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10CFR 50.90

July 13, 2007

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

Peach Bottom Atomic Power Station, Units 2 and 3
Renewed Facility Operating License Nos. DPR-44 and DPR-56
Docket Nos. 50-277 and 50-278

Subject: License Amendment Request – 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) Technical Specification Task Force (TSTF) Traveler, TSTF-51, "Revise Containment Requirements During Handling of Irradiated Fuel and Core Alterations," Revision 2

Pursuant to 10 CFR 50.67, "Accident source term," and 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC, (Exelon) hereby requests the following amendment to the Technical Specifications (TS), Appendix A, of Renewed Facility Operating License Nos. DPR-44 and DPR-56 for Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3. The proposed change is requested to support the application of Alternative Source Term (AST) methodology.

Exelon previously submitted a License Amendment Request (LAR) to adopt AST by letter dated July 14, 2003, for PBAPS, which was subsequently withdrawn as documented in a letter dated May 10, 2005. The technical issues associated with that LAR were discussed during a pre-submittal public meeting with the NRC on February 16, 2007. The issues pertaining to the original LAR have been addressed in the current application.

The proposed change is 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 changes to the current licensing basis for PBAPS include:

- Technical Specifications (TS) and associated Bases revisions to reflect implementation of AST assumptions.
- TS and associated Bases revisions to increase primary containment allowable leakage.
- TS and associated Bases revisions to increase main steam isolation valve allowable leakage.

Designate Original
John O. Hughey
A001
NRR
Rec'd 9/10/07

- TS and associated Bases revisions to change the applicability requirements for the following systems during movement of recently irradiated fuel assemblies in secondary containment and to reflect that these systems are no longer required to be operable during core alterations under these conditions:
    - Standby Gas Treatment (SGT),
    - Secondary Containment, and
    - Secondary Containment Isolation Valves.
- TS and associated Bases revisions to reflect use of the Standby Liquid Control (SLC) System to buffer suppression pool pH to prevent iodine re-evolution during a postulated radiological release.
- TS and associated Bases revisions to reflect a new specification to limit containment purge time to less than (or equal to) 90 hours per calendar year.
- TS and associated Bases to increase the secondary containment drawdown time from the exiting two (2) minutes to three (3) minutes.

The proposed changes related to the applicability requirements during movement of recently irradiated fuel assemblies are consistent with Technical Specification Task Force Traveler (TSTF) 51, Revision 2 (Reference 2).

Attachment 1 of this submittal provides a Description of Proposed Changes, Technical Analysis, and Regulatory Analysis. Attachment 2 provides the Markup of Technical Specification pages. Attachment 3 provides the Markup of Technical Specification Bases pages (for Information only). Attachment 4 provides the List of Commitments resulting from the proposed changes. Attachment 5 provides a compact disk (CD) containing PBAPS meteorological data for the calculation of the atmospheric dispersion factors (χ/Qs), as well as the revised dose analysis calculations. Attachment 6 provides the Mark-up of Updated Final Safety Analysis (UFSAR) Section 5.2.4.3.2 including Figure 5.2.16. Attachment 7 contains a table listing the degree of conformance to Regulatory Guide (RG) 1.183. Attachment 8 provides a CD containing the referenced AST calculations.

The proposed changes have been reviewed by the Plant Operations Review Committee and approved by the Nuclear Safety Review Board.

Exelon requests approval of the proposed amendment by July 13, 2008, with the amendment being implemented within 90 days of issuance. The requested approval date and implementation period will allow usage of the Technical Specification Amendment needed for the Unit 2 Fall 2008 Refueling Outage.

Pursuant to 10 CFR 50.91(b)(1), a copy of this License Amendment Request is being provided to the designated official of the Commonwealth of Pennsylvania.

If any additional information is needed, please contact Mr. Richard Gropp at 610-765-5557.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 13th day of July 2007.

Respectfully,

Pamela B. Cowan
Director, Licensing & Regulatory Affairs
Exelon Generation Company, LLC

Attachments:
1. Evaluation of Proposed Changes
2. Markup of Proposed Technical Specification pages
3. Markup of Proposed Technical Specification Bases pages *(Information only)*
4. List of Commitments
5. Compact Disk containing PBAPS Meteorological data *(Information only)*
6. UFSAR Section 5.2.4.3.2 Mark-Up and Supporting Technical Information
7. RG 1.183 Conformance Matrix
8. ~~Compact Disk~~ containing AST Calculations

Paper Copy Provided

cc: S. J. Collins, Administrator, Region I, USNRC
F. L. Bower, USNRC Senior Resident Inspector, PBAPS
J. Hughey, Project Manager, USNRC
R. R. Janati, Commonwealth of Pennsylvania

ATTACHMENT 1

Evaluation of Proposed Changes

PBAPS, Units 2 and 3
Renewed Facility Operating License Nos. DPR-44 and DPR-56

"PBAPS Alternative Source Term Implementation"

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 (Exelon) requests a change to Appendix A, Technical Specifications (TS), of Facility Operating License Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3. The proposed changes are requested to support application of an alternative source term (AST) methodology, with the exception that Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," (Reference 7.1) will continue to be used as the radiation dose basis for equipment qualification.

Exelon has performed radiological consequence analyses of the four Design Basis Accidents (DBAs) that result in offsite exposure (i.e., Loss of Coolant Accident (LOCA), Main Steam Line Break (MSLB), Fuel Handling Accident (FHA), and Control Rod Drop Accident (CRDA)) to support a full-scope implementation of AST as described in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Reference 7.2). The AST analyses for PBAPS were performed following the guidance in Reference 7.2 and Standard Review Plan 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms" (Reference 7.3), and 10 CFR 50.67, "Accident source term."

Approval of this change will provide a realistic source term for PBAPS that will result in a more accurate assessment of DBA radiological doses. This allows relaxation of some current licensing basis requirements as described in Section 2.0, Proposed Changes. Adopting the AST methodology may also support future evaluations and license amendments.

2.0 PROPOSED CHANGES

The proposed changes to the current licensing basis for PBAPS that are justified by the AST analyses include:

- Technical Specifications (TS) and associated Bases revisions to reflect implementation of AST assumptions.
- TS and associated Bases revisions to increase primary containment allowable leakage.
- TS and associated Bases revisions to increase main steam isolation valve allowable leakage.
- TS and associated Bases revisions to change the applicability requirements for the following systems during movement of recently irradiated fuel assemblies in secondary containment and to reflect that these systems are no longer required to be operable during core alterations under these conditions:
    - Standby Gas Treatment (SGT)
    - Secondary Containment
    - Secondary Containment Isolation Valves

- TS and associated Bases revisions to reflect use of the Standby Liquid Control (SLC) System to buffer suppression pool pH to prevent iodine re-evolution during a postulated radiological release.
- TS and associated Bases revisions to reflect a new specification to limit containment purge time to less than (or equal to) 90 hours per calendar year.
- TS and associated Bases to increase the secondary containment drawdown time from the existing two (2) minutes to three (3) minutes.

There is an additional proposed change to the UFSAR. It is proposed to revise UFSAR Section 5.2.4.3.2, "Minimum Containment Pressure Available," to reflect the impact of AST assumptions. Portions of the text and Figure 5.2.16, "Minimum Containment Pressure Available and Containment Overpressure License," were previously approved by the NRC in a Safety Evaluation dated August 14, 2000 (Reference 7.22). Refer to Attachment 6 of this submittal for additional information concerning the proposed UFSAR 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 (Reference 7.20). TSTF-51, Revision 2, was approved by the NRC on October 15, 1999. TSTF-51 changes the TS operability requirements for engineered safety features such that they are not required after sufficient radioactive decay has occurred to ensure that offsite doses remain within limits.

The proposes revisions to PBAPS TS include the following:

2.1 TS Section 1.1, "Definitions"

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," Second Printing, 1989. This change reflects the application of AST assumptions.

2.2 TS Section 1.1, "Recently Irradiated Fuel"

The proposed change revises Section 1.1 to add a new definition for RECENTLY IRRADIATED FUEL. RECENTLY IRRADIATED FUEL is fuel that has occupied part of a critical reactor core within the previous 84 hours. This 84-hour time period may be reduced to 24 hours if all outside secondary containment ground-level hatches (hatches H15 through H24 and Units 2 and 3 Torus room access hatches) are closed.

2.3 TS Section 3.1.7, "Standby Liquid Control (SLC) System"

The proposed change revises the Applicability of TS Section 3.1.7 to add the requirement for the LCO to be met in Mode 3. This change implements AST assumptions regarding the use of the SLC System to buffer the suppression pool following a LOCA involving significant fission product release. The required actions

for Condition D are being revised to add an additional requirement to be in Mode 4 with a completion time of 36 hours.

2.4 TS Section 3.3.6.1, "Primary Containment Isolation Instrumentation"

TS Section 3.3.6.1, Table 3.3.6.1-1 lists the applicability requirements for Primary Containment Isolation Instrumentation. The proposed change revises the applicability of the SLC System Initiation Function of the Reactor Water Cleanup System Isolation instrumentation to add the requirement for this function to be operable in Mode 3. The revised applicability for this function is consistent with the revised applicability for the SLC System. This change is supported by TSTF-51.

2.5 TS Section 3.3.6.2, "Secondary Containment Isolation Instrumentation"

The proposed change revises footnote (b) of TS Table 3.3.6.2-1 by deleting, "CORE ALTERATIONS, and during," which eliminates the requirement for Function 3 (i.e., Reactor Building Ventilation Exhaust Radiation – High) and Function 4 (i.e., Refueling Floor Ventilation Exhaust Radiation – High) of the Secondary Containment Isolation Instrumentation to be operable during core alterations. The proposed change also relaxes TS requirements to require these functions to be operable only when handling recently irradiated fuel. With the application of AST, secondary containment is not credited for the FHA after a 24-hour decay period. This change is supported by TSTF-51.

2.6 TS Section 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)"

The proposed change involves revising TS Section 3.6.1.3 to include new a Limiting Condition for Operation (LCO) requirement stipulating that the accumulated time a purge or vent flow path shall be open shall be limited to 90 hours per calendar year, with the reactor in MODES 1 and 2, and reactor pressure greater than 100 psig. In the event that flow paths are open greater than 90 hours in a calendar year, the penetration(s) must be isolated with 4 hours or the reactor shall be in MODE 3 within 12 hours, and in MODE 4 within 36 hours. This will limit the total time that a flow path exists through certain containment penetrations. Consequently, the impact on plant risks resulting from Emergency Core Cooling System (ECCS) net positive suction head (NPSH) during a LOCA while purging contribute little to the likelihood of ECCS equipment malfunctions.

The proposed change revises Surveillance Requirement (SR) 3.6.1.3.14 to increase the allowable limit for the combined leakage rate for all MSIV leakage paths. Currently, the allowable limit is less than or equal to 11.5 scfh for each MSIV when tested at greater than or equal to 25 psig. This limit will be increased to less than or equal to 204 scfh for all four main steam lines and less than or equal to 116 scfh for any one main steam line, when tested at greater than or equal to 25 psig. The revised SR 3.6.1.3.14 reads:

> Verify combined MSIV leakage rate for all four main steam lines is $\leq$ 204 scfh, and $\leq$116 scfh for any one steam line, when tested at $\geq$ 25 psig.

The Frequency for SR 3.6.1.3.14 is "In accordance with the Primary Containment Leakage Rate Testing Program," and this frequency is not being changed.

2.7 TS Section 3.6.4.1, "Secondary Containment"

The proposed change deletes "During CORE ALTERATIONS" from the applicability statement for TS LCO 3.6.4.1 and relaxes TS requirements to require LCO 3.6.4.1 to be applicable only when handling recently irradiated fuel. The proposed change revises Condition C, and associated required actions and completion times, to reflect the revision of the applicability requirements for LCO 3.6.4.1. With the application of AST, secondary containment is not credited for the FHA after a 24-hour decay period. The final proposed change to TS 3.6.4.1 is to SR 3.6.4.1.3, increasing the secondary containment drawdown time from less than or equal to 120 seconds to less than or equal to 180 seconds. This is change supported by TSTF-51.

2.8 TS Section 3.6.4.2, "Secondary Containment Isolation Valves (SCIVs)"

The proposed change deletes "During CORE ALTERATIONS" from the applicability statement for TS LCO 3.6.4.2 and relaxes TS requirements to require LCO 3.6.4.2 to be applicable only when handling recently irradiated fuel. The proposed change revises Condition D, and associated required actions and completion times, to reflect the revision of the applicability requirements for LCO 3.6.4.2. With the application of AST, closure of secondary containment isolation valves is not credited for the FHA after a 24-hour decay period. This change is supported by TSTF-51.

2.9 TS Section 3.6.4.3, "Standby Gas Treatment (SGT) System"

The proposed change deletes "During CORE ALTERATIONS" from the applicability statement for TS LCO 3.6.4.3 and relaxes TS requirements to require LCO 3.6.4.3 to be applicable only when handling recently irradiated fuel. The proposed change revises Condition C and Condition E, and associated required actions and completion times, to reflect the revision of the applicability requirements for LCO 3.6.4.3. These changes are being made to reflect that, with application of AST, the SGT System is no longer required to be operable during movement of irradiated fuel assemblies, that have decayed at least 24-hours, in the secondary containment, or during core alterations, since this system is not credited for the FHA after a 24-hour decay period. This change is supported by TSTF-51.

2.10 TS Section 5.5.12, "Primary Containment Leakage Rate Testing Program"

The proposed change increases the maximum allowable primary containment leakage rate, L_a, at P_a, from 0.5% to 0.7% of primary containment air weight per day. Application of AST supports the increase in maximum allowable primary containment leakage rate.

2.11 UFSAR Section 5.2.4.3.2, "Minimum Containment Pressure Available"

The proposed change to the UFSAR revises the Containment Overpressure License (COPL) described in UFSAR Figure 5.2.16 to increase the COPL; however,

adequate margin is maintained between the revised Minimum Containment Pressure Available (MCPA) and the proposed COPL.

3.0 BACKGROUND

On December 23, 1999, the NRC published regulation 10 CFR 50.67 in the Federal Register. This regulation provides a mechanism for operating license holders to revise the current accident source term used in design-basis radiological analyses with an AST. Regulatory guidance for the implementation of AST is provided in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors", July 2000 (Reference 7.2). This regulatory guide provides guidance on acceptable applications of ASTs. The use of AST changes only the regulatory assumptions regarding the analytical treatment of the design basis accidents (DBAs).

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 as discussed in Technical Information Document (TID) 14844, "Calculation of Distance Factors For Power and Test Reactor Sites," (Reference 7.1). Since the publication of Reference 7.1, 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 representing the progress of a severe accident in a LWR are described in NUREG-1465 as:

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 DBA evaluations; however, they are all assumed to occur instantaneously. Phases 4 and 5 are related to severe accident evaluations. Under the AST, the coolant activity release is assumed to occur instantaneously and end with the onset of the gap activity release.

The requested license amendment involves a full-scope application of the AST, addressing the composition and magnitude of the radioactive material, its chemical and physical form, and the timing of its release as described in Reference 7.2.

Exelon has performed radiological consequence analyses of the four DBAs that result in offsite exposure (i.e., LOCA, MSLB, FHA, and CRDA). These analyses were performed to support full scope implementation of AST. The AST analyses have been performed in accordance with the guidance in References 7.2 and 7.3. The implementation consisted of the following tasks:

- Identification of the AST based on plant-specific analysis of core fission product inventory
- Calculation of the release fractions for the four DBAs that could potentially result in control room and offsite doses (i.e., LOCA, MSLB, FHA, and CRDA)
- Analysis of the atmospheric dispersion for the radiological propagation pathways
- Calculation of fission product deposition rates and transport and removal mechanisms
- Calculation of offsite and control room personnel Total Effective Dose Equivalent (TEDE) doses
- Evaluation of suppression pool pH to ensure that the iodine deposited into the suppression pool during a DBA LOCA does not re-evolve and become airborne as elemental iodine

The analysis assumptions for the transport, reduction, and release of the radioactive material from the fuel and the reactor coolant are consistent with the guidance provided in applicable appendices of Reference 7.2 for the four analyzed DBAs.

Accordingly, Exelon, as a holder of an operating license issued prior to January 10, 1997, is requesting the use of AST for several areas of operational relief for systems used in the event of a Design Basis Accident (DBA), and without crediting the use of certain previously assumed safety systems/functions.

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 (Reference 7.20) 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 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, Exelon is committing to the applicable provisions of Nuclear Utilities Management and Resources Council (NUMARC) 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 3 (Reference 7.23), as described in TSTF-51. NUMARC 93-01 provides recommendations on the need to initiate actions to verify and/or re-establish secondary containment, and if needed, primary containment, in the event of a dropped fuel assembly. Note that at the time TSTF-51, Revision 2 was issued, a reference to Section 11.2.6 of Draft NUMARC 93-01, Revision 3, was to be made. The final NUMARC 93-01, Revision 3 (July 2000), does not have a section numbered 11.2.6. Therefore, Section 11.3.6.5 was used since the section title in TSTF-51 matches that in the July 2000 final version of NUMARC 93-01, Revision 3.

4.0 TECHNICAL ANALYSIS

4.1 INTRODUCTION

4.1.1 Attributes of the PBAPS AST

The Peach Bottom AST is based on major accidents, hypothesized for the purposes of design analyses or consideration of possible accidental events that could result in hazards not exceeded by those from other accidents considered credible. It addresses events that involve a substantial meltdown of the core with the subsequent release of appreciable quantities of fission products, the times and rates of appearance of radioactive fission products released into containment, the types and quantities of the radioactive species released, and the chemical forms of iodine released.

4.1.2 Accident Source Term

The inventory of fission products in the reactor core that is available for release to the containment is based on the maximum full power operation of the core with bounding values for fuel enrichment, and fuel burnup. The core power used in the analyses (3514 MWt) is the current licensed rated thermal power. The period of irradiation is of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. ORIGEN 2.1 (Reference 7.5) based methodology was used to determine core inventory. These source terms were evaluated at end-of-cycle and at beginning of cycle conditions (100 effective full power days (EFPD) to achieve equilibrium) and worst-case inventory used for the selected isotopes. These values were then divided by 3514 MWt to obtain activity in units of Ci/MWt. The Ci/MWt activities are subsequently multiplied in RADTRAD dose calculations by 3528 MWt, which is equivalent to the current licensed rated thermal power (3514 MWt) times the ECCS evaluation uncertainty (1.004). Source terms are based on a 2-year fuel cycle with a nominal 711 EFPD per cycle.

Sensitivity analyses performed using various enrichment levels (3.56% to 5%) and cycle lengths (351 EFPD up to 740 EFPD) have confirmed that the source term used produces bounding doses for Control Room (CR) and offsite locations for the assumed Loss of Coolant Accident.

For the DBA LOCA, all fuel assemblies in the core are assumed to be affected and the core average inventory is used. For DBA events that do not involve the entire core (CRDA and FHA), 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 are applied in determining the inventory of the damaged rods. For the MSLB event, no fuel damage is postulated to occur.

No adjustment to the fission product inventory is made for events postulated to occur during power operations at less than full rated power or those postulated to occur at the beginning of core life. For events postulated to occur while the facility is shutdown (e.g., a fuel handling accident) radioactive decay from the time of shutdown is modeled.

4.1.3 Release Fractions

The core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage phases for the DBA LOCA listed in RG 1.183 Table 1 for BWRs are used. These fractions are applied to the equilibrium core inventory developed for PBAPS.

For non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides in RG 1.183 Table 3 are used. These release fractions are used in conjunction with the fission product inventory calculated with the maximum core radial peaking factor.

The core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage for a Design Basis Accident (DBA) LOCA are those from Regulatory Guide 1.183, Tables 1 and 4. These release fractions are acceptable for use given that the peak fuel burnup meets the 62,000 MWD/MTU requirement specified in Regulatory Guide 1.183 Footnote 10. Reload checks are performed to ensure that PBAPS core designs meet this requirement.

4.1.4 Timing of Release Phases

RG 1.183 Table 4 tabulates the onset and duration of each sequential release phase for DBA LOCAs. 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 is 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 is assumed to occur instantaneously with the onset of the projected damage. The PBAPS AST analyses use these release phases.

4.1.5 Radionuclide Composition

The elements and radionuclide groups listed in RG 1.183 Table 5 are used in the PBAPS AST design basis accident analyses.

For the MSLB accident, coolant activities specified in Technical Specifications are assumed in lieu of fuel damage.

4.1.6 Chemical Form

Of the radioiodine released from the reactor coolant system (RCS) to the containment in a postulated accident, 95 percent of the iodine released is assumed to be cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. This includes releases from the gap and the fuel pellets. With the exception of elemental and organic iodine and noble gases, fission products are assumed to be in particulate form. The same chemical form is assumed in releases from fuel pins in fuel handling accidents (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 descriptions that follow provide additional details.

4.2 EVALUATION

4.2.1 Scope

4.2.1.1 Offsite Dose Consequences

The following assumptions are used in determining the Total Effective Dose Equivalent (TEDE) for the maximum exposed individual at EAB and LPZ locations:

- The offsite dose is determined as a TEDE, which is the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure from all radionuclides that are significant with regard to dose consequences and the released radioactivity. The RADTRAD computer code performs this summation to calculate the TEDE.

- The offsite dose analysis uses the Committed Effective Dose Equivalent (CEDE) Dose Conversion Factors (DCFs) for inhalation exposure. 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" (Reference 7.15), provides tables of conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the CEDE.

- Since RADTRAD calculates Deep Dose Equivalent (DDE) using whole body submergence in semi-infinite cloud with appropriate credit for attenuation by body tissue, the DDE can be assumed nominally equivalent to the Effective Dose Equivalent (EDE) from external exposure. Therefore, the offsite dose analysis uses EDE in lieu of DDE Dose Conversion Factors in determining external exposure. Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil" (Reference 7.16), provides external EDE conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the EDE.

- The maximum EAB TEDE for any two-hour period following the start of the radioactivity release is determined and used in determining compliance with the dose acceptance criteria in 10 CFR 50.67.

- TEDE is determined for the most limiting receptor at the outer boundary of the low population zone (LPZ) and is used in determining compliance with the dose criteria in 10 CFR 50.67. The breathing rates specified in RG 1.183 and/or SRP 6.4 (Reference 7.2 and 7.13, respectively) are used.

- No correction is made for depletion of the effluent plume by deposition on the ground.

4.2.1.2 Control Room Dose Consequences

The following guidance was used in determining the TEDE for maximum exposed individuals located in the control room:

- The CR TEDE analysis considers the following sources of radiation that will cause exposure to control room personnel.
    - Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the post-accident radioactive plume released from the facility (via CR air intake)
    - 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 (via CR unfiltered inleakage)
    - Radiation shine from the external radioactive plume released from the facility (external airborne cloud)
    - Radiation containment 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 ventilation filters (CR filter shine dose)
- The radioactivity releases and radiation levels used for the control room dose are determined using the same source term, transport, and release assumptions used for determining the exclusion area boundary (EAB) and the low population zone (LPZ) TEDE values.
- Credit for engineered safety features that mitigate airborne activity within the control room is taken for control room isolation / pressurization and intake filtration.
    - The MCREV system is conservatively assumed to be initiated at 30 minutes after a LOCA. The CR unfiltered inleakage is conservatively assumed to be 18,500 cfm during this 30-minute normal mode of CR HVAC operation based on a parametric study performed to determine the unfiltered inleakage to maximize CR dose as shown in Figure 1.
    - The modeled unfiltered inleakage rates include ingress / egress inleakage of 10 cfm.
    - The most limiting atmospheric dispersion factors X/Q generated for the CR intake are representative for control room inleakage.
    - The currently licensed value (as discussed in current TS 5.5.7) for MCREV filter bypass leakage (1%) is used in determining CR radiological dose. The AST analysis calculates a reduced filter efficiency to account for the currently licensed bypass leakage value. Therefore, the current licensing basis bypass value of 1% remains acceptable.
- No credit for KI pills or respiratory protection is taken.

4.2.2 Atmospheric Dispersion

4.2.2.1 Onsite Meteorological Measurements Program

The PBAPS meteorological measurement program meets the guidelines of Regulatory Guide 1.23 (Safety Guide 23) Proposed Revision 1, "Onsite Meteorological Programs" (Reference 7.11). The tower base areas are on natural surfaces (e.g., short natural vegetation) with towers

free from obstructions and micro-scale influences. This ensures that data is representative of the overall site area.

All sensors and related equipment on the towers are calibrated according to written procedures designed to ensure adherence to Regulatory Guide 1.23 guidelines for accuracy. Calibrations occur at least every six (6) months, with component checks and adjustments performed when required.

Inspections and maintenance of all equipment is accomplished in accordance with written procedures. Instrument checks occur at least once per week. Qualified technicians are available who are capable of performing maintenance if required. In the event that the required maintenance could affect the instrument's calibration, another calibration is performed prior to returning the instrument to service.

Data from the towers are digitized and transmitted to the control room and to an on-site computer for archive storage. Periodically, all digital and analog data are sent to the approved meteorological consultant for data processing and analysis. Upon receipt of the digital data, the consultant performs a quality check on system performance with the objective of identifying potential problems and to notify plant personnel as soon as possible in order to minimize down-time. This quality check consists of time continuity, instrument malfunction, directional switching problems, negative speeds, missing data, and digital/analog correlation.

Data are compared with other monitoring site or regional data for consistency. If deviations occur, they are evaluated and dispositioned as appropriate.

Meteorological data utilized for this evaluation were selected from the historical record of the PBAPS meteorological monitoring tower network. Monitoring records dating back to 1967 and extending through 2001 were reviewed. Examination of the release locations and configurations in conjunction with the sharply varying topography (both in the vicinity of the release and at the desired receptor locations to be addressed) resulted in the selection of three (3) different towers from which representative data for the χ/Q modeling analyses were used. It was desired that this calculation be based upon a continuous five-year period of data common to all 3 towers, and meet NRC Regulatory Guide 1.23 (Safety Guide 23) (Reference 7.11) specifications. Data from the period of 1984 through 1988 was selected because it meets these criteria.

The meteorological measurements program at PBAPS consists of monitoring wind direction, wind speed, temperature, dew point, and precipitation. The method used for determining atmospheric stability is delta temperature (T), which measures the vertical temperature difference. When the delta-T method is not available, the sigma theta (statistical variation in wind direction) method can be used. However, for the data used in the calculation, no sigma theta data was used. These data, referenced in ANSI/ANS-2.5-1984 (Reference 7.10), are used to determine the meteorological conditions prevailing at the plant site.

The meteorological towers are equipped with instrumentation that conforms to the system accuracy recommendations of References 7.10 and 7.11. The equipment is placed on booms oriented into the generally prevailing wind at the site.

Recorded meteorological data are used to generate joint frequency distributions of wind direction, wind speed, and atmospheric stability class used to provide estimates of airborne concentrations of gaseous effluents and projected offsite radiation dose. Better than 90% data recovery is attained from each measuring and recording system.

It should be noted that an upgrade to the meteorological monitoring system was completed in May 1983. For approximately the next year, strip charts were used to record the data. The accuracy of reading these charts was such that 1/10th of a mile per hour could not be readily seen. In the 1984/1985 timeframe, a new digital data logger was purchased and placed into service, thereby affording better accuracy. Additionally, prior to this, the older Climatronics wind cups, would report a wind speed of 0.5 mph if there were no motion of the cups. The use of the strip charts to record data also compounded this effect. Inclusion of the 1984 wind data with more frequent low wind speeds adds conservatism to the calculation.

4.2.2.2 Atmospheric Dispersion Factors

MSExcel© spreadsheet software was used to convert hour-by-hour delta-T data values recorded in "°F", as measured over a height range specified in "feet", into "delta-T/height" values in units of "°C/100 meters", which were then assigned the appropriate hourly stability class values as prescribed by Safety Guide 23 (Reference 7.11). Also, in order to provide wind speed data compatible with the ARCON96 (Reference 7.9) input requirement for "wind speed times 10", raw wind speed values were reformatted within MSExcel© by appropriately adjusting the decimal in the wind speed data, as applicable.

Wind rises and joint frequency distributions were reviewed for meteorological and climatological reasonableness and found to be acceptable prior to use. A review was also conducted on specific hourly data prior to the execution of the atmospheric dispersion calculations in PAVAN (Reference 7.8) and ARCON96. This consisted of manual spot checks of the MSExcel© spreadsheet reformatted data in comparison with the raw data provided by the vendor.

Atmospheric dispersion coefficients were calculated, for the identified release paths, based on site-specific meteorological data collected between 1984 through 1988. The dispersion coefficients developed represent a change to those used in the current UFSAR analyses. The values currently in the UFSAR are based on Regulatory Guide 1.3 (Reference 7.12). The Regulatory Guide 1.145 (Reference 7.7) results were used for the offsite atmospheric dispersion coefficients for the AST analyses.

Figure 2 illustrates the release-intake points for PBAPS. The χ/Q values for these release-intake combinations are summarized in Table 4.3-1.

Table 4.3-1 lists χ/Q values used for the control room dose assessments. The ground level release χ/Q values (i.e., LOCA-MSIV and FHA release) were calculated by the ARCON96 computer code. The elevated release χ/Q values (i.e., LOCA main stack release) were calculated using ARCON96, and also the PAVAN code in accordance with Regulatory Guide 1.145 methodology. The separate χ/Q results of each of these two models were then analyzed according to the methodology in RG 1.194 (Reference 7.19), and the appropriate controlling χ/Q values for the elevated release were determined and are given in Table 13. These results are based on site-specific hourly meteorological data in a five-year period of record.

Table 4.3-1 lists χ/Q values for the EAB and LPZ boundaries. These values, including an initial 30-minute fumigation period, were calculated using the PAVAN code and Regulatory Guide 1.145 guidance using the same five-year period of record of site hourly meteorological data.

PAVAN was used for releases from the main stack to the control room as suggested in Regulatory Guide 1.194, Section 3.2.2. For conservatism, stack grade (85.4 m) was assumed to be equal to plant grade (35.4 m). The effective release height was then determined in accordance with RG-1.194, which states, "the input parameters should be adjusted such that the effective release height is measured from the elevation of the control room outside air intake rather than plant grade." The stack top is 152.4 m above its assumed plant grade elevation and the control room intake is 21.0 m above plant grade elevation, which results in the following calculation:

Stack height – intake height = effective release height

152.4 m – 21.0 m = **131.4 m**

Therefore, 131.4 m is the release height that was utilized in the PAVAN analysis.

Evaluation of High Wind Speeds for PBAPS Secondary Containment dP

The wind speed exceeded only 5% of the time at PBAPS is 10 mph based on five years of measurements (1984-1988) at the 33 ft elevation of the primary meteorological monitoring tower (Tower 1A), consistent with the site's atmospheric dispersion (X/Q) calculation (Reference 7.32). This location (Tower 1A 33 ft elev.) is higher than the top of the reactor building roof, and is thus conservative for use in this evaluation.

Wind creates a negative pressure region along the roof and rear side and a positive pressure region on the upwind side of the reactor building. This evaluation shows that the effect of increasing wind speeds on secondary containment at PBAPS has been considered, and that adequate secondary containment vacuum can be maintained for a DBA.

PBAPS maintains negative pressure in secondary containment with increasing high wind speeds by means of the SGT system. The SGT system is a seismically qualified system consisting of dampers that control system in-leakage and subsequent system exhaust to a Technical Specifications value of less than or equal to 10,500 cfm. SGT system exhaust value is tech spec controlled to a value of less than or equal to 10,500 cfm, with concurrent dP greater than or equal to -0.25 inch.

The SGT system isolates the Secondary Containment, and establishes and maintains a negative pressure in either (or both) unit's Reactor Building following a DBA. The potential increases in inleakage associated with increasing wind speed are accommodated by changes in damper positioning. Changes in damper positioning maintain dP closer to the -0.25" value by ensuring that SGT exhausts an air flow less than or equal to 10,500 cfm.

The Reactor Building above the Refueling Floor is maintained at a negative pressure by the modulation of pneumatic damper PO-09A-2(3)0477-02. The pneumatic damper is controlled by controller DPC-2(3)0479-02. The controller maintains this negative pressure by comparing the outside atmosphere pressure from element PE-2(3)0003-02 with the Reactor Building above the Refueling Floor pressure from element PE-2(3)0003-04.

The Reactor Building below the Refueling Floor is maintained at a negative pressure by pneumatic damper PO-09A-2(3)0477-01, controller DPC-2(3)0479-01, outside atmosphere

pressure element PE-2(3)0003-01, and Reactor Building below the Refueling Floor pressure element PE-2(3)0003-03.

The modulation of the pneumatic dampers controls the amount of flow exhausted from the Secondary Containment to the SGT system. The current Technical Specification controlled values of SGT ensure the system's integrity during increased wind speeds.

In addition to the function of the SGT system, increased flow removes activity from the secondary containment at a faster rate, but this is offset by the X/Qs expected at these elevated wind speeds. These are impacted by the fact that wind speed is the denominator of the derivation equations, and high wind speeds are generally associated with atmospheric stability classes that result in higher dispersion.

The wind speed achieved only 5% of the time at PBAPS, 10 mph, does not challenge the ability of secondary containment to maintain a negative pressure with increasing wind speed.

4.2.2.3 Control Room

Inleakage of unfiltered air into the control room occurs through the control room boundary, system components, and backflow at the control room doors as a result of personnel ingress or egress to or from the control room.

The unfiltered inleakage allowance is assumed to be at control room intake equivalent conditions. The control room is maintained at a positive pressure with a once-through ventilation system. Unfiltered inleakage paths are therefore limited to small portions of the ventilation system and control room boundary that are found to be at a negative pressure. The Turbine Building surrounds the control room on 3 lateral sides with the remaining side facing the Radwaste Building. The infiltration paths were not specifically identified via inleakage testing via tracer gas. Since the control room was verified to be at a positive pressure with respect to adjacent areas, it is therefore assumed that the inleakage is through the ventilation system. Therefore, intake X/Qs are appropriate.

During emergency pressurized modes of operation, the control room ventilation system supplies 3,000 cfm of filtered, outdoor air to maintain the control room at 0.1-inch water column positive pressure with respect to the adjacent areas. Intentionally admitting outdoor air into the control room facilitates reduction of infiltration through the control room boundary by assuring that air is exfiltrating from the zone at an adequate velocity (i.e., a velocity through the control room boundary to develop and maintain a pressure of 0.1-inch water column).

During the isolation mode, infiltration through the control room boundary is initially negligible because the control room will be at a positive pressure at the time of system isolation. Infiltration following isolation is assumed to be 500 cfm of unfiltered inleakage, which includes impacts due to ingress and egress. This bounds the amount as measured by a tracer gas test.

Infiltration through the system components located outside the control room occurs through joints and seams in the ductwork, around damper shafts, through joints and penetrations in the air-handling units, and through the dampers that isolate the control room from non-habitable areas. The inleakage has been measured via tracer gas testing. This AST analysis assumes a value of 500 cfm, which is greater than the 369 cfm resulting from the tracer gas test.

The opening and closing of boundary doors can induce infiltration to the control room. The backflow infiltration is conservatively assumed at 10 cfm as recommended by Reference 7.13. This 10 cfm is included in the 500 cfm total unfiltered inleakage value assumed in the analysis. Since this inleakage is assumed to occur continuously (even though the access doors are not open continuously), the intake X/Qs are used in the analysis.

Potential adverse interactions between the control room and adjacent zones that may allow the transfer of toxic or radioactive gases into the control room are minimized by maintaining the control room at a positive pressure of 0.1-inch water column with respect to adjacent areas during emergency pressurized modes.

The standard breathing rates and occupancy factors used for control room dose assessments and for the offsite personnel are shown in Table 4.3-1, PBAPS AST Design Inputs Used in the LOCA Analysis.

Activity removal mechanisms are included in the applicable event-specific discussions.

4.2.3 Suppression Pool pH Control

PBAPS proposes to credit control of the pH in the suppression pool following a LOCA by means of injecting sodium pentaborate into the reactor core with the standby liquid control (SLC) system. The SLC system design was not previously reviewed for this safety function (pH control post-LOCA). To demonstrate that the SLC system is capable of performing the safety function assumed in the AST LOCA dose analysis, the following addresses previous NRC information requests concerning the Peach Bottom SLC system (Reference 7.18).

1. The SLC system should be classified as a safety-related system as defined in 10 CFR 50.2, and satisfy the regulatory requirements for such systems.

   A SLC system meeting items (a) – (e) below would result in its acceptance in support of a 10 CFR 50.67 request even if the system is not classified as safety-related. The PBAPS SLC System is a safety related system and meets the criteria of (a) – (e) below.

   (a) The SLC system should be provided with standby AC power supplemented by the emergency diesel generators.

   (b) The SLC system should be seismically qualified in accordance with Regulatory Guide 1.29 and Appendix A to 10 CFR Part 100 (or equivalent used for original licensing).

   (c) The SLC system should be incorporated into the plant's ASME Code ISI and IST Programs based upon the plant's code of record (10 CFR 50.55a).

   (d) The SLC system should be incorporated into the plant's Maintenance Rule program consistent with 10 CFR 50.65.

   (e) The SLC system should meet 10 CFR 50.49 and Appendix A to 10 CFR 50 (GDC 4, or equivalent used for original licensing).

2. The operators should have procedures for injecting the sodium pentaborate using the SLC system.

   (a) Have the SLC activation steps been placed in a safety-related plant procedure?

It is planned that SLC System activation steps credited for AST would be located in Severe Accident Management Plan procedure SAMP-1, "RPV and Primary Containment Flooding" (developed from the BWROG SAGs). Existing steps in SAMP-1 direct injection of Boron (sodium pentaborate) into the RPV with the SLC system shortly upon entry into the procedure and ensure injection of the entire contents of the SLC tank (until 0% tank level). It is planned that the Bases for these steps would be revised to discuss use of Boron injection for Torus pH control.

It is planned that Alarm Response Card procedure 00C226B A-3 (00C226D B-5), "Unit 2(3) Containment Radiation Monitor Hi-Rad" would be updated to direct injection of boron (sodium pentaborate) into the RPV with the SLC system if primary containment radiation levels reach 10,000 R/Hr. Injection will continue until the entire contents of the SLC tank is injected (until 0% tank level). These steps would provide direction to the operator in addition to the direction in the SAMP-1 procedure for injection of Boron for Torus pH control.

(b) Are the steps activated by parameters that are symptoms of imminent or actual core damage?

Entry into the Severe Accident Management Plan (SAMP) procedures is directed when the adequate core cooling requirements of the Emergency Operating Procedures are not satisfied or there is evidence of core damage occurring due to a loss of adequate core cooling. Entry conditions to the SAMP procedures are symptomatic of conditions of imminent or actual core damage due to a failure to maintain adequate core cooling.

It is planned that procedural steps located in the containment high radiation Alarm Response Card procedure would direct boron injection using the SLC system if containment radiation level reaches 10,000 R/hr. This radiation level is indicative of actual core damage based on previously approved plant calculations. Calculation PM-1056 (Reference 7.33) determines the suppression pool pH based on the expected radiation level in the containment following a large-break LOCA. This expected value is in the megarad range (100 times the 10,000 R/hr used here as a symptomatic trigger for the injection of SLC. This margin is sufficient to positively identify the fact that fuel damage has occurred, but is still 100 times lower than that dose where the majority of the radiolysis is occurring. Additionally, this dose rate would be seen very early on after the onset of a LOCA. Therefore, there is ample warning provided to inject SLC.

(c) Does the instrumentation relied upon to provide this indication meet the quality requirements for a Type E variable as defined in RG 1.97 Tables 1 and 2?

The instruments used to provide indication for entry into the SAMP procedures, such as containment pressure and reactor vessel water level (i.e., LOCA signature parameters) meet the quality requirements for a Type E variable as defined in RG 1.97 Tables 1 and 2 and are listed under Peach Bottom Tech Spec 3.3.3.1, "Post Accident Monitoring Instrumentation."

