence Analysis					
Item	**Parameter**	**Unit**	**Value**	**Reference**	**Notes**
6.7	Iodine Releases from steam generator to environment Elemental Organic	 % 	 97 3	RG 1.183	
6.8	Offsite Dose Acceptance Criteria TEDE	 rem	 2.5	RG 1.183	
6.9	CR Offsite Dose Acceptance Criteria TEDE	 rem	 5	10CFR 50.67	

7.0 Input Parameters for the MSLB Radiological Consequence Analysis					
Item	**Parameter**	**Unit**	**Value**	**Reference**	**Notes**
7.1	Steam Released to Environment Faulted SG (in 2 minutes) Intact SGs 0 – 2 hours 2 – 8 hours 8 – 40 hours Primary to Secondary Leakage Faulted SG Intact SG (each)	 lbm gpm gpm	 167,000 442,000 977,000 2,216,000 0.5 0.218	UFSAR Table 15.1-3	The total leakage to be evenly divided for the four steam generators is 1 gpm. For events involving a faulted steam generator, 0.5 gpm leak rate shall be used for faulted generator and 0.218 gpm shall be used for each of the intact generators. Assumed to be based on water at cold conditions. These assumptions are chosen to optimize accident induced leak rate limit and bound the values specified in Tech Spec LCO 3.4.13.
7.2	Duration of steam releases from intact SGs	hr	40	UFSAR Table 15.1-3	
7.3	Duration of activity release due to leakage of primary coolant to the faulted SG	hr	40	UFSAR Table 15.1-3	
7.4	Offsite Dose Acceptance Criteria Pre-accident Iodine Spike Case TEDE Accident initiated Iodine Spike Case TEDE	 rem rem	 25 2.5	RG 1.183	
7.5	CR Offsite Dose Acceptance Criteria Pre-accident Iodine Spike Case TEDE CR Accident initiated Iodine Spike Case TEDE	 rem rem	 5 5	10CFR 50.67	

7.0 Input Parameters for the MSLB Radiological Consequence Analysis					
Item	**Parameter**	**Unit**	**Value**	**Reference**	**Notes**
7.6	Accident Iodine Spike		Factor of 500	RG 1.183	In addition to pre-accident iodine spike case.
7.7	Primary and secondary coolant volumes: Primary Secondary Coolant – 3 Intact SGs Faulted SG	gm	2.063E8 1.017E8 7.575E7	See Notes	These values are consistent with the historic Power Uprate calcs.
7.8	Noble Gas releases through the faulted SG due to primary to secondary leakage: KR-85m KR-85 KR-87 KR-88 XE-131m XE-133m XE-133 XE-135m XE-135 XE-138	Curies	3.713E2 1.467E3 2.372E2 6.911E2 6.829E2 7.530E2 5.178E4 1.007E2 1.593E3 1.368E2	UFSAR Table 15.0-9 See Notes	These values, based on operation with 1% fuel defects, are consistent with the historic Power Uprate calcs, The values are also consistent with UFSAR Table 15.0-9 with a RCS "volume" of 2.063E8 gm.

Table 8.0 Input Parameters for SGTR Radiological Consequence Analysis					
Item	**Parameter**	**Unit**	**Value**	**Reference**	**Notes**
8.1	Accident mitigation recovery phases associated with the sequence of events: • Start of event until Reactor trip • Reactor Trip until ruptured SG PORV is opened • Ruptured SG PORV opening until ruptured SG PORV is isolated • Ruptured SG PORV Isolation until RCS cooldown is initiated • RCS cooldown initiation until Break Flow flashing ceases • End of Break Flow flashing until RCS depressurization initiation • RCS depressurization initiation until all Break Flow ceases	seconds	0 – 445 445 – 465 465 – 1700 1700 – 2800 2800 – 3000 3000 – 3500 3500 – 3590	See Notes	These values are consistent with the historic Power Uprate calcs.
8.2	Iodine Spike Release Period	hours	8	See Notes	This value is consistent with the historic Power Uprate calcs.

Table 8.0 Input Parameters for SGTR Radiological Consequence Analysis

Item	Parameter	Unit	Value	Reference	Notes
8.3	Ruptured SG steam releases and partition factors	lbm	See Table 8.1	See Notes	This value is consistent with and bounding for the historic Power Uprate calcs.
8.4	Intact SG steam releases and time period and partition factors	lbm	See Table 8.2	See Notes	This value is consistent with and bounding for the historic Power Uprate calcs.
8.5	Full power steam release through the Condenser until the time of reactor trip and loss of offsite power	lbm/hour	16.04E6	See Notes	This value is consistent with the historic Power Uprate calcs.
8.6	Primary and secondary coolant masses: Ruptured SG - initial Ruptured SG at break flow termination Intact SGs – initial Intact SGs at break flow termination RCS	 lbm	 7.03 E4 2.04 E5 2.11 E5 2.63 E5 5.02 E5	See Notes	These values are consistent with the historic Power Uprate calcs.

Table 8.1 Steam Releases and Partition Factors Inputs for SGTR (Faulted Steam Generator)

RCS to Ruptured SG Volume (Un-Flashed Break Flow)		
Time	Time	Adjusted Average Break Flow Rate
(sec)	(hrs)	(g/min)
0	0.000	1.462E+06
445	0.124	1.215E+06
465	0.129	1.454E+06
1700	0.472	1.534E+06
2800	0.778	1.426E+06
3000	0.833	1.140E+06
3500	0.972	6.505E+05
3590	0.997	0.000E+00
7200	2.000	0.000E+00
28800	8.000	0.000E+00

(Noble Gas) RCS to Environment (Flashed Break Flow)		
Time	Time	Total Primary to Secondary Flow Rate
(sec)	(hrs)	(g/min)
0	0.000	1.575E+06
445	0.124	1.243E+06
465	0.129	1.504E+06
1700	0.472	1.551E+06
2800	0.778	1.434E+06
3000	0.833	1.142E+06
3500	0.972	6.530E+05
3590	0.997	2.475E+03
7200	2.000	2.475E+03
28800	8.000	0.000E+00

(Iodine) RCS to Environment (Flashed Break Flow)				
Time	Time	Mass Flow Rate	Partition Factor	Iodine Transfer Rate
(sec)	(hrs)	(g/min)	(Iodine)	(g/min)
0	0.000	1.108E+05	0.01	1.1080E+03
445	0.124	2.632E+04	1	2.6320E+04
465	0.129	4.844E+04	1	4.8440E+04
1700	0.472	1.475E+04	1	1.4750E+04
2800	0.778	6.260E+03	1	6.2600E+03
3000	0.833	0.000E+00	1	0.0000E+00
3500	0.972	0.000E+00	1	0.0000E+00
3590	0.997	0.000E+00	1	0.0000E+00
7200	2.000	0.000E+00	1	0.0000E+00
28800	8.000	0.000E+00	1	0.0000E+00

Ruptured SG to Environment (Steam Release)				
Time	Time	Mass Flow Rate	Partition Factor	Iodine Transfer Rate
(sec)	(hrs)	(g/min)	(Iodine)	(g/min)
0	0	3.032E+07	0.0001	3.0320E+03
465	0.129	2.076E+06	0.01	2.0760E+04
1700	0.472	0.000E+00	0.01	0.0000E+00
7200	2	3.352E+04	0.01	3.3520E+02

Table 8.2 Steam Releases and Partition Factors Inputs for SGTR (Intact Steam Generator)

Intact SGs to Environment (Steam Release)				
Time	Time	Mass Flow Rate	Partition Factor	Iodine Transfer Rate
(sec)	(hrs)	(g/min)	(Iodine)	(g/min)
0	0.000	9.095E+07	0.0001	9.0950E+03
2800	0.778	2.115E+07	0.01	2.1150E+05
3500	0.972	0.000E+00	0.01	0.0000E+00
7200	2.000	1.474E+06	0.01	1.4740E+04
28800	8.000	0.000E+00	0.01	0.0000E+00

ATTACHMENT 7

Regulatory Guide Conformance Tables

NOTE:	In Tables A-H, the text shown in the "RG Position" columns is taken from RG 1.183; therefore, references to footnotes, tables, and numbered references, may be found in RG 1.183.
Table A	Conformance with Regulatory Guide 1.183 Main Sections
Table B	Conformance with Regulatory Guide 1.183 Appendix A (Loss-of-Coolant Accident)
Table C	Conformance with Regulatory Guide 1.183 Appendix B (Fuel Handling Accident)
Table D	Conformance with Regulatory Guide 1.183 Appendix H (PWR Rod Ejection Accident)
Table E	Conformance with Regulatory Guide 1.183 Appendix G (PWR Locked Rotor Accident)
Table F	Conformance with Regulatory Guide 1.183 Appendix E (PWR Main Steam Line Break)
Table G	Conformance with Regulatory Guide 1.183 Appendix F (PWR Steam Generator Tube Rupture Accident)
Table H	Conformance with Regulatory Guide 1.194 (Diffuse Area Source Guidance)