The instruments used to provide indication of primary containment radiation

levels are RR-8(9)103A,B, are also listed under Peach Bottom Technical Specifications Section 3.3.3.1, "Post Accident Monitoring Instrumentation." RR-8(9)103A,B are classified as Type E variable components as defined by Regulatory Guide 1.97, Table 1.

(d) Will plant personnel receive initial and periodic refresher training in the procedure?

Licensed Operators receive initial and continuing training on SAMP procedures and Alarm Response Card procedures including the Bases for actions directed by the SAMP procedures such as injection of Boron. It is planned that Licensed Operators would receive refresher training if the task is selected for continuing training. Simulator scenarios that result in containment radiation reaching the alarm response card procedure threshold for injection of Boron for Torus pH control would exercise the operators in performance of that task. Selected members of the Emergency Response Organization receive initial and continuing training on SAMP procedures including the Bases for actions directed by the procedures.

(e) Will other plant procedures (e.g., ERGs/SAGs) that call for termination of SLC as a reactivity control measure be appropriately revised to enable SLC injection for pH control?

It is planned that PBAPS EOP T-101, "RPV Control" would be revised to permit injection of Boron for Torus pH control to continue even if boron injection is not or no longer needed for reactivity control. T-101 procedure steps that currently direct termination of boron injection as a reactivity control measure, i.e. "WHEN an ATWS is no longer in progress, THEN terminate boron injection...." would be revised to permit injection to continue if boron injection is directed for pH control. It is planned that the Bases for the EOP step would be revised to include a discussion of use of Boron injection using the SLC system for Torus pH control as directed by the Containment High Radiation Alarm Response Card procedures as an exception to the direction to terminate boron injection.

SAMP (SAG) procedures have been reviewed and do not require revision. SAMP procedures only direct the termination of SLC when tank level drops to 0 % for the purpose of protecting SLC equipment. This ensures injection of the entire contents of the SLC tank independent of reactivity control considerations.

3. A sufficient concentration and quantity of sodium pentaborate should be available for injection into the reactor vessel to control pH in the suppression pool.

The source term analysis is tied to the plant's design basis accident, which is the large break LOCA, a break of a recirculation pipe.

<u>Part I - Suppression Pool pH Analysis Assumptions, Inputs, Methods, and Results</u>

The calculation of suppression pool pH is based on the previously approved methodology developed for the calculation done for the Grand Gulf Nuclear Station (GGNS), Unit 1, as revised December 2000. The accuracy of translation of the equations in these documents into spreadsheet cell formulas was verified by duplicating the Grand Gulf calculation.

Injection of sodium pentaborate solution by the Standby Liquid Control System is a required function in order to control post-LOCA pH in the suppression pool and prevent iodine re-evolution. Based on the worst-case beginning of cycle condition, injection should be completed within 24 hours after the start of the DBA-LOCA. Therefore, manual initiation is acceptable. Manual initiation of SLC is expected early in a DBA-LOCA as a result of emergency procedures for events resulting in fuel damage.

Acceptance Criteria: Per the guidance of Appendix A of Regulatory Guide 1.183, the Suppression Pool pH should be controlled at values of 7 or greater following loss of coolant accidents.

Assumptions

The Suppression Pool is assumed to be well mixed so that the pH at any time can be represented by a single value.

Cable parameters include the exposed termination length of what is in a raceway. As a conservative estimate of the cable lengths in free air, an additional 5% of the raceway's totals were assumed to be in free air. A 10% contingency on the cable surface is also included. Radiolysis of surface coatings on the steel and concrete surfaces in the Drywell and Containment would not be significant contributors, since the coatings utilize non-chlorinated polymers.

Temperature

Suppression Pool temperatures are taken from UFSAR Figures 14.6-12 and Figure 14.6.12A. Refer to the attached calculations for additional information (Attachment 8 – compact disk).

Sodium Pentaborate mass in SLC Tank

Per Technical Specification SR 3.1.7.7, the minimum B-10 stored in the SLC tank is 162.7 lbs. In order to prepare this calculation, total boron (B-10 and B-11) is needed. The highest enrichment used at Peach Bottom is 63.5 atom % B-10. For this calculation, 65 atom % B-10 enrichment is assumed. Since B-10 has an atomic weight of 10.01, this gives 7373 gram-atoms of B-10 and 11,342 gram atoms of total boron. Since the formula of Sodium Pentaborate is $Na_2B_{10}O_{16}{\cdot}10H_2O$, there are 1134 gram-mols of the pentaborate in the SLC Tank.

Suppression Pool Volume

The maximum suppression pool volume is limiting with respect to the calculation pH results. For this analysis a beyond design basis value of 151,488 cubic feet is used. This value is the "Maximum Pressure Suppression Primary Containment Water Level" that corresponds to the bottom of the vent ring header (17.1 ft) in the torus, and provides a limited additional margin for beyond design basis accident water additions. The above value exceeds that which corresponds to the total of the TS suppression pool maximum water level (14.9 ft) plus water in the vessel and attached piping, as discussed in and used for the response that follows on adequacy of early mixing of the SLC solution with suppression pool water.

Hydriodic Acid Production

Iodine, accompanied by Cesium, is released during the Gap Release and Early In-Vessel Release phases.

The following equation, valid during the Early Vessel Release Phase, includes the release during the Gap Release Phase.

Iodine and cesium core inventories are calculated for both beginning and end of cycle (BOC and EOC) conditions. Since EOC conditions result in increased inventory of both acidic (iodine) and basic (cesium) compounds, pH values are calculated for both conditions. For conservatism, the EOC radiation doses are used for the BOC calculation.

The hydriodic acid concentration is governed by the following equation:

$$[HI](t) = m_I / (120 * V_{POOL}) * [t - (0.5 + t_{gap})] + m_I / (400 * V_{POOL}) \quad \text{[Equation 1]}$$

where:

$[HI](t)$ = concentration of Hydriodic Acid at time t (moles/liter)
m_I = core iodine inventory (gram-moles)
V_{POOL} = Suppression Pool volume (liters)
t = time after start of accident (hrs) (includes t_{gap} + Gap Release [0.5 hrs] +
Early In-Vessel Release [1.5 hrs] durations for a t_{max} = 2.0336 hrs)
t_{gap} = time of onset of gap release = 121 seconds = 0.0336 hrs
t_{max} = 2.0336 hrs = end of Early In-Vessel Release

Nitric Acid Production

Nitric Acid is produced by radiolysis of the water in the Suppression Pool with a G value of 0.007 molecules HNO_3 / 100 eV absorbed dose or 7.3E-6 g moles / megarad- liter.

The nitric acid concentration is governed by the following equation:

$$[HNO_3](t) = 7.3E\text{-}6 * D(t)_{pool} \quad \text{[Equation 2]}$$

where:

$[HNO_3](t)$ = nitric acid concentration at time t (moles/liter)
$D(t)_{pool}$ = Total accumulated dose in Suppression Pool at time t (megarad)

Hydrochloric Acid Production

Hydrochloric Acid is produced by radiolysis of chlorinated polymer cable jacketing. Radiolysis of surface coatings on the steel and concrete surfaces in the Drywell and Containment would not be significant contributors, since the coatings utilize non-chlorinated polymers.

The calculation of the resulting concentration in the Suppression Pool is based on the equations in the GGNS Calculation. These equations are in turn based on the following G value for HCl production in Hypalon chlorinated polymer.

G_{HCl} = 2.115 molecules/100eV = 3.512E-20 g moles HCl / MeV

The hydrochloric acid concentration is governed by the following equations:

Doses from beta and gamma radiation are calculated separately.

$$[HCl]_\beta(t) = G_{HCl} / V_{POOL} * (S_{tray} / 2 + S_{fa}) / \mu_{\beta\ air} * D_\beta(t) \quad \text{[Equation 3]}$$

where the effective cable surface area for β dose is:

$$S_{tray} / 2 + S_{fa} = \pi * D_0 * (L_{tray} / 2 + L_{fa})$$

$$[HCl]_\gamma(t) = G_{HCl} / V_{POOL} * (S_{tray} + S_{fa}) * (1 - e^{-\mu\lambda\ air * r\lambda}) / \mu_{\gamma\ air} * (1 - e^{-\mu\lambda\ Hypalon * th}) * D_\gamma(t) \quad \text{[Equation 4]}$$

where:

$S_{tray} + S_{fa} = \pi * D_0 * (L_{tray} + L_{fa})$

$[HCl]_\beta(t)$ = HCl concentration from Beta radiation at time t (g moles/liter)

$[HCl]_\gamma(t)$ = HCl concentration from Gamma radiation at time t (g moles/liter)

D_0 = cable diameter (cm)

L_{tray} = cable length in trays (raceways) (cm)

L_{fa} = cable length in free air (cm)

$\mu_{\beta\ air}$ = linear beta absorption coefficient in air (1/cm)

$\mu_{\lambda\ air}$ = linear gamma absorption coefficient in air (1/cm)

r_λ = gamma free path (cm)

$\mu_{\lambda\ Hypalon}$ = linear gamma absorption coefficient in Hypalon (1/cm)

th = Hypalon jacket thickness (cm)

$D_\beta(t)$ = accumulated beta dose per unit volume at time t (MeV/cm^3)

$D_\gamma(t)$ = accumulated gamma dose per unit volume at time t (MeV/cm^3)

G_{HCl} = 3.512E-20 (g moles HCl / MeV)

V_{POOL} = Suppression Pool volume (Liters)

S_{tray} = Cable surface area in trays (cm^2)

S_{fa} = Cable surface area in free air (cm^2)

Cesium Hydroxide Production

Cesium, accompanied by iodine, is released during the Gap Release and Early In-Vessel Release phases. The following equation, valid during the Early Vessel Release Phase, includes the release during the Gap Release Phase.

Iodine and cesium core inventories are calculated for both beginning and end of cycle (BOC and EOC) conditions. Since EOC conditions result in increased inventory of both acidic (iodine) and basic (cesium) compounds, pH values are calculated for both conditions. For conservatism, the EOC radiation doses are used for the BOC calculation.

The cesium hydroxide concentration is governed by the following equation:

$$[CsOH](t) = (0.4 * m_{Cs} - 0.475 * m_I) / 3 * V_{POOL}) * [t - (0.5 + t_{gap})] + (0.05 * m_{Cs} - 0.0475 * m_I) / V_{POOL} \quad \text{[Equation 5]}$$

where:

[CsOH](t) = concentration of Cesium Hydroxide at time t (g moles/liter)

m_I = core Iodine inventory (gram-moles)

m_{Cs} = core Cesium inventory (gram-moles)

V_{POOL} = Suppression Pool volume (liters)

t = time after start of accident (hrs) (includes t_{gap} + Gap Release [0.5 hrs] + Early In-Vessel Release [1.5 hrs] durations for a t_{max} = 2.0336 hrs) [per, Table 4, page 1.183-15]

t_{gap} = time of onset of gap release = 121 seconds = 0.0336 hrs

t_{max} = 2.0336 hrs = end of Early In-Vessel Release

Part II– Adequacy of Mixing

The Emergency Core Cooling System (ECCS) takes water from the suppression pool and pumps it into the core region of the reactor vessel. Additionally, the SLC System will pump from the SLC Tank into the core region of the reactor vessel. This water will refill the Reactor Pressure Vessel (RPV) under post-LOCA conditions, and the mixed ECCS water and SLC solution will spill out of the break to the suppression pool. To illustrate the adequacy of SLC solution mixing in the suppression pool, bounding calculations are done for the maximum initial liquid volume of the suppression pool or 127,300 ft^3, per PBAPS UFSAR Table 5.2-1. An electrical failure is assumed resulting in one residual heat removal (RHR) system loop and one core spray (CS) system loop available for containment spray and RPV flooding. One RHR loop (2 pumps) in the Low-Pressure Coolant Injection System is capable of injecting a minimum of 20,000 gpm, and one CS loop (2 pumps) is capable of injecting a minimum of 6,250 gpm in the Low Pressure Core Spray System mode, per PBAPS UFSAR Table 6.3-1. A two-hour time delay before ECCS and SLC initiation to refill the RPV is conservatively assumed, consistent with the event timing in Regulatory Guide 1.183.

After 2 hours, simultaneous injection of sodium pentaborate solution and ECCS fill of the RPV take place. The PBAPS RPV and attached pipes have a volume of 14,000 ft^3. This volume would be flooded within 5.24 minutes utilizing only the minimum ECCS flows above. The minimum flow rate of the SLC pump is not credited in this fill time, but it is noted that 100% injection of the minimum sodium pentaborate solution would occur within 65 minutes. The bottom of the drywell would already be flooded from the RPV blowdown.

Based on the 127,300 $ft^{3\,s}$ suppression pool volume plus the entire RPV and attached pipes 14,000 ft^3 volume (for a total of 141,300 ft^3) and a total minimum 26,250 gpm (210,540 ft^3/hour) flow rate, a turnover of one suppression pool volume containing essentially 100% of the SLC injection is calculated to be completed in less than the assumed 2 hours after the LOCA, with subsequent suppression pool turnovers of the suppression pool volume taking place every 0.6 hours. Thus, the mixing assumption of the Part I analysis is assured.

4. The SLC system should not be rendered incapable of performing its AST function due to a single failure of an active component. For this purpose the check valve is considered an active device for AST since the check valve must open to inject sodium pentaborate for suppression pool pH control.

If the SLC system cannot be considered redundant with respect to its active components, this lack of redundancy may be offset by providing information in (a) below.

(a) Show acceptable quality and reliability of the non-redundant active components and/or compensatory actions in the event of failure of the non-redundant active components.

If you choose this option, provide the following information to justify the lack of redundancy of active components in the SLC system:

(1) Identify the non-redundant active components in the SLC system and provide their make, manufacturer, and model number.

(2) Provide the design-basis conditions for the component and the environmental and seismic conditions under which the component may be required to operate during a design-basis accident. Environmental conditions include design-basis pressure, temperature, relative humidity and radiation fields.

(3) Indicate whether the component was purchased in accordance with Appendix B to 10 CFR Part 50. If the component was not purchased in accordance with Appendix B, provide information on the quality standards under which it was purchased.

(4) Provide the performance history of the component both at the licensee's facility and in industry databases such as EPIX and NPRDS.

(5) Provide a description of the component's inspection and testing program, including standards, frequency, and acceptance criteria.

(6) Indicate potential compensating actions that could be taken within an acceptable time period to address the failure of the component. An example of a compensating action might be the ability to jumper a switch in the control room to overcome its failure. The staff reviewer will consider the availability of compensating actions and the likelihood of successful injection of the sodium pentaborate where non-redundant active components fail to perform their intended functions.

The PBAPS SLC System is functionally redundant to the control rods in achieving and maintaining the Reactor subcritical. Based on this functional redundancy the SLC System as a system by itself is not required to meet single failure criteria; therefore, the SLC System can not be considered redundant with respect to its active components.

Step 4 above states: " If the SLC system can not be considered redundant with respect to its active components, this lack of redundancy may be offset by providing information in (a) above. Therefore, information is provided to support for each of the identified components below.

I. <u>Check Valves - CHK-2(3)-11-16</u>

1. Identify the non-redundant active components in the SLC system and provide their make, manufacturer, and model number.

<u>Response</u>

Component: Check Valves - CHK-2(3)-11-16, "SLC DISCH HEADER TO RPV OUTBD CHK UPSTRM PENE N-42"

Make: 1-1/2", MARK N953M3RY

Manufacturer: FLOWSERVE CORPORATION

Model Number: W8421981 DWG

2. Provide the design-basis conditions for the component and the environmental and seismic conditions under which the component may be required to operate during a design-basis accident. Environmental conditions include design-basis pressure, temperature, relative humidity and radiation fields.

   Response

   Environmental Conditions:

   Maximum Post LOCA Temperature = 151 degrees F
   Worst Case HELB conditions = 185 degrees F
   Maximum Pressure = 15 psia
   Relative Humidity = 100%
   180 day LOCA dose = 2.29×10^5 rads

   Seismic Conditions:

   Maximum Credible Earthquake

3. Was the component purchased in accordance with Appendix B to 10 CFR Part 50?

   Response

   Yes, the component was purchased IAW Appendix B to 10 CFR Part 50.

4. Provide the performance history of the component both at the licensee's facility and in industry databases such as EPIX and NPRDS:

   Response

   Component Performance History at Peach Bottom:

   Local Leak Rate failures for containment isolation purposes occurred as follows:

   - 05/30/87 (reference Work Order C0086960)
   - 11/13/88 (reference Work Orders C0026199 and C0026200)
   - 10/08/95 (reference Work Order C0165642)

   Local leak rate failures are defined as leak rates that exceed 9,000 cc/min in the reverse direction. No failures occurred in the forward direction of the check valve.

   Component Performance History in Industry Databases:

   Search was performed in the following databases:
   SOER/SER/SEN/O&MR; LER 1990 – present; LER 1984 – 1989; Nuclear

Network Operating Experience; NRC; Plant Events (From Events Database). Search did not identify any adverse performance history.

In summary, check valve CHK-2(3)-11-16 has an excellent performance history.

5. Provide a description of the component's inspection and testing program, including standards, frequency, and acceptance criteria.

   Response

   Component testing:

   - SLC Injection Test, ST-O-011-400(5)-2(3), Standby Liquid Control System Loop Injection Test
     - Satisfies Peach Bottom Tech Spec. SR 3.1.7.9, "Ensures that there is a functioning flow path from the boron solution storage tank to the RPV..."
     - Confirm forward direction of check valve by verifying a flow rate of ≥ 43 gpm
     - Frequency = Once every 24 months
   - Local Leak Rate in accordance with 10CFR50 Appendix J, ST 20.11.2
     - Satisfies Peach Bottom TS SR 3.6.1.1.1, "Maintaining the primary containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Primary Containment Leakage Rate Testing Program"
     - Leakage must be less than or equal to 9000 cc/min
     - Frequency = In accordance with Primary Containment Leakage Rate Testing Program

6. Indicate potential compensating actions that could be taken within an acceptable time period to address the failure of the component. An example of a compensating action might be the ability to jumper a switch in the control room to overcome its failure. The staff reviewer will consider the availability of compensating actions and the likelihood of successful injection of the sodium pentaborate where non-redundant active components fail to perform their intended functions.

   Response

   In the unlikely event that a SLC injection path check valve fails to open, there are means of injecting Sodium Pentaborate utilizing two other systems (Reactor Water Cleanup or Control Rod Drive System) that are available for use currently for other accidents such as ATWS. Please note that, although potentially available for use, Peach Bottom is not taking credit for these alternate methods of injecting SLC for pH control. Exelon believes compensating actions are not warranted due to the reliability of the check valves.

II. Check Valves CHK-2(3)-11-17

1. Identify the non-redundant active components in the SLC system and provide their make, manufacturer, and model number.

   Response:

Component:	CHK-2(3)-11-17, " SLC DISCH HEADER TO RPV INBRD CHK DNSTRM PENE N-42"
Make:	1 1/2", MARK 952M3
Manufacturer:	POWELL , WILLIAM CO - USE 461P
Model Number:	2349A-SWE FIG

2. Provide the design-basis conditions for the component and the environmental and seismic conditions under which the component may be required to operate during a design-basis accident. Environmental conditions include design-basis pressure, temperature, relative humidity and radiation fields.

   Environmental Conditions:

   Maximum Post LOCA Temperature = 340 degrees F
   Maximum Pressure = 60.2 psia
   Relative Humidity = 100%
   180 day LOCA dose = 4.57×10^7 rads

   Seismic Conditions:

   Maximum Credible Earthquake

3. Was the component purchased in accordance with Appendix B to 10 CFR Part 50?

   Response

   Yes, the component was purchased IAW Appendix B to 10 CFR Part 50.

4. Provide the performance history of the component both at the licensee's facility and in industry databases such as EPIX and NPRDS:

   Response

   Component History at Peach Bottom:

   No failures identified in the forward or reverse direction.

Component Performance History in Industry Databases:

Search was performed in the following databases:
SOER/SER/SEN/O&MR; LER 1990 – present; LER 1984 – 1989; Nuclear Network Operating Experience; NRC; Plant Events (From Events Database). Search did not identify any adverse performance history.

In summary, check valve CHK-2(3)-11-17 has an excellent performance history.

5. Provide a description of the component's inspection and testing program, including standards, frequency, and acceptance criteria.

   Response

   Component testing:

   - SLC Injection Test, ST-O-011-400(5)-2(3), Standby Liquid Control System Loop Injection Test
     - Satisfies Peach Bottom Tech Spec. SR 3.1.7.9, "Ensures that there is a functioning flow path from the boron solution storage tank to the RPV..."
     - Confirm forward direction of check valve by verifying a flow rate of ≥ 43 gpm
     - Frequency = Once every 24 months
   - Local Leak Rate in accordance with 10CFR50 Appendix J, ST 20.11.2
     - Satisfies Peach Bottom Tech Spec. SR 3.6.1.1.1, "Maintaining the primary containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Primary Containment Leakage Rate Testing Program."
     - Leakage must be less than or equal to 9000 cc/min
     - Frequency = In accordance with Primary Containment Leakage Rate Testing Program

6. Indicate potential compensating actions that could be taken within an acceptable time period to address the failure of the component. An example of a compensating action might be the ability to jumper a switch in the control room to overcome its failure. The staff reviewer will consider the availability of compensating actions and the likelihood of successful injection of the sodium pentaborate where non-redundant active components fail to perform their intended functions.

   Response

   In the unlikely event that a SLC injection path check valve fails to open, there are means of injecting Sodium Pentaborate utilizing two other systems (Reactor Water Cleanup or Control Rod Drive System) that are available for use currently for other accidents such as ATWS. Please note that, although potentially available for use, Peach Bottom is not taking credit for these alternate methods of injecting SLC for pH control. Exelon believes compensating actions are not warranted due to the reliability of the check valves.

III. REMOTE SWITCH RMS-2(3)-11A-S001

1. Identify the non-redundant active components in the SLC system and provide their make, manufacturer, and model number.

   Response

Component:	RMS-2(3)-11A-S001, Remote switch for the "A & B SLC Pumps"
Make:	SB1 with Key Lock
Manufacturer:	General Electric
Model Number:	225A4959P001

2. Provide the design-basis conditions for the component and the environmental and seismic conditions under which the component may be required to operate during a design-basis accident. Environmental conditions include design-basis pressure, temperature, relative humidity and radiation fields.

   Response

   Switches RMS-(2)3-11A-S001 are located in the Main Control Room and are subject to a mild environment.

3. Was the component purchased in accordance with Appendix B to 10 CFR Part 50?

   Response

   Yes, the component was purchased IAW Appendix B to 10 CFR Part 50.

4. Provide the performance history of the component both at the licensee's facility and in industry databases such as EPIX and NPRDS:

   Response

   Component History at Peach Bottom:

   RMS-3-11A-S001 key failed to lock in the "STOP" position (reference Work Order C0183323). The system remained operable in this condition.

   Component Performance History in Industry Databases:

   Search was performed in the following databases:
   SOER/SER/SEN/O&MR; LER 1990 – present; LER 1984 – 1989; Nuclear Network Operating Experience; NRC; Plant Events (From Events Database). Search did not identify any adverse performance history.

   In summary, the remote switch RMS-2(3)-11A-S001 has an excellent performance history.

5. Provide a description of the component's inspection and testing program, including standards, frequency, and acceptance criteria.

   Response

   There is no inspection program currently in place for the control switch. However, this switch is functionally tested every refueling outage during

the performance of surveillance test (ST-O-11-400(5)-2(3)) for the SLC system.

6. Indicate potential compensating actions that could be taken within an acceptable time period to address the failure of the component. An example of a compensating action might be the ability to jumper a switch in the control room to overcome its failure. The staff reviewer will consider the availability of compensating actions and the likelihood of successful injection of the sodium pentaborate where non-redundant active components fail to perform their intended functions.

   Response

   As stated above, the control switch's performance history has been excellent. Additionally, this component is readily accessible with the ability to jumper this switch from the control room within the required injection time.

4.2.4 NUREG-0737 (Reference 7.21) Evaluation

A review of the applicability of the revised Alternative Source Term to the current Technical Specification bases and various licensee commitments in accordance with NUREG-0737 was completed. NUREG-0737, II.B.3, Post-Accident Sampling Capability, and Item II.F.1, Accident Monitoring Instrumentation, have been evaluated and will not be affected as a result of AST implementation.

NUREG-0737, Item II.B.2, Post-Accident Access Shielding, previously met the guidelines of GDC 19 dose limits (5 Rem whole body limit). To convert whole body and thyroid dose to TEDE, 3% of the thyroid dose, when added to the whole body dose, yields an approximate TEDE dose. Therefore, implementation of AST will bound the previously analyzed requirements, and item II.B.2 will meet the requirements of 10 CFR 50.67 (5 Rem TEDE).

NUREG-0737, Item III.D.1.1, Leakage Control, at PBAPS will continue to be controlled, tested and measured in accordance with the local leak rate test program. No changes will occur as a result of AST implementation.

NUREG-0737, Item III.A.1.2, Emergency Response Facilities, which include the design of the TSC, OSC and EOF, have been analyzed for AST applicability. The TSC is located onsite in the administration building that was previously the Unit 1 control room at a distance of approximately 700 feet from Peach Bottom, Units 2 and 3. The TSC at Peach Bottom is governed as part of the licensing basis of the Emergency Preparedness Plan, and is therefore, not affected by AST. For other areas requiring plant personnel access, a qualitative assessment of the regulatory positions on source terms indicates that, with no new operator actions required, radiation exposures would remain acceptable.

4.2.5 Environmental Qualification

Regulatory Position 6 in RG 1.183 states: "The NRC staff is assessing the effect of increased cesium releases on EQ doses to determine whether licensee action is warranted. Until such

time as this generic issue is resolved, licensees may use either the AST or the TID-14844 assumptions for performing the required EQ analyses. However, no plant modifications are required to address the impact of the difference in source term characteristics (i.e., AST vs. TID-14844) on EQ doses."

Therefore, PBAPS will continue to use Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites", as the radiation dose basis for equipment qualification.

4.3 LOSS OF COOLANT ACCIDENT (LOCA)

4.3.1 Background

Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 defines LOCAs as those postulated accidents that result from a loss of coolant inventory at rates that exceed the capability of the reactor coolant makeup system. Leaks up to a double-ended rupture of the largest pipe of the reactor coolant system are included. The LOCA, as with all design basis accidents (DBAs), is a conservative surrogate accident that is intended to challenge selective aspects of the facility design. Analyses are performed using a spectrum of break sizes to evaluate fuel and ECCS performance. With regard to radiological consequences, a large-break LOCA is assumed as the design basis case for evaluating the performance of release mitigation systems and the containment and for evaluating the proposed siting of a facility.

The Peach Bottom Atomic Power Station (PBAPS) design basis loss of coolant accident (LOCA) is analyzed using a conservative set of assumptions and as-built design input parameters compatible for AST and the TEDE dose criteria. The numeric values of the critical design inputs are conservatively selected to assure an appropriate prudent safety margin against unpredicted events in the course of an accident and compensate for large uncertainties in facility parameters, accident progression, radioactive material transport, and atmospheric dispersion.

Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and Control Room (CR) doses for PBAPS are calculated using the guidance in Regulatory Guide (RG) 1.183, and the Total Effective Dose Equivalent (TEDE) dose criteria.

In addition to direct shine to control room operators, the DBA LOCA calculation is performed for the following design basis post-LOCA release paths:

- Containment Leakage
- Engineered Safety Feature (ESF) Leakage
- Main Steam Isolation Valve (MSIV) Leakage

4.3.2 Assumptions On Transport In The Primary Containment - LOCA

The radioactivity released from the fuel is assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment (drywell). The radioactivity release into the drywell is assumed to terminate at the end of the early in-vessel phase, which occurs at

2 hrs after the onset of a LOCA. The reduction in drywell leakage activity by dilution in the reactor building (RB) and removal by the standby gas treatment (SGT) system filtration is not credited. The analysis dilutes the radioactivity released from the core into the drywell air volume during the first 2 hours of the LOCA, and then into the combined drywell plus suppression chamber air volume after 2 hours, at which time the containment volume becomes well mixed following the restoration of core cooling. The thermal-hydraulic conditions in the primary containment are expected to be quite active at this time due to a very high flow established between the drywell and wetwell as a result of steaming and condensing phenomenon.

4.3.3 Reduction In Airborne Activity Inside Containment – LOCA

The gravitational deposition of aerosols from the containment atmosphere is credited by using the RADTRAD "Powers Model" with a 10th percentile uncertainty distribution resulting in the lowest removal rate of aerosols from the containment. Iodine removal by suppression pool scrubbing is not credited since the bulk of core activity is released to containment well after the initial mass and energy release. Although containment sprays are not credited, the removal of the elemental iodine by wall deposition on wetted surfaces inside containment due to the iodine adsorption is modeled in the same way as containment spray iodine removal. The decontamination factor (DF) of elemental iodine is based on the Standard Review Plan (SRP) 6.5.2 (Reference 7.24) guidance and is limited to a DF of 200. This results in an elemental iodine removal rate of 3.36 hr^{-1} for the first 2.0 hours of the accident with 1.86 hr^{-1} until a DF of 200 is reached at 3.85 hrs.

A reactor building (RB) drawdown time of 3 minutes is used (increased from the previous 2 minutes). The post-LOCA MCREV system initiation time is conservatively delayed for 30 minutes. The use of a large CR unfiltered intake during this 30 minutes delay produces a conservative CR dose during the 3-minute RB drawdown time. A parametric study was performed to determine the CR unfiltered leakage during the isolation delay, which maximizes the CR dose. The results were plotted in Figure 1 and indicate that a CR unfiltered inleakage of 18,500 cfm results in a maximum CR dose due to this large containment leakage. This CR unfiltered inleakage rate is used in the analysis for the first 30 minutes.

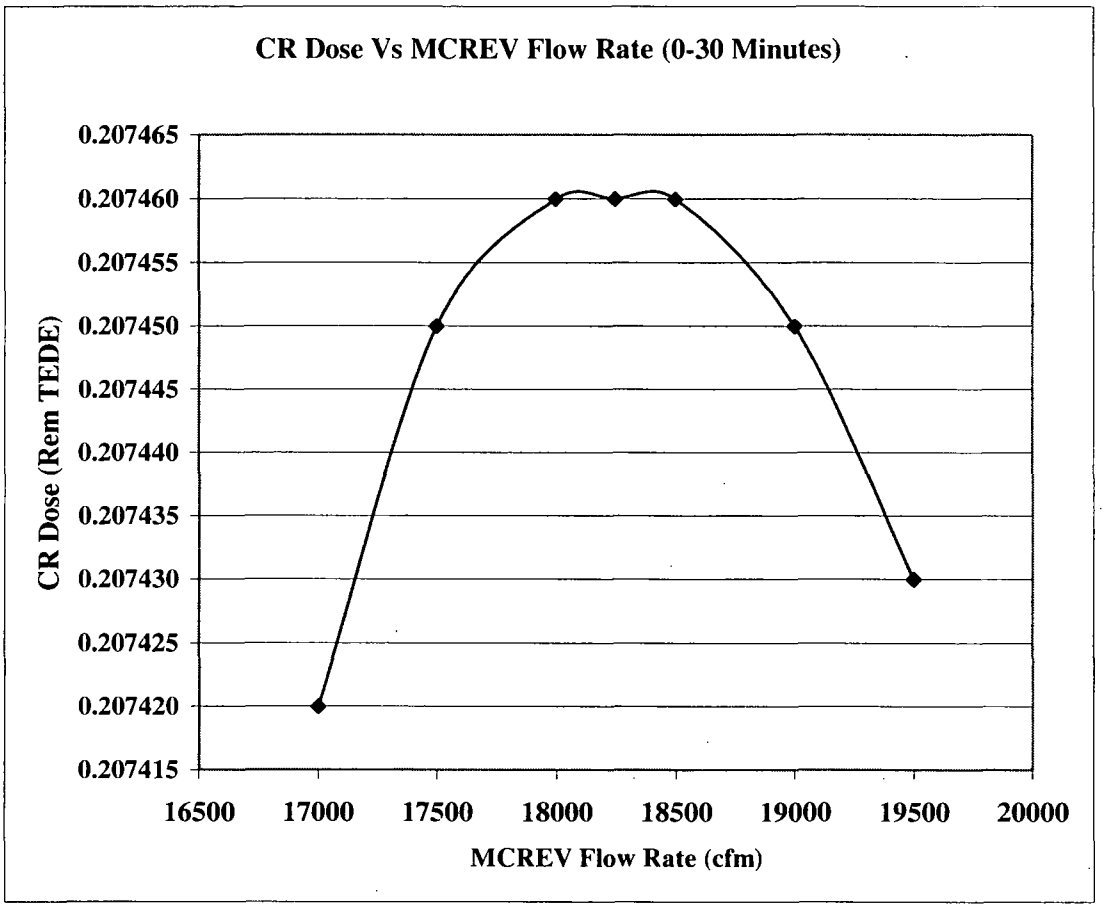


Figure 1: CR Dose vs. MCREV Flow Rate Before Pressurization

4.3.4 Post-LOCA Containment Leakage

A reduction in containment leakage after 24 hours is permitted in RG 1.183. The PBAPS analysis credits this reduction after 38 hours as determined in the plant design analysis (Reference 7.36).

The PBAPS, Units 2 and 3, licensing basis has previously included reliance on containment overpressure for the residual heat removal (RHR) and the core spray (CS) pumps following a postulated design basis loss of coolant accident (LOCA). An analysis is maintained that conservatively estimates the Minimum Containment Pressure Available (MCPA) following a postulated design basis LOCA. A brief discussion of this analysis is included in PBAPS UFSAR Section 5.2.4.3.2. Although the MCPA analysis is conservative, a change to the PBAPS Operating License was approved by the NRC (Reference 7.22) to define a Containment Overpressure License (COPL) less than the evaluated MCPA for the design basis LOCA. Both the MCPA for the design basis LOCA and the COPL are shown in PBAPS UFSAR Figure 5.2.16. The PBAPS OL permits credit for containment overpressure for any design basis event up to the MCPA for that event, but not greater than the COPL. Exelon has included a revision to PBAPS UFSAR Section 5.2.4.3.2 and Figure 5.2.16 in this License Amendment Request. The revised Figure 5.2.16 shows the revised MCPA and the proposed new COPL. Text

changes to PBAPS UFSAR Section 5.2.4.3.2 include revision to the discussion of containment leakage. The text now includes containment leakage assumptions from all leakage sources and identifies that these leakage values are consistent with the proposed changes to the PBAPS Technical Specifications. Refer to Attachment 6 of this submittal for additional information on the proposed UFSAR changes and updates to responses to previous requests for additional information dated April 23, 2004, and May 20, 2004 (References 7.17 and 7.18).

The containment leakage during and following the reactor building drawdown is assumed to be directly released to the atmosphere via the reactor building (RB) stack and main stack, respectively, without filtration through the SGTS charcoal and HEPA filters. Leakage from the primary containment is postulated to be released directly to the environment without mixing in the RB free air volume.

4.3.5 Containment Leakage Source Term – LOCA

The post-LOCA containment leakage model is shown in Figure 2. The BWR core inventory fractions listed in Regulatory Guide 1.183, Table 1 are released into the containment at the release timing shown in RG 1.183, Table 4. Since the post-LOCA minimum suppression chamber water pH is greater than 7.0 for the duration of the accident, the chemical form of radioiodine released into the containment is assumed to be 95% 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, the remaining fission products are assumed to be in particulate form. The plant-specific isotopic fission product core activities (in units of curies) were calculated and converted into Ci/MW_t using the core thermal power level.

4.3.6 Containment Purging

The PBAPS containment is not routinely purged for pressure control during power operations. However, the PBAPS containment is purged in preparation for outages. Containment purging is currently limited to 90 hours per calendar year in accordance with UFSAR requirements. A TS requirement to not purge the PBAPS, Units 2 and 3, containment greater than 90 hours per calendar year is a part of this License Amendment Request. Containment purge as a combustible gas or pressure control measure is not required within 30 days of a DBA LOCA. Therefore, the release from containment purge is not analyzed per RG 1.183, Section A.7. This submittal proposes to include dose requirements in the Technical Specifications in support of the proposed UFSAR changes regarding MCPA.

4.3.7 Post-LOCA ESF Leakage

The post-LOCA ESF leakage release model is shown in Figure 2. The ESF systems that recirculate suppression pool water outside of the primary containment are assumed to leak during their intended operation. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components. The radiological consequences from the postulated leakage are analyzed and combined with the radiological consequences from other fission product release paths to determine the total calculated radiological consequences from the LOCA. ESF components are located in the RB.

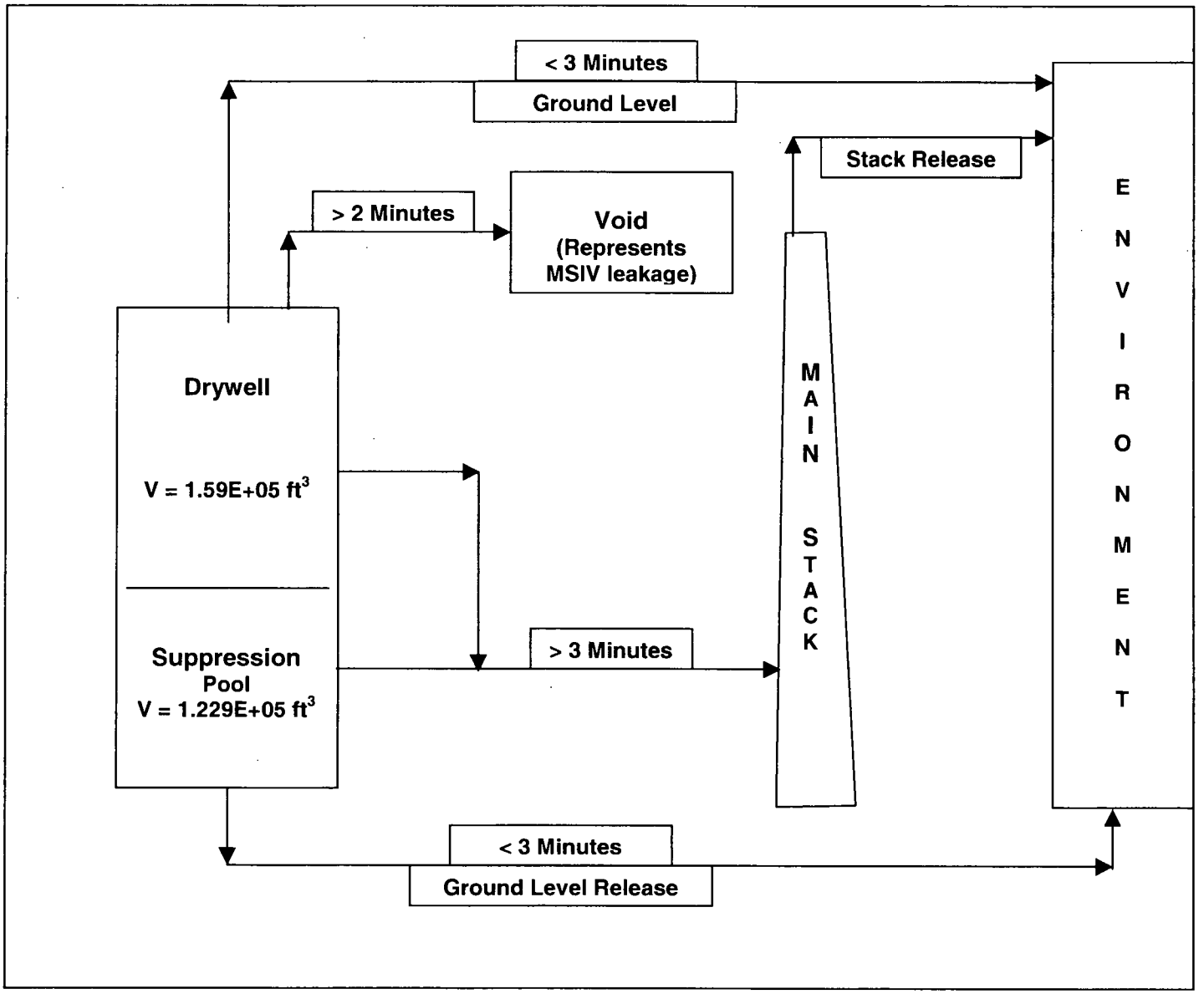


Figure 2: PBAPS Containment and ESF Leakage RADTRAD Nodalization

4.3.8 ESF Leakage Source Term

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 suppression pool water at the time of release from the core. The total ESF leakage from all components in the ESF systems is assumed to be 10.0 gpm, which is 2 times the expected leakage of 5.0 gpm and assumed to start immediately after the onset of a LOCA. With the exception of iodine, all

remaining fission products in the recirculating liquid are assumed to be retained in the pool water. Since the post-LOCA temperature of torus water recirculated through the ESF system is less than 212ºF, 10% of the iodine activity in the leaked ESF liquid is assumed to become airborne. The reduction in ESF leakage activity by dilution in the RB volume and removal by SGT filtration are not credited.