REGULATORY GUIDE 1.183 COMPARISON

Table A: Conformance with Regulatory Guide 1.183 Main Sections			
RG Section	RG Position	Analysis	Comments
3.1	The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty. 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 core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 or ORIGEN-ARP. Core inventory factors (Ci/MWt) provided in TID 14844 and used in some analysis computer codes were derived for low burnup, low enrichment fuel and should not be used with higher burnup and higher enrichment fuels.	Conforms	ORIGEN 2.1 based methodology was used to determine the bounding core inventory. These source terms were evaluated at end-of-cycle and at beginning of cycle (100 effective full power days (EFPD) to achieve equilibrium) conditions. The worst-case inventory was used for each of the selected 60 isotopes for the RADTRAD analyses. These values were then converted to units of Ci/MWt. Accident analyses are based on a 3658.3 MWt power level, based on the current accident analysis design basis allowance for instrument uncertainty. Source terms are based on an 18-month fuel cycle with 542.9 EFPD per cycle.
3.1	For the DBA LOCA, all fuel assemblies in the core are assumed to be affected and the core average inventory should be used. For DBA events that do not involve the entire core, the fission product inventory of each of the damaged fuel rods is determined by dividing the total core inventory by the number of fuel rods in the core. To account for differences in power level across the core, radial peaking factors from the facility's core operating limits report (COLR) or technical specifications should be applied in determining the inventory of the damaged rods.	Conforms	Peaking factors of 1.7 are used for DBA events that do not involve the entire core, with fission product inventories for damaged fuel rods determined by dividing the total core inventory by the number of fuel rods in the core.
3.1	No adjustment to the fission product inventory should be made for events postulated to occur during power operations at less than full rated power or those postulated to occur at the beginning of core	Conforms	No adjustments for less than full power are made in any analyses.

Table A: Conformance with Regulatory Guide 1.183 Main Sections

RG Section	RG Position	Analysis	Comments
	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.		
3.2	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 1 for BWRs and Table 2 for PWRs. These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1. **Table 2** **PWR Core Inventory Fraction Released Into Containment** **Group** / **Gap Release Phase** / **Early In-Vessel Phase** / **Total** Noble Gases 0.05 0.95 1.0 Halogens 0.05 0.35 0.4 Alkali Metals 0.05 0.25 0.3 Tellurium Metals 0.00 0.05 0.05 Ba, Sr 0.00 0.02 0.02 Noble Metals 0.00 0.0025 0.0025 Cerium Group 0.00 0.0005 0.0005 Lanthanides 0.00 0.0002 0.0002 Footnote 10: The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak rod burnup up to 62,000 MWD/MTU. The data in this section may not be applicable to cores containing mixed oxide (MOX) fuel.	Conforms	The release fractions from Regulatory Position 3.1, Table 1 are used. Footnote 10 criteria are met.

Table 2 (from RG Position 3.2 above):

Group	Gap Release Phase	Early In-Vessel Phase	Total
Noble Gases	0.05	0.95	1.0
Halogens	0.05	0.35	0.4
Alkali Metals	0.05	0.25	0.3
Tellurium Metals	0.00	0.05	0.05
Ba, Sr	0.00	0.02	0.02
Noble Metals	0.00	0.0025	0.0025
Cerium Group	0.00	0.0005	0.0005
Lanthanides	0.00	0.0002	0.0002

Table A: Conformance with Regulatory Guide 1.183 Main Sections

RG Section	RG Position	Analysis	Comments
3.2	For non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in Table 3. The release fractions from Table 3 are used in conjunction with the fission product inventory calculated with the maximum core radial peaking factor. **Table 3[11]** **Non-LOCA Fraction of Fission Product Inventory in Gap** **Group** — **Fraction** I-131 — 0.08 Kr-85 — 0.10 Other Noble Gases — 0.05 Other Halogens — 0.05 Alkali Metals — 0.12 Footnote 11: The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for rods with burnups that exceed 54 GWD/MTU. As an alternative, fission gas release calculations performed using NRC-approved methodologies may be considered on a case-by-case basis. To be acceptable, these calculations must use a projected power history that will bound the limiting projected plant-specific power history for the specific fuel load. For the BWR rod drop accident and the PWR rod ejection accident, the gap fractions are assumed to be 10% for iodines and noble gases.	Exception taken (as approved in a previous submittal by another Licensee)	The analysis does not fully comply with Note 11 of Table 3 since typical Byron and Braidwood core designs indicate that there are fuel assemblies that exceed the 6.3 kW/ft while >54GWD/MTU. Previous analyses (ANS 5.4) for TMI-1 have shown that those fuel assemblies exceeding these limits had no increase in gap release fractions of concern. Therefore, doubling of the gap fractions in Table 3 is conservative as used and approved in the Fort Calhoun AST submittal. Peaking factor of 1.7 used for DBA events that do not involve the entire core.

<table>
<tr><th colspan="4">Table A: Conformance with Regulatory Guide 1.183 Main Sections</th></tr>
<tr><th>RG Section</th><th>RG Position</th><th>Analysis</th><th>Comments</th></tr>
<tr><td>3.3</td><td>Table 4 tabulates the onset and duration of each sequential release phase for DBA LOCAs at PWRs and BWRs. The specified onset is the time following the initiation of the accident (i.e., time = 0). The early in-vessel phase immediately follows the gap release phase. The activity released from the core during each release phase should be modeled as increasing in a linear fashion over the duration of the phase. For non-LOCA DBAs, in which fuel damage is projected, the release from the fuel gap and the fuel pellet should be assumed to occur instantaneously with the onset of the projected damage.
Table 4
LOCA Release Phases
<table>
<tr><th></th><th colspan="2">PWRs</th><th colspan="2">BWRs</th></tr>
<tr><th>Phase</th><th>Onset</th><th>Duration</th><th>Onset</th><th>Duration</th></tr>
<tr><td>Gap Release</td><td>30 sec</td><td>0.5 hr</td><td>2 min</td><td>0.5 hr</td></tr>
<tr><td>Early In-Vessel</td><td>0.5 hr</td><td>1.3 hr</td><td>0.5 hr</td><td>1.5 hr</td></tr>
</table></td><td>Conforms</td><td>The PWR durations from Table 4 are used.

The LOCA activity released from the core is modeled in a linear fashion over the duration of the release phases.

Non-LOCA DBAs are modeled as an instantaneous release from the fuel.</td></tr>
<tr><td>3.3</td><td>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 for the specific facility. In the absence of approved alternatives, the gap release phase onsets in Table 4 should be used.</td><td>Not Applicable</td><td>Neither Byron nor Braidwood use leak-before-break methodology for DBA analyses.</td></tr>
</table>

Table A: Conformance with Regulatory Guide 1.183 Main Sections			
RG Section	**RG Position**	**Analysis**	**Comments**
3.4	Table 5 lists the elements in each radionuclide group that should be considered in design basis analyses. **Table 5** **Radionuclide Groups** **Group** — **Elements** Noble Gases — Xe, Kr Halogens — I, Br Alkali Metals — Cs, Rb Tellurium Group — Te, Sb, Se, Ba, Sr Noble Metals — Ru, Rh, Pd, Mo, Tc, Co Lanthanides — La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am Cerium — Ce, Pu, Np	Conforms	The nuclides used are the 60 identified as being potentially important dose contributors to total effective dose equivalent (TEDE) in the RADTRAD code, which encompasses those listed in RG 1.183, Table 5. The Co-58 and Co-60 values used are those from the RADTRAD defaults (activation products). All other isotope activities were determined using ORIGEN.
3.5	Of the radioiodine released from the reactor coolant system (RCS) to the containment in a postulated accident, 95 percent of the iodine released should be assumed to be cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. This includes releases from the gap and the fuel pellets. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form. The same chemical form is assumed in releases from fuel pins in FHAs and from releases from the fuel pins through the RCS in DBAs other than FHAs or LOCAs. However, the transport of these iodine species following release from the fuel may affect these assumed fractions. The accident-specific appendices to this regulatory guide provide additional details.	Conforms	This guidance was applied in the analyses. (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. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form.)

Table A: Conformance with Regulatory Guide 1.183 Main Sections			
RG Section	**RG Position**	**Analysis**	**Comments**
3.6	The amount of fuel damage caused by non-LOCA design basis events should be analyzed to determine, for the case resulting in the highest radioactivity release, the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and the fraction of fuel elements for which the fuel clad is breached. Although the NRC staff has traditionally relied upon the departure from nucleate boiling ratio (DNBR) as a fuel damage criterion, licensees may propose other methods to the NRC staff, such as those based upon enthalpy deposition, for estimating fuel damage for the purpose of establishing radioactivity releases.	Conforms	The currently licensed and approved assumptions regarding the amount of fuel damage for non-LOCA design basis events is used in the AST analyses.
4.1.1	The dose calculations should determine the TEDE. TEDE is the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure. The calculation of these two components of the TEDE should consider all radionuclides, including progeny from the decay of parent radionuclides that are significant with regard to dose consequences and the released radioactivity.	Conforms	TEDE is calculated, with significant progeny included.
4.1.2	The exposure-to-CEDE factors for inhalation of radioactive material should be derived from the data provided in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers" (Ref. 19). Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (Ref. 20), provides tables of conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the CEDE.	Conforms	Federal Guidance Report 11 dose conversion factors (DCFs) are used.

Table A: Conformance with Regulatory Guide 1.183 Main Sections			
RG Section	RG Position	Analysis	Comments
4.1.3	For the first 8 hours, the breathing rate of persons offsite should be assumed to be 3.5×10^{-4} cubic meters per second. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.8×10^{-4} cubic meters per second. After that and until the end of the accident, the rate should be assumed to be 2.3×10^{-4} cubic meters per second.	Conforms	The values that correspond to the rounded values in Section 4.1.3 of RG 1.183 are used.
4.1.4	The DDE should be calculated assuming submergence in semi-infinite cloud assumptions with appropriate credit for attenuation by body tissue. The DDE is nominally equivalent to the effective dose equivalent (EDE) from external exposure if the whole body is irradiated uniformly. Since this is a reasonable assumption for submergence exposure situations, EDE may be used in lieu of DDE in determining the contribution of external dose to the TEDE. Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil" (Ref. 21), provides external EDE conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the EDE.	Conforms	Federal Guidance Report 12 conversion factors are used.

Table A: Conformance with Regulatory Guide 1.183 Main Sections			
RG Section	**RG Position**	**Analysis**	**Comments**
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 (see also Table 6). Footnote 14: With regard to the EAB TEDE, the maximum two-hour value is the basis for screening and evaluation under 10 CFR 50.59. Changes to doses outside of the two-hour window are only considered in the context of their impact on the maximum two-hour EAB TEDE.	Conforms	The maximum two-hour LOCA EAB dose starts as follows: __Containment Leakage__: 11.01 rem TEDE 0.3 to 2.3 hours __ECCS Leakage__: 1.20 rem TEDE 1.8 to 3.8 hours Conservatively, the maximum 2-hour period dose was determined by adding the maximum 2-hour dose for each of the components listed above even though they do not occur simultaneously.
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	This guidance is applied in the analyses through the use of the RADTRAD computer code.
4.1.7	No correction should be made for depletion of the effluent plume by deposition on the ground.	Conforms	No such corrections are made in the analyses.
4.2.1	The TEDE analysis should consider all sources of radiation that will cause exposure to control room personnel. The applicable sources will vary from facility to facility, but typically will include: • Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility,	Conforms	The principal source of dose within the control room is due to airborne activity within the CR. The dose contributions from the other sources, such as direct shine, were also considered.

Table A: Conformance with Regulatory Guide 1.183 Main Sections			
RG Section	**RG Position**	**Analysis**	**Comments**
	• Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope, • Radiation shine from the external radioactive plume released from the facility, • Radiation shine from radioactive material in the reactor containment, • Radiation shine from radioactive material in systems and components inside or external to the control room envelope, e.g., radioactive material buildup in recirculation filters. •		
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	The source term, transport, and release methodology is the same for both the control room and offsite locations.
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	This guidance is applied in the analyses.
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 Post-accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear	Conforms	Engineered safety features that mitigate airborne radioactive material within the control room are credited. These features are qualified and acceptable per the referenced guidance. Control Room and intake and recirculation filtration are credited. Radiation isolation

Table A: Conformance with Regulatory Guide 1.183 Main Sections			
RG Section	**RG Position**	**Analysis**	**Comments**
	Power Plants" (Ref. 25), for guidance.		mode has been analyzed with manual initiation within 30 minutes. After this period, credit is taken for HEPA and charcoal adsorber efficiencies.
4.2.5	Credit should generally not be taken for the use of personal protective equipment or prophylactic drugs. Deviations may be considered on a case-by-case basis.	Conforms	Such credits are not taken.
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	Standard occupancy factors and breathing rate are used throughout the analyses.
4.2.7	Control room doses should be calculated using dose conversion factors identified in Regulatory Position 4.1 above for use in offsite dose analyses. The DDE from photons may be corrected for the difference between finite cloud geometry in the control room and the semi-infinite cloud assumption used in calculating the dose conversion factors. The following expression may be used to correct the semi-infinite cloud dose, DDE_{∞}, to a finite cloud dose, DDE_{finite}, where the control room is modeled as a hemisphere that has a volume, V, in cubic feet, equivalent to that of the control room (Ref. 22). $$DDE_{finite} = \frac{DDE_{\infty}V^{0.338}}{1173}$$	Conforms	The equation given is utilized for finite cloud correction when calculating external doses due to the airborne activity inside the control room.
4.3	The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-	Conforms	TSC habitability has been re-determined using AST and has been determined acceptable. The EOF is sufficiently far

Table A: Conformance with Regulatory Guide 1.183 Main Sections			
RG Section	RG Position	Analysis	Comments
	0737 (Ref. 2). Design envelope source terms provided in NUREG-0737 should be updated for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of TEDE. Integrated radiation exposure of plant equipment should be determined using the guidance of Appendix I of this guide.		away from the site (outside the LPZ) such that analysis is not required.
5.1.1	The evaluations required by 10 CFR 50.67 are re-analyses of the design basis safety analyses and evaluations required by 10 CFR 50.34; they are considered to be a significant input to the evaluations required by 10 CFR 50.92 or 10 CFR 50.59. These analyses should be prepared, reviewed, and maintained in accordance with quality assurance programs that comply with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.	Conforms	These analyses were prepared as specified in the guidance. These analyses have been prepared and reviewed in accordance with a quality assurance program that complies with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.
5.1.2	Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The single active component failure that results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences.	Conforms	Accident mitigation features credited in these analyses 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. Single active failures and loss of offsite power were also considered where required.

Table A: Conformance with Regulatory Guide 1.183 Main Sections

RG Section	RG Position	Analysis	Comments
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 assumptions are used. The effects of tolerance values were evaluated. Those values that produce the highest doses were used in the analyses.
5.1.4	Licensees should ensure that analysis assumptions and methods are compatible with the AST and the TEDE criteria.	Conforms	Analysis assumptions and methods are compatible with the AST and the TEDE criteria per this guidance.
5.3	Atmospheric dispersion values (χ/Q) for the EAB, the LPZ, and the control room that were approved by the staff during initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified by this guide. Methodologies that have been used for determining χ/Q values are documented in Regulatory Guides 1.3 and 1.4, Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and the paper, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19". References 22 *[Murphy – Campe]* and 28 *[RG 1.145]* (of RG 1.183) should be used if the FSAR X/Q values are to be revised or if values are to be determined for new release points or receptor distances. Fumigation should be considered where applicable for the EAB and LPZ. For the EAB, the assumed fumigation period should be timed to be included in the worst 2-hour exposure period. The NRC computer code PAVAN implements Regulatory Guide 1.145 and its use is acceptable to the NRC staff. The methodology of the NRC computer code ARCON96 is generally acceptable to the NRC staff for use in determining control room χ/Q values.	Conforms	New atmospheric dispersion values (χ/Q) for the EAB, the LPZ, control room, and the TSC were developed, using meteorological data for the years 1994-1998. ARCON96 and PAVAN were used with these data to determine control room, EAB, and LPZ atmospheric dispersion values. Since there is no tall stack, no fumigation is considered. Control room χ/Qs were developed in conformance with the guidance provided in RG-1.194.

Table B: Conformance with Regulatory Guide 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	**RG Position**	**Analysis**	**Comments**
1	Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.	Conforms (with exception relating to Table 3, footnote 11)	*Fission Product Inventory:* Bounding core source terms are developed using ORIGEN-2.1 based methodology. *Release Fractions:* Release fractions are per Table 2 of RG 1.183, and are implemented by RADTRAD. Non-LOCA Table 3 release fractions are doubled in the analyses to account for the effects of exceeding the LHGR value described in footnote 11. *Timing of Release Phases:* Release Phases are per Table 4 of RG 1.183, and are implemented by RADTRAD. *Radionuclide Composition:* Radionuclide grouping is per Table 5 of RG 1.183, as implemented in RADTRAD. *Chemical Form:* Treatment of release chemical form is per RG 1.183, Section 3.5.
2	If the sump or suppression pool pH is controlled at values of 7 or greater, the chemical form of radioiodine released to the containment should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine species, including those from iodine re-evolution, for sump or suppression pool pH values less than 7 will be evaluated on a case-by-case basis. Evaluations of pH should consider the effect of acids and bases created during the LOCA event, e.g., radiolysis products. With the exception of elemental and organic iodine and noble gases, fission products should	Conforms	The stated distributions of iodine chemical forms are used in the analyses. The post-LOCA containment sump pH has previously been evaluated, including consideration of the effects of acids and bases created during the LOCA event, the effects of key fission product releases, and the