4.3.9 Chemical Form

The radioiodine that is postulated to be available for release to the environment due to ESF leakage is assumed to be 97% elemental and 3% organic.

4.3.10 Post-LOCA MSIV Leakage

MSIV leakage is postulated to be released to the environment through an assumed MSIV failed steam line and one of the three remaining intact steam lines. Each release path consists of two well-mixed volume nodes consistent with the AEB 98-03 (Reference 7.25) two-segment nodalization. Piping between the Reactor Pressure Vessel (RPV) nozzle to outboard MSIV and that between the outboard MSIV to Turbine Stop Valve (TSV) at PBAPS is seismically qualified and credited in this analysis. The well-mixed two-volume nodes eliminate the potential variation of settling aerosol velocities in multiple volume nodes resulting from the different temperature/pressure boundary conditions and remove the in-series configuration of the aerosol and elemental iodine removal efficiencies in multiple volume nodes, which is believed to underestimate the resulting dose.

The post-LOCA MSIV Leakage model is shown in Figure 3. The four main steam lines, which penetrate the primary containment, are automatically isolated by the MSIVs in the event of a LOCA. There are two MSIVs on each steam line, one inside containment and one outside. The MSIVs are functionally part of the primary containment boundary and design leakage through these valves provides a leakage path for fission products that bypass the secondary containment and enter the environment as a ground-level release. Following the initial blowdown of the reactor pressure vessel, steaming in the RPV carries fission products to the containment. When core cooling is restored, the steam and the ESF flow carry fission products from the core to the primary containment via the severed recirculation line, resulting in well-mixed RPV dome and containment fission product concentrations. The main steam isolation valves (MSIVs) are postulated to leak at a total design leak rate of 360 scfh as calculated for the drywell peak pressure of 49.1 psig. The radiological consequences from postulated MSIV leakage are analyzed and combined with the radiological consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA.

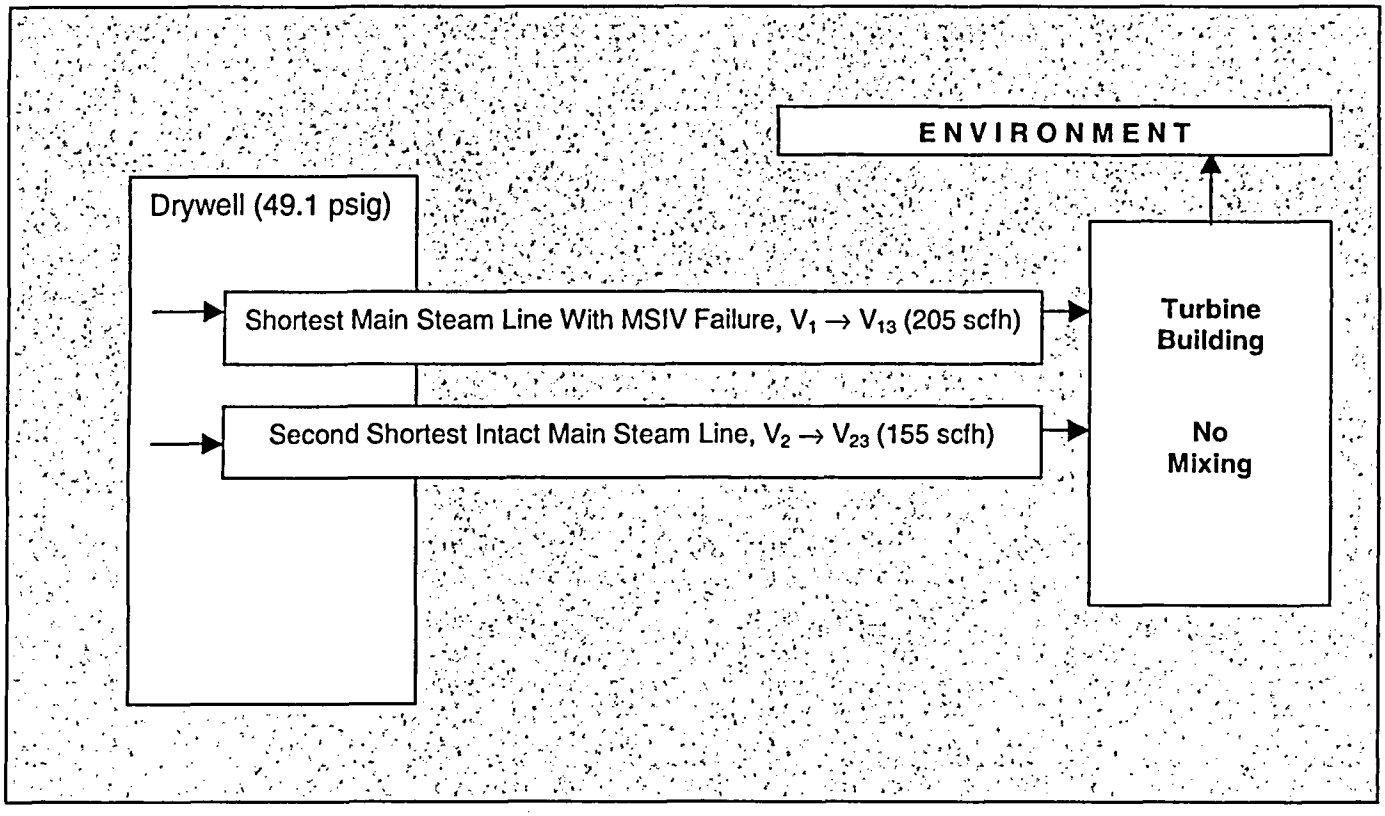


Figure 3: PBAPS MSIV Leakage Path Volumetric Distribution

4.3.11 MSIV Leakage Source Term

The activity available for release via MSIV leakage is assumed to be that activity released into the drywell for evaluating containment leakage per RG 1.183.

As shown in Figure 3, a total of 360 scfh (at 49.1 psig) MSIV leakage is assumed to occur. Steam line selection is based on the ability for deposition. The shortest line will have the least deposition. Therefore, 205 scfh (at 49.1 psig) is assumed to be associated through the shortest steam line. This line is modeled as having the failed inboard MSIV. Conservatively, the deposition of aerosol and removal of elemental iodine activities are not credited in the steam line between the RPV nozzle and the outboard MSIV (Volume V_1). The deposition of aerosols and removal of elemental iodine are conservatively credited in only the horizontal pipe between the outboard MSIV and turbine stop valve (TSV) for 0-96 hrs (Volume V_{13}). A leakage rate of 155 scfh (at 49.1 psig) is assumed to leak through the shortest of the three intact steam lines. The deposition of aerosol and removal of elemental iodine activities are similarly credited in only the horizontal pipe segments between the RPV nozzle and TSV for 0-96 hrs (Volumes V_2 and V_{23}). It is further assumed that 0 scfh passes through the second and third shortest intact steam lines. Note that if leakage through the valves in these lines occurs, as long as no one line leaks more than 116 scfh at 25 psig test pressure and 204 scfh total for all lines, doses will be acceptable (refer to Section 7.2.6 of Reference 7.37).

The aerosol deposition removal efficiencies for the main steam lines are determined based on the methodology in Appendix A of AEB-98-03 using only the horizontal pipe projected area (Diameter x Length). The natural removal efficiency for elemental iodine in each steam line volume is assumed to be 50% as recommended in the AEB 98-03, Appendix B, page B-3.

4.3.12 Determination of MSIV Leak Rates In Various Steam Line Volumes

The main steam system piping and isometric drawings for the drywell through the main steam tunnel were reviewed with regard to piping parameters. Isometric drawings indicated that the Unit 2 steam header "A" and the Unit 3 steam header "D" share the same shortest horizontal pipe length between the outboard MSIV and TSV, which provides the minimum horizontal pipe surface area and consequently results in the least aerosol deposition. Therefore, the selection of the two shortest steam headers in either unit is limited to those that result in the least aerosol removal efficiency, which in turn can be determined by the well-mixed volume. The main steam well-mixed volume and horizontal pipe surfaces were then calculated. The rate constant (λ_s) for different piping segments in the MSIV leakage release paths was calculated using the 40th percentile aerosol settling velocity (see AEB 98-03). The aerosol removal efficiencies due to gravitational depositions on the horizontal pipe surfaces were calculated using the mass balance equation for the well-mixed volumes.

The total MSIV leakage from all main steam lines is proposed to be increased from 46 scfh as measured at 25 psig to 360 scfh calculated at 49.1 psig, allowing a maximum of 205 scfh (at 49.1 psig) from any one of the 4 main steam lines. The total MSIV leakage of 360 scfh (at 49.1 psig) is converted using the Ideal Gas Law to determine the actual leakage (cfh) using the post-LOCA peak drywell temperature and pressure. Since the actual MSIV leak rate is reduced at the accident condition due to the combined effects of compression (due to the high pressure) and expansion (due to the high temperature), the increase in the MSIV leak rates to the environment from the TSVs are conservatively calculated using the Ideal Gas Law and drywell post-LOCA peak pressure and temperature. The MSIV leak rates are used in the analysis with aerosol removal efficiencies calculated based on the horizontal pipe surface areas calculated. To account for the assumed mixing between the wetwell and drywell after 2 hours and the resulting activity dilution, the flow rate through the MSIVs is reduced by the ratio of the drywell volume to the total volume at two hours.

4.3.13 Recirculation Line Rupture vs. Main Steam Line Rupture

Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 defines LOCAs as those postulated accidents that result from a loss of coolant inventory at rates that exceed the capability of the reactor coolant makeup system. Leaks up to a double-ended rupture of the largest pipe of the reactor coolant system are included. The LOCA, as with all design basis accidents (DBAs), is a conservative surrogate accident that is intended to challenge selective aspects of the facility design. Per UFSAR Section 6.2, the DBA for the safety related system design is a LOCA. This LOCA leads to a specific combination of dynamic, quasi-static, and static loads in time. The thermal transient due to other postulated events, including the steam line break inside the drywell, does not impose maximum challenge to the drywell pressure boundary and fuel integrity. The LOCA results in the maximum core damage and fission product release as shown in the RG 1.183, Table 1. Therefore, a recirculation line rupture is considered to be the limiting event with respect to radiological consequences.

RG 1.183, Appendix A, Section 6.5 allows reduction in MSIV releases that is due to holdup and deposition in main steam piping downstream of the MSIVs and in the main condenser, including the treatment of air ejector effluent by offgas systems, if the components and piping systems used in the release path are capable of performing their safety function during and following a safe shutdown earthquake (SSE). Although postulating a main steam line break in one steam line inside the drywell would maximize the dose contribution from the MSIV leakage, the steam line break is not a credible event during a LOCA, since the main steam piping is designed to withstand the SSE.

4.3.14. Control Room Model – LOCA

The CR shielding information is used in a conservative manner to calculate the CR doses.

The post-LOCA control room RADTRAD nodalization is shown in Figure 4. Post-LOCA radioactive releases that contribute to the CR TEDE dose are as follows:

- Post-LOCA Containment Leakage
- Post-LOCA ESF Leakage
- Post-LOCA MSIV Leakage

The radioactivity from the above sources is assumed to be released into the atmosphere and transported to the CR air intake, where it may leak into the CR envelope or be filtered by the CR intake filtration system prior to being distributed in the CR envelope. The four major radioactive sources which contribute to the CR TEDE dose are:

- Post-LOCA airborne activity inside the CR
- Post-LOCA airborne cloud external to CR
- Post-LOCA containment shine to CR
- Post-LOCA Main Control Room Emergency Ventilation (MCREV) filter shine

4.3.15 Post-LOCA Airborne Activity Inside CR

The post-LOCA radioactive releases from various sources are shown in Figures 2 and 3. The activities released from the various sources are diluted by atmospheric dispersion and carried to the CR air intake. Atmospheric dispersion factors were calculated for the containment, ESF, and MSIV leakage release points. The containment and ESF leakages have the same release point (Main Stack) and X/Qs. The CR RADTRAD dose model was developed using plant-specific design input parameters. The CR airborne TEDE dose contributions from the above post-LOCA airborne sources were calculated and tabulated.

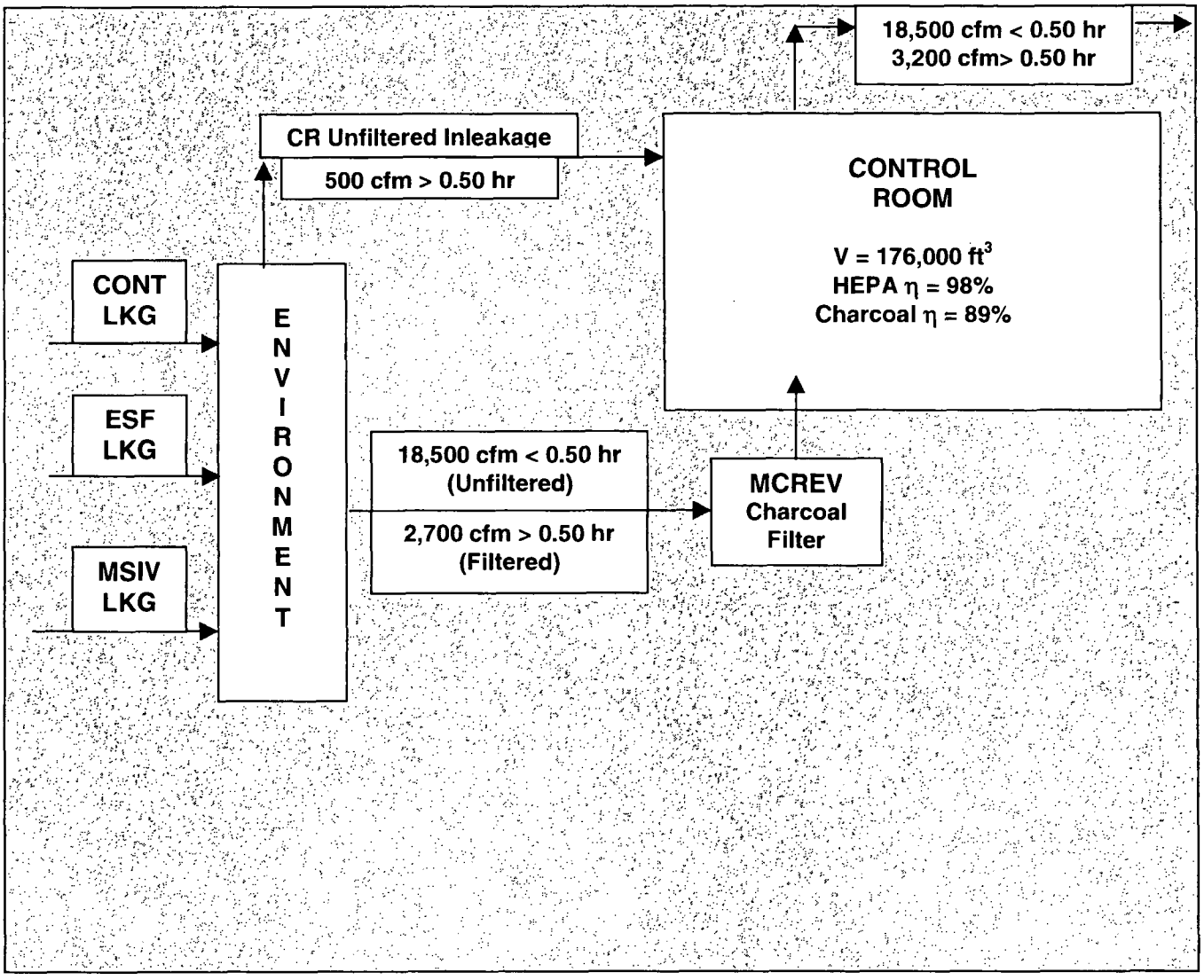


Figure 4: PBAPS Control Room Response RADTRAD Nodalization

4.3.16 Post-LOCA Airborne Cloud External to CR

The post-LOCA radioactive plume contains the radioactive sources from the containment, ESF, and MSIV leakages. The PBAPS combined (common) CR is located at the center of the plant in the turbine building at EL 165'-0". The gamma radiation due to the external radioactive plume shine to the CR personnel is attenuated by the 2'-6" minimum concrete ceiling thickness. The RADTRAD code calculates the whole body gamma dose based on the semi-infinite cloud immersion at the site boundary location. Since the containment and ESF leakages contribute insignificant direct shine whole body dose to the CR operator, they are not considered important sources for the external cloud dose. Therefore only the MSIV leakage path is evaluated to determine the external cloud dose to the CR operator. The χ/Qs for the LPZ receptor modeled in RADTRAD are modified by replacing them with the χ/Qs for the CR air intake location. The

resulting LPZ whole body dose is the semi-infinite gamma dose at the CR air intake. The total whole body gamma dose is 29.30 rem. Since this is a semi-infinite dose at the CR air intake, it is conservative to assign this dose to the CR roof. The gamma attenuation factor is calculated to be 0.000395 for a 1 MeV gamma emission. This attenuation factor includes the buildup due to multiple scattering. The resulting gamma dose from the external cloud shine would be 1.16E-02 rem (29.30 rem x 0.000395 = 1.16E-02 rem), which is added to the dose contributions from other post-LOCA sources.

4.3.17 Post-LOCA Containment Shine to CR

The post-LOCA airborne activity in the containment (drywell) is released into the reactor building (RB) via containment leakage through the penetrations and openings. The major drywell penetrations including the personnel airlock, equipment & CRD removal hatches, and large bore main steam and feedwater piping are located between elevations 135'-0" and 165'-0". The RB concrete shielding surrounding the penetration area varies from 3'-0" at EL 135'-0" to 3'-6" at EL 165'-0". The direct shine dose through the RB penetrations is insignificant due to the large distances between the penetrations and CR panels and the associated concrete shielding. The post-LOCA containment and ESF leakage activities are assumed to be uniformly distributed inside the RB. The airborne activity confined in the space above the operating floor of the RB contributes direct shine dose to the CR operators. The containment and ESF leakages are conservatively modeled to postulate the activity from the leakage is directly released to the environment without mixing in the RB. The activity from this leakage is released to and mixed in the RB volume and then released to the environment via the SGTS system at a rate of 10,500 cfm. The RADTRAD runs model the containment and ESF leakages released to the RB and then released to the environment via the SGTS at a rate of 10,500 cfm to calculate the post-LOCA activity in the RB. The post-LOCA containment leakage and ESF leakage time dependent isotopic activities inside the RB are obtained from the RADTRAD runs are and then combined in a table for time periods from 0 to 96 hrs. A total integrated dose is then derived from the summation of this table.

The concrete block wall/steel shielding on the RB operating floor and multiple concrete floor shadow shielding are conservatively not credited in the analysis. The containment shine dose is calculated for the Unit 2 CR, which is equally applicable to Unit 3 due to the symmetrical shielding geometry. The resulting time dependent CR containment shine dose is calculated based on the dose rates and integrated dose. The containment shine dose is calculated to be 2.80E-02 rem.

4.3.18 Post-LOCA MCREV Filter Shine Model

As previously stated, the PBAPS CR is located at the center of the plant in the turbine building at EL 165'-0". The MCREV charcoal filter is located in the radwaste building at EL 165'-0". The concrete wall between the MCREV charcoal filter and CR panel is 3'-0". The charcoal bed dimension is 2'6" x 6'-6" x 2'-0" (approximately) and is conservatively modeled as a rectangular source of 2'-0" x 2'-0" x 4'-0" at approximately 24'-0" from the concrete wall. The separation distance is approximately 28'-6". The post-LOCA CR doses indicate that MSIV leakage contributes the maximum dose to CR operators. Therefore, the MSIV leakage path is used to assess iodine and aerosol activity buildup on the CR charcoal filter. The resulting CR filter shine dose is adjusted for the containment and ESF leakage dose contributions.

The RADTRAD code calculates the cumulative elemental and organic iodine atoms and the aerosol mass released to the environment from the main steam lines due to MSIV leakage at various time steps. The activity released to the environment is dispersed to the control room HVAC intake louvers, where it is drawn into the MCREV System. The total elemental and organic iodine atoms and aerosol mass drawn into and retained on the MCREV charcoal and HEPA filters were calculated. Decay of the isotopes deposited on the MCREV filters is conservatively ignored.

4.3.19 Post-LOCA Iodine Activity On MCREV Charcoal Filter – MSIV Leakage

The iodine atom/curie relationship is established using RADTRAD, which is a typical relationship for all release paths. The total number of atoms accumulated on the charcoal filter is established based on the charcoal filter efficiency and MCREV intake flow rate. The MCREV charcoal filter efficiency is conservatively assumed to be 99% instead of 89% to maximize the filter shine dose. Knowing the iodine atom/curie relationship and the total number of elemental and organic iodine atoms on the charcoal filter due to the MSIV leakage, the total elemental iodine activity accumulated is then calculated. This calculation indicates the accumulation of iodine due to MSIV leakage is insignificant. This is due to the fact that most of the elemental iodine is removed by elemental deposition in the main steam piping before it is released to the environment and it is further reduced by air dilution before it migrates to the CR air intake.

4.3.20 Post-LOCA Aerosol Activity On CR HEPA Filter – MSIV Leakage

The total aerosol mass deposited on the MCREV HEPA filter due to the MSIV leakage is calculated based on the HEPA filter efficiency and MCREV intake flow rate. Knowing the aerosol mass/curie relationship, and the total mass of aerosols on the HEPA filter, the isotopic aerosol activities deposited on the MCREV filter due to the MSIV leakage is calculated. The resulting isotopic aerosol activity is insignificant due to the fact that most of the aerosols deposit out in the main steam piping horizontal surfaces before being released to the environment. The total post-LOCA iodine and aerosol activity accumulated on the MCREV charcoal and HEPA filters were then determined.

4.3.21 Shielding Analysis - MCREV Charcoal Filter

The MicroShield computer code (Reference 7.26) is used to calculate the post-LOCA MCREV charcoal filter shine dose to CR operators. The post-LOCA iodine and aerosol activity is uniformly distributed on the charcoal bed with a 3'-0" concrete wall located at 24'-0" from the charcoal bed with a dose point located 1'0" from the concrete wall at the center of the source to the CR. This MicroShield direct dose model is conservative with respect to the locations of MCREV charcoal filter, filter dimensions, and CR operator normal occupancy in the vicinity of the CR panels. The CR direct dose rate from the MCREV filter shine is calculated to be 1.92E-03 rem.

4.3.22 Assumptions for LOCA

The following assumptions used in evaluating the offsite and control room doses resulting from a Loss of Coolant Accident (LOCA) are based on the requirements in Regulatory Guide 1.183. These assumptions become the design inputs in Table 4.3-1 and are incorporated in the analyses.

4.3.22.1. Source Term Assumptions

Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are as follows.

4.3.22.1.1 Equilibrium Core Inventory

The assumed inventory of fission products in the reactor core and available for release to the containment is based on the maximum power level of 3,528 MWt, which includes 0.38% margin for instrument uncertainty relative to the rated thermal power of 3,514 MWt after the 1.62% thermal power optimization licensing change. The equilibrium core inventory is described in Table 4.3-1.

4.3.22.1.2 Release Fractions and Timing

The core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage for a Design Basis Accident (DBA) LOCA are those from Regulatory Guide 1.183, Tables 1 and 4. These release fractions are acceptable for use given that the peak fuel burnup meets the 62,000 MWD/MTU requirement specified in Regulatory Guide 1.183 Footnote 10. Reload checks are performed to ensure that PBAPS core designs meet this requirement.

4.3.22.1.3 Radionuclide Composition

The elements in each radionuclide group to be considered in design basis analyses are shown in Table 4.3-1.

4.3.22.1.4 Chemical Form

The long-term suppression pool water pH is greater than 7 during a LOCA. Consequently, the chemical forms of radioiodine released to the containment can be assumed to be 95% 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 are assumed to be in particulate form.

4.3.22.1.5 Assumptions for Activity Transport in Primary Containment

- The radioactivity released from the fuel is assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment. The radioactivity released from the fuel does not mix with the suppression pool air space until after two hours when adequate mixing becomes available.

- Reduction in airborne radioactivity in the containment by natural deposition within the containment is credited using the RADTRAD Powers' model for aerosol removal with a 10-percentile probability.

- The primary containment and the MSIVs are assumed to leak at the allowable Technical Specification peak pressure leak rate for the first 38 hours of the event. Subsequently, the leak is at 50% of the leakage value for the remainder of the event.

- A PBAPS, Units 2 and 3 TS annual limitation regarding containment purge is a part of this License Amendment Request for AST (< 90 hours per year). There is no requirement to relieve pressure or to reduce hydrogen concentration in the containment during a LOCA. Therefore, a release due to containment purge is not required to be analyzed.

4.3.23 Design Inputs – LOCA

4.3.23.1 General Considerations

Applicability of Prior Licensing Basis

The Peach Bottom Atomic Power Station specific design inputs and assumptions used in the TID-14844 analyses were assessed for their validity to represent the as-built condition of the plant and evaluated for their compatibility to meet the AST and TEDE methodology. The AST analyses ensure that assumptions, design inputs, and methods are compatible with the requirements of the AST and the TEDE criteria.

Credit for Engineered Safety Features

Credit is taken only for those 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 failures modeled for the LOCA are the MSIV in one main steam line failing to close and the MCREV system failing to automatically initiate. The single active failure criterion compliance for SLC is discussed in Section 4.2.3.

Assignment of Numeric Input Values

The numeric values that are chosen as inputs to analyses required by 10 CFR 50.67 are compatible to AST and TEDE dose criteria and selected with the objective of maximizing the postulated dose. As a conservative alternative, the limiting value applicable to each portion of the analysis is used in the evaluation of that portion. The use of a 30-minute delay in the CR Emergency Ventilation Mode initiation time, no aerosol and elemental iodine removal after 96 hrs, no aerosol and elemental iodine removal in the failed MSIV line between the RPV nozzle and outboard MSIV for entire duration of a LOCA, total MSIV leakage distribution among two lines instead of four lines, and use of ground release χ/Qs for the containment and ESF leakages during the drawdown time demonstrate the inherent conservatisms in the plant design and post-accident response.

Meteorology Considerations

Atmospheric dispersion factors (χ/Qs) for the onsite release points such as the main stack for containment and ESF leakage release path and the RB/TB exhaust vent for the MSIV leakage release path are developed using PBAPS plant specific meteorology and the NRC sponsored computer code ARCON96. The EAB and LPZ χ/Qs are developed using the same PBAPS plant specific meteorology with the NRC sponsored computer code PAVAN.

Accident-Specific Design Inputs / Assumptions

The design inputs/assumptions utilized in the EAB, LPZ, and CR habitability analyses are listed in the following sections. The design inputs are compatible with the requirements of the AST and TEDE dose criteria and the assumptions are consistent with those identified in Section 3 of Appendix A of RG 1.183. The design inputs and assumptions in the following sections represent the as-built design of the plant.

1. Aerosol Deposition On Horizontal Pipe Surfaces

PBAPS main steam piping from the reactor pressure vessel (RPV) nozzle to the TSV is seismically analyzed to assure the piping wall integrity during and after a seismic (safe shutdown earthquake [SSE]) event. The main steam lines credited in the MSIV leakage path are qualified to withstand the SSE; therefore, these lines are credited for the aerosol deposition in the following section:

PBAPS utilizes the aerosol deposition modeling discussed in AEB-98-03 (Reference 7.25). The Brockmann model for aerosol deposition is based on the plug flow model. The NRC staff concluded that the plug flow model for aerosol deposition in the main steam piping under-predicts the dose. The aerosol settling velocity in the well-mixed flow model depends on the variables having a large range of uncertainty. Therefore, the following aerosol deposition model is used. The NRC Staff performed a Monte Carlo analysis to determine the distribution of aerosol settling velocities for the main steam line during the in-vessel release phase. The results of the Monte Carlo analysis for settling velocity in the main steam line are given in the following table:

Percentile	Settling Velocity (m/sec)	Removal Rate Constant (hr^{-1})
60th (average)	0.00148	11.43
50th (median)	0.00117	9.04
40th	0.00081	6.26
10th	0.00021	1.62

The NRC staff concluded that use of a 10th percentile settling velocity with a well-mixed model is overly conservative and not appropriate. Instead, the Staff believes it is acceptable to utilize median values (i.e., 50th percentile settling velocity). This analysis is conservative relative to the Staff's recommendation, in that it models a 40th percentile settling velocity for aerosol deposition in the MSIV leakage.

2. ESF Leak Rates

Note: The RADTRAD runs model ESF leakage beginning at 2 minutes, which is coincident with the start of the gap release.

The expected ESF leakage is 5.0 gpm, which is doubled and converted into cfm as follows:

5.0 gallon/min × 2 × 1/7.4805 ft^3/gallon = 1.337 cfm
10% of ESF leakage becomes airborne = 0.10 × 1.337 = 0.1337 cfm

3. External Cloud Gamma Dose Attenuation Factor

The gamma attenuation for concrete shielding for an external cloud dose is conservatively calculated for an average gamma energy of 1.0 Mev.

The gamma radiation external radioactive plume shine to CR personnel is attenuated by at least 2'-6" of concrete shielding. This evaluation resulted in a direct shield attenuation factor of 3.95E-04.

4. Drywell Wetted Surface Area

The drywell wetted surface area is not readily available. However, the Hope Creek Generating Station (HCGS) is also a BWR/4, Mark I containment and is a sister plant of PBAPS sharing the same Containment Data Specification 22A6209 (Reference 7.27). Therefore, having the same original licensed power level and size of reactor and drywell, the HCGS drywell surface area information can be conservatively used to determine the PBAPS drywell wetted surface area. The use of a smaller wetted surface is conservative because it results in a smaller elemental removal coefficient and takes a longer time to reach an elemental iodine decontamination factor (DF) of 200, which allows the elemental iodine to remain airborne in the drywell atmosphere for release to the atmosphere via containment and MSIV leakage.

The surface areas below the drywell spray ring are subject to be wetted in the spray solution, which are listed as follows:

Estimated 25% of drywell lining surface	= 4,463 ft^2 (17,850 ft^2 /4 = 4, 463 ft^2)
Downcomer (to water level)	= 3,168 ft^2
Vent header and line	= 9,727 ft^2
Suppression chamber (to water level)	= 15,408 ft^2
Estimated 50% of major equipment	= 5,306 ft^2 (10,612 ft^2 /2 = 5,306 ft^2)
Estimated 50% of structures	= 6,181 ft^2 (12,361 ft^2 /2 = 6,181 ft^2)
Total estimated wetted drywell surface area	= 44,253 ft^2
75% of total estimated wetted drywell surface area	≈ 33,200 ft^2 (0.75 x 44,253 ft^2)

The above calculated wetted surface is compared with the actual calculated wetted surface areas of other Exelon fleet plants as follows:

Parameter	Dresden Units 2 & 3	Quad Cities Units 1 & 2	Peach Bottom Units 2 & 3
Thermal Power Level (MW_t)	3016	3016	3528
Total Drywell Volume (ft^3)	278,000	269,000	286,700
Wetted Surface Area (ft^2)	32,250	32,430	33,200

The comparison in the above table shows that the calculated PBAPS wetted surface area is conservative for the relatively larger drywell at the PBAPS.

The drywell wetted surface area of 33,200 ft^2 is used to calculate the elemental iodine removal coefficient by wall deposition.

5. Containment Elemental Iodine Removal Coefficient

Natural deposition on containment surfaces (plateout) of the elemental iodine released to containment is calculated using the methodology outlined in NUREG-0800, Standard Review Plan 6.5.2. The elemental iodine removal coefficient for time = 0 to 2.0 hrs is 3.36 hr^{-1}. The removal coefficient used for greater than 2.0 hrs is 1.86 hr^{-1}. The maximum DF of elemental iodine is 200.

The containment leakage case is analyzed in RADTRAD using the calculated elemental iodine removal coefficients to determine the cutoff time for terminating elemental iodine removal from the containment atmosphere. The cutoff time of 3.85 hrs was determined and is used in the containment and MSIV leakage path releases to the atmosphere.

6. CR Containment Shine Dose

The shielding geometry parameters (source volume, line of sight distance, and intercepting concrete shielding) are calculated. The shielding information along with the time dependent post-LOCA containment isotopic activities are input to the MicroShield computer code to calculate the time dependent containment shine gamma dose rates and integrated dose.

Containment shine dose due to Post-LOCA activity in the RB is 0.067 rem. Containment shine dose due to 42% of Post-LOCA activity above the operating floor in the RB is 0.028 rem.

7. CR Filter Efficiencies

Technical Specification 5.5.7, Ventilation Filter Testing Program (VFTP), requires routine testing of safety related filtration systems.

The current TS test acceptance criteria for in-place penetration and system bypass of the ESF HEPA filters is < 1.0% when tested in accordance with Regulatory Guide 1.52, Revision 2 (Reference 7.28), Section 5c, and ASME N510-1989 (Reference 7.29) at the specified system flow rate in TS 5.5.7.a. However, this existing requirement is an exception to the RG 1.52 requirement of <0.05%. PBAPS proposes to not change this value, maintaining an exemption to RG 1.52, Revision 2, Section 5c. In order to justify this 1% bypass value, the analysis was performed with an efficiency value that is one percentage point less than that which could normally be credited by RG 1.52 (see below). This method has been previously approved for the Byron, Braidwood, and Catawba AST amendments (ADAMS Accession Nos. ML062340420 and ML052730312).

HEPA filter efficiency = 99% (based on RG 1.52) - 1% (TS 5.5.7 bypass) = 98% credited in the analysis.

Similarly, the current TS test acceptance criteria for in-place penetration and system bypass of the ESF charcoal adsorber is < 1.0% when tested in accordance with Regulatory Guide 1.52, Revision 2, Section 5d, and ASME N510-1989 at the specified system flow rate in TS 5.5.7.b. However, this existing requirement is an exception to the RG 1.52 requirement of <0.05%. PBAPS proposes to not change this value, maintaining an exemption to RG 1.52, Revision 2, Section 5d. In order to justify this 1% bypass value, the analysis was performed with an efficiency value that is one percentage point less than that which could normally be credited by RG 1.52 (see below). This method has been previously approved for the Byron, Braidwood, and Catawba AST amendments.

Charcoal adsorber efficiency = 90% (based on RG 1.52 section 5.d) - 1% (TS 5.5.7 bypass) = 89% credited in the analysis.

As required by TS 5.5.7.c, laboratory testing of the charcoal adsorber samples, when obtained as described in Regulatory Guide 1.52, Revision 2, Section 6b, requires the methyl iodide penetration to be less than 5% when tested in accordance with the laboratory testing criteria of ASTM D3803-1989 (Reference 7.30) at a temperature of 30 degrees C [86 degrees F], with face velocity and relative humidity as specified in TS 5.5.7.c. There are no RG 1.52 exceptions taken related to laboratory testing of charcoal.

The test acceptance criteria for laboratory samples with methyl iodide is determined as follows:

Penetration (%) = (100% - η)/safety factor = (100% - η)/2

Where η = MCREV charcoal filter efficiency to be credited in the analysis.

This results in a penetration value of 10%, equating to a credited efficiency of 90%.

Therefore, assuming a 1% system bypass flow, MCREV charcoal efficiency, η = 90% - 1% = 89% as used in the analysis. A 1% bypass flow can be equated to a reduction in overall efficiency (by one percentage point).

8. Post-LOCA MCREV Filter Shine Dose

The MCREV filter shine gamma dose rate resulting from MSIV leakage is calculated to be 4.037E-03 mRem/hr. This results in a post-LOCA 30-day CR filter shine dose of 1.28E-03 Rem.

The MSIV leakage CR filter shine dose is increased by a factor 1.50 to include the filter shine dose contribution from containment and ESF leakages and results in a total CR Filter Shine Dose of 1.92E-03 Rem.

4.3.24 Results Summary and Conclusions

The results of the Post-LOCA AST analyses for the proposed licensing basis are summarized in the following table:

Post-LOCA Activity Release Path	**Post-LOCA TEDE Dose (Rem) Receptor Location**		
	Control Room	**EAB**	**LPZ**
Containment Leakage	2.07E-01	2.60E+00 (Occurs @ 1.0 hr)	2.82E+00
ESF Leakage	4.60E-02	5.66E-01 (Occurs @ 2.0 hr)	3.31E+00
MSIV Leakage	4.36E+00	3.48E+00 (Occurs @ 4.6 hr)	9.94E-01
Containment Shine	2.80E-02	Negligible	Negligible
External Cloud	1.16E-02	Negligible	Negligible
CR Filter Shine	1.92E-03	Negligible	Negligible
Total	**4.66E+00**	**6.65E+00**	**7.13E+00**
Allowable TEDE Limit	**5.00E+00**	**2.50E+01**	**2.50E+01**

4.3.25 Conclusions – LOCA

The results of this analysis, using conservative as-built design inputs and assumptions that reflect the proposed AST implementation indicate that the EAB, LPZ, and CR doses are within their allowable TEDE limits.