Table B: Conformance with Regulatory Guide 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	Analysis	Comments
	be assumed to be in particulate form.		impact of NaOH injection. Containment sump pH remains above 7 for at least 30 days.
3.1	The radioactivity released from the fuel should be assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment in PWRs or the drywell in BWRs as it is released. This distribution should be adjusted if there are internal compartments that have limited ventilation exchange. The suppression pool free air volume may be included provided there is a mechanism to ensure mixing between the drywell to the wetwell. The release into the containment or drywell should be assumed to terminate at the end of the early in-vessel phase.	Conforms	The radioactivity release from the fuel is assumed to instantaneously and homogeneously mix throughout the containment air space as it is released. Recirculation fans provide a mixing mechanism within the containment.
3.2	Reduction in airborne radioactivity in the containment by natural deposition within the containment may be credited. Acceptable models for removal of iodine and aerosols are described in Chapter 6.5.2, "Containment Spray as a Fission Product Cleanup System," of the Standard Review Plan (SRP), NUREG-0800 (Ref. A-1) and in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments" (Ref. A-2). The latter model is incorporated into the analysis code RADTRAD (Ref. A-3).	Conforms	For B/B, the RADTRAD computer program, including the Powers Natural Deposition algorithm based on NUREG/CR-6189, is used for modeling aerosol deposition in Containment. No natural deposition is assumed for elemental or organic iodine. The lower bound (10%) level of deposition credit is used.
3.3	Reduction in airborne radioactivity in the containment by containment spray systems that have been designed and are maintained in accordance with Chapter 6.5.2 of the SRP (Ref. A-1) may be credited. Acceptable models for the removal of iodine and aerosols are described in Chapter 6.5.2 of the SRP and NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays"1 (Ref. A-4). This simplified model is incorporated into the analysis code RADTRAD (Refs. A-1 to A-3).	Conforms	A qualified Containment Spray System is an available design feature at both Byron and Braidwood. The conservatively analyzed containment volume is 2.85E6 cubic feet, with 82.5% of this volume sprayed. The sprayed volume is 2.35125E6 cubic feet, unsprayed

Table B: Conformance with Regulatory Guide 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	Analysis	Comments
	The evaluation of the containment sprays should address areas within the primary containment that are not covered by the spray drops. The mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building, provided that adequate flow exists between these regions, is assumed to be two turnovers of the unsprayed regions per hour, unless other rates are justified. The containment building atmosphere may be considered a single, well-mixed volume if the spray covers at least 90% of the volume and if adequate mixing of unsprayed compartments can be shown. The SRP sets forth a maximum decontamination factor (DF) for elemental iodine based on the maximum iodine activity in the primary containment atmosphere when the sprays actuate, divided by the activity of iodine remaining at some time after decontamination. The SRP also states that the particulate iodine removal rate should be reduced by a factor of 10 when a DF of 50 is reached. The reduction in the removal rate is not required if the removal rate is based on the calculated time-dependent airborne aerosol mass. There is no specified maximum DF for aerosol removal by sprays. The maximum activity to be used in determining the DF is defined as the iodine activity in the columns labeled "Total" in Tables 1 and 2 of this guide multiplied by 0.05 for elemental iodine and by 0.95 for particulate iodine (i.e., aerosol treated as particulate in SRP methodology).		volume is 4.9875E5 cubic feet. Transfer between these two volumes is provided by the Containment Fan Coolers. The flow rate is 65,000 cfm per fan for a total of 130,000 cfm. It is assumed that after the end of the core activity release process the aerosols would continue to be removed at a λ of 6.0 hr^{-1} until an overall DF of 50 is achieved. The current SRP 6.5.2 based assessment of elemental iodine removal coefficients during containment spray will continue to be used. The spray removal coefficient was determined to be 30.3 hr^{-1}. Per SRP 6.5.2 this value is reduced to 20 hr^{-1}. For aerosol removal the DF of 50 is reached at 2.21 hours. From that point until 8 hours, the removal coefficient of 0.6 hr^{-1} is used. For elemental iodine removal, the DF of 100 is reached at 1.926 hours.
3.4	Reduction in airborne radioactivity in the containment by in-containment recirculation filter systems may be credited if these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. A-5	Not Applicable	Not applicable for Byron or Braidwood. In-containment recirculation filters are not credited in

Table B: Conformance with Regulatory Guide 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	**RG Position**	**Analysis**	**Comments**
	and A-6). The filter media loading caused by the increased aerosol release associated with the revised source term should be addressed.		the analyses.
3.5	Reduction in airborne radioactivity in the containment by suppression pool scrubbing in BWRs should generally not be credited. However, the staff may consider such reduction on an individual case basis. The evaluation should consider the relative timing of the blowdown and the fission product release from the fuel, the force driving the release through the pool, and the potential for any bypass of the suppression pool (Ref. 7). Analyses should consider iodine re-evolution if the suppression pool liquid pH is not maintained greater than 7.	Not Applicable	Not applicable for a PWR
3.6	Reduction in airborne radioactivity in the containment by retention in ice condensers, or other engineering safety features not addressed above, should be evaluated on an individual case basis. See Section 6.5.4 of the SRP (Ref. A-1).	Not Applicable	Neither Byron nor Braidwood have ice condensers. No other removal mechanisms are credited other than natural deposition.
3.7	The primary containment (i.e., drywell for Mark I and II containment designs) should be assumed to leak at the peak pressure technical specification leak rate for the first 24 hours. For PWRs, the leak rate may be reduced after the first 24 hours to 50% of the technical specification leak rate. For BWRs, leakage may be reduced after the first 24 hours, if supported by plant configuration and analyses, to a value not less than 50% of the technical specification leak rate. Leakage from subatmospheric containments is assumed to terminate when the containment is brought to and maintained at a subatmospheric condition as defined by technical specifications. For BWRs with Mark III containments, the leakage from the drywell into the primary containment should be based on the steaming rate of the heated reactor core, with no credit for core debris relocation. This leakage should be assumed during the two-hour period between the initial blowdown and termination of the fuel radioactivity release (gap and early in-vessel release phases). After two hours, the radioactivity is	Conforms	The analyses follow the guidance for PWRs (the analyzed leak rate may be reduced after the first 24 hours to 50% of the technical specification leak rate). Neither Byron nor Braidwood have subatmospheric containments.

Table B: Conformance with Regulatory Guide 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	Analysis	Comments
	assumed to be uniformly distributed throughout the drywell and the primary containment.		
3.8	If the primary containment is routinely purged during power operations, releases via the purge system prior to containment isolation should be analyzed and the resulting doses summed with the postulated doses from other release paths. The purge release evaluation should assume that 100% of the radionuclide inventory in the reactor coolant system liquid is released to the containment at the initiation of the LOCA. This inventory should be based on the technical specification reactor coolant system equilibrium activity. Iodine spikes need not be considered. If the purge system is not isolated before the onset of the gap release phase, the release fractions associated with the gap release and early in-vessel phases should be considered as applicable.	Conforms	The Byron and Braidwood containments can be considered to be routinely purged during power operation. Therefore, the resulting purge dose contribution is summed with the postulated doses from other release paths.
4.1	Leakage from the primary containment should be considered to be collected, processed by engineered safety feature (ESF) filters, if any, and released to the environment via the secondary containment exhaust system during periods in which the secondary containment has a negative pressure as defined in technical specifications. Credit for an elevated release should be assumed only if the point of physical release is more than two and one-half times the height of any adjacent structure.	Conforms	No leakage is assumed to be collected for processing. Containment leakage is assumed to be released as a diffuse area source per RG 1.194. Since neither Byron nor Braidwood have a "tall stack," elevated releases are not assumed.
4.2	Leakage from the primary containment is assumed to be released directly to the environment as a ground-level release during any period in which the secondary containment does not have a negative pressure as defined in technical specifications.	Conforms	For EAB and LPZ doses, ground level releases are assumed. For Control Room doses, releases are based on zero-velocity vent release assumptions (ground-level equivalent).

Table B: Conformance with Regulatory Guide 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	**RG Position**	**Analysis**	**Comments**
4.3	The effect of high wind speeds on the ability of the secondary containment to maintain a negative pressure should be evaluated on an individual case basis. The wind speed to be assumed is the 1-hour average value that is exceeded only 5% of the total number of hours in the data set. Ambient temperatures used in these assessments should be the 1-hour average value that is exceeded only 5% or 95% of the total numbers of hours in the data set, whichever is conservative for the intended use (e.g., if high temperatures are limiting, use those exceeded only 5%).	Conforms	Although B/B are single-containment PWRs (no secondary containments), the evaluation was performed relative to the Aux Building. The bounding 250 foot elevation wind speed exceeded only 5% of the time at B/B is approximately 25.2 mph. Based on representative average surface pressure coefficients for rectangular buildings, a wind speed of greater than 32.2 mph would be required before the TS 3.7.12 minimum negative 0.25 inches water gauge Aux Building pressure would be positive relative to outside air pressures at any building surface.
4.4	Credit for dilution in the secondary containment may be allowed when adequate means to cause mixing can be demonstrated. Otherwise, the leakage from the primary containment should be assumed to be transported directly to exhaust systems without mixing. Credit for mixing, if found to be appropriate, should generally be limited to 50%. This evaluation should consider the magnitude of the containment leakage in relation to contiguous building volume or exhaust rate, the location of exhaust plenums relative to projected release locations, the recirculation ventilation systems, and internal walls and floors that impede stream flow between the release and the exhaust.	N/A	Byron & Braidwood are PWRs with no secondary containment.