Table 4.3-1
Peach Bottom Atomic Power Station
AST Design Inputs Used in the LOCA Analysis

Design Input Parameter	Value Assigned	Comments
Containment Leakage Model Parameters		
Source Term		
Thermal Power Level	3,528 MWt (includes 0.38% margin relative to rated thermal power of 3,514 MWt)	Unchanged from currently licensed value
Extended Cycle Fuel Burnup	37.7 GWD/MTU	Unchanged from currently licensed value

Isotopic Core Inventory (Ci/MWt) (New values calculated using ORIGEN 2.1 for AST)

Isotope	Ci/MW$_t$	Isotope	Ci/MW$_t$	Isotope	Ci/MW$_t$
CO-58*	1.529E+02	RU-103	4.202E+04	CS-136	2.027E+03
CO-60*	1.830E+02	RU-105	2.908E+04	CS-137	4.538E+03
KR-85	3.946E+02	RU-106	1.730E+04	BA-139	5.084E+04
KR-85M	8.313E+03	RH-105	2.752E+04	BA-140	4.896E+04
KR-87	1.633E+04	SB-127	2.896E+03	LA-140	5.019E+04
KR-88	2.303E+04	SB-129	8.638E+03	LA-141	4.640E+04
RB-86	6.518E+01	TE-127	2.873E+03	LA-142	4.532E+04
SR-89	2.798E+04	TE-127M	3.855E+02	CE-141	4.492E+04
SR-90	3.178E+03	TE-129	8.501E+03	CE-143	4.427E+04
SR-91	3.801E+04	TE-129M	1.267E+03	CE-144	3.596E+04
SR-92	4.017E+04	TE-131M	3.869E+03	PR-143	4.293E+04
Y-90	3.272E+03	TE-132	3.821E+04	ND-147	1.838E+04
Y-91	3.448E+04	I-131	2.687E+04	NP-239	5.397E+05
Y-92	4.029E+04	I-132	3.881E+04	PU-238	1.796E+02
Y-93	4.526E+04	I-133	5.556E+04	PU-239	1.200E+01
ZR-95	4.489E+04	I-134	6.165E+04	PU-240	1.288E+01
ZR-97	4.657E+04	I-135	5.192E+04	PU-241	6.182E+03
NB-95	4.512E+04	XE-133	5.491E+04	AM-241	9.528E+00
MO-99	5.078E+04	XE-135	2.228E+04	CM-242	2.388E+03
TC-99M	4.447E+04	CS-134	7.280E+03	CM-244	2.602E+02

* CO-58 & CO-60 activities are obtained from RADTRAD User's Manual, Table 1.4.3.2-3 (NUREG/CR-6604)

Radionuclide Composition

Group	Elements	
Noble Gases	Xe, Kr	New assumption per RG 1.183, Section 3.4, Table 5
Halogens	I, Br	
Alkali Metals	Cs, Rb	
Tellurium Group	Te, Sb, Se	
Barium, Strontium	Ba, Sr	
Noble Metals	Ru, Rh, Pd, Mo, Tc, Co	

Design Input Parameter	Value Assigned	Comments
Lanthanides	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am	
Cerium	Ce, Pu, Np	
Timing of Release Phase (New Assumption per RG 1.183, Section 3.3, Table 4)		
Phase	**Onset**	**Duration**
Gap Release	2 min	0.5 hr
Early In-Vessel Release	0.5 hr	1.5 hr
Release Fractions (New Assumption per RG 1.183, Section 3.2, Table 1)		
BWR Core Inventory Fraction Released Into Containment (New Assumption per RG 1.183, Section 3.2, Table 1)		
Group	**Gap Release Phase**	**Early In-Vessel Release Phase**
Noble Gases	0.05	0.95
Halogens	0.05	0.25
Alkali Metals	0.05	0.20
Tellurium Metals	0.00	0.05
Ba, Sr	0.00	0.02
Noble Metals	0.00	0.0025
Cerium Group	0.00	0.0005
Lanthanides	0.00	0.0002
Iodine Chemical Form Released to the Containment		
Aerosol (CsI)	95%	New assumption per RG 1.183, Sections 3.5 and A.2
Elemental	4.85%	
Organic	0.15%	
Post-LOCA Drywell Pressure	49.1 psig	Unchanged from currently licensed value
Post-LOCA Drywell Temperature	280^0F	Reference 7.36, Section 8.3.4 Unchanged from currently licensed value
Activity Transport in Primary Containment		
Minimum Drywell Air Volume	159,000 ft^3	UFSAR Table 5.2.1 Unchanged from currently licensed value
Minimum Suppression Chamber Free Air Volume	127,700 ft^3	UFSAR Table 5.2.1 Unchanged from currently licensed value
Drywell plus Suppression Chamber Free Air Volume	286,700 ft^3 (159,000 ft^3 + 127,700 ft^3)	Unchanged from currently licensed value
Containment Elemental Iodine Removal Model	Standard Review Plan 6.5.2	New assumption per SRP 6.5.2, Page 6.5.2-10
Drywell Surface Area for Deposition/Plateout Model	33,200 ft^2	New assumption per Reference 7.37, Section 7.7
Particulate (Aerosol) Deposition/Plateout Model	Powers' 10 percentile model	New assumption per NUREG/CR-6604

Design Input Parameter	Value Assigned	Comments
Containment Leak Rate into Reactor Building	0.700 w%/day for 2 min to 38 hrs 0.350 w%/day for > 38 hrs	New requirement per TS 5.5.12 as justified in Reference 7.37.
Containment Drawdown Time	≤ 180 seconds (3 minutes)	New assumption as justified in Reference 7.37
SGT System Flow Rate	10,500 cfm	TS SR 3.6.4.1.4 Unchanged from currently licensed value
Reactor Building Volume	2,500,000 ft^3	Unchanged from currently licensed value
ESF Leakage Model Parameters		
Minimum Suppression Pool Water Volume	122,900 ft^3	Unchanged from currently licensed value
Sump Water Activity (New Assumption per RG 1.183, Sections A.5.1, A.5.3 & Tables 1 & 4)		
Group	**Gap Release Phase**	**Early In-Vessel Release Phase**
Timing Duration (Hrs)	2 min – 0.50 Hr (Conservatively earlier than actual end time of 0.52 hours)	0.50 – 2.0 Hr
Halogen	0.05	0.25
ESF Leakage Rate	10.0 gal/min (= 2 × 5.0 gal/min expected leakage rate)	New assumption to establish a new design basis (as used in Reference 7.37) and applied per RG 1.183, Section A.5.2
ESF Leakage Initiation Time and Duration	0 to 30 days	New conservative assumption used in Reference 7.37
Suppression Pool Scrubbing	Not credited	RG 1.183, Section A.3.5
Long-Term Suppression Pool Water pH	> 7.0	New Calculated value PM-1056, Rev. 0 (Reference 7.33), page 11 and 9.1, Section A.2
ESF Leakage Maximum Temperature	< 212^{0}F	Unchanged from currently licensed value
Fraction of Iodine in ESF Leakage that becomes Airborne	0.10	New assumption per RG 1.183 and Torus water temperature < 212^{0}F
Chemical Form of Iodine in ESF Leakage		
Elemental	97%	New assumption per RG 1.183, Section A.5.6
Organic	3%	
MSIV Leakage Model Parameters		
Total MSIV Leak Rate Through All Four Lines	360 scfh for < 38 hrs @ 49.1 psig (180 scfh for > 38 hrs)	New assumption justified in Reference 7.37
MSIV Leak Rate Through One Line With MSIV Failed	205 scfh for < 38 hrs @ 49.1 psig (102.5 scfh for >38 hrs)	New assumption justified in Reference 7.37 - maximum leakage rate through any one line
MSIV Leak Rate Through Three Intact Lines		

Design Input Parameter	Value Assigned	Comments
First Intact Line	155 scfh for < 38 hrs @ 49.1 psig (77.5 scfh for > 38 hrs)	New assumption justified in Reference 7.37 - remainder of unallocated leakage
Second Intact Line	0 scfh for < 30 days @ 49.1 psig	New assumption justified in Reference 7.37 - remainder of unallocated leakage
Third Intact Line	0 scfh for < 30 days @ 49.1 psig	New assumption justified in Reference 7.37 - remainder of unallocated leakage
Natural Removal Efficiency For Elemental Iodine In Each Steam Line Volume	50 percent	New assumption justified in Reference 7.37 per AEB 98-03, Appendix B, page B-3
Control Room Model Parameters		
CR Envelope Pressure Boundary Free Volume	176,000 ft^3	Unchanged from currently licensed value
MCREV Filtration System Actuation Time Following a LOCA	30 minutes	New conservative assumption used in Reference 7.37
CR Emergency Ventilation Mode Air Intake Rate	3,000 cfm ± 10% 2,700 cfm	Unchanged from currently licensed value (TS 5.5.7) Conservatively Modeled
CR Unfiltered Inleakage during Normal Operation (< 0.5 hr)	18,500 cfm (includes ingress/egress inleakage of 10 cfm)	New conservative assumption used in Reference 7.37. Based on Parametric Study (see Attachment 1, Figure 1).
CR Unfiltered Inleakage during Emergency Ventilation Mode (> 0.5 hr)	369 cfm by tracer gas testing 500 cfm (includes ingress/egress inleakage of 10 cfm)	NCS Corporation Report (Reference 7.38), Control Room Envelope Inleakage Testing At Peach Bottom Atomic Power Station, 2004, Table 16 New conservative assumption used in Reference 7.37.
CR Emergency Ventilation Mode Intake Charcoal and HEPA Filter Efficiencies		
Elemental Iodine	89%	New conservative assumption used in Reference 7.37, Section 7.11.
Organic Iodide	89%	
Particulate Aerosols	98%	
CR χ/Qs For Containment & ESF Leakage Release Via Off-Gas Stack Release		
Time	**X/Q (sec/m^3)**	
0-2	2.72E-06	Calculation PM-1055, Rev. 0 (Reference 7.32), Table 4-1 24-96 hrs χ/Q value is conservatively used for 2-24 hrs χ/Q values
2-8	1.46E-08	
8-24	1.46E-08	
24-96	1.46E-08	
96-720	4.21E-09	

Design Input Parameter	Value Assigned	Comments
CR X/Qs For MSIV Leakage Release Via Unit 2 TB/RB Exhaust Vent		
Time	**X/Q (sec/m^3)**	
0-2	1.18E-03	Calculation PM-1055, Rev. 0 (Reference 7.32), Table 4-1 for Unit 2 RB Stack χ/Q values, which are conservative for Unit 3
2-8	9.08E-04	
8-24	4.14E-04	
24-96	2.90E-04	
96-720	2.26E-04	
CR Occupancy Factors and Breathing Rate		
Time (Hr)	**%**	
0-24	100	No change from currently used (per RG 1.183, Section 4.2.6)
24-96	60	
96-720	40	
CR Breathing Rate	3.5E-04 m^3/sec	New value per RG 1.183, Section 4.2.6
Offsite Dose Receptor Release Model Parameters		
EAB X/Qs For Containment & ESF Leakage Release Via Off-Gas Stack Release		
Time (hrs)	**X/Q (sec/m^3)**	
0-0.5	5.30E-05	Calculation PM-1055, Rev. 0 Reference 7.32), Table 4-1
0.5-2	8.89E-06	
2-720	8.89E-06	New values calculated in Reference 7.32 and RG 1.183, Section 4.1.5 (0.5-2 hr χ/Q value conservatively modeled after 2 hours)
EAB X/Q For MSIV Leakage Release		
Time (hrs)	**X/Q (sec/m^3)**	
0-2	4.25E-04	Calculation PM-1055, Rev. 0 (Reference 7.32), Table 4-1
2-720	4.25E-04	RG 1.183, Section 4.1.5 (0-2 hr χ/Q value conservatively modeled after 2 hours)
EAB Breathing Rate	3.5E-04 m^3/sec	RG 1.183, Sections 4.1.3 & 4.1.5
LPZ X/Qs For Containment & ESF Leakage Release Via Off-Gas Stack Release		
Time (hrs)	**X/Q (sec/m^3)**	
0-0.5	1.75E-05	New calculated values from Calculation PM-1055, Rev. 0 (Reference 7.32), Table 4-1
0.5-2	8.87E-06	
2-8	3.94E-06	
8-24	2.62E-06	
24-96	1.09E-06	

Design Input Parameter	Value Assigned	Comments
96-720	3.06E-07	
LPZ X/Qs For MSIV Leakage Release Via Unit 2 TB/RB Exhaust Vent		
Time (hrs)	**X/Q (sec/m^3)**	
0-2	4.81E-05	New calculated values from Calculation PM-1055, Rev. 0 (Reference 7.32), Table 4-1
2-8	2.08E-05	
8-24	1.37E-05	
24-96	5.49E-06	
96-720	1.49E-06	
LPZ Breathing Rates		
Time (hrs)	**BR (m^3/sec)**	
0-8	3.5E-04	New assumption per RG 1.183, Sections 4.1.3 and 4.4
8-24	1.8E-04	
24-720	2.3E-04	

4.4 FUEL HANDLING ACCIDENT

4.4.1 Background

The Alternative Source Term (AST) methodology is being applied to the analysis of the design basis Fuel Handling Accident (FHA) for Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3. This analysis considers normal (unfiltered) exhaust through the building ventilation stack and other openings. This analysis supports changes to the current PBAPS, Units 2 and 3, Technical Specifications regarding the operability of Standby Gas Treatment (SGT) system and other systems previously required to mitigate the radiological consequences of fuel handling accidents.

Not having to consider secondary containment integrity and filtration requirements of SGT in support of refueling activities has the potential to significantly improve the flexibility and duration of scheduled plant outage activities.

Various release points based on the $^{\chi}/_{Q}$ values for all potential openings in secondary containment are evaluated for various decay periods before fuel movement is assumed to take place. Plant walk-downs and extensive drawing reviews have been completed to ensure that the most restrictive release locations have been identified and included in this evaluation. Post-shutdown decay periods are included in this evaluation that would support changes to Technical Specifications for the relaxation of secondary containment integrity.

Guidance in TSTF-51 suggests that "recently irradiated fuel" parameters be developed to identify the point in time after shutdown when secondary containment features are not required. Therefore, this evaluation supports the proposed definition of RECENTLY IRRADIATED FUEL as being:

RECENTLY IRRADIATED FUEL is fuel that has occupied part of a critical reactor core within the previous 84 hours. This 84-hour time period may be reduced to 24 hours if all secondary containment ground-level hatches (hatches H15 through H24 and Torus room access hatches) are closed.

Analyses of radiation transport and dose assessment are performed using RADTRAD v. 3.03. RADTRAD is a simplified model of RADionuclide Transport and Removal And Dose Estimation developed for the NRC and endorsed by the NRC as an acceptable methodology for reanalysis of the radiological consequences of design basis accidents. The technical basis for the RADTRAD code is documented in Section 2 of NUREG/CR-6604 (Reference 7.6). The methodologies significant to this analysis are the dose consequence analysis (NUREG/CR-6604, Section 2.3) and the Radioactive Decay Calculations (NUREG/CR-6604, Section 2.4).

4.4.2 Accident Source Term - FHA

The source term nuclide inventory used for the Fuel Handling Accident (FHA) is the same as that used for the LOCA.

Regulatory Guidance for DBA fuel handling accidents is such that 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 are applied in determining the inventory of the damaged rods.

For PBAPS, the fraction of the core fuel damaged is based on the GESTAR II (Reference 7.41) limiting case of damaging 172 fuel pins (based on a "Heavy Mast" design; i.e., the "NF500 mast") from GE12 or GE14 10x10 fuel bundle arrays with the analyzed equivalent of 87.33 pins per bundle, and with all of the damaged fuel assumed to have a limiting Peaking Factor (PF) of 1.7. This analysis is for an assembly and mast drop from the maximum height allowed by the refueling platform over the reactor well onto fuel in the reactor, and bounds all locations in terms of fuel damage potential. PBAPS, Units 2 and 3, core loads consist of all GE14 10x10 fuel assemblies, and therefore, the results of the GESTAR II limiting case are consistent with PBAPS core loading.

Bundle Type	Fuel Array	Pins in Bundle	Failed Pins	Damaged Core Fraction Assuming Core is All Specified Bundle Type	PF	Damaged Core Fraction with PF
GE12&GE14	10x10	87.33	172	0.002578	1.7	0.004382

With a 1.7 radial peaking factor, the associated power of the damaged fuel = 3528 MWth * 0.002578 * 1.7 = 15.46 MWth.

No adjustment to the fission product inventory is made for events postulated to occur during power operations at less than full rated power or those postulated to occur at the beginning of core life.

Because of radioactivity decay, the worst-case fuel handling accident is that associated with handling fuel that has most recently been part of a critical core operating at full power. In accordance with RG 1.25 based fuel handling accident analysis, movement of irradiated fuel will not occur less than 24 hours after the associated reactor shutdown, and therefore, a 24-hour delay period is the minimum decay period assumed. This value continues to be a very conservative assumption for BWRs, given the operations necessary before commencing fuel movement.

Radioactive decay from the time of shutdown is modeled for periods of 24 hours and 84 hours, to support the opening of the most restrictive secondary containment potential release points. Ground-level hatches that are located outside in proximity to the MCREV intake have been determined to be the most restrictive openings for potential radioactive release. With the exception of these ground-level hatches (hatches H15 through H24 and Torus room access hatches), all other openings assumed to be potential release points can be opened after the conservative 24-hour decay period.

4.4.3 Transport - FHA

This evaluation is applicable to fuel whose burnup and power limits are bounded by those specified in RG 1.183, footnote 11. This allows application of the gap activity fractions for LOCA events per RG 1.183, Table 3, which are as follows:

- 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

An assessment of water coverage and effective DFs considering FHAs over the reactor well and the spent fuel pool was performed. It identified that the drop over the reactor well is more limiting. Water coverage over the reactor well is approximately 52 feet. However, no additional credit is taken for water depth greater than 23 feet in order to maintain conservatism and consistency with regulatory guidance.

For a drop over the spent fuel pool, coverage over the dropped assemblies and the plenum in struck assemblies are slightly less than 23 feet. However, coverage is sufficient to maintain a 200 DF based on the Regulatory Guide 1.183 Appendix C recommended 500 DF for inorganic iodines, and the recommended inorganic/organic iodine ratios. Even if a lower inorganic iodine DF is selected to force an overall 200 DF with 23 feet of pool coverage, the reduction in DF is offset by reductions in fuel damage due to a much shorter drop (i.e., less than approximately 4 feet) over fuel in the spent fuel pool.

Release modeling uses the RADTRAD computer program. The compartments are the Refuel Floor Air Space (in the Reactor Building), the Environment (EAB and LPZ), and the Control Room. The refuel floor exhaust rate is set artificially high at 6 air changes per hour. This results in 99.9999% of the contained radioactivity being exhausted within two hours. The exhaust point that is postulated under the no filtration condition is the TB/RB Ventilation Stack. This particular

release point results in specific dispersion characteristics which are defined by unique dispersion factors, or χ/Q's.

4.4.4 Dispersion Model

Dispersion factors at given increments in time for the TB/RB Ventilation Stack and ground-level hatch release points are calculated. The χ/Q's are based on RG 1.194 methodology as implemented by ARCON96 for onsite locations (Control Room) and on the RG 1.145 methodology as implemented by PAVAN for offsite locations (EAB and LPZ).

4.4.5 Control Room Model

The Control Room, as analyzed for the TB/RB Ventilation Stack release FHA analysis cases, is unfiltered. Normal intake flow rates, plus an assumed unfiltered inleakage flow, are used with a corresponding exhaust rate on a once-through basis.

An additional case was analyzed for the release pathway through the worst-case hatch location, with the Control Room filter in operation at the worst case (3000 – 10% = 2700 cfm, see Table 4.4-2) and an additional assumed 500 cfm of unfiltered inleakage. This case is assumed to be the most limiting, and therefore no changes to the MCREV TS are being requested as part of this LAR.

4.4.6 Dose Acceptance Criteria - FHA

Dose acceptance criteria are per 10CFR50.67 and RG 1.183 guidance.

Table 4.4-1 lists the regulatory limits for accidental dose to 1) a control room operator, 2) a person at the EAB, and 3) a person at the LPZ boundary.

Table 4.4-1
Regulatory Dose Limits (Rem TEDE)

CR (30 days)	EAB (2 hours)	LPZ (30 days)
5	6.3	6.3

4.4.7 Design Inputs

The design inputs used for this evaluation were extracted from extensive review of PBAPS, Units 2 and 3, Licensing documents, existing calculations, and regulatory guidance documents. These parameters are summarized in Table 4.4-2 below:

Table 4.4-2
Parameters Applicable to AST Fuel Handling Accident Dose Considerations for Peach Bottom Atomic Power Station

Parameter or Method	AST Value	Comments
Reactor Power	3528 MWth	No change. Includes 0.38% margin for instrument uncertainty relative to the rated thermal power of 3,514 MWt after the 1.62% thermal power optimization uprate.
Fuel Assembly Configuration and properties	10 x 10 in an 87.33 fuel pin bundle and 172 pins damaged	Bounding assumptions for current PBAPS licensing basis FHA. Movement of irradiated fuel will not occur less than 24 hours after the associated reactor shutdown to support opening of secondary containment, with the exception of the ground-level hatches (hatches H15 through H24 and Torus access room hatches) that require 84 hours of decay before opening.
Radial Peaking Factor	1.7	New conservative bounding assumption
Allowable Fuel Burnup and non-LOCA gap fractions	RG 1.183, Table 3. Fuel bundle peak burnup will not exceed 62 GWD/MTU. For fuel exceeding 54 GWD/MTU, the maximum linear heat generation rate will not exceed 6.3 kW/ft.	The current design basis bounding fuel damage assessment scenario, which is associated with a drop over the reactor core, is used and meets the requirements of RG 1.183 Footnote 11.
FHA Radionuclide Inventory	From the 60 isotopes forming the standard RADTRAD library, with decay to 24 hours. Gap activities are per R.G. 1.183.	New ORIGEN 2.1 values calculated. Spent fuel source terms are based on the same bounding reactor core source terms as were used for the LOCA analysis.
Underwater Decontamination Factor	Noble Gases: 1 Particulate (cesiums and rubidiums): infinity Iodine: 200, corresponding to a 23-ft water depth	For conservatism, the effective minimum depth of 23 feet is assumed to be the water coverage over the reactor core. This is the worst-case location for a fuel drop FHA to take place (significantly more damage than a drop in the fuel pool).
Iodine chemical distribution	95% CsI, instantaneously dissociating in the pool water and re-evolving as elemental iodine. Since the pH of the pool water is not maintained	New assumption per RG 1.183.

Parameter or Method	AST Value	Comments
	above 7, iodine is assumed to be 4.85% elemental and 0.15% organic.	
Activity Transport to the Environment	Activity reaching the refuel floor airspace will essentially be all exhausted within 2 hours by using an artificially high exhaust rate. This also provides an allowance for uneven mixing in the refuel floor airspace.	New assumption per RG 1.183.
Release Pathways	The release pathways are through TB/RB Ventilation Stack or through worst-case potentially open ground-level hatch location (hatches H15 through H24 and Torus room access hatches). No credit is taken for filtration by the SGTS, or the elevated release resulting from exhaust through the PBAPS Main Stack.	New conservative assumptions.
Dose Conversion Factors	EPA Federal Guidance Reports 11 and 12	New assumptions per RG 1.183
Offsite Dose Limit	6.3 rem TEDE	New values per 10CFR50.67 and RG 1.183
Control Room Dose Limit	5 rem TEDE for the duration of the accident	Per 10CFR50 App. A, GDC 19 and 10CFR50.67
CR Volume	1600 cfm Volume 176,000 ft^3	
MCREV Operation for release from ground hatch	3000 – 10% cfm filtered intake (sensitivity analyses show the lower bound of the 10% flow uncertainty is conservative) plus an allowance of 500 cfm for unfiltered inleakage. MCREV operability is required in order to account for the most limiting for grade level hatch plugs that could be opened during movement of irradiated fuel, as long as it has been at	New conservative assumption used in Reference 7.31.
Refuel Floor Normal Ventilation rate and volume	6 air changes per hour with an artificial value of 100 ft^3 is used for simplicity. This evacuates 99.9999% of all activity within 2-hours.	New conservative assumption used in Reference 7.31.

Parameter or Method	AST Value	Comments
CR Potential Release Points and Limiting χ/Qs (0 – 2 hr) Personnel Access Doors Railroad Bay Doors RB Roof Scuttle Ground Level Hatch 4	 1.04E-03 sec/m^3 (Unit 3) 4.54e-04 sec/m^3 (Unit 2) 1.90E-03 sec/m^3 (Unit 2) 1.28E-02 sec/m^3 (Unit 3)	New calculated values from Reference 7.31.
EAB Release Point Basis and Distance to EAB Limiting Dispersion Factors 0 – 2 hr	 Normal RB exhaust stack and 1040 m (considered as applicable to all release locations) 4.25E-04 sec/m^3	Conservative assumption No change New calculated value from Reference 7.31
LPZ Release Point Basis and Distance to LPZ Limiting Dispersion Factors 0 – 8 hr	 Normal RB exhaust stack and 7300 m (considered as applicable to all release locations) 4.81E-05 sec/m^3	Conservative assumption No change New calculated value from Reference 7.31

4.4.8 Summary of Results - FHA

The RADTRAD code was used to examine the effect of the alternative source term release on offsite and CR doses. Shown below are the results.

Location	Dose (Rem TEDE)
LIMITS	**CR = 5.0; EAB & LPZ = 6.3**
EAB	**1.16**
LPZ	**0.132**
CR	**2.35**

Case 1: Dose Limits and Calculated Doses for TB/RB Stack Release Pathway
Decay Time = 24 Hours and No MCREV Operation, for conservatism

The following release paths assume neither SGTS filtration nor elevated release through the main stack, but with releases through various worst-case secondary containment building opening pathways, which could be open during the movement of the recently irradiated fuel (with decay greater than 24 hours) during an outage. The worst-case χ/Q values are listed for these release points in Table 4.4-2 above. For the given post-FHA source term and CR response, the resulting CR dose is directly proportional to the dilution of source term resulting

from the atmospheric dispersion factors (χ/Q_s). The review of the χ/Q values for the releases from the openings, which could be open during the refueling outage in Table 4.4-2 indicates that the χ/Q for the worst case ground-level hatch # 4 release is most limiting of all releases. The post-FHA doses for the two most limiting releases (RB Roof Scuttle and any outside ground-level hatch # 4) are shown in the following table.

Dose limits and Calculated Doses for Worst-Case Secondary Containment Building Opening Release Pathways for Assumed Decay Periods

Case	Most Limiting Release Point	Assumed Decay Time (Hours)	MCREV Credit	EAB Dose (Rem TEDE) [Limit = 6.3]	LPZ Dose (Rem TEDE) [Limit = 6.3]	CR Dose (Rem TEDE) [Limit = 5.0]
2	RB Roof Scuttle	24	No (for conservatism)	1.16	0.132	3.85
3	Limiting Grade Level Hatch	84 (3.5 d)	Yes	0.714	0.081	4.56

Case 2 results support Technical Specifications changes that include the definition of recently irradiated fuel, and relaxation of technical specifications of secondary containment integrity during fuel handling operations 24 hours after shutdown. Case 3 is included as the bounding case that would allow ground-level hatches (H15 through H24) to the west of the Reactor/Radwaste Buildings to be opened, with a minimum decay time of 84 hours after shutdown and MCREV operable.

4.4.9 Conclusions

1. For postulated releases through reactor building roof penetrations, personnel and equipment access doors, and railroad bay doors:

    a. Movement of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core in the previous 24 hours) can safely be accomplished without secondary containment integrity.

2. For postulated releases through grade level Torus room hatch openings (i.e., torus room concrete plug penetrations), movement of irradiated fuel (i.e., fuel that has occupied part of a critical reactor core in the previous 24 hours) cannot be accomplished without exceeding the CR dose limits specified in 10 CFR 50.67. Therefore, in order to keep these grade level hatch plugs open to allow fuel movement, the following restriction must be made:

    a. The grade level hatch plugs should not be opened until 84 hours (3.5 days) after reactor shutdown.

    The FHA analyses for releases through various RB openings do not include the MCREV delay of 30 minutes and reduced conservative MCREV charcoal and HEPA filter efficiencies due to the bypass flows. The 1% reduction in MCREV filter efficiency due to the existing bypass has minimal effect on CR dose.

4.5 CONTROL ROD DROP ACCIDENT (CRDA)

4.5.1 Objective

The radiological consequences of a Control Rod Drop Accident (CRDA) are based on the use of Alternative Source Terms (AST) as defined in RG 1.183. The design basis CRDA results in the release of radioactivity to the Condenser.

4.5.2 Background

An isolated Condenser is assumed to exhaust at a rate of 1% per day. However, during operating conditions there are forced flow paths from the turbine / condenser. For instance, the CRDA can occur during mechanical vacuum pump (MVP) operation, which can exhaust unprocessed non-condensable material from the condenser at a significantly larger rate.

A second forced flow path from the condenser is associated with maintaining condenser vacuum using the Steam Jet Air Ejectors (SJAE). The Peach Bottom Atomic Power Station (PBAPS) off-gas system provides SJAE flow processing which would eliminate iodine releases and greatly delay noble gas releases allowing for decay even with normal off-gas flow rates. This pathway is addressed in this calculation.

Thirdly, under normal operation, steam that contains activity is released to the Gland Sealing Steam System. This release pathway is considered, and subsequently ruled out, in this calculation as well.

The Main Steam Line Radiation Monitors (MSLRMs) provide an isolation function for the MSIVs. This prevents any of the aforementioned forced flow paths from facilitating activity release following a CRDA. The maximum set-points on which the MSLRMs trip assure that the regulatory limits for use of AST are not exceeded in the Control Room (CR), Exclusion Area Boundary (EAB), and Low Population Zone (LPZ).

4.5.3 Methodology and Acceptance Criteria - CRDA

Following a CRDA, radioisotopes postulated to be released will be transported through the Main Steam Lines (MSLs) directly to the Main Steam Condenser. From there, for a CRDA assumed to occur during MVP operation, it is expected that the MVP action, and all other forced flow paths, will automatically cease due to a trip of the MSLRM, which results from high radiation levels. This ensures that the only significant activity release will be from Condenser leakage. The Condenser is assumed to leak into the Turbine Building (TB) at a rate of 1% per day, and subsequently be released to the environment through one of two Turbine Building exhaust stacks, without filtration, and at that same rate. The dispersion that is modeled for this release pathway is defined by the derived χ/Q's. The doses from either accident scenario are not expected to exceed the acceptance criteria of the applicable regulatory guidance. PBAPS Design Analysis PM-1057 (Reference 7.34) contains additional information.

4.5.4 Core Source Term

For conservatism, the CRDA core source terms are those associated with a DBA power level of 3528 MWth.

4.5.5 Fuel Damage Assessment - CRDA

The current design basis for fuel damage from a CRDA is based on PBAPS 10x10 fuel in an 87.33 equivalent fuel pin array.

There are 1,200 fuel rods breached with melting in 0.77% of the fuel contained in the breached rods. A conservative radial peaking factor of 1.7 is used. Of the 0.77% of the fuel that melts during the CRDA, 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 addition to noble gas and iodine releases, releases of 12% of the core inventory of Cesium (an alkali metal, per Table 5 in Regulatory Position 3 of RG 1.183) is assumed, based on Table 3 in Regulatory Position 3 of RG 1.183. Radionuclide grouping is per Table 5 in Regulatory Position 3 of RG 1.183, as implemented in RADTRAD.

4.5.6 Radioactivity Transport

Release and transport fractions are per RG 1.183 Table 3 and its associated Appendix C.

4.5.7 Release Pathways

1. The Main Condenser is assumed to leak activity into the Turbine Building at a rate of 1% per day. This activity is then released, unfiltered, to the environment by way of the RB/TB Exhaust Ventilation Stacks, taking no credit for holdup in the TB.

2. When in operation, the Steam Jet Air Ejector's discharge is to the augmented off-gas system. Upon detection of high radiation levels by the MSLRM, the MSIVs are isolated and the SJAEs are shutdown; therefore, this forced release path need not be considered.

3. Sealing Steam System: As in the case of the SJAE, forced flow from a sealing steam exhauster is stopped following automatic isolation of the MSIVs. The CRDA is postulated to release available radioactivity to balance-of-plant systems before MSIV isolation. MSIV isolation eliminates available driving steam. Even if the steam seal exhauster continues to operate, the steam piping downstream of the MSIVs will be quickly depressurized. The approximately 1400 feet of 24 inch piping between the plant and the Main Stack will effectively contain the radioactivity. Therefore, this component is treated as an extension of the turbine / condenser volume that is assumed to leak at 1% per day.

4. Mechanical Vacuum Pump: The operation of the Mechanical Vacuum Pump as well as forced flow from it is ceased by trips initiated upon detection of high radiation levels by the MSLRM. Therefore, any activity in this system is held up in the condenser, and this forced release path need not be considered.

4.5.8 Dose Conversion Factors

The Dose Conversion Factors (DCFs) from the U.S. Federal Guidance Report 11 and 12 are used for this analysis. The RADTRAD code inputs these values directly from its internal database, and when used in the calculation.

4.5.9 Control Room Dose Model - CRDA

For this analysis, as performed using the RADTRAD code, the Peach Bottom, Units 2 and 3, Control Room is modeled as a closed volume of 176,000 ft^3. Normal maximum flow into the CR is used with an appropriate allowance for tolerance and unfiltered in-leakage. No credit is taken for any filtration of flows into the CR.

The air that enters the CR originates from a source that is characterized by a dispersion factor, calculated using ARCON96 (Reference 7.9). Following a CRDA, the MVP and all other force flow paths are immediately de-energized, isolating the MSIVs. The remaining activity, which is assumed to have all accumulated in the Condenser, leaks into the Turbine Building at a rate of 1% per day. The subsequent release into the environment from the Turbine Building is postulated to escape through the worst of two RB / TB Ventilation Exhaust Stacks. The total dose in the Control Room over the 24-hour period is the result of the released activities that enter through the air intake. The methodologies significant to this analysis are the dose consequence analysis in NUREG/CR-6604, Section 2.3, and the Radioactive Decay Calculations.

4.5.10 EAB and LPZ Dose Model

The EAB and LPZ χ/Q's have been determined, and are located, respectively, 1040 m and 7300 m from the postulated release points. Having determined these dispersion factors, the total dose is modeled in RADTRAD 3.03 using the same nodal breakdown as used in determining the CR total dose.

4.5.11 Acceptance Criteria - CRDA

Radiological doses resulting from a design basis CRDA for the control room operator and a person located at the EAB or LPZ are to be less than the regulatory dose limits as given in Table 4.5-1.

Table 4.5.1

Regulatory Dose Limits

	Control Room	EAB and LPZ
Rem TEDE	5[a]	6.3[b]

Notes:
[a] 10 CFR 50.67
[b] SRP 15.0.1 (Reference 7.2, RG 1.183)

4.5.12 Assumptions - CRDA

1. Core inventory was based on a DBA power level of 3528 MWth to account for uncertainty in the Rated Thermal Power Level of 3514 MWth.

2. An average power peaking factor of 1.7 per pin was assumed. 10% of the core inventory of noble gases and iodines are released from the fuel gap. Release fractions of other nuclide groups contained in the fuel gap are detailed in Table 3 of RG 1.183.

3. 0.77% of the fuel will melt during the CRDA. 100% of noble gases and 50% of the iodines contained in the melted fuel fraction are assumed to be released to the reactor coolant. Fractions of other nuclides released from the melted fuel are used from Table 1 of RG 1.183.

4. The activity released from the fuel from either the gap or from fuel pellets is assumed to instantaneously mix with the reactor coolant within the pressure vessel.

5. 100% of all noble gases, 10% of the iodines, and 1% of remaining nuclides are transported to the turbine / condenser.

6. Of the activity that reaches the turbine and condenser, 100% of the noble gases, 10% of the iodine, and 1% of the particulate nuclides are available for release to the environment.

7. The MVP, SJAE, and off gas systems are all immediately shutdown due to the automatic MSIV isolation function of the MSLRM caused by the high radiation levels following a CRDA.

8. Once all forced flow paths are automatically disabled, all leakage from the main steam turbine condenser leaks to the atmosphere from the RB/TB Ventilation Exhaust Stack at a rate of 1% per day, for a period of 24 hours. This becomes the only release of concern to this design basis accident.

9. The control room occupancy factor is 1 because the duration of this analysis of the CRDA is 24 hours.

4.5.13 Design Input - CRDA

4.5.13.1 Atmospheric Dispersion

The CR χ/Q values input to RADTRAD were taken from the ARCON96 results of the design calculation (PM-1055) in Reference 7.32. The χ/Q's calculated by ARCON96 are calculated from the worst-case RB/TB Exhaust Ventilation Stack release to the Control Room normal fresh air intake.

The CR atmospheric relative concentrations used are as follows:

χ/Q =1.18E-03 sec/m^3 (0-2 hours)
χ/Q =9.08E-04 sec/m^3 (2-8 hours)
χ/Q =4.14E-04 sec/m^3 (8-24 hours)

The EAB and LPZ PAVAN calculated χ/Q values input to RADTRAD, were also taken from Reference 7.32 (PM-1055). The EAB/LPZ atmospheric relative concentrations used are as follows:

EAB χ/Q =4.25E-04 sec/m^3 (0-2 hours)

LPZ χ/Q =2.08E-05 sec/m^3 (0-8 hours)
χ/Q =1.37E-05 sec/m^3 (8-24 hours)

4.5.13.2 Plant Data

- DBA Power Level — 3528 MWth
- Radial Peaking Factor — 1.7
- Number of Failed Fuel Rods (bounding case for 10x10 bundle type) — 1200
- Isotopic Release Fractions — RG 1.183

4.5.13.3 Control Room Data

- Volume of Control Room, ft^3 (UFSAR 15.6.5) — 176,000
- Control Room Normal Intake Flow, scfm — 20,600
- Assumed Unfiltered In-leakage, scfm — 1600

4.5.14 Calculations - CRDA

4.5.14.1 Source Term Calculation

The AST values used in this analysis were derived using guidance outlined in Regulatory Guide 1.183. A list of 60 core isotopic nuclides and their curie per megawatt activities were extracted from Attachment A of Reference 7.31 (PM-1059) for input into the RADTRAD "NIF." The release fractions associated with all of these nuclide groups, as detailed in Regulatory Guide 1.183, were applied to their given groups and subsequently input into the RADTRAD "RFT". RADTRAD uses these two files combined with the power of 3528 MWth to develop the source terms for this CRDA.

4.5.14.2 Dose Calculations

The RADTRAD v. 3.03 computer code is used to determine PBAPS< Units 2 and 3, CRDA doses at the three dose points cited in RG 1.183 (the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and Control Room). RADTRAD is a simplified model of RADionuclide Transport and Removal And Dose Estimation developed for the NRC and endorsed by the NRC as an acceptable methodology for reanalysis of the radiological consequences of design basis accidents.

RADTRAD estimates the releases using the reference Alternate Source Term source terms and assumptions. The RADTRAD code uses a combination of tables and/or numerical models of

source term reduction phenomena to determine the time-dependent dose at user-specified locations for a given accident scenario. The code system also provides the inventory, decay chain, and dose conversion factor tables needed for the dose calculation. The technical basis for the RADTRAD code is documented in Section 2 of NUREG/CR-6604.

4.5.15 Summary And Conclusions - CRDA

Table 4.5-2 provides the results from the RADTRAD code, as well as the dose acceptance criteria.

Table 4.5-2
RADTRAD Analysis Results and Comparisons to the Acceptance Criteria – CRDA

	EAB	LPZ	CR
Dose Limits (TEDE) / Basis Document	**6.25 rem / RG 1.183**	**6.25 rem / RG 1.183**	**5 rem / 10CFR50.67**
RADTRAD Analysis Results (1% of the Condenser free volume leakage per day)	**0.065 rem**	**0.012 rem**	**0.302 rem**

For the case analyzed in this calculation assuming automatic isolation of the MSIVs upon MSLRM trip, no SGT, and no MCREV credited at any point during the 24-hour accident, the limiting CR dose is **0.302 rem**. This limiting dose is well below the acceptance criteria, so it is verified that no off-gas or Control Room intake filtration is needed following a Control Rod Drop Accident.

4.6 MAIN STEAM LINE BREAK ACCIDENT (MSLB)

4.6.1 Objective - MSLB

The purpose of this evaluation is to determine the Control Room (CR), Exclusion Area Boundary (EAB), and Low Population Zone (LPZ) doses following a Main Steam Line Break (MSLB) Accident based on the assumptions on the break and resulting radiological releases to the Turbine Building as discussed in UFSAR Sections 14.6.5 and 14.9.2.3, and the additional assumptions for use of Alternative Source Terms (AST) contained in Appendix D of Regulatory Guide 1.183.

Inhalation Committed Effective Dose Equivalent (CEDE) Dose Conversion Factors (DCFs) from Federal Guidance Report No. 11 are used for calculation of normalized Iodine-131 Dose Equivalent activity in this calculation.