Table B: Conformance with Regulatory Guide 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	**RG Position**	**Analysis**	**Comments**
4.5	Primary containment leakage that bypasses the secondary containment should be evaluated at the bypass leak rate incorporated in the Technical Specifications. If the bypass leakage is through water, e.g., via a filled piping run that is maintained full, credit for retention of iodine and aerosols may be considered on a case-by-case basis. Similarly, deposition of aerosol radioactivity in gas-filled lines may be considered on a case-by-case basis.	N/A	Byron & Braidwood are PWRs with no secondary containment. Therefore, all containment leakage is released directly to the environment.
4.6	Reduction in the amount of radioactive material released from the secondary containment because of ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).	Conforms	Credited ESF ventilation systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02.
5.1	With the exception of noble gases, all the fission products released from the fuel to the containment (as defined in Tables 1 and 2 of this guide) should be assumed to instantaneously and homogeneously mix in the primary containment sump water (in PWRs) or suppression pool (in BWRs) at the time of release from the core. In lieu of this deterministic approach, suitably conservative mechanistic models for the transport of airborne activity in containment to the sump water may be used. Note that many of the parameters that make spray and deposition models conservative with regard to containment airborne leakage are non-conservative with regard to the buildup of sump activity.	Conforms	With the exception of noble gases, all the fission products released from the fuel to the containment are assumed to instantaneously and homogeneously mix in the reactor building sump water at the time of release from the core.
5.2	The leakage should be taken as two times the sum of the simultaneous leakage from all components in the ESF recirculation systems above which the technical specifications, or licensee commitments to item III.D.1.1 of NUREG-0737 (Ref. A-8), would require declaring such systems inoperable. The leakage should be assumed to start at the earliest time the recirculation flow occurs in these systems and end at the latest time the releases from these systems are terminated. Consideration should also be given to design leakage through valves	Conforms	ECCS leakage is analyzed at a rate twice that allowed. ECCS leakage is a minor contributor to LOCA doses from the Byron and Braidwood plants. ECCS leakage is administratively controlled to a value of 3910 cc/hour, a value determined

Table B: Conformance with Regulatory Guide 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	Analysis	Comments
	isolating ESF recirculation systems from tanks vented to atmosphere, e.g., emergency core cooling system (ECCS) pump miniflow return to the refueling water storage tank.		based on an engineering evaluation of expected leakage. The accident analysis basis is 276,000 cc/hour. This leak rate is considered an upper bound that would still allow ECCS operability after 30 days, without makeup (a 12% inventory loss). Principal differences between pre-AST and post-AST accident analyses are: ECCS leakage is assumed to be 276,000 cc/hour rather than just 2 times the administrative limit (7820 cc/hour) and ECCS leakage flashing fractions are assumed to be 10% for the duration of the accident.
5.3	With the exception of iodine, all radioactive materials in the recirculating liquid should be assumed to be retained in the liquid phase.	Conforms	With the exception of iodine, all radioactive materials in ECCS liquids are assumed to be retained in the liquid phase.
5.4	If the temperature of the leakage exceeds 212°F, the fraction of total iodine in the liquid that becomes airborne should be assumed equal to the fraction of the leakage that flashes to vapor. This flash fraction, FF, should be determined using a constant enthalpy, h, process, based on the maximum time-dependent temperature of the sump water circulating outside the containment: $FF = \frac{h_{f1} - h_{f2}}{h_{fg}}$ *Where:* h_{f1} is the enthalpy of liquid at system design temperature and pressure; h_{f2} is the enthalpy of liquid at saturation conditions (14.7 psia,	Conforms	The temperature of the leakage exceeds 212°F for a period of less than 24 hours. Therefore, a flashing factor of 10% is assumed.

Table B: Conformance with Regulatory Guide 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	**RG Position**	**Analysis**	**Comments**
	212°F); and h_{fg} is the heat of vaporization at 212°F.		
5.5	If the temperature of the leakage is less than 212°F or the calculated flash fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid, unless a smaller amount can be justified based on the actual sump pH history and area ventilation rates.	Conforms	ECCS leakage flashing fractions are assumed to be 10% for the duration of the accident.
5.6	The radioiodine that is postulated to be available for release to the environment is assumed to be 97% elemental and 3% organic. Reduction in release activity by dilution or holdup within buildings, or by ESF ventilation filtration systems, may be credited where applicable. Filter systems used in these applications should be evaluated against the guidance of Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).	Conforms	The credited Control Room intake charcoal and HEPA filters meet the requirements of RG 1.52 and Generic Letter 99-02. These are credited at 95% efficiency for elemental and organic iodines. Aerosol removal efficiencies are assumed to be 99% based on the HEPA/charcoal combination. The filter efficiency for the Auxiliary Building exhaust is 90.0%. These filters meet the requirements of RG 1.52 and Generic Letter 99-02
7.0	The radiological consequences from post-LOCA primary containment purging as a combustible gas or pressure control measure should be analyzed. If the installed containment purging capabilities are maintained for purposes of severe accident management and are not credited in any design basis analysis, radiological consequences need not be evaluated. If the primary containment purging is required within 30 days of the LOCA, the results of this analysis should be combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. Reduction in the amount of radioactive material released via	Conforms	Although the current FSAR discusses containment purge for hydrogen control, this will not be considered as a release pathway during a LOCA. There is a redundant hydrogen recombiner available for use.

Table B: Conformance with Regulatory Guide 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	Analysis	Comments
	ESF filter systems may be taken into account provided that these systems meet the guidance in Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).		

Table C: Conformance with Regulatory Guide 1.183 Appendix B (Fuel Handling Accident)			
RG Section	**RG Position**	**Analysis**	**Comments**
1	Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.	Conforms	See Table A for conformance with Regulatory Guide 1.183 Appendix B (Fuel Handling Accident). The bounding inventory of fission products in the reactor core and available for release to the containment is based on the maximum full power operation of the core with current licensed values for fuel enrichment, fuel burnup, and a core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty. Additional conservatisms are added as discussed in Table A of this document to ensure the bounding source term was determined.
1.1	The number of fuel rods damaged during the accident should be based on a conservative analysis that considers the most limiting case. This analysis should consider parameters such as the weight of the dropped heavy load or the weight of a dropped fuel assembly (plus any attached handling grapples), the height of the drop, and the compression, torsion, and shear stresses on the irradiated fuel rods. Damage to adjacent fuel assemblies, if applicable (e.g., events over the reactor vessel), should be considered.	Conforms	The number of fuel rods damaged is equal to one fuel assembly. As currently described in the B/B UFSAR Section 15.7.4 (Fuel Handling Accidents), the accident is defined as the drop of a spent fuel assembly (SFA) onto the spent fuel pool floor or the core, resulting in the postulated rupture of the cladding of all fuel rods in one assembly.

Table C: Conformance with Regulatory Guide 1.183 Appendix B (Fuel Handling Accident)			
RG Section	**RG Position**	**Analysis**	**Comments**
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.	Exception taken (alternative treatment used)	Since several fuel assemblies exceed the guidance outlined in Footnote 11, the gap release fractions are doubled for conservatism. This treatment (previously approved for Fort Calhoun is conservative as previously discussed in Table A of this Compliance Table (Item 3.2).
1.3	The chemical form of radioiodine released from the fuel to the spent fuel pool should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. The CsI released from the fuel is assumed to completely dissociate in the pool water. Because of the low pH of the pool water, the iodine re-evolves as elemental iodine. This is assumed to occur instantaneously. The NRC staff will consider, on a case-by-case basis, justifiable mechanistic treatment of the iodine release from the pool.	Conforms	All iodine added to the reactor vessel or spent fuel pool is assumed to instantaneously dissociate and re-evolve as elemental iodine and treated appropriately with regard to pool pH and is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide.
2	If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 500 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85%) and organic iodine (0.15%) species results in the iodine above the water being composed of 57% elemental and 43% organic species. If the depth of water is not 23 feet, the decontamination factor will have to be determined on a case-by-case method (Ref. B-1).	Conforms	The analyzed water depth above damaged fuel is 23 feet. This value corresponds to the minimum depth of water coverage over the top of irradiated fuel assemblies seated in the spent fuel pool racks within the spent fuel pool, as per TS 3.7.14. Therefore, an overall DF of 200 is used per this guidance. Iodine above the water is assumed to be composed of 57% elemental and 43% organic species.