As per UFSAR Section 14.6.5, this event involves the postulation that one main steam line instantaneously and circumferentially breaks outside the secondary containment at a location downstream of the outermost isolation valve. Closure of the Main Steam Isolation Valves (MSIVs) terminates the mass loss when the full closure is reached. No operator actions are assumed to be taken during the accident, so the normal air intake into the Control Room continues unfiltered during the duration of the event.

Although the TS Surveillance Requirement 3.6.1.3.9 specifies an MSIV closure time of less than or equal to 5 seconds, the analysis conservatively assumes an MSIV closure time of 10.5 seconds. For the assumed 10.5 seconds, the mass of coolant released during the MSLB was obtained from UFSAR Section 14.6.5.

4.6.2 Methodology - MSLB

The methodology used to assess the radiological consequences of a Main Steam Line Break (MSLB) accident is the same as that used by Exelon in previous MSLB accident assessments previously approved by the NRC for the Dresden Nuclear Power Station Units 2 and 3, Quad Cities Nuclear Power Station, Units 1 and 2, Limerick Generating Station, Units 1 and 2, and Clinton Power Station AST submittals (Reference ADAMS Accession Nos. ML062070290, ML062210214, ML031040096). PBAPS Design Analysis PM-1058 (Reference 7.35) contains additional information.

The postulated MSLB accident assumes a double-ended break of one main steam line outside the primary containment with displacement of the pipe ends that permits maximum blowdown rates. The break mass released includes the amount of steam in the steam line and connecting lines at the time of the break, plus the amount of steam that passes through the valves prior to closure.

The analysis assumes the MSLB accident to be an instantaneous ground level release. Two models are considered for assessment of MSLB accident radiological consequences. One is for assessing control room dose and the other is for assessing offsite dose consequences.

In the control room model, the released reactor coolant and steam at operating temperature and pressure is conservatively assumed to expand to a hemispheric volume at atmospheric pressure and temperature. No credit is taken for dilution of the steam cloud by the air into which the steam is ejected. Neither the Turbine Building structure nor its ventilation system is assumed to have an effect on the cloud resulting from the MSLB accident. This hemisphere is then assumed to move at a speed of 1 meter per second downwind past the control room intake. No credit is taken for buoyant rise of the steam cloud or for decay in transit. Dilution (i.e., dispersion) of the activity in the plume in transit was also conservatively ignored.

For offsite locations, the buoyant rise of the steam cloud is similarly ignored, and the ground level dispersion is based on the conservative and simplified Regulatory Guide 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors," methodology. The "instantaneous release" of the MSLB accident is converted to an equivalent curie release. Since no credit is taken for decay over this release time, or in transit, the calculation accurately models an instantaneous release.

In summary, the following assumptions were used in the control room and offsite dose evaluations for a MSLB accident.

- The release from the break to the environment is assumed to be instantaneous. No holdup in the Turbine Building or dilution by mixing with Turbine Building air volume is credited.

- The steam cloud is assumed to consist solely of the initial steam blowdown and that portion of the liquid reactor coolant release that flashed to steam.
- The released reactor coolant and steam is assumed to expand to a hemispheric volume at atmospheric pressure and temperature consistent with an assumption of no Turbine Building credit.
- This hemisphere is then assumed to move at a speed of 1 meter per second downwind past the control room intake.
- No credit is taken for buoyant rise of the steam cloud or for decay, and dispersion of the activity of the plume was conservatively ignored.
- For offsite locations, the buoyant rise of the steam cloud is similarly ignored, and the ground level dispersion is based on the conservative and simplified Regulatory Guide 1.5 methodology.

The radiological consequences resulting from a design basis MSLB accident to a person at the EAB; to a person at the LPZ; and to an operator in the Control Room following an MSLB accident were performed using a Microsoft EXCEL spreadsheet.

4.6.2.1 Source Term Model - MSLB

No fuel damage is expected to result from a MSLB. Therefore, the activity available for release from the break is that present in the reactor coolant and steam lines prior to the break, with two cases analyzed, corresponding to the Reactor Coolant System Specific Activity limits in Technical Specification 3.4.6 and its Bases. Case 1 is for continued full power operation with a maximum equilibrium coolant concentration of 0.2 uCi/gm dose equivalent I-131. Case 2 is for a maximum coolant concentration of 4.0 uCi/gm dose equivalent I-131, based on a pre-accident iodine spike potentially caused by power changes. In determining I-131 dose equivalent, inhalation CEDE DCFs from Federal Guidance Report No. 11 (Reference 7.15) are used. This accident source term basis meets the guidance in RG 1.183 for analysis of this event.

4.6.2.2 Release Model - MSLB

The mass of reactor coolant and steam released is that contained in the pipe before the break occurs and prior to MSIV closure. Reactor coolant radioactivity is based on the above reactor coolant concentrations.

Releases are assumed to be instantaneous and no credit is taken for dilution in turbine building air.

4.6.2.3 Dispersion Model - MSLB

Offsite and Onsite X/Q determinations are handled differently, but conservatively in both cases.

4.6.2.3.1 EAB and LPZ - MSLB

EAB and LPZ X/Q's are determined using the original methodology in RG 1.5. Specifically:

$$\frac{\chi}{Q} = \frac{0.0133}{\sigma_y u}$$

where

σ_y = horizontal standard deviation of the plume (meters)

u = wind velocity (meters/second)

Horizontal standard deviations are taken from the PAVAN outputs for the EAB and LPZ included in Calculation PM-1055 (Reference 7.32). Per RG 1.5, F stability class and a 1 meter/sec wind speed are used.

4.6.2.3.2 Control Room Transport - MSLB

For control room dose calculations, the plume was modeled as a hemispherical volume, the dimensions of which are determined based on the initial steam blowdown and that portion of the liquid reactor coolant release that flashed to steam.

Activity release is conservatively assumed to effectively occur at the Control Room intake elevation and, again conservatively, no credit is taken for plume buoyancy. A conservative translation time of the plume over the intake is assumed.

The activity of the cloud is based on the total mass of water released from the break, not just the portion that flashes to steam. This assumption is conservative because it considers the maximum release of fission products.

4.6.2.4 Dose Model - MSLB

Dose models for both onsite and offsite are simplified and meet R.G. 1.183 requirements, providing results in units of Total Effective Dose Equivalent (TEDE). Dose conversion factors are based on Federal Guidance Reports 11 and 12.

4.6.2.4.1 EAB and LPZ - MSLB

Doses at the EAB and LPZ for the MSLB are based on the following formulas:

$$\text{Dose}_{CEDE}\ (\text{rem}) = \text{Release (Curies)} * \frac{\chi}{Q}(\text{sec/m}^3) * \text{Breathing Rate (m}^3\text{/sec)} * \text{Inhalation DCF (rem}_{CEDE}\text{/Ci inhaled)}$$

and

$$\text{Dose}_{EDE}\ (\text{rem}) = \text{Release (Curies)} * \frac{\chi}{Q}(\text{sec/m}^3) * \text{Submersion DCF (rem}_{EDE}\text{ - m}^3\text{/ Ci - sec)}$$

and finally,

$$\text{Dose}_{\text{TEDE}}\ (\text{rem}) = \text{Dose}_{\text{CEDE}}\ (\text{rem}) + \text{Dose}_{\text{EDE}}\ (\text{rem})$$

4.6.2.4.1 Control Room - MSLB

CR operator doses are determined somewhat differently, because steam cloud concentrations are used, rather than X/Q times a curie release rate. No CR filter credit is taken and, therefore, for inhalation, a dose for a location outside of the CR can be and is used. For cloud submersion, a geometry factor is used to credit the reduced plume size seen in the control room. This is a conservative implementation of RG 1.183 guidance. The formulas used are:

$$\text{Dose}_{\text{CEDE}}\ (\text{rem}) = \text{Plume Concentration}\ (\text{Ci/m}^3) * \text{Transit Duration}\ (\text{sec}) * \text{Breathing Rate}\ (\text{m}^3/\text{sec}) * \text{Inhalation DCF}\ (\text{rem}_{\text{CEDE}}/\text{Ci inhaled})$$

and

$$\text{Dose}_{\text{EDE}}\ (\text{rem}) = \text{Plume Concentration}\ (\text{Ci/m}^3) * \text{Transit Duration}\ (\text{sec}) * \text{Submersion DCF}\ (\text{rem}_{\text{EDE}} - \text{m}^3/\text{Ci-sec})$$

and finally,

$$\text{Dose}_{\text{TEDE}}\ (\text{rem}) = \text{Dose}_{\text{CEDE}}\ (\text{rem}) + \text{Dose}_{\text{EDE}}\ (\text{rem})$$

4.6.3 Acceptance Criteria - MSLB

Dose acceptance criteria are per 10CFR50.67 and RG 1.183 guidance.

Table 4.6-1 lists the regulatory limits for accidental dose to 1) a control room operator, 2) a person at the EAB, and 3) a person at the LPZ boundary.

Table 4.6-1.
Regulatory Dose Limits (Rem TEDE)

I-131 Dose Equivalent	CR (30 days)	EAB (2 hours)	LPZ (30 days)
Normal Equilibrium	5	2.5	2.5
Iodine Spike	5	25	25

4.6.4 Assumptions - MSLB

4.6.4.1 Activity Release and Transport Models

- Iodine activity distribution in the coolant is taken from UFSAR Section 14.6.5.2.1, assumption 2.
- Total release quantities from the break are taken from UFSAR Section 14.9.1.5, with Section 14.6.5 break flow and MSIV closure characteristics, including the

conservative10-second valve closure time in comparison to the less than or equal to 5 seconds isolation time limit of Surveillance Requirement 3.6.1.3.9. A 0.5-second instrument response time is added in the calculation.

- Release from the break to the environment is assumed instantaneous. No holdup in the Turbine Building or dilution by mixing with Turbine Building air volume is credited.
- The steam cloud is assumed to consist of the initial steam blowdown and that portion of the liquid reactor coolant release that flashed to steam.
- The activity of the cloud is based on the total mass of water released from the break, not just the portion that flashes to steam. This assumption is conservative because it considers the maximum release of fission products.
- The fraction of liquid water contained in steam, which carries activity into the cloud, is assumed to be 2%, a conservatively high value consistent with current Boiling Water Reactor practice.
- A conservatively high flashing fraction of liquid water released of 40% is assumed. However, all activity in the water is assumed to be released.
- For offsite dose calculations, the release is treated per Regulatory Guide 1.5. Buoyancy effect of the cloud was conservatively ignored.
- For the control room dose calculations:

    - The plume is modeled as a hemispherical volume. This is consistent with the assumption of no Turbine Building credit. It is also reasonable for the more likely release paths through multiple large openings above the Turbine Building operating deck.
    - Dispersion of the activity of the plume is conservatively ignored.
    - The cloud is assumed to be carried away by a wind of speed 1 meter/second. Credit is not taken for decay.

4.6.4.2 Control Room Model

- No credit is taken for the operation of the control room emergency filtration systems during the MSLB.
- Inhalation doses are determined based on concentrations at the intake, and exposures for the duration of plume traverse.
- External exposures are determined based on concentrations at the intake, exposures for the duration of plume traverse, and a geometry factor credit based on the control room envelope volume of 176,000 cubic feet.

4.6.5 Design Input

4.6.5.1 Mass Release Data

- The mass of steam released is 25,800 lb. [Section 14.9.1.5 of UFSAR]
- The mass of liquid water released is 165,120 lb. [Section 14.9.1.5 of UFSAR]

4.6.5.1 Iodine Distribution

The PBAPS UFSAR Section 14.6.5.2.1 provides the following design basis concentrations of significant radionuclides contained in the coolant:

Iodine Isotope	Activity (μCi/cc)
I-131	0.17
I-132	1.02
I-133	1.04
I-134	1.47
I-135	1.30

4.6.5.2 Noble Gas Distribution

The Power Rerate MSLB analysis (Reference 7.40) provided the following Noble Gas concentrations for potentially significant radionuclides contained in the coolant:

Noble Gas Isotope	Concentration uCi/g
Kr-83M	1.92E-03
Kr-85M	3.44E-03
Kr-85	1.13E-05
Kr-87	1.13E-02
Kr-88	1.13E-02
Kr-89	7.33E-02
Xe-131M	8.46E-06
Xe-133M	1.63E-04
Xe-133	4.62E-03
Xe-135M	1.47E-02
Xe-135	1.24E-02
Xe-137	8.46E-02
Xe-138	5.02E-02

4.6.5.3 Cesium Distribution

Cesium activity is not considered for the following reasons.

- RG 1.183 Appendix D, Section 2 requires the maximum coolant activity allowed by TS to be used if no or minimal fuel damage is postulated.
- At PBAPS, there is no maximum TS activity for cesium.
- RG 1.183 Appendix D, Section 4.4 states that the iodine species released from the main steam line should be assumed to be 95% CsI as an aerosol, 4.85% elemental, and 0.15% organic. This breakdown is provided to allow the licensee to determine which filter is applicable to the release. Since no filtration is assumed, this breakdown of species is ignored.
    - Since the specific cesium isotopes are not provided, it is unrealistic to apply cesium to the analysis. Previous analyses have shown that cesium doses are very minor.

4.6.5.4 Control Room Data

- Control Room Envelope = 176,000 ft^3.
- No Emergency Filtration Credit taken.

4.6.5.5 EAB and LPZ Data

- EAB Distance from Release = 915 m
- LPZ Distance from Release = 7300 m

4.6.6 Calculations - MSLB

No or minimal fuel damage is expected for the limiting MSLB. As discussed in section 2, two iodine concentrations will be used (0.2 µCi/g and 4.0 µCi/g) when determining the consequences of the main steam line break. All of the radioactivity in the released coolant is assumed to be released to the atmosphere instantaneously as a ground-level release. No credit is taken for plateout, holdup, or dilution within facility buildings.

The spreadsheets perform this analysis using data and formulations discussed above. The following summarizes parameters and their treatment in the spreadsheet.

4.6.6.1 Cloud Volumes, Masses, and Control Room Intake Transit Times

The cloud is assumed to consist of the initial steam blowdown and that portion of the liquid reactor coolant release that flashes to steam. The flashing fraction (FF) is derived as follows:

FF x (steam enthalpy at 212° F) + (1-FF) x (liquid enthalpy at 212° F) =

(liquid enthalpy at temperature of steam at reactor vessel outlet)

A 548° F vessel outlet temperature is used, with liquid enthalpy of 546.9 BTU/lb.

At 212° F, a steam enthalpy of 1150.5 BTU/lb and a liquid enthalpy of 180.17 BTU/lb are used (these enthalpies are taken from the ASME Steam Tables).

Substituting,

FF = (546.9 - 180.17) / [(1150.5 – 180.17)] = 0.378

For conservatism, a value of 0.40 or 40% is used below.

Mass of water carrying activity into the cloud is calculated as the sum of the fraction of water in the steam and the liquid blowdown.

The mass of steam released	= 25,800 lb
The mass of liquid water released	= 165,120 lb
Flashing fraction for calculating cloud volume	= 40%
The mass of water contained in steam released	= (25,800 lb) * 2%

	= 516 lb
The mass of water carrying activity into the cloud	= 516 + 165,120 lb
	= 165,636 lb
	= (165,636 lb)(453.59 g/lb)
	= 7.5131E7 g
The mass of steam in the cloud	= (25,800 - 516) + 40%*165,120 lb
	=25,284 + 66,048
	= 91,332 lb

The release is assumed to be a hemisphere with a uniform concentration. The cloud dimensions (based on 91,332 lb of steam at 14.7 psi and 212 °F, v_g = 26.799 ft^3/lb) are calculated as follows:

Volume = (91,332 lb)(26.799 ft^3/lb)
= 2,447,600 ft^3
= (2,447,600 ft^3)/(35.3 ft^3/m^3)
= 69,337 m^3

The volume of a hemisphere is $\pi d^3/12$. Thus, the diameter of the hemispherical cloud is 64.2 meters.

The period of time required for the cloud to pass over the control room intake, assuming a wind speed of 1 m/s is 64.2 s (=(64.2 m)/(1 m/s)). Note: The units "m/s" equals meters/second.

Therefore, at a wind speed of 1 m/s, the base of the hemispherical cloud will pass over the control room intake in 64.2 seconds.

4.6.6.2 Dispersion for Offsite Dose Assessment

The following formulation was used for Offsite Dose X/Q assessment, with Class F Pasquill Stability and a 1 meters/sec wind speed.

$$\frac{\chi}{Q} = \frac{0.0133}{\sigma_y u}$$

where

σ_y = horizontal standard deviation of the plume (meters)

u = wind velocity (meters/second)

As calculated in the PAVAN run, at the conservatively assumed 823 meter EAB distance σ_y is 38.3, and at the 7300 meter LPZ distance σ_y is 222.6. The resulting EAB and LPZ X/Qs are 3.47E-04 and 5.97E-05 sec/m^3, respectively.

4.6.6.3 Release Isotopics and Quantification

The iodine isotopic distribution given in Section 4.6.5.1 is used. The concentrations of this mix are adjusted to I-131 equivalence, using the inhalation Committed Effective Dose Equivalent (CEDE) Dose Conversion Factors (DCFs) from Federal Guidance Report No. 11. This is a more conservative set of DCF assumptions for Control Room and off-site dose calculation than the use of ICRP 2 DCFs. It is also more conservative for these calculations than use of RG 1.109 (Reference 7.39) or Federal Guidance Report No. 12 DCFs.

This I-131 equivalent mix is adjusted to the activity yielding the two design basis MSLB accident reactor coolant activities of 0.2 μCi/cc and 4.0 μCi/cc. The released activities are these concentrations times the 7.51E+07 grams of water carrying activity released, with the assumption that TS activities are based on laboratory temperature and pressure conditions.

For the Noble Gases, the isotopic distribution given in Section 4.3 is used. The released activities are these concentrations times the 25,800 lb mass of steam released, converted to 1.17E+07 grams using the 453.59 g/lb conversion factor.

4.6.3.4 Dose Assessment - MSLB

Doses at the EAB and LPZ distances, and in the Control Room were calculated using the formulas cited above. Concentrations at the receptor locations are those in the steam plume for the Control Room or based on the release times the applicable X/Q for the EAB and LPZ.

Doses are calculated for inhalation (rem CEDE) and plume submersion (rem EDE) and totaled to yield rem TEDE. The breathing rate of 3.47E-04 m^3/sec is used. The resulting calculated doses are summarized below.

4.6.4 Summary and Conclusions - MSLB

Doses from a design basis MSLB were calculated for the control room operator, a person at the EAB, and a person at LPZ. The results are summarized in the Table below. The doses at the Control Room, EAB, and LPZ resulting from a postulated design basis MSLB do not exceed the regulatory limits.

Location	Case 1 (Normal equilibrium Limit of 0.2 μCi) Dose (rem TEDE)	Case 2 (Iodine spike Limit of 4.0 μCi) Dose (rem TEDE)
LIMITS	CR: 5.0; EAB&LPZ: 2.5	CR: 5.0; EAB&LPZ: 25
EAB	7.99E-02	1.60E+00
LPZ	1.38E-02	2.75E-01
CR	1.62E-01	3.23 E+00

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

Exelon Generation Company, LLC (Exelon) is requesting a revision to the Facility Operating Licenses for Peach Bottom Atomic Power Station, Units 2 and 3. Specifically, Exelon is requesting a revision to the Technical Specifications and licensing and design bases to reflect the application of alternative source term (AST) assumptions.

The AST analyses were performed in accordance with the guidance in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000, and Standard Review Plan Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms."

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.

5.1.1 The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The implementation of alternative source term (AST) assumptions has been evaluated in revisions to the analyses of the following limiting design basis accidents (DBAs) at Peach Bottom Atomic Power Station (PBAPS):

- Loss-of-Coolant Accident,
- Fuel Handling Accident,
- Control Rod Drop Accident, and
- Main Steam Line Break 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. This guidance is presented in 10 CFR 50.67 and associated Regulatory Guide 1.183, and Standard Review Plan Section 15.0.1. The Alternative Source Term 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 radiation released from the fuel. It does, however, better represent the physical characteristics of the release, so that

appropriate mitigation techniques may be applied. Therefore, the consequences of an accident previously evaluated are not significantly increased.

The equipment affected by the proposed changes is mitigative in nature, and relied upon after an accident has been initiated. Application of the Alternative Source Term (AST) does not involve any physical changes to the plant design. While the operation of various systems do change as a result of these proposed changes, these systems are not accident initiators. Application of the AST is not an initiator of a design basis accident. The proposed changes to the Technical Specifications (TS), while they revise certain performance requirements, do not involve any physical modifications 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. As such, removal of operability requirements during the specified conditions will not significantly increase the probability of occurrence for an accident previously analyzed. Since design basis accident initiators are not being altered by adoption of the Alternative Source Term analyses, the probability of an accident previously evaluated is not affected.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

5.1.2 The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendment does not involve a physical alteration of the plant (no new or different type of equipment will be installed and there are no physical modifications to existing equipment associated with the proposed changes). Similarly, it does not physically change any structures, systems or components involved in the mitigation of any accidents; thus, 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 amendment.

As such, the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

5.1.3 The proposed changes do not involve a significant reduction in a margin of safety.

Safety margins and analytical conservatisms have been evaluated and have been found acceptable. The analyzed events have been carefully selected and margin has been retained to ensure that the analyses adequately bound postulated event scenarios. The dose consequences due to design basis accidents comply with the requirements of 10 CFR 50.67 and the guidance of Regulatory Guide 1.183.

The proposed amendment is associated with the implementation of a new licensing basis for PBAPS Design Basis Accidents (DBAs). Approval of the change from the original source term to a new source term taken from Regulatory Guide 1.183 is being requested. The results of the accident analyses, revised in support of the proposed license amendment, are subject to revised acceptance criteria. The analyses have been performed using conservative methodologies, as specified in Regulatory Guide 1.183. Safety margins have been evaluated and analytical conservatism has been utilized to ensure that the analyses adequately bound the

postulated limiting event scenario. The dose consequences of these DBAs remain within the acceptance criteria presented in 10 CFR 50.67, "Accident Source Term", and Regulatory Guide 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 corresponding regulatory limits.

Therefore, operation of PBAPS in accordance with the proposed changes will not involve a significant reduction in a margin of safety.

Conclusion

Exelon Generation Company, LLC (Exelon) concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of no significant hazards consideration is justified.

5.2 Applicable Regulatory Requirements/Criteria

The NRC's traditional methods (prior to the AST) for calculating the radiological consequences of design basis accidents are described in a series of regulatory guides 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 ASTs and with the Total Effective Dose Equivalent (TEDE) criteria provided in 10 CFR 50.67. Regulatory Guide 1.183 provides assumptions and methods that are acceptable to the NRC staff for performing design basis radiological analyses using an AST. This guidance supersedes corresponding radiological analysis assumptions provided in the older regulatory guides and SRP chapters when used in conjunction with an approved AST and the TEDE criteria provided in 10 CFR 50.67.

Also, the NRC published a new SRP section to address AST. It is Standard Review Plan Section 15.0.1, Rev. 0, entitled "Radiological Consequence Analyses Using Alternative Source Terms". It provides guidance on which NRC branches will review various aspects of an AST license amendment request, but otherwise is consistent with the guidance found in Regulatory Guide 1.183. The plant-specific information provided in this license amendment request adequately addresses 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 EVALUATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10CFR 20, "Standards for Protection Against Radiation," or would change an inspection or surveillance requirement. However, the proposed amendment does not involve: (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the

amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10CFR 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). Therefore, pursuant to 10CFR 51.22, Paragraph (b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

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

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"

7.11 U. S. Nuclear Regulatory Commission Regulatory Guide 1.23 (Safety Guide 23) Proposed Revision 1, "Onsite Meteorological Programs," February 17, 1972

7.12 U. S. Nuclear Regulatory Commission Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling 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 U. S. Nuclear Regulatory Commission Regulatory Guide 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors," March 1971

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 (Second Printing, 1989).

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

7.17 Letter from Mr. J. P. Gallagher (Exelon Nuclear) to U.S. NRC dated April 23, 2004, "Supplement to the Request for License Amendments Related to Application of Alternative Source Term," dated July 14, 2003.

7.18 Letter from Mr. K. Jury (Exelon Nuclear) to U.S. NRC dated May 20, 2004, "Supplement to the Request for License Amendments Related to Application of Alternative Source Term," dated July 14, 2003.

7.19 Regulatory Guide 1.194; Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants; U.S. Nuclear Regulatory Commission; June 2003.

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 Letter from U.S. NRC to Mr. J. A. Hutton, PECO Energy Company, "Peach Bottom Atomic Power Station, Unit Nos. 2 and 3 – Issuance of Amendment Regarding Crediting of Containment Overpressure for Net Positive Suction Head Calculations for Emergency Core Cooling Pumps", dated August 14, 2000).

7.23 Nuclear Utilities Management and Resources Council (NUMARC) 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 3

7.24 NUREG-0800, Standard Review Plan, "Containment Spray as a Fission Product Cleanup System," SRP 6.5.2, Revision 2, 1988

7.25 AEB 98-03, Assessment of Radiological Consequences for the Perry Pilot Plant Application Using The Revised (NUREG-1465) Source Term

7.26 MicroShield Computer Code, V&V Version 5.05, Grove Engineering

7.27 General Electric Specification No. 22A6209, Rev 1, "Containment Data"

7.28 Regulatory Guide 1.52, Revision 2, "Design, Testing, and Maintenance Criteria For Post Accident Engineered Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants", March 1978.

7.29 ASME N510-1989

7.30 ASTM D3803-1989

7.31 PBAPS Design Analysis PM-1059, Rev. 2, "Re-analysis of Fuel Handling Accident (FHA) Using Alternative Source Terms"

7.32 PBAPS Design Analysis PM-1055, Rev. 0, "Calculation of Alternative Source Term (AST) Onsite and Offsite χ/Q Values"

7.33 PBAPS Design Analysis PM-1056, Rev. 0, "Suppression Pool pH Calculation for Alternative Source Terms"

7.34 PBAPS Design Analysis PM-1057, Rev.1, "Re-analysis of Control Rod Drop Accident (CRDA) Using Alternative Source Terms."

7.35 PBAPS Calculation No. PM-1058, Rev 0, "Re-analysis of Main Steam Line Break Accident (MSLB) Using Alternative Source Terms."

7.36 PBAPS Calculation No. PM-1061, Rev 0, Determination of Reduced Primary Containment Leakage Rate for AST Implementation

7.37 PBAPS Calculation No. PM-1077, Rev 0, "Post-LOCA EAB, LPZ, and CR Doses Using Alternative Source Term (AST)."

7.38 NCS Corporation Report, Control Room Envelope Inleakage Testing At Peach Bottom Atomic Power Station, 2004.

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

7.40 PBAPS Calculation No. PM-0765, "Technical Support Center Analysis for Power Rerate."

7.41 NEDE-24011-P-A-15, September 2005, General Electric Standard Application for Reactor Fuel (GESTAR II)

ATTACHMENT 2

PBAPS, Units 2 and 3
Renewed Facility Operating License Nos. DPR-44 and DPR-56

"PBAPS Alternative Source Term Implementation"

Markup of Proposed Technical Specification Pages

REVISED TS PAGES

UNIT 2

1.1-2
1.1-5
3.1-20 to 21
3.3-54
3.3-58
3.6-12
3.6-16
3.6-34 to 36
3.6-38
3.6-40 to 41
5.0-17

UNIT 3

1.1-2
1.1-5
3.1-20 to 21
3.3-54
3.3-58
3.6-12
3.6-16
3.6-34 to 36
3.6-38
3.6-40 to 41
5.0-17

1.1 Definitions (continued)

CHANNEL FUNCTIONAL TEST	A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarm, interlock, display, and trip functions, and channel failure trips. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS: a. Movement of wide range neutron monitors, local power range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and b. Control rod movement, provided there are no fuel assemblies in the associated core cell. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same ~~thyroid~~ dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The ~~thyroid~~ dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites" or Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989.

(continued)

1.1 Definitions

PHYSICS TESTS (continued)	b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3514 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from the opening of the sensor contact up to and including the opening of the trip actuator contacts.
RECENTLY IRRADIATED FUEL	**RECENTLY IRRADIATED FUEL is fuel that has occupied part of a critical reactor core within the previous 84 hours. This 84-hour time period may be reduced to 24 hours if all outside secondary containment ground-level hatches (hatches H15 through H24 and Units 2 and 3 Torus room access hatches) are closed.**
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that: a. The reactor is xenon free; b. The moderator temperature is 68°F; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

(continued)

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1 ~~and 2~~ ,2, and 3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Concentration of boron in solution > 9.82% weight.	A.1 Verify the concentration and temperature of boron in solution and pump suction piping temperature are within the limits of Figure 3.1.7-1.	8 hours AND Once per 12 hours thereafter
	AND	
	A.2 Restore concentration of boron in solution to ≤ 9.82% weight.	72 hours AND 10 days from discovery of failure to meet the LCO
B. One SLC subsystem inoperable for reasons other than Condition A.	B.1 Restore SLC subsystem to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two SLC subsystems inoperable for reasons other than Condition A.	C.1 Restore one SLC subsystem to OPERABLE status.	8 hours
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify level of sodium pentaborate solution in the SLC tank is ≥ 46%.	24 hours
SR 3.1.7.2	Verify temperature of sodium pentaborate solution is ≥ 53°F.	24 hours
SR 3.1.7.3	Verify temperature of pump suction piping is ≥ 53°F.	24 hours
SR 3.1.7.4	Verify continuity of explosive charge.	31 days

(continued)

Table 3.3.6.1-1 (page 3 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. Reactor Water Cleanup (RWCU) System Isolation					
a. RWCU Flow-High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.7	≤ 125% rated flow (23.0 in-wc)
b. SLC System Initiation	1,2,3	1	H	SR 3.3.6.1.7	NA
c. Reactor Vessel Water Level-Low (Level 3)	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.7	≥ 1.0 inches
6. RHR Shutdown Cooling System Isolation					
a. Reactor Pressure-High	1,2,3	1	F	SR 3.3.6.1.3 SR 3.3.6.1.7	≤ 70.0 psig
b. Reactor Vessel Water Level-Low (Level 3)	3,4,5	2[a]	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.7	≥ 1.0 inches
7. Feedwater Recirculation Isolation					
a. Reactor Pressure-High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 600 psig
8. Traversing Incore Probe Isolation					
a. Reactor Vessel Water Level-Low (Level 3)	1,2,3	2	J	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.7	≥ 1.0 inches
b. Drywell Pressure-High	1,2,3	2	J	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 2.0 psig

(a) In MODES 4 and 5, provided RHR Shutdown Cooling System integrity is maintained, only one channel per trip system with an isolation signal available to one shutdown cooling pump suction isolation valve is required.

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level —Low (Level 3)	1,2,3, (a)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≥ 1.0 inches
2. Drywell Pressure —High	1,2,3	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 2.0 psig
3. Reactor Building Ventilation Exhaust Radiation —High	1,2,3, (a),(b)	2	SR 3.3.6.2.1 SR 3.3.6.2.3 SR 3.3.6.2.5	≤ 16.0 mR/hr
4. Refueling Floor Ventilation Exhaust Radiation —High	1,2,3, (a),(b)	2	SR 3.3.6.2.1 SR 3.3.6.2.3 SR 3.3.6.2.5	≤ 16.0 mR/hr

(a) During operations with a potential for draining the reactor vessel.

(b) During ~~CORE ALTERATIONS, and during~~ movement of ~~irradiated fuel~~ assemblies in secondary containment.

RECENTLY IRRADIATED FUEL

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Purge/Vent flowpath open for an accumulated time of greater than 90 hours for the calendar year while in MODE 1 or 2 with Reactor Pressure greater than 100 psig.	**E.1 Isolate the penetration** **OR** **E.2 Be in MODE 3.** **AND** **E.3 Be in MODE 4**	**4 hours** **12 hours** **36 hours**
~~E~~F. Required Action and associated Completion Time of Condition A, B, C, or D not met in MODE 1, 2, or 3.	~~E~~F.1 Be in MODE 3. AND ~~E~~F.2 Be in MODE 4.	12 hours 36 hours
~~F~~G. Required Action and associated Completion Time of Condition A, B, C, or D not met for PCIV(s) required to be OPERABLE during MODE 4 or 5.	~~F~~G.1 Initiate action to suspend operations with a potential for draining the reactor vessel. OR ~~F~~G.2 Initiate action to restore valve(s) to OPERABLE status.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.1	Verify Containment Atmospheric Dilution (CAD) System liquid nitrogen storage tank level is ≥ 16 inches water column.	24 hours
SR 3.6.1.3.2	Verify Safety Grade Instrument Gas (SGIG) System header pressure is ≥ 80 psig.	24 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.14	~~Verify leakage rate through each MSIV is ≤ 11.5 scfh when tested at ≥ 25 psig.~~	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.15	Verify each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve is blocked to restrict opening greater than the required maximum opening angle.	24 months
SR 3.6.1.3.16	Replace the inflatable seal of each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve.	96 months

Verify combined MSIV leakage rate for all four main steam lines is ≤ 204 scfh and ≤ 116 scfh for any one steam line, when tested at ≥ 25 psig.

3.6 CONTAINMENT SYSTEMS

3.6.4.1 Secondary Containment

LCO 3.6.4.1 The secondary containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of ~~irradiated fuel~~ RECENTLY IRRADIATED FUEL assemblies in the secondary containment,
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Secondary containment inoperable in MODE 1, 2, or 3.	A.1 Restore secondary containment to OPERABLE status.	4 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours
C. Secondary containment inoperable during movement of ~~irradiated fuel~~ RECENTLY IRRADIATED FUEL assemblies in the secondary containment, ~~during CORE ALTERATIONS,~~ or during OPDRVs.	C.1 --------NOTE-------- LCO 3.0.3 is not applicable. ---------- Suspend movement of ~~irradiated fuel~~ assemblies in the secondary containment. AND	Immediately

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	~~C.2 Suspend CORE ALTERATIONS.~~ ~~AND~~ C.~~3~~ 2 Initiate action to suspend OPDRVs.	~~Immediately~~ Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.1	Verify all secondary containment equipment hatches are closed and sealed.	31 days
SR 3.6.4.1.2	Verify one secondary containment access door in each access opening is closed.	31 days
SR 3.6.4.1.3	Verify secondary containment can be drawn down to ≥ 0.25 inch of vacuum water gauge in ≤ ~~120~~ 180 seconds using one standby gas treatment (SGT) subsystem.	24 months on a STAGGERED TEST BASIS for each subsystem
SR 3.6.4.1.4	Verify the secondary containment can be maintained ≥ 0.25 inch of vacuum water gauge for 1 hour using one SGT subsystem at a flow rate ≤ 10,500 cfm.	24 months on a STAGGERED TEST BASIS for each subsystem

3.6 CONTAINMENT SYSTEMS

3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

LCO 3.6.4.2 Each SCIV shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of ~~irradiated fuel~~ RECENTLY IRRADIATED FUEL assemblies in the secondary containment,
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

----------NOTES----------

1. Penetration flow paths may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each penetration flow path.

3. Enter applicable Conditions and Required Actions for systems made inoperable by SCIVs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more penetration flow paths with one SCIV inoperable.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange. AND	8 hours (continued)

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 in the secondary containment, ~~during CORE ALTERATIONS,~~ or during OPDRVs.	D.1 --------NOTE--------- LCO 3.0.3 is not applicable. ---------- Suspend movement of ~~irradiated fuel~~ assemblies in the secondary containment.	Immediately
RECENTLY IRRADIATED FUEL	~~AND~~ ~~D.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	AND D.~~3~~ 2 Initiate action to suspend OPDRVs.	Immediately

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Two SGT subsystems shall be OPERABLE.

RECENTLY IRRADIATED FUEL

APPLICABILITY: MODES 1, 2, and 3,
During movement of ~~irradiated fuel~~ assemblies in the secondary containment,
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SGT subsystem inoperable.	A.1 Restore SGT subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met during movement of ~~irradiated fuel~~ assemblies in the secondary containment, ~~during CORE ALTERATIONS,~~ or during OPDRVs.	----------NOTE---------- LCO 3.0.3 is not applicable. ---------- C.1 Place OPERABLE SGT subsystem in operation. OR	Immediately

(continued)

RECENTLY IRRADIATED FUEL

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2.1 Suspend movement of ~~irradiated fuel~~ assemblies in secondary containment.	Immediately
	~~AND~~	
	~~C.2.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	AND	
	C.2.~~3~~ 2 Initiate action to suspend OPDRVs.	Immediately
D. Two SGT subsystems inoperable in MODE 1, 2, or 3.	D.1 Enter LCO 3.0.3	Immediately
E. Two SGT subsystems inoperable during movement of ~~irradiated fuel~~ assemblies in the secondary containment ~~during CORE ALTERATIONS,~~ or during OPDRVs.	E.1 --------NOTE--------- LCO 3.0.3 is not applicable. ----------	
	Suspend movement of ~~irradiated fuel~~ assemblies in secondary containment.	Immediately
	~~AND~~	
	~~E.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	AND	
	E.~~3~~ 2 Initiate action to suspend OPDRVs.	Immediately

RECENTLY IRRADIATED FUEL

RECENTLY IRRADIATED FUEL

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

1. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

2. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

3. A required system redundant to support system(s) for the supported systems (b.1) and (b.2) above is also inoperable.

c. 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.

5.5.12 Primary 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, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Section 10.2:

a. MSIV leakage is excluded from the combined total of 0.6 L_a for the Type B and C tests.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 49.1 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be ~~0.5%~~ of primary containment air weight per day.

0.7%

Leakage Rate acceptance criteria are:

a. Primary Containment leakage rate acceptance criterion is ≤ 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 Type C tests and ≤ 0.75 L_a for Type A tests;

(continued)

1.1 Definitions (continued)

CHANNEL FUNCTIONAL TEST	A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarm, interlock, display, and trip functions, and channel failure trips. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS: a. Movement of wide range neutron monitors, local power range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and b. Control rod movement, provided there are no fuel assemblies in the associated core cell. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same ~~thyroid~~ dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The ~~thyroid~~ dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites" or Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989.

(continued)

1.1 Definitions

PHYSICS TESTS (continued)	b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3514 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from the opening of the sensor contact up to and including the opening of the trip actuator contacts.
RECENTLY IRRADIATED FUEL	**RECENTLY IRRADIATED FUEL is fuel that has occupied part of a critical reactor core within the previous 84 hours. This 84-hour time period may be reduced to 24 hours if all outside secondary containment ground-level hatches (hatches H15 through H24 and Units 2 and 3 Torus room access hatches) are closed.**
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that: a. The reactor is xenon free; b. The moderator temperature is 68°F; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

(continued)

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Concentration of boron in solution > 9.82% weight.	A.1 Verify the concentration and temperature of boron in solution and pump suction piping temperature are within the limits of Figure 3.1.7-1.	8 hours AND Once per 12 hours thereafter
	AND	
	A.2 Restore concentration of boron in solution to ≤ 9.82% weight.	72 hours AND 10 days from discovery of failure to meet the LCO
B. One SLC subsystem inoperable for reasons other than Condition A.	B.1 Restore SLC subsystem to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two SLC subsystems inoperable for reasons other than Condition A.	C.1 Restore one SLC subsystem to OPERABLE status.	8 hours
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify level of sodium pentaborate solution in the SLC tank is ≥ 46%.	24 hours
SR 3.1.7.2	Verify temperature of sodium pentaborate solution is ≥ 53°F.	24 hours
SR 3.1.7.3	Verify temperature of pump suction piping is ≥ 53°F.	24 hours
SR 3.1.7.4	Verify continuity of explosive charge.	31 days

(continued)

Table 3.3.6.1-1 (page 3 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. Reactor Water Cleanup (RWCU) System Isolation					
a. RWCU Flow – High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.7	≤ 125% rated flow (23.0 in-wc)
b. SLC System Initiation	1,2,3	1	H	SR 3.3.6.1.7	NA
c. Reactor Vessel Water Level – Low (Level 3)	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.7	≥ 1.0 inches
6. RHR Shutdown Cooling System Isolation					
a. Reactor Pressure – High	1,2,3	1	F	SR 3.3.6.1.3 SR 3.3.6.1.7	≤ 70.0 psig
b. Reactor Vessel Water Level – Low (Level 3)	3,4,5	2[a]	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.7	≥ 1.0 inches
7. Feedwater Recirculation Isolation					
a. Reactor Pressure – High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 600 psig
8. Traversing Incore Probe Isolation					
a. Reactor Vessel Water Level-Low (Level 3)	1,2,3	2	J	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.7	≥ 1.0 inches
b. Drywell Pressure-High	1,2,3	2	J	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 2.0 psig

(a) In MODES 4 and 5, provided RHR Shutdown Cooling System integrity is maintained, only one channel per trip system with an isolation signal available to one shutdown cooling pump suction isolation valve is required.