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

RG Section	RG Position	Analysis	Comments
3	The retention of noble gases in the water in the fuel pool or reactor cavity is negligible (i.e., decontamination factor of 1). Particulate radionuclides are assumed to be retained by the water in the fuel pool or reactor cavity (i.e., infinite decontamination factor).	Conforms	DF = 1 for noble gas isotopes; DF = infinite for particulate radionuclides.
4.1	The radioactive material that escapes from the fuel pool to the fuel building is assumed to be released to the environment over a 2-hour time period.	Conforms	The release is assumed to occur over a two hour period.
4.2	A reduction in the amount of radioactive material released from the fuel pool by engineered safety feature (ESF) filter systems may be taken into account provided these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. B-2, B-3). Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system should be determined and accounted for in the radioactivity release analyses.	Conforms	All ESF filtration systems credited in the analyses are qualified in accordance with the references cited in this section.
4.3	The radioactivity release from the fuel pool should be assumed to be drawn into the ESF filtration system without mixing or dilution in the fuel building. If mixing can be demonstrated, credit for mixing and dilution may be considered on a case-by-case basis. This evaluation should consider the magnitude of the building volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the pool, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the pool and the exhaust plenums.	Conforms	As per RG 1.183, the release from the fuel building to the environment is assumed over a 2-hour time period. To assure this, the refueling floor exhaust rate is set artificially high at 5 times this value or 0.118 air changes per minute during Control Room (emergency) Mode 2 operation.
5.1	If the containment is isolated during fuel handling operations, no radiological consequences need to be analyzed.	Not Applicable	Containment isolation is not credited in the analysis.

Table C: Conformance with Regulatory Guide 1.183 Appendix B (Fuel Handling Accident)			
RG Section	**RG Position**	**Analysis**	**Comments**
5.2	If the containment is open during fuel handling operations, but designed to automatically isolate in the event of a fuel handling accident, the release duration should be based on delays in radiation detection and completion of containment isolation. If it can be shown that containment isolation occurs before radioactivity is released to the environment, no radiological consequences need to be analyzed.	Conforms	Automatic Containment isolation is not credited. Therefore, a radiological consequence analysis is performed.
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. Note 3: *The staff will generally require that technical specifications allowing such operations include administrative controls to close the airlock, hatch, or penetrations within 30 minutes. Such administrative controls will generally require that a dedicated individual be present, with the necessary equipment available, to restore containment closure should a fuel handling accident occur. Radiological analyses should generally not credit this manual isolation.*	Conforms (with site-specific exceptions as noted in Attach 5 to this submittal)	The radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period.
5.4	A reduction in the amount of radioactive material released from the containment by ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. B-2 and B-3). Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system should be determined and accounted for in the radioactivity release analyses.	Conforms	For non-Recently Irradiated Fuel, no filtration of the radioactive gas released from the pool or automatic isolation of the accident location is assumed, with essentially all of the activity reaching the refueling floor airspace exhausted to the environment within two hours after the accident. For Recently Irradiated Fuel, an additional FHA analysis was performed

Table C: Conformance with Regulatory Guide 1.183 Appendix B (Fuel Handling Accident)			
RG Section	RG Position	Analysis	Comments
			with containment closure established or with the FHB ventilation system operable. The results of this analysis also met the limits of 10 CFR 50.67 assuming a minimum decay time of six hours. The six-hour minimum decay time is inconsequential as it is physically impossible to remove the reactor head and move fuel within the first six hours after the reactor is subcritical.
5.5	Credit for dilution or mixing of the activity released from the reactor cavity by natural or forced convection inside the containment may be considered on a case-by-case basis. Such credit is generally limited to 50% of the containment free volume. This evaluation should consider the magnitude of the containment volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the reactor cavity, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the reactor cavity and the exhaust plenums.	Conforms	The activity is instantaneously released from the fuel into the containment and is assumed to mix with 100% of the containment volume to calculate a hypothetical release rate with which to remove nearly all the activity within a two-hour period. This creates a conservative release rate over the two-hour release period.

Table D: Conformance with Regulatory Guide 1.183 Appendix H (PWR Rod Ejection Accident)			
RG Section	**RG Position**	**Analysis**	**Comments**
1	Assumptions acceptable to the NRC staff regarding core inventory are in Regulatory Position 3 of this guide. For the rod ejection accident, the release from the breached fuel is based on the estimate of the number of fuel rods breached and the assumption that 10% of the core inventory of the noble gases and iodines is in the fuel gap. The release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and the assumption that 100% of the noble gases and 25% of the iodines contained in that fraction are available for release from containment. For the secondary system release pathway, 100% of the noble gases and 50% of the iodines in that fraction are released to the reactor coolant.	Conforms	The CREA core source terms are those associated with a DBA power level of 3658.3 MWth, which includes an additional 2% power over that of the full licensed power to account for uncertainty. The sudden rod ejection and localized temperature spike associated with the CREA results in the damage of 10% of the core. Only 2.5 % of the damaged core releases melted fuel activity, i.e., 0.00250 of the total core melts. Therefore, the source term available for release is associated with this fraction of melted fuel and the fraction of core activity existing in the gap. A peaking factor of 1.7 is also applied.
2	If no fuel damage is postulated for the limiting event, a radiological analysis is not required as the consequences of this event are bounded by the consequences projected for the 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 Case 1, 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

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

RG Section	RG Position	Analysis	Comments
			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 is available for release to the environment via a Containment leak rate limit, or L_a. For Case 2, no breach of the RPV is assumed following the rod ejection. In this case, reactor coolant system (RCS), integrity is maintained and all activity from damaged fuel that has been mixed with the RCS leaks to the secondary side coolant through the Steam Generator (SG) tubes via the Tech. Spec. primary to secondary coolant leakage rate of 1.0 gpm. From here, activity is available for release to the environment by steaming of the SG Power-Operated Relief Valves (PORVs). In addition to the activity released from the primary to secondary coolant, pre-existing Tech. Spec. iodine activity in the secondary coolant system is assumed to also be released.

Table D: Conformance with Regulatory Guide 1.183 Appendix H (PWR Rod Ejection Accident)			
RG Section	**RG Position**	**Analysis**	**Comments**
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 (Conservative)	All iodine released from the SGs is conservatively assumed to be of the elemental species. This is done for RADTRAD simulation considerations, and is consistent with the RG 1.183, because elemental and organic iodine are identically treated by the computer model
5	Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic.	Conforms	All iodine released from the SGs is assumed to be of the elemental species. This is done for RADTRAD simulation considerations, and is consistent with the RG 1.183 specification of 97% elemental and 3% organic, because elemental and organic iodine are identically treated by the computer model
6	Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material in and from the containment are as follows.	Conforms	(See sections 6.1 and 6.2 below)

Table D: Conformance with Regulatory Guide 1.183 Appendix H (PWR Rod Ejection Accident)			
RG Section	**RG Position**	**Analysis**	**Comments**
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	The RADTRAD computer program, including the Powers Natural Deposition algorithm based on NUREG/CR-6189, is used for modeling aerosol deposition in Containment. No natural deposition is assumed for elemental or organic iodine. The lower bound (10%) level of deposition credit is used. Decay of radioactivity is credited in all compartments, prior to release. This is implemented in RADTRAD using the half-lives in the Nuclide Inventory File (NIF). The RADTRAD decay plus daughter option is used. In reality, daughter products such as xenon from iodines or iodines from tellurium are unlikely to readily escape from the fuel matrix in which the parent iodine or tellurium is contained. Nevertheless, the RADTRAD feature to include daughter effects is selected for conservatism. No credit for containment spray is taken.

Table D: Conformance with Regulatory Guide 1.183 Appendix H (PWR Rod Ejection Accident)			
RG Section	**RG Position**	**Analysis**	**Comments**
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 is 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.
7.1	A leak rate equivalent to the primary-to-secondary leak rate limiting condition for operation specified in the technical specifications should be assumed to exist until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms	The leak rate equivalent to the primary-to-secondary leak rate limiting condition for operation specified in the technical specifications is assumed to exist until shutdown cooling is in operation and releases from the steam generators have been terminated.
7.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests typically are based on cooled liquid. The facility's instrumentation used to determine leakage typically is located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft3).	Conforms	The density is assumed to be 1.0 gm/cc (62.4 lbm/ft3)
7.3	All noble gas radionuclides released to the secondary system are assumed to be released to the environment without reduction or mitigation.	Conforms	Noble gases are released without reduction or mitigation.
7.4	The transport model described in assumptions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.	Conforms	The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E was utilized for iodine and particulates.

Table E: Conformance with Regulatory Guide 1.183 Appendix G (PWR Locked Rotor Accident)			
RG Section	RG Position	Analysis	Comments
1	Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are in Regulatory Position 3 of this regulatory guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.	A conservative exception is taken regarding release fractions	The analysis does not fully comply with Note 11 of Table 3 since typical Byron and Braidwood core designs indicate that there are fuel assemblies that exceed the 6.3 kW/ft while >54GWD/MTU. Previous analyses (ANS 5.4) for TMI-1 have shown that those fuel assemblies exceeding these limits had no increase in gap release fractions of concern. Therefore, doubling of the "Other Noble Gases', "Other Halogens", and "Alkali Metals" gap fractions in Table 3 is conservative as used and approved in the Fort Calhoun AST submittal. Additionally, a peaking factor of 1.7 is used for DBA events that do not involve the entire core.
2	If no fuel damage is postulated for the limiting event, a radiological analysis is not required as the consequences of this event are bounded by the consequences projected for the main steam line break outside containment.	Not Applicable	Fuel damage is assumed. Therefore, a specific analysis is performed.
3	The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant.	Conforms	The activity is assumed to be released instantaneously and homogeneously through the primary coolant.

Table E: Conformance with Regulatory Guide 1.183 Appendix G (PWR Locked Rotor Accident)			
RG Section	**RG Position**	**Analysis**	**Comments**
4	The chemical form of radioiodine released from the fuel should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic. These fractions apply to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking.	Conforms	Iodine chemical form is in accordance with this guidance (97% elemental, 3% organic iodines).
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	Neither Byron nor Braidwood have implemented alternative repair criteria. Therefore, the primary-to-secondary leak rate in the steam generators are assumed to be the leak rate limiting condition for operation specified in the technical specifications. The design basis leak rate is 0.218 gpm per intact SG, totaling 0.654 gpm.
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/ft3).	Conforms	The density is assumed to be 1.0 gm/cc (62.4 lbm/ft3)

Table E: Conformance with Regulatory Guide 1.183 Appendix G (PWR Locked Rotor Accident)			
RG Section	**RG Position**	**Analysis**	**Comments**
5.3	The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212° F). The release of radioactivity should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms	The steaming release and primary-to-secondary coolant leakage is postulated to end at 40 hours, when the RCS and secondary loop have equilibrated.
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.
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.
5.6	The transport model described in assumptions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.	Conforms	The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E was utilized for iodine and particulates.

Table F: Conformance with Regulatory Guide 1.183 Appendix E (PWR Main Steam Line Break)			
RG Section	**RG Position**	**Analysis**	**Comments**
1	Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this regulatory guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. The fuel damage estimate should assume that the highest worth control rod is stuck at its fully withdrawn position.	Conforms	No fuel damage is postulated to occur during the MSLB (see Section 2 below).
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 (DE I-131), only the radioiodine associated with normal operations or iodine spikes is included.
2.1	A reactor transient has occurred prior to the postulated main steam line break (MSLB) and has raised the primary coolant iodine concentration to the maximum value (typically 60 µCi/gm DE I-131) permitted by the technical specifications (i.e., a pre-accident iodine spike case).	Conforms	This analyzed case involves a 60 µCi/gm pre-accident Iodine spike, consistent with the B/B Technical Specification operational Reactor Coolant System (RCS) activity concentration limit for an assumed spike. All of the spike activity is homogeneously mixed in the primary coolant, prior to accident initiation.

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

RG Section	RG Position	Analysis	Comments
2.2	The primary system transient associated with the MSLB causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (typically 1.0 µCi/gm DE I-131) specified in technical specifications (i.e., concurrent iodine spike case). A concurrent iodine spike need not be considered if fuel damage is postulated. The assumed iodine spike duration should be 8 hours. Shorter spike durations may be considered on a case-by-case basis if it can be shown that the activity released by the 8-hour spike exceeds that available for release from the fuel gap of all fuel pins.	Conforms (Iodine spike has been determined to last for 6 hours instead of 8)	This case involves an accident initiated iodine spike that occurs concurrently with the release of fluid from the primary and secondary coolant systems. This spike results in a release rate from the operating limit defective fuel fraction that is 500 times the normal rate. Conservative B/B analyses have shown that after 6 hours the total iodine gap activity of the defective fuel will have been completely released into the primary coolant.
3	The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant.	Conforms	The released activity is assumed to be released instantaneously and homogeneously through the primary coolant.
4	The chemical form of radioiodine released from the fuel should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic. These fractions apply to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking.	Conforms	Iodine chemical form is in accordance with this guidance (i.e., 97% elemental and 3% organic).

Table F: Conformance with Regulatory Guide 1.183 Appendix E (PWR Main Steam Line Break)			
RG Section	**RG Position**	**Analysis**	**Comments**
5.1	For facilities that have not implemented alternative repair criteria (see Ref. E-1, DG-1074), the primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate limiting condition for operation specified in the technical specifications. For facilities with traditional generator specifications (both per generator and total of all generators), the leakage should be apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized.	Conforms	Neither Byron nor Braidwood have implemented alternative repair criteria. Therefore, 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. Activity that originates in the primary RCS is released to the secondary coolant by means of the primary-to-secondary coolant leak rate. This design basis leak rate value is 0.218 gpm, per intact SG, totaling 0.654 gpm, and 0.5 gpm for the faulted SG with the broken steam line.
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/ft3).	Conforms	The density is assumed to be 1.0 gm/cc (62.4 lbm/ft3).
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 40 hours, when the RCS and secondary loop have equilibrated.

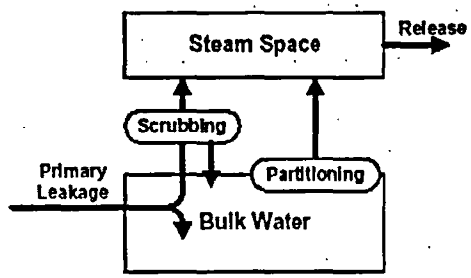
Table F: Conformance with Regulatory Guide 1.183 Appendix E (PWR Main Steam Line Break)			
RG Section	**RG Position**	**Analysis**	**Comments**
5.4	All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.	Conforms	Noble gases are released without reduction or mitigation.
5.5	The transport model described in this section should be utilized for iodine and particulate releases from the steam generators. This model is shown in Figure E-1 and summarized below: **Figure E-1** **Transport Model** 	Conforms	The transport model described in this section is utilized for iodine and particulate releases from the steam generators.
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	Conforms	Primary to secondary coolant leakage through the faulted steam generator 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.

Table F: Conformance with Regulatory Guide 1.183 Appendix E (PWR Main Steam Line Break)			
RG Section	**RG Position**	**Analysis**	**Comments**
	during periods of total tube submergence.		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, 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 0.01 was taken directly from RG 1.183. 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 the volatility of noble gases, no partitioning is assumed for any such isotopes.
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.	Conforms	See Comments for Section 5.5.1 above.
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.

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

RG Section	RG Position	Analysis	Comments
5.5.4	The radioactivity in the bulk water is assumed to become vapor at a rate that is the function of the steaming rate and the partition coefficient. A partition coefficient for iodine of 100 may be assumed. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators.	Conforms	The specified partition coefficient is used in the analysis.
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.	Conforms	See Comments for Section 5.5.1 above.

Table G: Conformance with Regulatory Guide 1.183 Appendix F (PWR Steam Generator Tube Rupture Accident)			
RG Section	**RG Position**	**Analysis**	**Comments**
1	Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are in Regulatory Position 3 of this guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.	Conforms	See Table A for conformance with Regulatory Guide 1.183 Appendix F (PWR Steam Generator Tube Rupture Accident). The bounding inventory of fission products in the reactor core and available for release to the containment is based on the maximum full power operation of the core with current licensed values for fuel enrichment, fuel burnup, and a core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty. Additional conservatisms are added as discussed in Table A of this document to ensure the bounding source term was determined.
2	If no or minimal[2] fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by technical specification. Two cases of iodine spiking should be assumed. 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 historical design basis at the Byron and Braidwood generating stations assumes no fuel damage for the postulated SGTR event. For this SGTR accident, the source terms are defined by the Tech Spec activity release rates from a maximum failed fuel fraction assumed during operation, which are characterized by the equilibrium 1.0 µCi/gm Dose Equivalent (DE) I-131 iodine activity concentration in the primary reactor coolant system. The noble gas

Table G: Conformance with Regulatory Guide 1.183 Appendix F (PWR Steam Generator Tube Rupture Accident)			
RG Section	RG Position	Analysis	Comments
			inventory in the RCS is based on operation with a conservative worst-case 1% core fuel defects. Because no fuel damage is assumed for this accident, only iodine and noble gas isotopes are modeled to contribute to dose. To identify the worst-case SGTR accident, however, two different cases of iodine spiking are analyzed, per regulatory guidance (Pre-Accident Iodine Spike and Concurrent Iodine Spike).
2.1	A reactor transient has occurred prior to the postulated steam generator tube rupture (SGTR) and has raised the primary coolant iodine concentration to the maximum value (typically 60 µCi/gm DE I-131) permitted by the technical specifications (i.e., a pre-accident iodine spike case).	Conforms	This analyzed case involves a 60 µCi/gm pre-accident Iodine spike, consistent with the B/B Technical Specification operational Reactor Coolant System (RCS) activity concentration limit for an assumed spike. All of the spike activity is homogeneously mixed in the primary coolant, prior to accident initiation.
2.2	The primary system transient associated with the SGTR causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 335 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (typically 1.0 µCi/gm DE I-131) specified in technical specifications (i.e., concurrent iodine spike case). A concurrent iodine spike need not be considered if fuel damage is	Conforms	The second analyzed case involves an accident initiated iodine spike that occurs concurrently with the release of fluid from the primary and secondary coolant systems. This spike results in a release rate from the operating limit defective fuel fraction (~1%) that is 335 times the normal rate, and lasts for an 8-hour