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level —Low (Level 3)	1,2,3, (a)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≥ 1.0 inches
2. Drywell Pressure —High	1,2,3	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 2.0 psig
3. Reactor Building Ventilation Exhaust Radiation —High	1,2,3, (a),(b)	2	SR 3.3.6.2.1 SR 3.3.6.2.3 SR 3.3.6.2.5	≤ 16.0 mR/hr
4. Refueling Floor Ventilation Exhaust Radiation —High	1,2,3, (a),(b)	2	SR 3.3.6.2.1 SR 3.3.6.2.3 SR 3.3.6.2.5	≤ 16.0 mR/hr

(a) During operations with a potential for draining the reactor vessel.

(b) During ~~CORE ALTERATIONS, and during~~ movement of ~~irradiated fuel~~ assemblies in secondary containment.

RECENTLY IRRADIATED FUEL

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Purge/Vent flowpath open for an accumulated time greater than 90 hours for the calendar year while in MODE 1 or 2 with Reactor Pressure greater than 100 psig.	**E.1 Isolate the penetration** **OR** **E.2 Be in MODE 3.** **AND** **E.3 Be in MODE 4.**	**4 hours** **12 hours** **36 hours**
~~E~~F. Required Action and associated Completion Time of Condition A, B, C, or D not met in MODE 1, 2, or 3.	~~E~~F.1 Be in MODE 3. AND ~~E~~F.2 Be in MODE 4.	12 hours 36 hours
~~F~~G. Required Action and associated Completion Time of Condition A, B, C, or D not met for PCIV(s) required to be OPERABLE during MODE 4 or 5.	~~F~~G.1 Initiate action to suspend operations with a potential for draining the reactor vessel. OR ~~F~~G.2 Initiate action to restore valve(s) to OPERABLE status.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.1	Verify Containment Atmospheric Dilution (CAD) System liquid nitrogen storage tank level is ≥ 16 inches water column.	24 hours
SR 3.6.1.3.2	Verify Safety Grade Instrument Gas (SGIG) System header pressure is ≥ 80 psig.	24 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.14	~~Verify leakage rate through each MSIV is ≤ 11.5 scfh when tested at ≥ 25 psig.~~	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.15	Verify each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve is blocked to restrict opening greater than the required maximum opening angle.	24 months
SR 3.6.1.3.16	Replace the inflatable seal of each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve.	96 months

Verify combined MSIV leakage rate for all four main steam lines is ≤ 204 scfh and ≤ 116 scfh for any one steam line, when tested ≥ 25 psig.

3.6 CONTAINMENT SYSTEMS

3.6.4.1 Secondary Containment

LCO 3.6.4.1 The secondary containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of ~~irradiated fuel~~ RECENTLY IRRADIATED FUEL assemblies in the secondary containment,
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Secondary containment inoperable in MODE 1, 2, or 3.	A.1 Restore secondary containment to OPERABLE status.	4 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours
C. Secondary containment inoperable during movement of ~~irradiated fuel~~ RECENTLY IRRADIATED FUEL assemblies in the secondary containment, ~~during CORE ALTERATIONS,~~ or during OPDRVs.	C.1 --------NOTE--------- LCO 3.0.3 is not applicable. ---------- Suspend movement of ~~irradiated fuel~~ assemblies in the secondary containment. AND	Immediately

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	~~C.2 Suspend CORE ALTERATIONS.~~ ~~AND~~ C.~~3~~ 2 Initiate action to suspend OPDRVs.	~~Immediately~~ Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.1	Verify all secondary containment equipment hatches are closed and sealed.	31 days
SR 3.6.4.1.2	Verify one secondary containment access door in each access opening is closed.	31 days
SR 3.6.4.1.3	Verify secondary containment can be drawn down to ≥ 0.25 inch of vacuum water gauge in ≤ ~~120~~ 180 seconds using one standby gas treatment (SGT) subsystem.	24 months on a STAGGERED TEST BASIS for each subsystem
SR 3.6.4.1.4	Verify the secondary containment can be maintained ≥ 0.25 inch of vacuum water gauge for 1 hour using one SGT subsystem at a flow rate ≤ 10,500 cfm.	24 months on a STAGGERED TEST BASIS for each subsystem

3.6 CONTAINMENT SYSTEMS

3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

LCO 3.6.4.2 Each SCIV shall be OPERABLE.

RECENTLY IRRADIATED FUEL

APPLICABILITY: MODES 1, 2, and 3,
During movement of ~~irradiated fuel~~ assemblies in the secondary containment,
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

----------NOTES----------

1. Penetration flow paths may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each penetration flow path.

3. Enter applicable Conditions and Required Actions for systems made inoperable by SCIVs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more penetration flow paths with one SCIV inoperable.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange. AND	8 hours (continued)

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~~ RECENTLY IRRADIATED FUEL assemblies in the secondary containment, ~~during CORE ALTERATIONS,~~ or during OPDRVs.	D.1 --------NOTE--------- LCO 3.0.3 is not applicable. ---------- Suspend movement of ~~irradiated fuel~~ assemblies in the secondary containment.	Immediately
	~~AND~~ ~~D.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	AND D 3 2 Initiate action to suspend OPDRVs.	Immediately

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Two SGT subsystems shall be OPERABLE.

RECENTLY IRRADIATED FUEL

APPLICABILITY: MODES 1, 2, and 3,
During movement of ~~irradiated fuel~~ assemblies in the secondary containment,
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SGT subsystem inoperable.	A.1 Restore SGT subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met during movement of ~~irradiated fuel~~ assemblies in the secondary containment, ~~during CORE ALTERATIONS,~~ or during OPDRVs.	----------NOTE---------- LCO 3.0.3 is not applicable. ---------- C.1 Place OPERABLE SGT subsystem in operation. OR	Immediately

(continued)

RECENTLY IRRADIATED FUEL

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2.1 Suspend movement of ~~irradiated fuel~~ assemblies in secondary containment.	Immediately
	~~AND~~	
	~~C.2.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	AND	
	C.2.~~3~~ 2 Initiate action to suspend OPDRVs.	Immediately
D. Two SGT subsystems inoperable in MODE 1, 2, or 3.	D.1 Enter LCO 3.0.3	Immediately
E. Two SGT subsystems inoperable during movement of ~~irradiated fuel~~ assemblies in the secondary containment ~~during CORE ALTERATIONS,~~ or during OPDRVs.	E.1 --------NOTE--------- LCO 3.0.3 is not applicable. ----------	
	Suspend movement of ~~irradiated fuel~~ assemblies in secondary containment.	Immediately
	~~AND~~	
	~~E.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	AND	
	E.~~3~~ 2 Initiate action to suspend OPDRVs.	Immediately

RECENTLY IRRADIATED FUEL

RECENTLY IRRADIATED FUEL

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

1. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

2. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

3. A required system redundant to support system(s) for the supported systems (b.1) and (b.2) above is also inoperable.

c. 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.

5.5.12 Primary 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, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exceptions to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J":

a. Section 10.2: MSIV leakage is excluded from the combined total of 0.6 L_a for the Type B and C tests.

b. Section 9.2.3: The first Type A test performed after the December, 1991 Type A test shall be performed no later than December, 2006.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 49.1 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be ~~0.5%~~ of primary containment air weight per day.

0.7%

Leakage Rate acceptance criteria are:

a. Primary Containment leakage rate acceptance criterion is ≤ 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 Type C tests and ≤ 0.75 L_a for Type A tests;

(continued)

ATTACHMENT 3

PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 AND 3

Renewed Facility Operating License Nos. DPR-44 and DPR-56

"PBAPS Alternative Source Term Implementation"

Markup of Proposed Technical Specification Bases Pages
(For Information Only)

REVISED TS BASES PAGES

UNITS 2 and 3

B 2.0-5 to 7
B 2.0-9 to 10
B 3.1-34
B 3.1-37
B 3.1-39 to 41
B 3.1-44
B 3.1-47 to 48
B 3.1-52
B 3.2-11
B 3.3-147 to 150 (Unit 2 only)
B 3.3-148 to 151 (Unit 3 only)
B 3.3-156 (Unit 2 only)
B 3.3-157 (Unit 3 only)
B 3.3-174
B 3.4-29 to 32
B 3.6-2
B 3.6-7
B 3.6-23
B 3.6-27
B 3.6-29
B 3.6-73 to 76
B 3.6-78 to 79
B 3.6-82
B 3.6-84
B 3.6-87 to B 3.6-89
B 3.7-29 to 30
B 3.9-17
B 3.9-19

BASES (continued)

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.

APPLICABILITY

SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.

SAFETY LIMIT VIOLATIONS

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 2) **and 10 CFR 50.67, "Accident Source Term," for accidents analyzed using AST (Ref 3).** Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

(continued)

BASES

REFERENCES

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," latest approved revision.

2. 10 CFR 100.

3. **10 CFR 50.67.**

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on reactor steam dome pressure protects 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. Establishing an upper limit on reactor steam dome pressure ensures continued RCS integrity with regard to pressure excursions. Per the UFSAR (Ref. 1), the reactor coolant pressure boundary (RCPB) shall be designed with sufficient margin to ensure that the design conditions are not exceeded during normal operation and abnormal operational transients.

During normal operation and abnormal operational transients, 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, in accordance with ASME Code requirements, prior to initial operation when there is no fuel in the core. Any further hydrostatic testing with fuel in the core may be done under LCO 3.10.1, "Inservice Leak and Hydrostatic Testing Operation." Following inception of unit operation, RCS components shall be pressure tested in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB reducing the number of protective barriers designed to prevent radioactive releases from exceeding the limits specified in ~~10 CFR 100, "Reactor Site Criteria"~~**10 CFR 50.67, "Accident Source Term"** (Ref. 4). If this occurred in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere.

APPLICABLE SAFETY ANALYSES

The RCS safety/relief valves and the Reactor Protection System Reactor Pressure—High Function have settings established to ensure that the RCS pressure SL will not be exceeded.

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of ~~10 CFR 100, "Reactor Site Criteria,"~~**10 CFR 50.67, "Accident Source Term"** limits (Ref. 4). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also assures that the probability of an accident occurring during the period is minimal.

REFERENCES

1. UFSAR, Section 1.5.2.2.

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

(continued)

BASES

REFERENCES (continued)

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

4. ~~10 CFR 100.~~**10 CFR 50.67**

5. ASME, Boiler and Pressure Vessel Code, Section III, 1965 Edition, including Addenda to winter of 1965.

6. ASME, Boiler and Pressure Vessel Code, Section III, 1980 Edition, Addenda to winter of 1981.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Rod Pattern Control

BASES

BACKGROUND

Control rod patterns during startup conditions are controlled by the operator and the rod worth minimizer (RWM) (LCO 3.3.2.1, "Control Rod Block Instrumentation"), so that only specified control rod sequences and relative positions are allowed over the operating range of all control rods inserted to 10% RTP. The sequences limit the potential amount of reactivity addition that could occur in the event of a Control Rod Drop Accident (CRDA).

This Specification assures that the control rod patterns are consistent with the assumptions of the CRDA analyses of References 1 and 2.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the CRDA are summarized in References 1 and 2. CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. These sequences define the potential initial conditions for the CRDA analysis. The RWM (LCO 3.3.2.1) provides backup to operator control of the withdrawal sequences to ensure that the initial conditions of the CRDA analysis are not violated.

Prevention or mitigation of positive reactivity insertion events is necessary to limit the energy deposition in the fuel, thereby preventing significant fuel damage which could result in the undue release of radioactivity. Since the failure consequences for UO_2 have been shown to be insignificant below fuel energy depositions of 300 cal/gm (Ref. 3), the fuel damage limit of 280 cal/gm provides a margin of safety from significant core damage which would result in release of radioactivity (Ref~~s. 4 and~~ 5). Generic evaluations (Refs. 1 and 6) of a design basis CRDA (i.e., a CRDA resulting in a peak fuel energy deposition of 280 cal/gm) have shown that if the peak fuel enthalpy remains below 280 cal/gm, then the maximum reactor pressure will be less than the required ASME Code limits (Ref. 7) and the calculated offsite doses will be well within the required limits (Ref. 5).

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

control rods has less impact on control rod worth than withdrawals have. Required Action B.1 is modified by a Note which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position.

LCO 3.3.2.1 requires verification of control rod movement by a second licensed operator or a qualified member of the technical staff.

When nine or more OPERABLE control rods are not in compliance with BPWS, the reactor mode switch must be placed in the shutdown position within 1 hour. With the mode switch in shutdown, the reactor is shut down, and as such, does not meet the applicability requirements of this LCO. The allowed Completion Time of 1 hour is reasonable to allow insertion of control rods to restore compliance, and is appropriate relative to the low probability of a CRDA occurring with the control rods out of sequence.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

The control rod pattern is verified to be in compliance with the BPWS at a 24 hour Frequency to ensure the assumptions of the CRDA analyses are met. The 24 hour Frequency was developed considering that the primary check on compliance with the BPWS is performed by the RWM (LCO 3.3.2.1), which provides control rod blocks to enforce the required sequence and is required to be OPERABLE when operating at ≤ 10% RTP.

REFERENCES

1. NEDE-24011-P-A-10-US, "General Electric Standard Application for Reactor Fuel, Supplement for United States," Section 2.2.3.1, February 1991.

2. Letter (BWROG-8644) from T. Pickens (BWROG) to G. C. Lainas (NRC), "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A."

3. UFSAR, Section 14.6.2.3.

4. **Deleted** ~~NUREG-0800, Section 15.4.9, Revision 2, July 1981.~~

5. ~~10 CFR 100.11~~**10 CFR 50.67.**

(continued)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Standby Liquid Control (SLC) System

BASES

BACKGROUND

The SLC System is designed to provide the capability of bringing the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory (which is at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive, xenon free state without taking credit for control rod movement. The SLC System satisfies the requirements of 10 CFR 50.62 (Ref. 1) on anticipated transient without scram using enriched boron.

The SLC System is also used to maintain suppression pool pH at or above 7 following a loss of coolant accident (LOCA) involving significant fission product releases. Maintaining suppression pool pH levels at or above 7 following an accident ensures that sufficient iodine will be retained in the suppression pool water.

Reference 1 requires a SLC System with a minimum flow capacity and boron content equivalent in control capacity to 86 gpm of 13 weight percent sodium pentaborate solution. Natural sodium pentaborate solution is 19.8% atom Boron-10. Therefore, the system parameters of concern, boron concentration (C), SLC pump flow rate (Q), and Boron-10 enrichment (E), may be expressed as a multiple of ratios. The expression is as follows:

$$\frac{C}{13\%\ \text{weight}} \times \frac{Q}{86\ \text{gpm}} \times \frac{E}{19.8\%\ \text{atom}}$$

If the product of this expression is ≥ 1, then the SLC System satisfies the criteria of Reference 1. As such, the equation forms the basis for acceptance criteria for the surveillances of concentration, flow rate, and boron enrichment and is presented in Table 3.1.7-1.

The SLC System consists of a boron solution storage tank, two positive displacement pumps, two explosive valves that are provided in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). The borated solution is discharged near the bottom of the core shroud, where it then mixes with the cooling water rising through the core. A smaller tank containing demineralized water is provided for testing purposes.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The SLC System is manually initiated from the main control room, as directed by the emergency operating procedures, if the operator believes the reactor cannot be shut down, or kept shut down, with the control rods. The SLC System is used in the event that enough control rods cannot be inserted to accomplish shutdown and cooldown in the normal manner. The SLC System injects borated water into the reactor core to add negative reactivity to compensate for all of the various reactivity effects that could occur during plant operations. To meet this objective, it is necessary to inject a quantity of boron, which produces a concentration of 660 ppm of natural boron, in the reactor coolant at 68°F. To allow for potential leakage and imperfect mixing in the reactor system, an additional amount of boron equal to 25% of the amount cited above is added (Ref. 2). The minimum mass of Boron-10 (162.7 lbm) needed for injection is calculated such that the required quantity is achieved accounting for dilution in the RPV with normal water level and including the water volume in the residual heat removal shutdown cooling piping and in the recirculation loop piping. This quantity of borated solution is the amount that is above the pump suction shutoff level in the boron solution storage tank. No credit is taken for the portion of the tank volume that cannot be injected. The maximum concentration of sodium pentaborate listed in Table 3.1.7-1 has been established to ensure that the solution saturation temperature does not exceed 43°F.

The sodium pentaborate solution in the SLC System is also used, post-LOCA, to maintain suppression pool pH at or above 7. The system parameters used in the calculation are the Boron-10 minimum mass of 162.7 lbm, and an upper bound Boron-10 enrichment of 65%.

The SLC System satisfies Criterion 4 of the NRC Policy Statement.

LCO

The OPERABILITY of the SLC System provides backup capability for reactivity control independent of normal reactivity control provisions provided by the control rods. The OPERABILITY of the SLC System is based on the conditions of the borated solution in the storage tank and the availability of a flow path to the RPV, including the OPERABILITY of the pumps and valves. Two SLC subsystems are required to be OPERABLE; each contains an OPERABLE pump, an explosive valve, and associated piping, valves, and instruments and controls to ensure an OPERABLE flow path.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1 and 2, shutdown capability is required. **In MODES 1, 2, and 3 SLC System injection capability is required in order to maintain post DBA LOCA suppression pool pH.** In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate controls to ensure that the reactor remains subcritical. In MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Demonstration of adequate SDM (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") ensures that the reactor will not become critical. Therefore, the SLC System is not required to be OPERABLE when only a single control rod can be withdrawn.

In MODES 1, 2, and 3, the SLC System must be OPERABLE to ensure that offsite doses remain within 10 CFR 50.67 (Ref. 3) limits following a LOCA involving significant fission product releases. The SLC System is designed to maintain suppression pool pH at or above 7 following a LOCA involving significant fission product releases to ensure that sufficient iodine will be retained in the suppression pool water.

ACTIONS

A.1 and A.2

If the boron solution concentration is > 9.82% weight but the concentration and temperature of boron in solution and pump suction piping temperature are within the limits of Figure 3.1.7-1, operation is permitted for a limited period since the SLC subsystems are capable of performing the intended function. It is not necessary under these conditions to declare both SLC subsystems inoperable since the SLC subsystems are capable of performing their intended function.

The concentration and temperature of boron in solution and pump suction piping temperature must be verified to be within the limits of Figure 3.1.7-1 within 8 hours and once per 12 hours thereafter (Required Action A.1). The temperature versus concentration curve of Figure 3.1.7-1 ensures a 10°F margin will be maintained above the saturation temperature. This verification ensures that boron does not precipitate out of solution in the storage tank or in the pump suction piping due to low boron solution temperature (below the saturation temperature for the given concentration). The Completion Time for performing Required Action A.1 is considered acceptable given the low probability of a Design Basis Accident (DBA) or transient occurring concurrent with the failure of the control rods to shut down the reactor and operating experience which has shown there are relatively slow variations in the measured parameters of concentration and temperature over these time periods.

(continued)

BASES

ACTIONS

D.1 **and D.2** (continued)

brought to MODE 3 within 12 hours **and MODE 4 within 36 hours.** The allowed Completion Time**s are** ~~of 12 hours is~~ reasonable, based on operating experience, to reach ~~MODE 3~~**the required MODES** from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1, SR 3.1.7.2, and SR 3.1.7.3

SR 3.1.7.1 through SR 3.1.7.3 are 24 hour Surveillances verifying certain characteristics of the SLC System (e.g., the level and temperature of the borated solution in the storage tank), thereby ensuring SLC System OPERABILITY without disturbing normal plant operation. These Surveillances ensure that the proper borated solution level and temperature, including the temperature of the pump suction piping, are maintained. Maintaining a minimum specified borated solution temperature is important in ensuring that the boron remains in solution and does not precipitate out in the storage tank or in the pump suction piping. The temperature limit specified in SR 3.1.7.2 and SR 3.1.7.3 and the maximum sodium pentaborate concentration specified in Table 3.1.7-1 ensures that a 10°F margin will be maintained above the saturation temperature. Control room alarms for low SLC storage tank temperature and low SLC System piping temperature are available and are set at 55°F. As such, SR 3.1.7.2 and SR 3.1.7.3 may be satisfied by verifying the absence of low temperature alarms for the SLC storage tank and SLC System piping. The 24 hour Frequency is based on operating experience and has shown there are relatively slow variations in the measured parameters of level and temperature.

SR 3.1.7.4 and SR 3.1.7.6

SR 3.1.7.4 verifies the continuity of the explosive charges in the injection valves to ensure that proper operation will occur if required. Other administrative controls, such as those that limit the shelf life of the explosive charges, must be followed. The 31 day Frequency is based on operating experience and has demonstrated the reliability of the explosive charge continuity.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.9 (continued)

Surveillance when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.1.7.10

Enriched sodium pentaborate solution is made by mixing granular, enriched sodium pentaborate with water. In order to ensure the proper B-10 atom percentage (in accordance with Table 3.1.7-1) is being used, calculations must be performed to verify the actual B-10 enrichment within 8 hours after addition of the solution to the SLC tank. The calculations may be performed using the results of isotopic tests on the granular sodium pentaborate or vendor certification documents. The Frequency is acceptable considering that boron enrichment is verified during the procurement process and any time boron is added to the SLC tank.

REFERENCES

1. 10 CFR 50.62.

2. UFSAR, Section 3.8.4.

3. **10 CFR 50.67.**

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

BASES

BACKGROUND

The SDV vent and drain valves are normally open and discharge any accumulated water in the SDV to ensure that sufficient volume is available at all times to allow a complete scram. During a scram, the SDV vent and drain valves close to contain reactor water. As discussed in Reference 1, the SDV vent and drain valves need not be considered primary containment isolation valves (PCIVs) for the Scram Discharge System. (However, at PBAPS, these valves are considered PCIVs.) The SDV is a volume of header piping that connects to each hydraulic control unit (HCU) and drains into an instrument volume. There are two SDVs (headers) and a common instrument volume that receives all of the control rod drive (CRD) discharges. The instrument volume is connected to a common drain line with two valves in series. Each header is connected to a common vent line with two valves in series for a total of four vent valves. The header piping is sized to receive and contain all the water discharged by the CRDs during a scram. The design and functions of the SDV are described in Reference 2.

APPLICABLE

The Design Basis Accident and transient analyses assume all SAFETY ANALYSES of the control rods are capable of scramming. The acceptance criteria for the SDV vent and drain valves are that they operate automatically to close during scram to limit the amount of reactor coolant discharged so that adequate core cooling is maintained and offsite doses remain within the limits of ~~10 CFR 100~~**10 CFR 50.67** (Ref. 3).

Isolation of the SDV can also be accomplished by manual closure of the SDV valves. Additionally, the discharge of reactor coolant to the SDV can be terminated by scram reset or closure of the HCU manual isolation valves. For a bounding leakage case, the offsite doses are well within the limits of ~~10 CFR 100~~**10 CFR 50.67** (Ref. 3), and adequate core cooling is maintained (Ref. 1). The SDV vent and drain valves allow continuous drainage of the SDV during normal plant operation to ensure that the SDV has sufficient capacity to contain the reactor coolant discharge during a full core scram. To automatically ensure this capacity, a reactor scram (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation") is initiated if the SDV water level in the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.8.3 (continued)

unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components will pass the Surveillance when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," August 1981.

2. UFSAR, Sections 3.4.5.3.1 and 7.2.3.6.

3. ~~10 CFR 100~~**10 CFR 50.67.**

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

BASES

BACKGROUND

The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including abnormal operational transients. Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure, or inability to cool the fuel does not occur during the anticipated operating conditions identified in Reference 1.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel system design are presented in References 1, 2, 3, 4, 5, 6, 7, and 8. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation, and protection system) that fuel damage will not result in the release of radioactive materials in excess of the guidelines of 10 CFR, Parts 20, 50, and 100, **as applicable**. The mechanisms that could cause fuel damage during operational transients and that are considered in fuel evaluations are:

a. Rupture of the fuel rod cladding caused by strain from the relative expansion of the UO_2 pellet; and

b. Severe overheating of the fuel rod cladding caused by inadequate cooling.

A value of 1% plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur (Ref. 9).

Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit is not exceeded during continuous operation with LHGRs up to the operating limit specified in the COLR. The analysis also

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.a. Reactor Vessel Water Level–Low Low Low (Level 1) (continued)

The Reactor Vessel Water Level–Low Low Low (Level 1) Allowable Value is chosen to be the same as the ECCS Level 1 Allowable Value (LCO 3.3.5.1) to ensure that the MSLs isolate on a potential loss of coolant accident (LOCA) to prevent offsite doses from exceeding ~~10 CFR 100~~**10 CFR 50.67** limits.

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

1.b. Main Steam Line Pressure–Low

Low MSL pressure indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure–Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 3). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 785 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 25% RTP.)

The MSL low pressure signals are initiated from four transmitters that are connected to the MSL header. The transmitters are arranged such that, even though physically separated from each other, each transmitter is able to detect low MSL pressure. Four channels of Main Steam Line Pressure –Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure–Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 1).

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.c. Main Steam Line Flow-High

Main Steam Line Flow-High is provided to detect a break of the MSL and to initiate closure of the MSIVs. If the steam were allowed to continue flowing out of the break, the reactor would depressurize and the core could uncover. If the RPV water level decreases too far, fuel damage could occur. Therefore, the isolation is initiated on high flow to prevent or minimize core damage. The Main Steam Line Flow-High Function is directly assumed in the analysis of the main steam line break (MSLB) (Ref. 3). The isolation action, along with the scram function of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46 and offsite doses do not exceed the ~~10 CFR 100~~**10 CFR 50.67** limits.

The MSL flow signals are initiated from 16 transmitters that are connected to the four MSLs. The transmitters are arranged such that, even though physically separated from each other, all four connected to one MSL would be able to detect the high flow. Four channels of Main Steam Line Flow-High Function for each MSL (two channels per trip system) are available and are required to be OPERABLE so that no single instrument failure will preclude detecting a break in any individual MSL.

The Allowable Value is chosen to ensure that offsite dose limits are not exceeded due to the break.

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

1.d. Main Steam Line-High Radiation

The Main Steam Line-High Radiation Function is provided to detect gross release of fission products from the fuel and to initiate closure of the MSIVs. The trip setting is set low enough so that a high radiation trip results from a design basis rod drop accident and high enough above background radiation levels in the vicinity of the main steam lines so that spurious trips at rated power are avoided. The Main Steam Line-High Radiation Function is directly assumed in the analysis of the control rod drop accident (Ref. 3).

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.f Reactor Building Main Steam Tunnel Temperature-High (continued)

The Allowable Value is chosen to detect a leak equivalent to between 1% and 10% rated steam flow.

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

Primary Containment Isolation

2.a. Reactor Vessel Water Level-Low (Level 3)

Low RPV water level indicates that the capability to cool the fuel may be threatened. The valves whose penetrations communicate with the primary containment are isolated to limit the release of fission products. The isolation of the primary containment on Level 3 supports actions to ensure that offsite dose limits of ~~10 CFR 100~~**10 CFR 50.67** are not exceeded.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.a. Reactor Vessel Water Level–Low (Level 3) (continued)

The Reactor Vessel Water Level–Low (Level 3) Function associated with isolation is implicitly assumed in the UFSAR analysis as these leakage paths are assumed to be isolated post LOCA.

Reactor Vessel Water Level–Low (Level 3) signals are initiated from level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level–Low (Level 3) Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level–Low (Level 3) Allowable Value was chosen to be the same as the RPS Level 3 scram Allowable Value (LCO 3.3.1.1), since isolation of these valves is not critical to orderly plant shutdown.

This Function isolates the Group II(A) valves listed in Reference 1 with the exception of RWCU isolation valves and RHR shutdown cooling pump suction valves which are addressed in Functions 5.c and 6.b, respectively.

2.b. Drywell Pressure–High

High drywell pressure can indicate a break in the RCPB inside the primary containment. The isolation of some of the primary containment isolation valves on high drywell pressure supports actions to ensure that offsite dose limits of ~~10 CFR 100~~**10 CFR 50.67** are not exceeded. The Drywell Pressure–High Function, associated with isolation of the primary containment, is implicitly assumed in the UFSAR accident analysis as these leakage paths are assumed to be isolated post LOCA.

High drywell pressure signals are initiated from pressure transmitters that sense the pressure in the drywell. Four channels of Drywell Pressure–High are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

5.a. RWCU Flow-High (continued)

The high RWCU flow signals are initiated from transmitters that are connected to the pump suction line of the RWCU System. Two channels of RWCU Flow-High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The RWCU Flow-High Allowable Value ensures that a break of the RWCU piping is detected.

This Function isolates the inboard and outboard RWCU pump suction penetration and the outboard valve at the RWCU connection to reactor feedwater.

5.b. Standby Liquid Control (SLC) System Initiation

The isolation of the RWCU System is required when the SLC System has been initiated to prevent dilution and removal of the boron solution by the RWCU System (Ref. 5). SLC System initiation signals are initiated from the remote SLC System start switch.

There is no Allowable Value associated with this Function since the channels are mechanically actuated based solely on the position of the SLC System initiation switch.

For reactivity insertion accidents, ~~T~~two channels of the SLC System Initiation Function are available and are required to be OPERABLE ~~only~~ in MODES 1 and 2, since these are the only MODES where the reactor can be critical.~~, and these~~ **In addition, for accidents involving significant fission product releases, both channels are required to be OPERABLE in MODES 1, 2, and 3. The SLC System is designed to maintain suppression pool pH at or above 7 following a LOCA to ensure that sufficient iodine will be retained in the suppression pool water. These** MODES are consistent with the Applicability for the SLC System (LCO 3.1.7).

This Function isolates the inboard and outboard RWCU pump suction penetration and the outboard valve at the RWCU connection to reactor feedwater.

5.c. Reactor Vessel Water Level-Low (Level 3)

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of some interfaces with the reactor vessel occurs to isolate the potential sources of a break. The isolation of the RWCU System on Level 3 supports actions to ensure that the fuel

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3, 4. Reactor Building Ventilation and Refueling Floor Ventilation Exhaust Radiation—High (continued)

channels of Reactor Building Ventilation Exhaust Radiation—High Function and four channels of Refueling Floor Ventilation Exhaust Radiation—High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Values are chosen to promptly detect gross failure of the fuel cladding.

The Reactor Building Ventilation and Refueling Floor Ventilation Exhaust Radiation—High Functions are required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists; thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. In MODES 4 and 5, the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES; thus, these Functions are not required. In addition, the Functions are also required to be OPERABLE during ~~CORE ALTERATIONS,~~ OPDRVs~~,~~ and movement of ~~irradiated fuel~~**RECENTLY IRRADIATED FUEL** assemblies in the secondary containment, because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery or dropped fuel assemblies) must be provided to ensure that offsite dose limits are not exceeded.

ACTIONS

A Note has been provided to modify the ACTIONS related to secondary containment isolation instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable secondary containment isolation instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable secondary containment isolation instrumentation channel.

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Specific Activity

BASES

BACKGROUND

During circulation, the reactor coolant acquires radioactive materials due to release of fission products from fuel leaks into the reactor coolant and activation of corrosion products in the reactor coolant. These radioactive materials in the reactor coolant can plate out in the RCS, and, at times, an accumulation will break away to spike the normal level of radioactivity. The release of coolant during a Design Basis Accident (DBA) could send radioactive materials into the environment.

~~———~~Limits on the maximum allowable level of radioactivity in the reactor coolant are established to ensure that in the event of a release of any radioactive material to the environment during a DBA, radiation doses are maintained within the limits of ~~10 CFR 100~~**10 CFR 50.67** (Ref. 1).

This LCO contains the iodine specific activity limits. The iodine isotopic activities per gram of reactor coolant are expressed in terms of a DOSE EQUIVALENT I-131. The allowable level is intended to limit the **maximum** 2 hour radiation dose to an individual at the site boundary to ~~well~~ within the ~~10 CFR 100~~**10 CFR 50.67** limit **as modified in Regulatory Guide 1.183, Table 6.**

APPLICABLE SAFETY ANALYSES

Analytical methods and assumptions involving radioactive material in the primary coolant are presented in the UFSAR (Ref. 2). The specific activity in the reactor coolant (the source term) is an initial condition for evaluation of the consequences of an accident due to a main steam line break (MSLB) outside containment. No fuel damage is postulated in the MSLB accident, and the release of radioactive material to the environment is assumed to end when the main steam isolation valves (MSIVs) close completely.

~~———~~This MSLB release forms the basis for determining offsite doses (Ref. 2). The limits on the specific activity of the primary coolant ensure that the **maximum** 2 hour ~~thyroid and whole body~~**TEDE** doses at the site boundary, resulting from an MSLB ~~—~~outside containment during steady state operation, will not exceed the dose guidelines of ~~10 CFR 100~~**10 CFR 50.67 as modified in Regulatory Guide 1.183, Table 6.**

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limits on specific activity are values from a parametric evaluation of typical site locations. These limits are conservative because the evaluation considered more restrictive parameters than for a specific site, such as the location of the site boundary and the meteorological conditions of the site.

RCS specific activity satisfies Criterion 2 of the NRC Policy Statement.

LCO

The specific iodine activity is limited to ≤ 0.2 μCi/gm DOSE EQUIVALENT I-131. This limit ensures the source term assumed in the safety analysis for the MSLB is not exceeded, so any release of radioactivity to the environment during an MSLB is ~~well~~ within the ~~10 CFR 100~~**10 CFR 50.67** limits **as modified in Regulatory Guide 1.183, Table 6.**

APPLICABILITY

In MODE 1, and MODES 2 and 3 with any main steam line not isolated, limits on the primary coolant radioactivity are applicable since there is an escape path for release of radioactive material from the primary coolant to the environment in the event of an MSLB outside of primary containment.

In MODES 2 and 3 with the main steam lines isolated, such limits do not apply since an escape path does not exist. In MODES 4 and 5, no limits are required since the reactor is not pressurized and the potential for leakage is reduced.

ACTIONS

A.1 and A.2

When the reactor coolant specific activity exceeds the LCO DOSE EQUIVALENT I-131 limit, but is ≤ 4.0 μCi/gm, samples must be analyzed for DOSE EQUIVALENT I-131 at least once every 4 hours. In addition, the specific activity must be restored to the LCO limit within 48 hours. The Completion Time of once every 4 hours is based on the time needed to take and analyze a sample. The 48 hour Completion Time to restore the activity level provides a reasonable time for temporary coolant activity increases (iodine spikes) to be cleaned up with the normal processing systems.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S) while relying on the ACTIONS. This allowance 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 plant remains at, or proceeds to, power operation.

B.1, B.2.1, B.2.2.1, and B.2.2.2

If the DOSE EQUIVALENT I-131 cannot be restored to ≤ 0.2 μCi/gm within 48 hours, or if at any time it is > 4.0 μCi/gm, it must be determined at least once every 4 hours and all the main steam lines must be isolated within 12 hours. Isolating the main steam lines precludes the possibility of releasing radioactive material to the environment in an amount that is more than ~~a small fraction of~~ the requirements of ~~10 CFR 100~~**10 CFR 50.67 as modified in Regulatory Guide 1.183, Table 6**, during a postulated MSLB accident.

Alternatively, the plant can be placed in MODE 3 within 12 hours and in MODE 4 within 36 hours. This option is provided for those instances when isolation of main steam lines is not desired (e.g., due to the decay heat loads). In MODE 4, the requirements of the LCO are no longer applicable.

The Completion Time of once every 4 hours is the time needed to take and analyze a sample. The 12 hour Completion Time is reasonable, based on operating experience, to isolate the main steam lines in an orderly manner and without challenging plant systems. Also, the allowed Completion Times for Required Actions B.2.2.1 and B.2.2.2 for placing the unit in MODES 3 and 4 are reasonable, based on operating experience, to achieve the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This Surveillance is performed to ensure iodine remains within limit during normal operation. The 7 day Frequency is adequate to trend changes in the iodine activity level.

This SR is modified by a Note that requires this Surveillance to be performed only in MODE 1 because the level of fission products generated in other MODES is much less.

REFERENCES

1. ~~10 CFR 100.11, 1973~~**10 CFR 50.67.**
2. UFSAR, Section 14.6.5.

BASES (continued)

APPLICABLE SAFETY ANALYSES

The safety design basis for the primary containment is that it must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE such that release of fission products to the environment is controlled by the rate of primary containment leakage.

Analytical methods and assumptions involving the primary containment are presented in Reference 1. The safety analyses assume a nonmechanistic fission product release following a DBA, which forms the basis for determination of offsite doses. The fission product release is, in turn, based on an assumed leakage rate from the primary containment. OPERABILITY of the primary containment ensures that the leakage rate assumed in the safety analyses is not exceeded.

The maximum allowable leakage rate for the primary containment (L_a) is ~~0.5~~**0.7%** by weight of the containment air per 24 hours at the design basis LOCA maximum peak containment pressure (P_a) of 49.1 psig. The value of P_a (49.1 psig) is conservative with respect to the current calculated peak drywell pressure of 47.2 psig (Ref. 2). This value is 47.8 psig for operation with 90°F Final Feedwater Temperature Reduction (Ref. 7).

Primary containment satisfies Criterion 3 of the NRC Policy Statement.

LCO

Primary containment OPERABILITY is maintained by limiting leakage to ≤ 1.0 L_a, except prior to the first startup after performing a required Primary Containment Leakage Rate Testing Program leakage test. At this time, applicable leakage limits must be met. In addition, the leakage from the drywell to the suppression chamber must be limited to ensure the pressure suppression function is accomplished and the suppression chamber pressure does not exceed design limits. Compliance with this LCO will ensure a primary containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analyses.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE, such that release of fission products to the environment is controlled by the rate of primary containment leakage. The primary containment is designed with a maximum allowable leakage rate (L_a) of ~~0.5~~**0.7%** by weight of the containment air per 24 hours at the maximum peak containment pressure (P_a) of 49.1 psig. The value of P_a (49.1 psig) is conservative with respect to the current calculated peak drywell pressure of 47.2 psig (Ref. 3).
This value is 47.8 psig for operation with 90°F Final Feedwater Temperature Reduction (Ref. 4). This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock.

Primary containment air lock OPERABILITY is also required to minimize the amount of fission product gases that may escape primary containment through the air lock and contaminate and pressurize the secondary containment.

The primary containment air lock satisfies Criterion 3 of the NRC Policy Statement.

LCO

As part of primary containment, the air lock's safety function is related to control of containment leakage rates following a DBA. Thus, the air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

The primary containment 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 to be opened at a time. This provision ensures that a gross breach of primary containment does not exist when primary 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 and exit from primary containment.

(continued)

BASES

ACTIONS

D.1 (continued)

rate for the isolated penetration is assumed to be the actual pathway leakage through the isolation device. If two isolation devices are used to isolate the penetration, the leakage rate is assumed to be the lesser actual pathway leakage of the two devices. The 8 hour Completion Time is reasonable considering the time required to restore the leakage by isolating the penetration, the fact that MSIV closure will result in isolation of the main steam line and a potential for plant shutdown, and the relative importance of MSIV leakage to the overall containment function.

E.1, E.2, and E.3

The accumulated time that the large containment purge and/or vent valves (6" and 18" vent valves) are open, when reactor pressure is greater than 100 psig and the reactor is in MODES 1 or 2, is limited to 90 hours per calendar year. This will limit the total time that a flow path exists through certain containment penetrations and ensures that there is no significant threat to the design analysis for crediting containment overpressure for ECCS NPSH. Consequently, there exists minimal impact on plant risk resulting from challenges to ECCS NPSH during a LOCA while purging. The 4-hour Completion Time to isolate the penetration is considered a reasonable amount of time to ensure compliance with the design analysis for containment overpressure. If the penetration is not isolated within the specified 4-hour time period, then the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

~~E~~F.1 and ~~E~~F.2

If any Required Action and associated Completion Time cannot be met in MODE 1, 2, or 3, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

~~F~~G.1 and ~~F~~G.2

If any Required Action and associated Completion Time cannot be met for PCIV(s) required to be OPERABLE during MODE 4 or 5, the unit must be placed in a condition in which the LCO does not apply. Action must be immediately initiated to suspend operations with a potential for draining the reactor vessel (OPDRVs) to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended and valve(s) are restored to OPERABLE status. If suspending an OPDRV would result in closing the residual heat removal (RHR) shutdown cooling isolation valves, an alternative Required Action is provided to immediately initiate action to restore the valve(s) to OPERABLE status. This allows RHR to remain in service while actions are being taken to restore the valve.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.7 (continued)

position, since these valves were verified to be in the correct position prior to locking or securing. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.6.1.3.8

Verifying the isolation time of each power operated automatic PCIV is within limits is required to demonstrate OPERABILITY. MSIVs may be excluded from this SR since MSIV full closure isolation time is demonstrated by SR 3.6.1.3.9. The isolation time test ensures that the valve will isolate in a time period less than or equal to that assumed in the safety analyses. The isolation time is in accordance with Reference 2 or the requirements of the Inservice Testing Program which ever is more conservative. The Frequency of this SR is in accordance with the requirements of the Inservice Testing Program.

SR 3.6.1.3.9

Verifying that the isolation time of each MSIV is within the specified limits is required to demonstrate OPERABILITY. The isolation time test ensures that the MSIV will isolate in a time period that does not exceed the times assumed in the DBA analyses. This ensures that the calculated radiological consequences of these events remain within ~~10 CFR 100~~**10 CFR 50.67** limits **as modified in Regulatory Guide 1.183, Table 6.** The Frequency of this SR is in accordance with the requirements of the Inservice Testing Program.

SR 3.6.1.3.10

Automatic PCIVs close on a primary containment isolation signal to prevent leakage of radioactive material from primary containment following a DBA. This SR ensures that each automatic PCIV will actuate to its isolation position on a primary containment isolation signal. The LOGIC SYSTEM

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.13

This SR ensures that in case the non-safety grade instrument air system is unavailable, the SGIG System will perform its design function to supply nitrogen gas at the required pressure for valve operators and valve seals supported by the SGIG System. The 24 month Frequency was developed considering it is prudent that this Surveillance be performed only during a plant outage. Operating experience has shown that these components will usually pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.3.14

~~Leakage through each MSIV must be ≤ 11.5 scfh when tested at ≥ P_t (25 psig). The analyses in Reference 1 are based on treatment of MSIV leakage as a secondary containment bypass leakage, independent of a primary to secondary containment leakage analyzed at 1.27 L_a. In the Reference 1 analysis all 4 steam lines are assumed to leak at the TS Limit. This ensures that MSIV leakage is properly accounted for in determining the overall impacts of primary containment leakage. The Frequency is required by the Primary Containment Leakage Rate Testing Program.~~**Total leakage through all four main steam lines must be ≤ 204 scfh, and ≤ 116 scfh for any one steam line, when tested at ≥ 25 psig. The analysis in Reference 1 is based on treatment of MSIV leakage as secondary containment bypass leakage, independent of the primary to secondary leakage analyzed at L_a. The Frequency is in accordance with the Primary Containment Leakage Rate Testing Program.**

SR 3.6.1.3.15

Verifying the opening of each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve is restricted by a blocking device to less than or equal to the required maximum opening angle specified in the UFSAR (Ref. 4) is required to ensure that the valves can close under DBA conditions within the times in the analysis of Reference 1. If a LOCA occurs, the purge and exhaust valves must close to maintain primary containment leakage within the values assumed in the accident analysis. At other times pressurization concerns are not present, thus the purge and exhaust valves can be fully open. The 24 month Frequency is appropriate because the blocking devices may be removed during a refueling outage.

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.1 Secondary Containment

BASES

BACKGROUND

The function of the secondary containment is to contain and hold up fission products that may leak from primary containment following a Design Basis Accident (DBA). In conjunction with operation of the Standby Gas Treatment (SGT) System and closure of certain valves whose lines penetrate the secondary containment, the secondary containment is designed to reduce the activity level of the fission products prior to release to the environment and to isolate and contain fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be OPERABLE, or that take place outside primary containment.

The secondary containment is a structure that completely encloses the primary containment and those components that may be postulated to contain primary system fluid. This structure forms a control volume that serves to hold up and dilute the fission products. It is possible for the pressure in the control volume to rise relative to the environmental pressure (e.g., due to pump and motor heat load additions). To prevent ground level exfiltration while allowing the secondary containment to be designed as a conventional structure, the secondary containment requires support systems to maintain the control volume pressure at less than the external pressure. Requirements for these systems are specified separately in LCO 3.6.4.2, "Secondary Containment Isolation Valves (SCIVs)," and LCO 3.6.4.3, "Standby Gas Treatment (SGT) System."

APPLICABLE SAFETY ANALYSES

There are two principal accidents for which credit is taken for secondary containment OPERABILITY. These are a loss of coolant accident (LOCA) (Ref. 1) and a fuel handling accident inside secondary containment (Ref. 2) **involving RECENTLY IRRADIATED FUEL.** The secondary containment performs no active function in response to each of these limiting events; however, its leak

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	tightness is required to ensure that fission products entrapped within the secondary containment structure will be treated by the SGT System prior to discharge to the environment. Secondary containment satisfies Criterion 3 of the NRC Policy Statement.
LCO	An OPERABLE secondary containment provides a control volume into which fission products that leak from primary containment, or are released from the reactor coolant pressure boundary components located in secondary containment, can be processed prior to release to the environment. For the secondary containment to be considered OPERABLE, it must have adequate leak tightness to ensure that the required vacuum can be established and maintained.
APPLICABILITY	In MODES 1, 2, and 3, a LOCA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, secondary containment OPERABILITY is required during the same operating conditions that require primary containment OPERABILITY. In MODES 4 and 5, the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining secondary containment OPERABLE is not required in MODE 4 or 5~~ to ensure a control volume~~, except for other situations for which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs), ~~during CORE ALTERATIONS~~, or during movement of ~~irradiated fuel~~**RECENTLY IRRADIATED FUEL** assemblies in the secondary containment. **However, outside ground level hatches (hatches H15 through H24 and Torus room access hatches) may not be opened during movement of irradiated fuel unless a decay period of 84 hours after shutdown has occurred. This will maintain CR doses acceptable.**
ACTIONS	A.1 If secondary containment is inoperable, it must be restored to OPERABLE status within 4 hours. The 4 hour Completion Time provides a period of time to correct the problem that is commensurate with the importance of maintaining secondary containment during MODES 1, 2, and 3. This time period also ensures that the probability of an accident (requiring secondary containment OPERABILITY) occurring during periods where secondary containment is inoperable is minimal.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If secondary containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1~~, and C.3~~ **and** C.2

Movement of ~~irradiated fuel~~**RECENTLY IRRADIATED FUEL** assemblies in the secondary containment~~, CORE ALTERATIONS,~~ and OPDRVs can be postulated to cause fission product release to the secondary containment. In such cases, the secondary containment is the only barrier to release of fission products to the environment. ~~CORE ALTERATIONS and~~**Therefore**, movement of ~~irradiated fuel~~**RECENTLY IRRADIATED FUEL** assemblies must be immediately suspended if the secondary containment is inoperable.

Suspension of these activities shall not preclude completing an action that involves moving a component to a safe position. Also, action must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

Required Action C.1 has been modified by a Note stating that LCO 3.0.3 is not applicable~~. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown~~, **since the movement of RECENTLY IRRADIATED FUEL can only be performed in MODES 4 and 5.**

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.1 and SR 3.6.4.1.2

Verifying that secondary containment equipment hatches and one access door in each access opening are closed ensures that the infiltration of outside air of such a magnitude as to prevent maintaining the desired negative pressure does not occur. Verifying that all such openings are closed provides adequate assurance that exfiltration from the secondary containment will not occur. In this application, the term "sealed" has no connotation of leak tightness. Maintaining secondary containment OPERABILITY requires verifying one door in the access opening is closed. An access opening contains one inner and one outer door. In some cases, secondary containment access openings are shared such that a secondary containment barrier may have multiple inner or multiple outer doors. The intent is to not breach secondary containment at any time when secondary containment is required. This is achieved by maintaining the inner or outer portion of the barrier closed at all times. However, all secondary containment access doors are normally kept closed, except when the access opening is being used for entry and exit or when maintenance is being performed on an access opening. The 31 day Frequency for these SRs has been shown to be adequate, based on operating experience, and is considered adequate in view of the other indications of door and hatch status that are available to the operator.

SR 3.6.4.1.3 and SR 3.6.4.1.4

The SGT System exhausts the secondary containment atmosphere to the environment through appropriate treatment equipment. Each SGT subsystem is designed to draw down pressure in the secondary containment to ≥ 0.25 inches of vacuum water gauge in ≤ ~~120~~ **180** seconds and maintain pressure in the secondary containment at ≥ 0.25 inches of vacuum water gauge for 1 hour at a flow rate ≤ 10,500 cfm. To ensure that all fission products released to the secondary containment are treated, SR 3.6.4.1.3 and SR 3.6.4.1.4 verify that a pressure in the secondary containment that is less than the lowest postulated pressure external to the secondary containment boundary can rapidly be established and maintained. When the SGT System is operating as designed, the establishment and maintenance of secondary containment pressure cannot be accomplished if the secondary containment boundary is not intact. Establishment of this pressure is confirmed by SR 3.6.4.1.3 which demonstrates that the secondary containment can be drawn down to ≥ 0.25 inches of vacuum water gauge in ≤ ~~120~~ **180**

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

BASES

BACKGROUND

The function of the SCIVs, in combination with other accident mitigation systems, is to ~~limit~~ **control** fission product release during and following postulated Design Basis Accidents (DBAs) (Refs. 1 and 2). Secondary containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that fission products that leak from primary containment following a DBA, or that are released during certain operations when primary containment is not required to be OPERABLE or take place outside primary containment, are maintained within the secondary containment boundary.

The OPERABILITY requirements for SCIVs help ensure that an adequate secondary containment boundary is maintained during and after an accident by minimizing potential paths to the environment. These isolation devices consist of either passive devices or active (automatic) devices. Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), and blind flanges are considered passive devices.

Automatic SCIVs close on a secondary containment isolation signal to establish a boundary for untreated radioactive material within secondary containment following a DBA or other accidents.

Other penetrations are isolated by the use of valves in the closed position or blind flanges.

APPLICABLE SAFETY ANALYSES

The SCIVs must be OPERABLE to ensure the secondary containment barrier to fission product releases is established. The principal accidents for which the secondary containment boundary is required are a loss of coolant accident (Ref. 1) and a fuel handling accident inside secondary containment (Ref. 2) **involving RECENTLY IRRADIATED FUEL.** The secondary containment performs no active function in response to either of these limiting events, but the boundary

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

established by SCIVs is required to ensure that leakage from the primary containment is processed by the Standby Gas Treatment (SGT) System before being released to the environment.

Maintaining SCIVs OPERABLE with isolation times within limits ensures that fission products will remain trapped inside secondary containment so that they can be treated by the SGT System prior to discharge to the environment.

SCIVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

SCIVs form a part of the secondary containment boundary. The SCIV safety function is related to control of offsite radiation releases resulting from DBAs.

The power operated automatic isolation valves are considered OPERABLE when their isolation times are within limits and the valves actuate on an automatic isolation signal. The valves covered by this LCO, along with their associated stroke times, are listed in Reference ~~3~~**2**.

The normally closed isolation valves or blind flanges are considered OPERABLE when manual valves are closed or open in accordance with appropriate administrative controls, automatic SCIVs are de-activated and secured in their closed position, and blind flanges are in place. These passive isolation valves or devices are listed in Reference ~~3~~**2**.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could lead to a fission product release to the primary containment that leaks to the secondary containment. Therefore, the OPERABILITY of SCIVs is required.

In MODES 4 and 5, the probability and consequences of these events are reduced due to pressure and temperature limitations in these MODES. Therefore, maintaining SCIVs OPERABLE is not required in MODE 4 or 5, except for other situations under which significant radioactive releases can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs)~~, during CORE ALTERATIONS,~~ or during movement of ~~irradiated fuel~~**RECENTLY IRRADIATED FUEL** assemblies in the secondary containment. **SCIVs are only required to be OPERABLE during OPDRVs or handling RECENTLY IRRADIATED FUEL.** Moving irradiated fuel assemblies in the secondary containment may also occur in MODES 1, 2, and 3.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

If any Required Action and associated Completion Time cannot be met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1~~,~~ **and** D.2~~, and D.3~~

If any Required Action and associated Completion Time are not met, the plant must be placed in a condition in which the LCO does not apply. If applicable, ~~CORE ALTERATIONS and~~ the movement of ~~irradiated fuel~~**RECENTLY IRRADIATED FUEL** assemblies in the secondary containment must be immediately suspended. Suspension of ~~these activities~~**this activity** shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be immediately initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

Required Action D.1 has been modified by a Note stating that LCO 3.0.3 is not applicable~~. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving fuel while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.~~, **since the movement of RECENTLY IRRADIATED FUEL can only be performed in MODES 4 and 5.**

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.1

This SR verifies that each secondary containment manual isolation valve and blind flange that is not locked, sealed, or otherwise secured and is required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the secondary containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification that those SCIVs in secondary containment that are capable of being mispositioned are in the correct position.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.3 (continued)

under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components will usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section ~~14.6.3~~**14.9.2.**

~~2. UFSAR, Section 14.6.4.~~

~~3~~**2.** Technical Requirements Manual.

BASES

LCO (continued)

For Unit 2, one SGT subsystem is OPERABLE when one charcoal filter train, one fan (0AV020) and associated ductwork, dampers, valves, and controls are OPERABLE. The second SGT subsystem is OPERABLE when the other charcoal filter train, one fan (0BV020) and associated ductwork, damper, valves, and controls are OPERABLE.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, SGT System OPERABILITY is required during these MODES.

In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the SGT System in OPERABLE status is not required in MODE 4 or 5, except for other situations under which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs), ~~during CORE ALTERATIONS,~~ or during movement of ~~irradiated fuel~~**RECENTLY IRRADIATED FUEL** assemblies in the secondary containment. **The SGT System is only required to be OPERABLE during OPDRVs or handling of RECENTLY IRRADIATED FUEL.**

ACTIONS

A.1

With one SGT subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status in 7 days. In this Condition, the remaining OPERABLE SGT subsystem is adequate to perform the required radioactivity release control function. However, the overall system reliability is reduced because a single failure in the OPERABLE subsystem could result in the radioactivity release control function not being adequately performed. The 7 day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant SGT subsystem and the low probability of a DBA occurring during this period.

B.1 and B.2

If the SGT subsystem cannot be restored to OPERABLE status within the required Completion Time in MODE 1, 2, or 3, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1, C.2.1, ~~C.2.2, and C.2.3~~**and C.2.2**

During movement of ~~irradiated fuel~~**RECENTLY IRRADIATED FUEL** assemblies, in the secondary containment~~, during CORE ALTERATIONS,~~ or during OPDRVs, when Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE SGT subsystem should immediately be placed in operation. This action ensures that the remaining subsystem is OPERABLE, that no failures that could prevent automatic actuation have occurred, and that any other failure would be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that represent a potential for releasing radioactive material to the secondary containment, thus placing the plant in a condition that minimizes risk. If applicable, ~~CORE ALTERATIONS and~~ movement of ~~irradiated fuel~~**RECENTLY IRRADIATED** assemblies must immediately be suspended. Suspension of ~~these activities~~**this activity** must not preclude completion of movement of a component to a safe position. Also, if applicable, actions must immediately be initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

The Required Actions of Condition C have been modified by a Note stating that LCO 3.0.3 is not applicable~~. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.~~, **since the movement of RECENTLY IRRADIATED FUEL can only be performed in MODES 4 and 5.**

(continued)

BASES

ACTIONS (continued)

D.1

If both SGT subsystems are inoperable in MODE 1, 2, or 3, the SGT System may not be capable of supporting the required radioactivity release control function. Therefore, actions are required to enter LCO 3.0.3 immediately.

E.1~~,~~ **and** E.2~~, and E.3~~

When two SGT subsystems are inoperable, if applicable, ~~CORE ALTERATIONS and~~ movement of ~~irradiated fuel~~**RECENTLY IRRADIATED FUEL** assemblies in secondary containment must immediately be suspended. Suspension of ~~these activities~~**this activity** shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must immediately be initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

Required Action E.1 has been modified by a Note stating that LCO 3.0.3 is not applicable~~. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.~~, **since the movement of RECENTLY IRRADIATED FUEL can only be performed in MODES 4 and 5.**

SURVEILLANCE REQUIREMENTS

SR 3.6.4.3.1

Operating each SGT subsystem (including each filter train fan) for ≥ 15 minutes ensures that both subsystems are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. Operation with the heaters on (automatic heater cycling to maintain temperature) for ≥ 15 minutes every 31 days is sufficient to eliminate moisture on the adsorbers and HEPA filters since during idle periods instrument air is injected into the filter plenum to keep the filters dry. The 31 day Frequency was developed in consideration of the known reliability of fan motors and controls and the redundancy available in the system.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.7 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident.

A general description of the spent fuel storage pool design is found in the UFSAR, Section 10.3 (Ref. 1). The assumptions of the fuel handling accident are found in the UFSAR, Section 14.6.4 (Ref. 2).

APPLICABLE SAFETY ANALYSES

The water level above the irradiated fuel assemblies is an implicit assumption of the fuel handling accident. A fuel handling accident is evaluated to ensure that the radiological consequences ~~(calculated whole body and thyroid doses at the site boundary)~~ are ~~well~~ below the guidelines set forth in ~~10 CFR 100~~**10 CFR 50.67** (Ref. 3) **as modified by Regulatory Guide 1.183, Table 6.** A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in Reference 2.

The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto the reactor core. The consequences of a fuel handling accident over the spent fuel storage pool are ~~no more~~**less** severe than those of the fuel handling accident over the reactor core. The water level in the spent fuel storage pool provides for absorption of water soluble fission product gases ~~and transport delays of soluble and insoluble gases that must pass through the water~~ before being released to the secondary containment atmosphere. ~~This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident.~~**Noble gases are not retained in the water and particulates are retained in the water (RG 1.183, Appendix B, Item 3).**

The spent fuel storage pool water level satisfies Criteria 2 and 3 of the NRC Policy Statement.

LCO

The specified water level (232 ft 3 inches plant elevation, which is equivalent to 22 ft over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks) preserves the assumptions of the fuel handling accident analysis (Ref. 2). As such, it is the minimum required for fuel movement within the spent fuel storage pool.

(continued)

BASES (continued)

APPLICABILITY

This LCO applies during movement of fuel assemblies in the spent fuel storage pool since the potential for a release of fission products exists.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. If moving fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of fuel assemblies is not a sufficient reason to require a reactor shutdown.

When the initial conditions for an accident cannot be met, action must be taken to preclude the accident from occurring. If the spent fuel storage pool level is less than required, the movement of fuel assemblies in the spent fuel storage pool is suspended immediately. Suspension of this activity shall not preclude completion of movement of a fuel assembly to a safe position. This effectively precludes a spent fuel handling accident from occurring.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

This SR verifies that sufficient water is available in the event of a fuel handling accident. The water level in the spent fuel storage pool must be checked periodically. The 7 day Frequency is acceptable, based on operating experience, considering that the water volume in the pool is normally stable, and all water level changes are controlled by unit procedures.

REFERENCES

1. UFSAR, Section 10.3.
2. UFSAR, Section 14.6.4.
3. ~~10 CFR 100~~**10 CFR 50.67.**

B 3.9 REFUELING OPERATIONS

B 3.9.6 Reactor Pressure Vessel (RPV) Water Level

BASES

BACKGROUND

The movement of fuel assemblies or handling of control rods within the RPV requires a minimum water level of 458 inches above RPV instrument zero. During refueling, this maintains a sufficient water level in the reactor vessel cavity and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to ~~well~~ below the guidelines set forth in ~~10 CFR 100~~**10 CFR 50.67** (Ref. 3) **as modified by Regulatory Guide 1.183, Table 6.**

APPLICABLE SAFETY ANALYSES

During movement of fuel assemblies or handling of control rods, the water level in the RPV and the spent fuel pool is an implicit initial condition design parameter in the analysis of a fuel handling accident in containment postulated in Reference 1. A minimum water level of 20 ft 11 inches above the top of the RPV flange allows a partition factor of 100 to be used in the accident analysis for halogens (Ref. 1).

Analysis of the fuel handling accident inside containment is described in Reference 1. With a minimum water level of 458 inches above RPV instrument zero (20 ft 11 inches above the top of the RPV flange) and a minimum decay time of 24 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and that offsite doses are maintained within allowable limits (Ref. 3).

While the worst case assumptions include the dropping of an irradiated fuel assembly onto the reactor core, the possibility exists of the dropped assembly striking the RPV flange and releasing fission products. Therefore, the minimum depth for water coverage to ensure acceptable radiological consequences is specified from the RPV flange. Since the worst case event results in failed fuel assemblies seated in the core, as well as the dropped assembly,

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level of 458 inches above RPV instrument zero ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is met. Water at the required level limits the consequences of damaged fuel rods, which are postulated to result from a fuel handling accident in containment (Ref. 1).

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 on valve positions, which make significant unplanned level changes unlikely.

REFERENCES

1. UFSAR, Section 14.6.4.

2. UFSAR, Section 10.3.

3. ~~10 CFR 100.11~~**10 CFR 50.67.**

BASES (continued)

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.

APPLICABILITY

SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.

SAFETY LIMIT VIOLATIONS

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 3) **and 10 CFR 50.67, "Accident Source Term," for accidents analyzed using AST (Ref. 4).** Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

(continued)

BASES

REFERENCES

1. DELETED

2. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," latest approved revision.

3. 10 CFR 100.

4. **10 CFR 50.67.**

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on reactor steam dome pressure protects 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. Establishing an upper limit on reactor steam dome pressure ensures continued RCS integrity with regard to pressure excursions. Per the UFSAR (Ref. 1), the reactor coolant pressure boundary (RCPB) shall be designed with sufficient margin to ensure that the design conditions are not exceeded during normal operation and abnormal operational transients.

During normal operation and abnormal operational transients, 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, in accordance with ASME Code requirements, prior to initial operation when there is no fuel in the core. Any further hydrostatic testing with fuel in the core may be done under LCO 3.10.1, "Inservice Leak and Hydrostatic Testing Operation." Following inception of unit operation, RCS components shall be pressure tested in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB reducing the number of protective barriers designed to prevent radioactive releases from exceeding the limits specified in ~~10 CFR 100, "Reactor Site Criteria"~~**10 CFR 50.67, "Accident Source Term"** (Ref. 4). If this occurred in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere.

APPLICABLE SAFETY ANALYSES

The RCS safety/relief valves and the Reactor Protection System Reactor Pressure–High Function have settings established to ensure that the RCS pressure SL will not be exceeded.

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of ~~10 CFR 100, "Reactor Site Criteria,"~~**10 CFR 50.67, "Accident Source Term,"** limits (Ref. 4). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also assures that the probability of an accident occurring during the period is minimal.

REFERENCES

1. UFSAR, Section 1.5.2.2.

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

(continued)

BASES

REFERENCES (continued)

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

4. ~~10 CFR 100~~**10 CFR 50.67.**

5. ASME, Boiler and Pressure Vessel Code, Section III, 1965 Edition, including Addenda to summer of 1966.

6. ASME, Boiler and Pressure Vessel Code, Section III, 1980 Edition, Addenda to winter of 1981.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Rod Pattern Control

BASES

BACKGROUND

Control rod patterns during startup conditions are controlled by the operator and the rod worth minimizer (RWM) (LCO 3.3.2.1, "Control Rod Block Instrumentation"), so that only specified control rod sequences and relative positions are allowed over the operating range of all control rods inserted to 10% RTP. The sequences limit the potential amount of reactivity addition that could occur in the event of a Control Rod Drop Accident (CRDA).

This Specification assures that the control rod patterns are consistent with the assumptions of the CRDA analyses of References 1 and 2.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the CRDA are summarized in References 1 and 2. CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. These sequences define the potential initial conditions for the CRDA analysis. The RWM (LCO 3.3.2.1) provides backup to operator control of the withdrawal sequences to ensure that the initial conditions of the CRDA analysis are not violated.

Prevention or mitigation of positive reactivity insertion events is necessary to limit the energy deposition in the fuel, thereby preventing significant fuel damage which could result in the undue release of radioactivity. Since the failure consequences for UO_2 have been shown to be insignificant below fuel energy depositions of 300 cal/gm (Ref. 3), the fuel damage limit of 280 cal/gm provides a margin of safety from significant core damage which would result in release of radioactivity (Ref~~s. 4 and~~ 5). Generic evaluations (Refs. 1 and 6) of a design basis CRDA (i.e., a CRDA resulting in a peak fuel energy deposition of 280 cal/gm) have shown that if the peak fuel enthalpy remains below 280 cal/gm, then the maximum reactor pressure will be less than the required ASME Code limits (Ref. 7) and the calculated offsite doses will be well within the required limits (Ref. 5).

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

control rods has less impact on control rod worth than withdrawals have. Required Action B.1 is modified by a Note which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position.

LCO 3.3.2.1 requires verification of control rod movement by a second licensed operator or a qualified member of the technical staff.

When nine or more OPERABLE control rods are not in compliance with BPWS, the reactor mode switch must be placed in the shutdown position within 1 hour. With the mode switch in shutdown, the reactor is shut down, and as such, does not meet the applicability requirements of this LCO. The allowed Completion Time of 1 hour is reasonable to allow insertion of control rods to restore compliance, and is appropriate relative to the low probability of a CRDA occurring with the control rods out of sequence.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

The control rod pattern is verified to be in compliance with the BPWS at a 24 hour Frequency to ensure the assumptions of the CRDA analyses are met. The 24 hour Frequency was developed considering that the primary check on compliance with the BPWS is performed by the RWM (LCO 3.3.2.1), which provides control rod blocks to enforce the required sequence and is required to be OPERABLE when operating at ≤ 10% RTP.

REFERENCES

1. NEDE-24011-P-A-10-US, "General Electric Standard Application for Reactor Fuel, Supplement for United States," Section 2.2.3.1, February 1991.

2. Letter (BWROG-8644) from T. Pickens (BWROG) to G. C. Lainas (NRC), "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A."

3. UFSAR, Section 14.6.2.3.

4. **Deleted**~~NUREG-0800, Section 15.4.9, Revision 2, July 1981.~~

5. ~~10 CFR 100.11~~**10 CFR 50.67.**

(continued)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Standby Liquid Control (SLC) System

BASES

BACKGROUND

The SLC System is designed to provide the capability of bringing the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory (which is at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive, xenon free state without taking credit for control rod movement. The SLC System satisfies the requirements of 10 CFR 50.62 (Ref. 1) on anticipated transient without scram using enriched boron.

The SLC System is used to maintain suppression pool pH at or above 7 following a loss of coolant accident (LOCA) involving significant fission product releases. Maintaining suppression pool pH at or below 7 following an accident ensures that iodine will be retained in the suppression pool water.

Reference 1 requires a SLC System with a minimum flow capacity and boron content equivalent in control capacity to 86 gpm of 13 weight percent sodium pentaborate solution. Natural sodium pentaborate solution is 19.8% atom Boron-10. Therefore, the system parameters of concern, boron concentration (C), SLC pump flow rate (Q), and Boron-10 enrichment (E), may be expressed as a multiple of ratios. The expression is as follows:

$$\frac{C}{13\%\ \text{weight}} \times \frac{Q}{86\ \text{gpm}} \times \frac{E}{19.8\%\ \text{atom}}$$

If the product of this expression is ≥ 1, then the SLC System satisfies the criteria of Reference 1. As such, the equation forms the basis for acceptance criteria for the surveillances of concentration, flow rate, and boron enrichment and is presented in Table 3.1.7-1.

The SLC System consists of a boron solution storage tank, two positive displacement pumps, two explosive valves that are provided in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). The borated solution is discharged near the bottom of the core shroud, where it then mixes with the cooling water rising through the core. A smaller tank containing demineralized water is provided for testing purposes.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The SLC System is manually initiated from the main control room, as directed by the emergency operating procedures, if the operator believes the reactor cannot be shut down, or kept shut down, with the control rods. The SLC System is used in the event that enough control rods cannot be inserted to accomplish shutdown and cooldown in the normal manner. The SLC System injects borated water into the reactor core to add negative reactivity to compensate for all of the various reactivity effects that could occur during plant operations. To meet this objective, it is necessary to inject a quantity of boron, which produces a concentration of 660 ppm of natural boron, in the reactor coolant at 68°F. To allow for potential leakage and imperfect mixing in the reactor system, an additional amount of boron equal to 25% of the amount cited above is added (Ref. 2). The minimum mass of Boron-10 (162.7 lbm) needed for injection is calculated such that the required quantity is achieved accounting for dilution in the RPV with normal water level and including the water volume in the residual heat removal shutdown cooling piping and in the recirculation loop piping. This quantity of borated solution is the amount that is above the pump suction shutoff level in the boron solution storage tank. No credit is taken for the portion of the tank volume that cannot be injected. The maximum concentration of sodium pentaborate listed in Table 3.1.7-1 has been established to ensure that the solution saturation temperature does not exceed 43°F.

The sodium pentaborate solution in the SLC System is also used, post-LOCA, to maintain suppression pool pH at or above 7. The system parameters used in the calculation are the Boron-10 minimum mass of 162.7 lbm, and an upper bound Boron-10 enrichment of 65%.

The SLC System satisfies Criterion 4 of the NRC Policy Statement.

LCO

The OPERABILITY of the SLC System provides backup capability for reactivity control independent of normal reactivity control provisions provided by the control rods. The OPERABILITY of the SLC System is based on the conditions of the borated solution in the storage tank and the availability of a flow path to the RPV, including the OPERABILITY of the pumps and valves. Two SLC subsystems are required to be OPERABLE; each contains an OPERABLE pump, an explosive valve, and associated piping, valves, and instruments and controls to ensure an OPERABLE flow path.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1 and 2, shutdown capability is required. **In MODES 1, 2, and 3 SLC System injection capability is required in order to maintain post DBA LOCA suppression pool pH.** In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate controls to ensure that the reactor remains subcritical. In MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Demonstration of adequate SDM (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") ensures that the reactor will not become critical. Therefore, the SLC System is not required to be OPERABLE when only a single control rod can be withdrawn.

In MODES 1, 2, and 3, the SLC System must be OPERABLE to ensure that offsite does remain within 10 CFR 50.67 (Ref. 3) limits following a LOCA involving significant fission product releases. The SLC System is designed to maintain suppression pool pH at or above 7 following a LOCA involving significant fission product releases to ensure that iodine will be retained in the suppression pool water.

ACTIONS

A.1 and A.2

If the boron solution concentration is > 9.82% weight but the concentration and temperature of boron in solution and pump suction piping temperature are within the limits of Figure 3.1.7-1, operation is permitted for a limited period since the SLC subsystems are capable of performing the intended function. It is not necessary under these conditions to declare both SLC subsystems inoperable since the SLC subsystems are capable of performing their intended function.

The concentration and temperature of boron in solution and pump suction piping temperature must be verified to be within the limits of Figure 3.1.7-1 within 8 hours and once per 12 hours thereafter (Required Action A.1). The temperature versus concentration curve of Figure 3.1.7-1 ensures a 10°F margin will be maintained above the saturation temperature. This verification ensures that boron does not precipitate out of solution in the storage tank or in the pump suction piping due to low boron solution temperature (below the saturation temperature for the given concentration). The Completion Time for performing Required Action A.1 is considered acceptable given the low probability of a Design Basis Accident (DBA) or transient occurring concurrent with the failure of the control rods to shut down the reactor and operating experience which has shown there are relatively slow variations in the measured parameters of concentration and temperature over these time periods.

(continued)

BASES

ACTIONS

D.1 **and D.2** (continued)

brought to MODE 3 within 12 hours **and MODE 4 within 36 hours.** The allowed Completion Time**s are** ~~of 12 hours is~~ reasonable, based on operating experience, to reach ~~MODE 3~~**the required MODES** from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1, SR 3.1.7.2, and SR 3.1.7.3

SR 3.1.7.1 through SR 3.1.7.3 are 24 hour Surveillances verifying certain characteristics of the SLC System (e.g., the level and temperature of the borated solution in the storage tank), thereby ensuring SLC System OPERABILITY without disturbing normal plant operation. These Surveillances ensure that the proper borated solution level and temperature, including the temperature of the pump suction piping, are maintained. Maintaining a minimum specified borated solution temperature is important in ensuring that the boron remains in solution and does not precipitate out in the storage tank or in the pump suction piping. The temperature limit specified in SR 3.1.7.2 and SR 3.1.7.3 and the maximum sodium pentaborate concentration specified in Table 3.1.7-1 ensures that a 10°F margin will be maintained above the saturation temperature. Control room alarms for low SLC storage tank temperature and low SLC System piping temperature are available and are set at 55°F. As such, SR 3.1.7.2 and SR 3.1.7.3 may be satisfied by verifying the absence of low temperature alarms for the SLC storage tank and SLC System piping. The 24 hour Frequency is based on operating experience and has shown there are relatively slow variations in the measured parameters of level and temperature.

SR 3.1.7.4 and SR 3.1.7.6

SR 3.1.7.4 verifies the continuity of the explosive charges in the injection valves to ensure that proper operation will occur if required. Other administrative controls, such as those that limit the shelf life of the explosive charges, must be followed. The 31 day Frequency is based on operating experience and has demonstrated the reliability of the explosive charge continuity.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.9 (continued)

Surveillance when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.1.7.10

Enriched sodium pentaborate solution is made by mixing granular, enriched sodium pentaborate with water. In order to ensure the proper B-10 atom percentage (in accordance with Table 3.1.7-1) is being used, calculations must be performed to verify the actual B-10 enrichment within 8 hours after addition of the solution to the SLC tank. The calculations may be performed using the results of isotopic tests on the granular sodium pentaborate or vendor certification documents. The Frequency is acceptable considering that boron enrichment is verified during the procurement process and any time boron is added to the SLC tank.

REFERENCES

1. 10 CFR 50.62.

2. UFSAR, Section 3.8.4.

3. **10 CFR 50.67**

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

BASES

BACKGROUND

The SDV vent and drain valves are normally open and discharge any accumulated water in the SDV to ensure that sufficient volume is available at all times to allow a complete scram. During a scram, the SDV vent and drain valves close to contain reactor water. As discussed in Reference 1, the SDV vent and drain valves need not be considered primary containment isolation valves (PCIVs) for the Scram Discharge System. (However, at PBAPS, these valves are considered PCIVs.) The SDV is a volume of header piping that connects to each hydraulic control unit (HCU) and drains into an instrument volume. There are two SDVs (headers) and a common instrument volume that receives all of the control rod drive (CRD) discharges. The instrument volume is connected to a common drain line with two valves in series. Each header is connected to a common vent line with two valves in series for a total of four vent valves. The header piping is sized to receive and contain all the water discharged by the CRDs during a scram. The design and functions of the SDV are described in Reference 2.

APPLICABLE SAFETY ANALYSES

The Design Basis Accident and transient analyses assume all of the control rods are capable of scramming. The acceptance criteria for the SDV vent and drain valves are that they operate automatically to close during scram to limit the amount of reactor coolant discharged so that adequate core cooling is maintained and offsite doses remain within the limits of ~~10 CFR 100~~**10 CFR 50.67** (Ref. 3).

Isolation of the SDV can also be accomplished by manual closure of the SDV valves. Additionally, the discharge of reactor coolant to the SDV can be terminated by scram reset or closure of the HCU manual isolation valves. For a bounding leakage case, the offsite doses are well within the limits of ~~10 CFR 100~~**10 CFR 50.67** (Ref. 3), and adequate core cooling is maintained (Ref. 1). The SDV vent and drain valves allow continuous drainage of the SDV during normal plant operation to ensure that the SDV has sufficient capacity to contain the reactor coolant discharge during a full core scram. To automatically ensure this capacity, a reactor scram (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation") is initiated if the SDV water level in the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.8.3 (continued)

unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components will pass the Surveillance when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," August 1981.

2. UFSAR, Sections 3.4.5.3.1 and 7.2.3.6.

3. ~~10 CFR 100~~**10 CFR 50.67.**

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

BASES

BACKGROUND

The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including abnormal operational transients. Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure, or inability to cool the fuel does not occur during the anticipated operating conditions identified in Reference 1.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel system design are presented in References 1, 2, 3, 4, 5, 6, 7, and 8. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation, and protection system) that fuel damage will not result in the release of radioactive materials in excess of the guidelines of 10 CFR, Parts 20, 50, and 100, **as applicable**. The mechanisms that could cause fuel damage during operational transients and that are considered in fuel evaluations are:

a. Rupture of the fuel rod cladding caused by strain from the relative expansion of the UO_2 pellet; and

b. Severe overheating of the fuel rod cladding caused by inadequate cooling.

A value of 1% plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur (Ref. 10).

Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit is not exceeded during continuous operation with LHGRs up to the operating limit specified in the COLR. The analysis also

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.a. Reactor Vessel Water Level – Low Low Low (Level 1) (continued)

The Reactor Vessel Water Level – Low Low Low (Level 1) Allowable Value is chosen to be the same as the ECCS Level 1 Allowable Value (LCO 3.3.5.1) to ensure that the MSLs isolate on a potential loss of coolant accident (LOCA) to prevent offsite doses from exceeding ~~10 CFR 100~~**10 CFR 50.67** limits.

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

1.b. Main Steam Line Pressure – Low

Low MSL pressure indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure – Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 3). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 785 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 25% RTP.)

The MSL low pressure signals are initiated from four transmitters that are connected to the MSL header. The transmitters are arranged such that, even though physically separated from each other, each transmitter is able to detect low MSL pressure. Four channels of Main Steam Line Pressure – Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure – Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 1).

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.c. Main Steam Line Flow-High

Main Steam Line Flow-High is provided to detect a break of the MSL and to initiate closure of the MSIVs. If the steam were allowed to continue flowing out of the break, the reactor would depressurize and the core could uncover. If the RPV water level decreases too far, fuel damage could occur. Therefore, the isolation is initiated on high flow to prevent or minimize core damage. The Main Steam Line Flow-High Function is directly assumed in the analysis of the main steam line break (MSLB) (Ref. 3). The isolation action, along with the scram function of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46 and offsite doses do not exceed the ~~10 CFR 100~~**10 CFR 50.67** limits.

The MSL flow signals are initiated from 16 transmitters that are connected to the four MSLs. The transmitters are arranged such that, even though physically separated from each other, all four connected to one MSL would be able to detect the high flow. Four channels of Main Steam Line Flow-High Function for each MSL (two channels per trip system) are available and are required to be OPERABLE so that no single instrument failure will preclude detecting a break in any individual MSL.