Table G: Conformance with Regulatory Guide 1.183 Appendix F (PWR Steam Generator Tube Rupture Accident)			
RG Section	**RG Position**	**Analysis**	**Comments**
	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.		duration.
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 instantly and homogeneously.
4	Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic.	Conforms	*Such iodine releases are assumed to* be 97% elemental and 3% organic.
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	Activity that originates in the primary RCS is released to the secondary coolant by means of the primary-to-secondary coolant leak rate. This design basis leak rate value is 0.218 gpm per intact SG, totaling 0.654 gpm.
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/ft3).	Conforms	The density is assumed to be 1.0 gm/cc (62.4 lbm/ft3)

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

RG Section	RG Position	Analysis	Comments
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 when shutdown cooling has been established.
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.
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.
5.6	The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.	Conforms	The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E was utilized for iodine and particulates.

Table H: Conformance with Regulatory Guide 1.194 (Diffuse Area Source Guidance)			
RG Section	**RG Position**	**B/B Analysis**	**Comments**
3.2.4	Examples of possible area sources are postulated releases from the surface of a reactor or a secondary containment building. A reasonable approach (is) to model the building surface as a vertical planar area source. This approach is not intended to address dispersion resulting from building-induced turbulence. Treatment of a release as a diffuse source will be acceptable for design basis calculations if the guidance herein is followed.	Conforms	Introductory information excerpted – no requirements.
3.2.4.1	Diffuse source modeling should be used only for those situations in which the activity being released is homogeneously distributed throughout the building and when the assumed release rate from the building surface would be reasonably constant over the surface of the building. For example, steam releases within a turbine building with roof ventilators or louvered walls would generally not be suitable for modeling as a diffuse source. (See Regulatory Positions 3.2.4.7 and 3.2.4.8.)	Conforms	Used only for Containment Building, where the situation would comply with this guidance.
3.2.4.2	Since leakage is more likely to occur at a penetration, analysts must consider the potential impact of building penetrations exposed to the environment* within this modeled area. If the penetration release would be more limiting, the diffuse area source model should not be used. Releases from personnel air locks and equipment hatches exposed to the environment, or containment purge releases prior to containment isolation, may need to be treated differently. It may be necessary to consider several cases to ensure that the X/Q value for the most limiting location is identified. *Penetrations that are enclosed within safety-related structures need not be considered in this evaluation if the release would be captured and released via a plant ventilation system, as ventilation system releases should have already been addressed as a separate release point.	Conforms	Containment radioactivity releases through penetrations and personnel/equipment hatches into the Auxiliary Building are served by otherwise un-credited HEPA filters and charcoal adsorbers before release through the Plant Vent. This filtration more than offsets differences between Plant Vent and Containment diffuse area source X/Qs. Therefore, unfiltered containment diffuse area treatment can be conservatively applied to these penetrations. All leakage through the secondary personnel/equipment hatch would be unfiltered, but the hatch is located on the far

Table H: Conformance with Regulatory Guide 1.194 (Diffuse Area Source Guidance)			
RG Section	**RG Position**	**B/B Analysis**	**Comments**
			side of the Containment Buildings with respect to the Control Room intakes, with X/Qs more favorable than the Containment Building diffuse source X/Qs. Containment vent penetrations are exhausted through the plant vent, but such leakage, if not directly into the auxiliary building, would be past minflow or normal ventilation HEPA filters, or the post-LOCA purge HEPA filters and charcoal adsorbers. Therefore, unfiltered containment diffuse area treatment can be conservatively applied to these penetrations. Containment purge supply penetration leakage, if not directly into the auxiliary building, would be to the auxiliary building supply air intake such that it would be drawn back into the auxiliary building. Therefore, unfiltered containment diffuse area treatment can be conservatively applied to these penetrations. All penetration leakage into the steam tunnel are not from the containment atmosphere, due to the barrier provided by the steam generators.

Table H: Conformance with Regulatory Guide 1.194 (Diffuse Area Source Guidance)			
RG Section	**RG Position**	**B/B Analysis**	**Comments**
3.2.4.3	The total release rate (e.g., Ci/second) from the building atmosphere is to be used in conjunction with the diffuse area source X/Q in assessments. This release rate is assumed to be equally distributed over the entire diffuse source area from which the radioactivity release can enter the environment. For freestanding containments, this would be the entire periphery above grade or above a building that surrounds the lower elevations of the containment. When a licensee can justify assuming collection of a portion of the release from the containment within the surrounding building, the total release from the containment may be apportioned between the exposed and enclosed building surfaces. Similarly, if the building atmosphere release is modeled through more than one simultaneous pathway (e.g., drywell leakage and main steam safety valve leakage in a BWR), only that portion of the total release released through the building surface should be used with the diffuse area X/Q, The release rate should not be averaged or otherwise apportioned over the surface area of the building. For example, reducing the release rate by 50 percent because only 50 percent of the surface faces the control room intake would be inappropriate.	Conforms	The Containment Buildings are freestanding containments, so the source area is the entire periphery above grade.

Table H: Conformance with Regulatory Guide 1.194 (Diffuse Area Source Guidance)			
RG Section	**RG Position**	**B/B Analysis**	**Comments**
3.2.4.4	ARCON96 uses two initial diffusion coefficients entered by the user to represent the area source. There are insufficient field measurements to mechanistically model these initial diffusion coefficients. The following deterministic equations should be used in the absence of site-specific empirical data.* $\text{Sigma } Y_o = \frac{\text{Width source area}}{6}$ $\text{Sigma } Z_o = \frac{\text{Height source area}}{6}$ *See Regulatory Position 7 regarding the use of site-specific empirical measurements.	Conforms	ARCON96 and the two equations are utilized.
3.2.4.5	The height and width of the area source (e.g., the building surface) are taken as the maximum vertical and horizontal dimensions of the above-grade building cross-sectional area perpendicular to the line of sight from the building center to the control room intake (see Figure 2). These dimensions are projected onto a vertical plane perpendicular to the line of sight and located at the closest point on the building surface to the control room intake. The release height is set at the vertical center of the projected plane. The source-to-receptor distance (slant path) is measured from this point to the control room intake.	Conforms	
3.2.4.6	Intentional releases from a secondary containment (e.g., standby gas treatment systems (SGTS) at BWR reactors) or annulus ventilation systems in dual containment structures should be treated as a ground-level release or an elevated stack release, as appropriate. The diffuse area source model may be appropriate for time intervals for which the secondary containment or annulus	Not applicable	

Table H: Conformance with Regulatory Guide 1.194 (Diffuse Area Source Guidance)			
RG Section	**RG Position**	**B/B Analysis**	**Comments**
	ventilation system is not capable of maintaining the requisite negative pressure differential specified in technical specifications or in the FSAR. Secondary containment bypass leakage (i.e., leakage from the primary containment that bypasses the secondary containment and is not collected by the SGTS) should be treated as a ground-level release or an elevated stack release, as appropriate.		
3.2.4.7	A second possible application of the diffuse area source model is determining a X/Q value for multiple (i.e., 3 or more) roof vents. This treatment would be appropriate for configurations in which (1) the vents are in a close arrangement, (2) no individual vent is significantly* closer to the control room intake than the center of the area source, (3) the release rate from each vent is approximately the same, and (4) no credit is taken for plume rise. The distance to the receptor is measured from the closest point on the perimeter of the assumed area source. For assumed areas that are not circular, the area width is measured perpendicular to the line of sight from the center of the assumed source to the control room intake. The initial diffusion coefficient sigma Yo is found by Equation 3; sigma Z_o is assumed to be 0.0. * The degree of significance will depend on the radius or width of the assumed area and the proximity of the vent cluster to the control room intake. As the radius decreases or the distance from the cluster to the control room intake increases, the less significance the position of any one vent has.	Not applicable	

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

ATTACHMENT 8

BYRON STATION
UNITS 1 AND 2

Docket Nos. 50-454 and 50-455

License Nos. NPF-37 and NPF-66

and

BRAIDWOOD STATION
UNITS 1 AND 2

Docket Nos. 50-456 and 50-457

License Nos. NPF-72 and NPF-77

License Amendment Request
"Alternative Source Term Implementation"

Compact Disk Containing Meteorological Data