The Allowable Value is chosen to ensure that offsite dose limits are not exceeded due to the break.

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

1.d. Main Steam Line-High Radiation

The Main Steam Line-High Radiation Function is provided to detect gross release of fission products from the fuel and to initiate closure of the MSIVs. The trip setting is set low enough so that a high radiation trip results from a design basis rod drop accident and high enough above background radiation levels in the vicinity of the main steam lines so that spurious trips at rated power are avoided. The Main Steam Line-High Radiation Function is directly assumed in the analysis of the control rod drop accident (Ref. 3).

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.d. Main Steam Line–High Radiation (continued)

The Main Steam Line–High Radiation signals are initiated from four gamma sensitive instruments. Four channels are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value is chosen to ensure that offsite dose limits are not exceeded.

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

1.e. Main Steam Tunnel Temperature–High

The Main Steam Tunnel Temperature Function is provided to detect a break in a main steam line and provides diversity to the high flow instrumentation.

Main Steam Tunnel Temperature signals are initiated from resistance temperature detectors (RTDs) located along the main steam line between the drywell wall and the turbine. Sixteen channels of Main Steam Tunnel Temperature–High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value is chosen to detect a leak equivalent to between 1% and 10% rated steam flow.

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

This Function in Unit 3 combines Unit 2 Functions 1.e. and 1.f.

Primary Containment Isolation

2.a. Reactor Vessel Water Level–Low (Level 3)

Low RPV water level indicates that the capability to cool the fuel may be threatened. The valves whose penetrations communicate with the primary containment are isolated to limit the release of fission products. The isolation of the primary containment on Level 3 supports actions to ensure that offsite dose limits of ~~10 CFR 100~~**10 CFR 50.67** are not exceeded.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.a. Reactor Vessel Water Level–Low (Level 3) (continued)

The Reactor Vessel Water Level–Low (Level 3) Function associated with isolation is implicitly assumed in the UFSAR analysis as these leakage paths are assumed to be isolated post LOCA.

Reactor Vessel Water Level–Low (Level 3) signals are initiated from level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level–Low (Level 3) Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level–Low (Level 3) Allowable Value was chosen to be the same as the RPS Level 3 scram Allowable Value (LCO 3.3.1.1), since isolation of these valves is not critical to orderly plant shutdown.

This Function isolates the Group II(A) valves listed in Reference 1 with the exception of RWCU isolation valves and RHR shutdown cooling pump suction valves which are addressed in Functions 5.c and 6.b, respectively.

2.b. Drywell Pressure–High

High drywell pressure can indicate a break in the RCPB inside the primary containment. The isolation of some of the primary containment isolation valves on high drywell pressure supports actions to ensure that offsite dose limits of ~~10 CFR 100~~**10 CFR 50.67** are not exceeded. The Drywell Pressure–High Function, associated with isolation of the primary containment, is implicitly assumed in the UFSAR accident analysis as these leakage paths are assumed to be isolated post LOCA.

High drywell pressure signals are initiated from pressure transmitters that sense the pressure in the drywell. Four channels of Drywell Pressure–High are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

5.a. RWCU Flow-High (continued)

The high RWCU flow signals are initiated from transmitters that are connected to the pump suction line of the RWCU System. Two channels of RWCU Flow-High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The RWCU Flow-High Allowable Value ensures that a break of the RWCU piping is detected.

This Function isolates the inboard and outboard RWCU pump suction penetration and the outboard valve at the RWCU connection to reactor feedwater.

5.b. Standby Liquid Control (SLC) System Initiation

The isolation of the RWCU System is required when the SLC System has been initiated to prevent dilution and removal of the boron solution by the RWCU System (Ref. 5). SLC System initiation signals are initiated from the remote SLC System start switch.

There is no Allowable Value associated with this Function since the channels are mechanically actuated based solely on the position of the SLC System initiation switch.

For reactivity insertion accidents, ~~Two~~ **two** channels of the SLC System Initiation Function are available and are required to be OPERABLE only in MODES 1 and 2, since these are the only MODES where the reactor can be critical.~~, and these~~ **In addition, for accidents involving significant fission product releases, both channels are also required to be OPERABLE in MODES 1, 2, and 3. The SLC System is designed to maintain suppression pool pH at or above 7 following a LOCA to ensure that iodine will be retained in the suppression pool water. These** MODES are consistent with the Applicability for the SLC System (LCO 3.1.7).

This Function isolates the inboard and outboard RWCU pump suction penetration and the outboard valve at the RWCU connection to reactor feedwater.

5.c. Reactor Vessel Water Level-Low (Level 3)

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of some interfaces with the reactor vessel occurs to isolate the potential sources of a break. The isolation of the RWCU System on Level 3 supports actions to ensure that the fuel

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3, 4. Reactor Building Ventilation and Refueling Floor Ventilation Exhaust Radiation–High (continued)

channels of Reactor Building Ventilation Exhaust Radiation–High Function and four channels of Refueling Floor Ventilation Exhaust Radiation–High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Values are chosen to promptly detect gross failure of the fuel cladding.

The Reactor Building Ventilation and Refueling Floor Ventilation Exhaust Radiation–High Functions are required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists; thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. In MODES 4 and 5, the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES; thus, these Functions are not required. In addition, the Functions are also required to be OPERABLE during ~~CORE ALTERATIONS,~~ OPDRVs~~,~~ and movement of ~~irradiated fuel~~**RECENTLY IRRADIATED FUEL** assemblies in the secondary containment, because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery or dropped fuel assemblies) must be provided to ensure that offsite dose limits are not exceeded.

ACTIONS

A Note has been provided to modify the ACTIONS related to secondary containment isolation instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable secondary containment isolation instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable secondary containment isolation instrumentation channel.

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Specific Activity

BASES

BACKGROUND

During circulation, the reactor coolant acquires radioactive materials due to release of fission products from fuel leaks into the reactor coolant and activation of corrosion products in the reactor coolant. These radioactive materials in the reactor coolant can plate out in the RCS, and, at times, an accumulation will break away to spike the normal level of radioactivity. The release of coolant during a Design Basis Accident (DBA) could send radioactive materials into the environment.

Limits on the maximum allowable level of radioactivity in the reactor coolant are established to ensure that in the event of a release of any radioactive material to the environment during a DBA, radiation doses are maintained within the limits of ~~10 CFR 100~~**10 CFR 50.67** (Ref. 1).

This LCO contains the iodine specific activity limits. The iodine isotopic activities per gram of reactor coolant are expressed in terms of a DOSE EQUIVALENT I-131. The allowable level is intended to limit the **maximum** 2 hour radiation dose to an individual at the site boundary to well within the ~~10 CFR 100~~**10 CFR 50.67** limit **as modified in Regulatory Guide 1.183, Table 6.**

APPLICABLE SAFETY ANALYSES

Analytical methods and assumptions involving radioactive material in the primary coolant are presented in the UFSAR (Ref. 2). The specific activity in the reactor coolant (the source term) is an initial condition for evaluation of the consequences of an accident due to a main steam line break (MSLB) outside containment. No fuel damage is postulated in the MSLB accident, and the release of radioactive material to the environment is assumed to end when the main steam isolation valves (MSIVs) close completely.

This MSLB release forms the basis for determining offsite doses (Ref. 2). The limits on the specific activity of the primary coolant ensure that the **maximum** 2 hour ~~thyroid and whole body~~**TEDE** doses at the site boundary, resulting from an MSLB outside containment during steady state operation, will not exceed the dose guidelines of ~~10 CFR 100~~**10 CFR 50.67 as modified in Regulatory Guide 1.183, Table 6.** .

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limits on specific activity are values from a parametric evaluation of typical site locations. These limits are conservative because the evaluation considered more restrictive parameters than for a specific site, such as the location of the site boundary and the meteorological conditions of the site.

RCS specific activity satisfies Criterion 2 of the NRC Policy Statement.

LCO

The specific iodine activity is limited to ≤ 0.2 μCi/gm DOSE EQUIVALENT I-131. This limit ensures the source term assumed in the safety analysis for the MSLB is not exceeded, so any release of radioactivity to the environment during an MSLB is well within the ~~10 CFR 100~~**10 CFR 50.67** limits **as modified in Regulatory Guide 1.183, Table 6.**

APPLICABILITY

In MODE 1, and MODES 2 and 3 with any main steam line not isolated, limits on the primary coolant radioactivity are applicable since there is an escape path for release of radioactive material from the primary coolant to the environment in the event of an MSLB outside of primary containment.

In MODES 2 and 3 with the main steam lines isolated, such limits do not apply since an escape path does not exist. In MODES 4 and 5, no limits are required since the reactor is not pressurized and the potential for leakage is reduced.

ACTIONS

A.1 and A.2

When the reactor coolant specific activity exceeds the LCO DOSE EQUIVALENT I-131 limit, but is ≤ 4.0 μCi/gm, samples must be analyzed for DOSE EQUIVALENT I-131 at least once every 4 hours. In addition, the specific activity must be restored to the LCO limit within 48 hours. The Completion Time of once every 4 hours is based on the time needed to take and analyze a sample. The 48 hour Completion Time to restore the activity level provides a reasonable time for temporary coolant activity increases (iodine spikes) to be cleaned up with the normal processing systems.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S) while relying on the ACTIONS. This allowance 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 plant remains at, or proceeds to, power operation.

B.1, B.2.1, B.2.2.1, and B.2.2.2

If the DOSE EQUIVALENT I-131 cannot be restored to ≤ 0.2 μCi/gm within 48 hours, or if at any time it is > 4.0 μCi/gm, it must be determined at least once every 4 hours and all the main steam lines must be isolated within 12 hours. Isolating the main steam lines precludes the possibility of releasing radioactive material to the environment in an amount that is more than a small fraction of the requirements of ~~10 CFR 100~~**10 CFR 50.67 as modified in Regulatory Guide 1.1.83**, **Table 6**, during a postulated MSLB accident.

Alternatively, the plant can be placed in MODE 3 within 12 hours and in MODE 4 within 36 hours. This option is provided for those instances when isolation of main steam lines is not desired (e.g., due to the decay heat loads). In MODE 4, the requirements of the LCO are no longer applicable.

The Completion Time of once every 4 hours is the time needed to take and analyze a sample. The 12 hour Completion Time is reasonable, based on operating experience, to isolate the main steam lines in an orderly manner and without challenging plant systems. Also, the allowed Completion Times for Required Actions B.2.2.1 and B.2.2.2 for placing the unit in MODES 3 and 4 are reasonable, based on operating experience, to achieve the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This Surveillance is performed to ensure iodine remains within limit during normal operation. The 7 day Frequency is adequate to trend changes in the iodine activity level.

This SR is modified by a Note that requires this Surveillance to be performed only in MODE 1 because the level of fission products generated in other MODES is much less.

REFERENCES

1. ~~10 CFR 100.11, 1973~~**10 CFR 50.67.**
2. UFSAR, Section 14.6.5.

BASES (continued)

APPLICABLE SAFETY ANALYSES

The safety design basis for the primary containment is that it must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE such that release of fission products to the environment is controlled by the rate of primary containment leakage.

Analytical methods and assumptions involving the primary containment are presented in Reference 1. The safety analyses assume a nonmechanistic fission product release following a DBA, which forms the basis for determination of offsite doses. The fission product release is, in turn, based on an assumed leakage rate from the primary containment. OPERABILITY of the primary containment ensures that the leakage rate assumed in the safety analyses is not exceeded.

The maximum allowable leakage rate for the primary containment (L_a) is ~~0.5~~**0.7%** by weight of the containment air per 24 hours at the design basis LOCA maximum peak containment pressure (P_a) of 49.1 psig. The value of P_a (49.1 psig) is conservative with respect to the current calculated peak drywell pressure of 47.2 psig (Ref. 2). This value is 47.8 psig for operation with 90°F Final Feedwater Temperature Reduction (Ref. 7).

Primary containment satisfies Criterion 3 of the NRC Policy Statement.

LCO

Primary containment OPERABILITY is maintained by limiting leakage to ≤ 1.0 L_a, except prior to the first startup after performing a required Primary Containment Leakage Rate Testing Program leakage test. At this time, applicable leakage limits must be met. In addition, the leakage from the drywell to the suppression chamber must be limited to ensure the pressure suppression function is accomplished and the suppression chamber pressure does not exceed design limits. Compliance with this LCO will ensure a primary containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analyses.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE, such that release of fission products to the environment is controlled by the rate of primary containment leakage. The primary containment is designed with a maximum allowable leakage rate (L_a) of ~~0.5~~**0.7%** by weight of the containment air per 24 hours at the maximum peak containment pressure (P_a) of 49.1 psig. The value of P_a (49.1 psig) is conservative with respect to the current calculated peak drywell pressure of 47.2 psig (Ref. 3). This value is 47.8 psig for operation with 90°F Final Feedwater Temperature Reduction (Ref. 4). This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock.

Primary containment air lock OPERABILITY is also required to minimize the amount of fission product gases that may escape primary containment through the air lock and contaminate and pressurize the secondary containment.

The primary containment air lock satisfies Criterion 3 of the NRC Policy Statement.

LCO

As part of primary containment, the air lock's safety function is related to control of containment leakage rates following a DBA. Thus, the air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

The primary containment 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 to be opened at a time. This provision ensures that a gross breach of primary containment does not exist when primary 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 and exit from primary containment.

(continued)

BASES

ACTIONS

D.1 (continued)

rate for the isolated penetration is assumed to be the actual pathway leakage through the isolation device. If two isolation devices are used to isolate the penetration, the leakage rate is assumed to be the lesser actual pathway leakage of the two devices. The 8 hour Completion Time is reasonable considering the time required to restore the leakage by isolating the penetration, the fact that MSIV closure will result in isolation of the main steam line and a potential for plant shutdown, and the relative importance of MSIV leakage to the overall containment function.

E.1, E.2 and E.2

The accumulated time that the large containment purge and/or vent valves 6" and 18" vent valves) are open, when reactor pressure is greater than 100 psig and the reactor is in MODES 1 or 2, is limited to 90 hours per calendar year. This will limit the total time that a flow path exists through certain containment penetrations and ensures that there is no threat to the design analysis for crediting containment overpressure for ECCS NPSH. Consequently, there exists minimal impact on plant risk resulting challenges to ECCS NPSH during a LOCA while purging. The 4-hour Completion Time to isolate the penetration is considered a reasonable amount of time to ensure compliance with the design analysis for containment overpressure. If the penetration is not isolated within the specified 4-hour time period, then the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

~~E~~F.1 and ~~E~~F.2

If any Required Action and associated Completion Time cannot be met in MODE 1, 2, or 3, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

~~F~~G.1 and ~~F~~G.2

If any Required Action and associated Completion Time cannot be met for PCIV(s) required to be OPERABLE during MODE 4 or 5, the unit must be placed in a condition in which the LCO does not apply. Action must be immediately initiated to suspend operations with a potential for draining the reactor vessel (OPDRVs) to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended and valve(s) are restored to OPERABLE status. If suspending an OPDRV would result in closing the residual heat removal (RHR) shutdown cooling isolation valves, an alternative Required Action is provided to immediately initiate action to restore the valve(s) to OPERABLE status. This allows RHR to remain in service while actions are being taken to restore the valve.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.7 (continued)

position, since these valves were verified to be in the correct position prior to locking or securing. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.6.1.3.8

Verifying the isolation time of each power operated automatic PCIV is within limits is required to demonstrate OPERABILITY. MSIVs may be excluded from this SR since MSIV full closure isolation time is demonstrated by SR 3.6.1.3.9. The isolation time test ensures that the valve will isolate in a time period less than or equal to that assumed in the safety analyses. The isolation time is in accordance with Reference 2 or the requirements of the Inservice Testing Program which ever is more conservative. The Frequency of this SR is in accordance with the requirements of the Inservice Testing Program.

SR 3.6.1.3.9

Verifying that the isolation time of each MSIV is within the specified limits is required to demonstrate OPERABILITY. The isolation time test ensures that the MSIV will isolate in a time period that does not exceed the times assumed in the DBA analyses. This ensures that the calculated radiological consequences of these events remain within ~~10 CFR 100~~**10 CFR 50.67** limits **as modified in Regulatory Guide 1.183, Table 6**. The Frequency of this SR is in accordance with the requirements of the Inservice Testing Program.

SR 3.6.1.3.10

Automatic PCIVs close on a primary containment isolation signal to prevent leakage of radioactive material from primary containment following a DBA. This SR ensures that each automatic PCIV will actuate to its isolation position on a primary containment isolation signal. The LOGIC SYSTEM

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.13

This SR ensures that in case the non-safety grade instrument air system is unavailable, the SGIG System will perform its design function to supply nitrogen gas at the required pressure for valve operators and valve seals supported by the SGIG System. The 24 month Frequency was developed considering it is prudent that this Surveillance be performed only during a plant outage. Operating experience has shown that these components will usually pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.3.14

~~Leakage through each MSIV must be ≤ 11.5 scfh when tested at ≥ P_t (25 psig). The analyses in Reference 1 are based on treatment of MSIV leakage as a secondary containment bypass leakage, independent of a primary to secondary containment leakage analyzed at 1.27 L_a. In the Reference 1 analysis all 4 steam lines are assumed to leak at the TS Limit. This ensures that MSIV leakage is properly accounted for in determining the overall impacts of primary containment leakage. The Frequency is required by the Primary Containment Leakage Rate Testing Program.~~**Total leakage through all four main steam lines must be ≤ 200 scfh, and ≤ 115 scfh for any one steam line, when tested at ≥ 25 psig. The analysis in Reference 1 is based on treatment of MSIV leakage as secondary containment bypass leakage, independent of the primary to secondary leakage analyzed at L_a. The Frequency is in accordance with the Primary Containment Leakage Rate Testing Program.**

SR 3.6.1.3.15

Verifying the opening of each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve is restricted by a blocking device to less than or equal to the required maximum opening angle specified in the UFSAR (Ref. 4) is required to ensure that the valves can close under DBA conditions within the times in the analysis of Reference 1. If a LOCA occurs, the purge and exhaust valves must close to maintain primary containment leakage within the values assumed in the accident analysis. At other times pressurization concerns are not present, thus the purge and exhaust valves can be fully open. The 24 month Frequency is appropriate because the blocking devices may be removed during a refueling outage.

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.1 Secondary Containment

BASES

BACKGROUND

The function of the secondary containment is to contain and hold up fission products that may leak from primary containment following a Design Basis Accident (DBA). In conjunction with operation of the Standby Gas Treatment (SGT) System and closure of certain valves whose lines penetrate the secondary containment, the secondary containment is designed to reduce the activity level of the fission products prior to release to the environment and to isolate and contain fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be OPERABLE, or that take place outside primary containment.

The secondary containment is a structure that completely encloses the primary containment and those components that may be postulated to contain primary system fluid. This structure forms a control volume that serves to hold up and dilute the fission products. It is possible for the pressure in the control volume to rise relative to the environmental pressure (e.g., due to pump and motor heat load additions). To prevent ground level exfiltration while allowing the secondary containment to be designed as a conventional structure, the secondary containment requires support systems to maintain the control volume pressure at less than the external pressure. Requirements for these systems are specified separately in LCO 3.6.4.2, "Secondary Containment Isolation Valves (SCIVs)," and LCO 3.6.4.3, "Standby Gas Treatment (SGT) System."

APPLICABLE SAFETY ANALYSES

There are two principal accidents for which credit is taken for secondary containment OPERABILITY. These are a loss of coolant accident (LOCA) (Ref. 1) and a fuel handling accident inside secondary containment (Ref. 2)**involving RECENTLY IRRADIATED FUEL**. The secondary containment performs no active function in response to each of these limiting events; however, its leak

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	tightness is required to ensure that fission products entrapped within the secondary containment structure will be treated by the SGT System prior to discharge to the environment. Secondary containment satisfies Criterion 3 of the NRC Policy Statement.
LCO	An OPERABLE secondary containment provides a control volume into which fission products that leak from primary containment, or are released from the reactor coolant pressure boundary components located in secondary containment, can be processed prior to release to the environment. For the secondary containment to be considered OPERABLE, it must have adequate leak tightness to ensure that the required vacuum can be established and maintained.
APPLICABILITY	In MODES 1, 2, and 3, a LOCA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, secondary containment OPERABILITY is required during the same operating conditions that require primary containment OPERABILITY. In MODES 4 and 5, the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining secondary containment OPERABLE is not required in MODE 4 or 5 ~~to ensure a control volume~~, except for other situations for which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs), ~~during CORE ALTERATIONS~~, or during movement of ~~irradiated fuel~~ **RECENTLY IRRADIATED FUEL** assemblies in the secondary containment. **However, outside ground level hatches (hatches H15 through H24 and Torus room access hatches) may not be opened during movement of irradiated fuel unless a decay period of 84 hours after shutdown has occurred. This will maintain CR does acceptable.**
ACTIONS	A.1 If secondary containment is inoperable, it must be restored to OPERABLE status within 4 hours. The 4 hour Completion Time provides a period of time to correct the problem that is commensurate with the importance of maintaining secondary containment during MODES 1, 2, and 3. This time period also ensures that the probability of an accident (requiring secondary containment OPERABILITY) occurring during periods where secondary containment is inoperable is minimal.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If secondary containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1~~,~~**and** C.2~~, and C.3~~

Movement of ~~irradiated fuel~~**RECENTLY IRRADIATED FUEL** assemblies in the secondary containment~~, CORE ALTERATIONS,~~ and OPDRVs can be postulated to cause fission product release to the secondary containment. In such cases, the secondary containment is the only barrier to release of fission products to the environment. ~~CORE ALTERATIONS and~~**Therefore**, movement of ~~irradiated fuel~~**RECENTLY IRRADIATED FUEL** assemblies must be immediately suspended if the secondary containment is inoperable.

Suspension of these activities shall not preclude completing an action that involves moving a component to a safe position. Also, action must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

Required Action C.1 has been modified by a Note stating that LCO 3.0.3 is not applicable~~. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.~~, **since the movement of RECENTLY IRRADIATED FUEL can only be performed in MODES 4 and 5.**

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.1 and SR 3.6.4.1.2

Verifying that secondary containment equipment hatches and one access door in each access opening are closed ensures that the infiltration of outside air of such a magnitude as to prevent maintaining the desired negative pressure does not occur. Verifying that all such openings are closed provides adequate assurance that exfiltration from the secondary containment will not occur. In this application, the term "sealed" has no connotation of leak tightness. Maintaining secondary containment OPERABILITY requires verifying one door in the access opening is closed. An access opening contains one inner and one outer door. In some cases, secondary containment access openings are shared such that a secondary containment barrier may have multiple inner or multiple outer doors. The intent is to not breach secondary containment at any time when secondary containment is required. This is achieved by maintaining the inner or outer portion of the barrier closed at all times. However, all secondary containment access doors are normally kept closed, except when the access opening is being used for entry and exit or when maintenance is being performed on an access opening. The 31 day Frequency for these SRs has been shown to be adequate, based on operating experience, and is considered adequate in view of the other indications of door and hatch status that are available to the operator.

SR 3.6.4.1.3 and SR 3.6.4.1.4

The SGT System exhausts the secondary containment atmosphere to the environment through appropriate treatment equipment. Each SGT subsystem is designed to draw down pressure in the secondary containment to ≥ 0.25 inches of vacuum water gauge in ≤ ~~120~~ **180** seconds and maintain pressure in the secondary containment at ≥ 0.25 inches of vacuum water gauge for 1 hour at a flow rate ≤ 10,500 cfm. To ensure that all fission products released to the secondary containment are treated, SR 3.6.4.1.3 and SR 3.6.4.1.4 verify that a pressure in the secondary containment this is less than the lowest postulated pressure external to the secondary containment boundary can rapidly be established and maintained. When the SGT System is operating as designed, the establishment and maintenance of secondary containment pressure cannot be accomplished if the secondary containment boundary is not intact. Establishment of this pressure is confirmed by SR 3.6.4.1.3 which demonstrates that the secondary containment can be drawn down to ≥ 0.25 inches of vacuum water gauge in ≤ ~~120~~ **180**

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

BASES

BACKGROUND

The function of the SCIVs, in combination with other accident mitigation systems, is to ~~limit~~ **control** fission product release during and following postulated Design Basis Accidents (DBAs) (Refs. 1 and 2). Secondary containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that fission products that leak from primary containment following a DBA, or that are released during certain operations when primary containment is not required to be OPERABLE or take place outside primary containment, are maintained within the secondary containment boundary.

The OPERABILITY requirements for SCIVs help ensure that an adequate secondary containment boundary is maintained during and after an accident by minimizing potential paths to the environment. These isolation devices consist of either passive devices or active (automatic) devices. Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), and blind flanges are considered passive devices.

Automatic SCIVs close on a secondary containment isolation signal to establish a boundary for untreated radioactive material within secondary containment following a DBA or other accidents.

Other penetrations are isolated by the use of valves in the closed position or blind flanges.

APPLICABLE SAFETY ANALYSES

The SCIVs must be OPERABLE to ensure the secondary containment barrier to fission product releases is established. The principal accidents for which the secondary containment boundary is required are a loss of coolant accident (Ref. 1) and a fuel handling accident inside secondary containment (Ref. 2) **involving RECENTLY IRRAIDATED FUEL.** The secondary containment performs no active function in response to either of these limiting events, but the boundary

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	established by SCIVs is required to ensure that leakage from the primary containment is processed by the Standby Gas Treatment (SGT) System before being released to the environment. Maintaining SCIVs OPERABLE with isolation times within limits ensures that fission products will remain trapped inside secondary containment so that they can be treated by the SGT System prior to discharge to the environment. SCIVs satisfy Criterion 3 of the NRC Policy Statement.
LCO	SCIVs form a part of the secondary containment boundary. The SCIV safety function is related to control of offsite radiation releases resulting from DBAs. The power operated automatic isolation valves are considered OPERABLE when their isolation times are within limits and the valves actuate on an automatic isolation signal. The valves covered by this LCO, along with their associated stroke times, are listed in Reference ~~3~~**2**. The normally closed isolation valves or blind flanges are considered OPERABLE when manual valves are closed or open in accordance with appropriate administrative controls, automatic SCIVs are de-activated and secured in their closed position, and blind flanges are in place. These passive isolation valves or devices are listed in Reference ~~3~~**2**.
APPLICABILITY	In MODES 1, 2, and 3, a DBA could lead to a fission product release to the primary containment that leaks to the secondary containment. Therefore, the OPERABILITY of SCIVs is required. In MODES 4 and 5, the probability and consequences of these events are reduced due to pressure and temperature limitations in these MODES. Therefore, maintaining SCIVs OPERABLE is not required in MODE 4 or 5, except for other situations under which significant radioactive releases can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs)~~, during CORE ALTERATIONS,~~ or during movement of ~~irradiated fuel~~ **RECENTLY IRRADIATED FUEL** assemblies in the secondary containment. **SCIVs are only required to be OPERABLE during handling RECENTLY IRRADIATED FUEL.** Moving irradiated fuel assemblies in the secondary containment may also occur in MODES 1, 2, and 3.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

If any Required Action and associated Completion Time cannot be met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1~~,~~ **and** D.2~~, and D.3~~

If any Required Action and associated Completion Time are not met, the plant must be placed in a condition in which the LCO does not apply. If applicable, ~~CORE ALTERATIONS and~~ the movement of ~~irradiated fuel~~**RECENTLY IRRADIATED FUEL** assemblies in the secondary containment must be immediately suspended. Suspension of ~~these activities~~**this activity** shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be immediately initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

Required Action D.1 has been modified by a Note stating that LCO 3.0.3 is not applicable~~. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving fuel while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.~~, **since the movement of RECENTLY IRRADIATED FUEL can only be performed in MODES 4 and 5.**

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.1

This SR verifies that each secondary containment manual isolation valve and blind flange that is not locked, sealed, or otherwise secured and is required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the secondary containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification that those SCIVs in secondary containment that are capable of being mispositioned are in the correct position.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.3 (continued)

under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components will usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section ~~14.6.3~~**14.9.2**.

~~2. UFSAR, Section 14.6.4.~~

~~3~~**2**. Technical Requirements Manual.

BASES

LCO (continued)

For Unit 3, one SGT subsystem is OPERABLE when one charcoal filter train, one fan (0CV020) and associated ductwork, dampers, valves, and controls are OPERABLE. The second SGT subsystem is OPERABLE when the other charcoal filter train, one fan (0BV020) and associated ductwork, damper, valves, and controls are OPERABLE.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, SGT System OPERABILITY is required during these MODES.

In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the SGT System in OPERABLE status is not required in MODE 4 or 5, except for other situations under which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs)~~, during CORE ALTERATIONS,~~ or during movement of ~~irradiated fuel~~**RECENTLY IRRADIATED FUEL** assemblies in the secondary containment. **The SGT System is only required to be OPERABLE during OPRDVs or handling of RECENTLY IRRADIATED FUEL.**

ACTIONS

A.1

With one SGT subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status in 7 days. In this Condition, the remaining OPERABLE SGT subsystem is adequate to perform the required radioactivity release control function. However, the overall system reliability is reduced because a single failure in the OPERABLE subsystem could result in the radioactivity release control function not being adequately performed. The 7 day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant SGT subsystem and the low probability of a DBA occurring during this period.

B.1 and B.2

If the SGT subsystem cannot be restored to OPERABLE status within the required Completion Time in MODE 1, 2, or 3, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1, C.2.1, ~~C.2.2, and C.2.3~~**and C.2.2**

During movement of ~~irradiated fuel~~**RECENTLY IRRADIATED FUEL** assemblies, in the secondary containment~~, during CORE ALTERATIONS,~~ or during OPDRVs, when Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE SGT subsystem should immediately be placed in operation. This action ensures that the remaining subsystem is OPERABLE, that no failures that could prevent automatic actuation have occurred, and that any other failure would be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that represent a potential for releasing radioactive material to the secondary containment, thus placing the plant in a condition that minimizes risk. If applicable, ~~CORE ALTERATIONS and~~ movement of ~~irradiated fuel~~**RECENTLY IRRADIATED FUEL** assemblies must immediately be suspended. Suspension of ~~these activities~~**this activity** must not preclude completion of movement of a component to a safe position. Also, if applicable, actions must immediately be initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

The Required Actions of Condition C have been modified by a Note stating that LCO 3.0.3 is not applicable. ~~If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.~~, **since the movement of RECENTLY IRRADIATED FUEL can only be performed in MODES 4 and 5.**

(continued)

BASES

ACTIONS (continued)

D.1

If both SGT subsystems are inoperable in MODE 1, 2, or 3, the SGT System may not be capable of supporting the required radioactivity release control function. Therefore, actions are required to enter LCO 3.0.3 immediately.

E.1~~, and E.2, and E.3~~ **and** E.2

When two SGT subsystems are inoperable, if applicable, ~~CORE ALTERATIONS and~~ movement of ~~irradiated fuel~~**RECENTLY IRRADIATED FUEL** assemblies in secondary containment must immediately be suspended. Suspension of ~~these activities~~**this activity** shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must immediately be initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

Required Action E.1 has been modified by a Note stating that LCO 3.0.3 is not applicable~~. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.~~, **since the movement of RECENTLY ITTADIATED FUEL can only be performed in MODES 4 and 5.**

SURVEILLANCE REQUIREMENTS

SR 3.6.4.3.1

Operating each SGT subsystem (including each filter train fan) for ≥ 15 minutes ensures that both subsystems are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. Operation with the heaters on (automatic heater cycling to maintain temperature) for ≥ 15 minutes every 31 days is sufficient to eliminate moisture on the adsorbers and HEPA filters since during idle periods instrument air is injected into the filter plenum to keep the filters dry. The 31 day Frequency was developed in consideration of the known reliability of fan motors and controls and the redundancy available in the system.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.7 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident.

A general description of the spent fuel storage pool design is found in the UFSAR, Section 10.3 (Ref. 1). The assumptions of the fuel handling accident are found in the UFSAR, Section 14.6.4 (Ref. 2).

APPLICABLE SAFETY ANALYSES

The water level above the irradiated fuel assemblies is an implicit assumption of the fuel handling accident. A fuel handling accident is evaluated to ensure that the radiological consequences ~~(calculated whole-body and thyroid doses at the site boundary)~~ are well below the guidelines set forth in ~~10 CFR 100~~**10 CFR 50.67** (Ref. 3) **as modified in Regulatory Guide 1.183, Table 6.** A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in Reference 2.

The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto the reactor core. The consequences of a fuel handling accident over the spent fuel storage pool are ~~no more~~**less** severe than those of the fuel handling accident over the reactor core. The water level in the spent fuel storage pool provides for absorption of water soluble fission product gases ~~and transport delays of soluble and insoluble gases that must pass through the water~~ before being released to the secondary containment atmosphere. ~~This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident.~~**Noble gases are not retained in the water and particulates are retained in the water (RG 1.185, Appendix B, Item 3).**

The spent fuel storage pool water level satisfies Criteria 2 and 3 of the NRC Policy Statement.

LCO

The specified water level (232 ft 3 inches plant elevation, which is equivalent to 22 ft over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks) preserves the assumptions of the fuel handling accident analysis (Ref. 2). As such, it is the minimum required for fuel movement within the spent fuel storage pool.

(continued)

BASES (continued)

APPLICABILITY

This LCO applies during movement of fuel assemblies in the spent fuel storage pool since the potential for a release of fission products exists.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. If moving fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of fuel assemblies is not a sufficient reason to require a reactor shutdown.

When the initial conditions for an accident cannot be met, action must be taken to preclude the accident from occurring. If the spent fuel storage pool level is less than required, the movement of fuel assemblies in the spent fuel storage pool is suspended immediately. Suspension of this activity shall not preclude completion of movement of a fuel assembly to a safe position. This effectively precludes a spent fuel handling accident from occurring.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

This SR verifies that sufficient water is available in the event of a fuel handling accident. The water level in the spent fuel storage pool must be checked periodically. The 7 day Frequency is acceptable, based on operating experience, considering that the water volume in the pool is normally stable, and all water level changes are controlled by unit procedures.

REFERENCES

1. UFSAR, Section 10.3.
2. UFSAR, Section 14.6.4.
3. ~~10 CFR 100~~**10 CFR 50.67.**

B 3.9 REFUELING OPERATIONS

B 3.9.6 Reactor Pressure Vessel (RPV) Water Level

BASES

BACKGROUND

The movement of fuel assemblies or handling of control rods within the RPV requires a minimum water level of 458 inches above RPV instrument zero. During refueling, this maintains a sufficient water level in the reactor vessel cavity and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to well below the guidelines set forth in ~~10 CFR 100~~**10 CFR 50.67** (Ref. 3) **as modified in Regulatory Guide 1.1.83, Table 6.**

APPLICABLE SAFETY ANALYSES

During movement of fuel assemblies or handling of control rods, the water level in the RPV and the spent fuel pool is an implicit initial condition design parameter in the analysis of a fuel handling accident in containment postulated in Reference 1. A minimum water level of 20 ft 11 inches above the top of the RPV flange allows a partition factor of 100 to be used in the accident analysis for halogens (Ref. 1).

Analysis of the fuel handling accident inside containment is described in Reference 1. With a minimum water level of 458 inches above RPV instrument zero (20 ft 11 inches above the top of the RPV flange) and a minimum decay time of 24 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and that offsite doses are maintained within allowable limits (Ref. 3).

While the worst case assumptions include the dropping of an irradiated fuel assembly onto the reactor core, the possibility exists of the dropped assembly striking the RPV flange and releasing fission products. Therefore, the minimum depth for water coverage to ensure acceptable radiological consequences is specified from the RPV flange. Since the worst case event results in failed fuel assemblies seated in the core, as well as the dropped assembly,

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level of 458 inches above RPV instrument zero ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is met. Water at the required level limits the consequences of damaged fuel rods, which are postulated to result from a fuel handling accident in containment (Ref. 1).

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 on valve positions, which make significant unplanned level changes unlikely.

REFERENCES

1. UFSAR, Section 14.6.4.

2. UFSAR, Section 10.3.

3. ~~10 CFR 100.11~~**10 CFR 50.67.**

ATTACHMENT 4

PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 AND 3

Docket Nos. 50-277
50-278

Renewed License Nos. DPR-44
DPR-56

License Amendment Request
"PBAPS Alternative Source Term Implementation"

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	COMMITTED DATE OR "OUTAGE"	COMMITMENT TYPE	
		One-Time Action (Yes/No)	Programmatic (Yes/No)
1) Per TSTF-51, Revision 2, licensees adding the term "recently" must make the following commitment which is consistent with 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)". Exelon makes a commitment to the following NUMARC 93-01 section: *"In addition to the guidance in NUMARC 91-06, for plants which obtain license amendments to utilize shutdown safety administrative controls in lieu of Technical Specification requirements on primary or secondary containment operability or ventilation system operability, during fuel handling or core alterations, the following guidelines should be included in the assessment of systems removed from service:* *-During Fuel Handling/Core Alterations, ventilation system and radiation monitor availability (as defined in NUMARC 91-06) should be assessed, with respect to filtration and monitoring of releases from the fuel. Following shutdown, radioactivity in the RCS decays away fairly rapidly. The basis of the Technical Specification operability amendment is the reduction in doses due to such decay. The goal of maintaining ventilation system and radiation monitoring availability is to reduce doses even further below that provided by the natural decay and to avoid unmonitored releases.* *-A single normal or contingency method to promptly close primary or secondary containment penetrations should be developed. Such prompt methods need not completely block the penetration or be capable of resisting pressure. The purpose of this is to enable ventilation systems to draw the release from a postulated fuel handling accident in the proper direction such that it can be treated and monitored."* Peach Bottom is defining the definition of prompt in this context to mean being accomplished within 1 hour.	Upon Implementation	No	Yes

ATTACHMENT 5

PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 AND 3

Docket Nos. 50-277
50-278

Renewed License Nos. DPR-44
DPR-56

License Amendment Request
"PBAPS Alternative Source Term Implementation"

Compact Disk Containing PBAPS Meteorological Data
(For Information Only)

ATTACHMENT 6

PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 AND 3

Docket Nos. 50-277
50-278

Renewed License Nos. DPR-44
DPR-56

License Amendment Request
"PBAPS Alternative Source Term Implementation"

UFSAR Section 5.2.4.3.2 Mark-Up and Supporting Technical Information

UFSAR Section 5.2.4.3.2 Mark-Up and Supporting Technical Information

By letter dated July 14, 2003, Exelon submitted proposed changes to the Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3, Updated Final Safety Analysis Report (UFSAR) in support of a proposed License Amendment Request (LAR) to implement Alternative Source Term (AST) methodology. During its review of the proposed LAR, the NRC requested additional information regarding the proposed UFSAR changes by letter dated January 16, 2004. Exelon responded to the NRC requests for information by letters dated April 23, 2004, and May 20, 2004 (References 1 2). Exelon subsequently withdrew the LAR and the UFSAR changes were not implemented.

However, in support of this LAR submittal, Exelon is resubmitting updated information regarding the proposed UFSAR changes in support of the implementation of AST methodology. To help facilitate the NRC's review of the proposed UFSAR changes, we have restated the specific requests for additional information identified in the earlier NRC correspondence followed by our updated response.

January 16, 2004, NRC letter - RAI Question 10

Question:

The proposed change to the UFSAR (Section 8 of the LAR) states that the temperature profile presented in UFSAR Figure 14.6.12A includes a 2σ adder for decay heat. The figure is identified in the UFSAR as Revision 15 dated 04/98. During an amendment review in 2000, it was stated that the figure did not include this adder and additional information was provided to the staff to justify adequate conservatism in the MCPA calculation without the adder at that time (Letter from B.C. Buckley, Sr., USNRC to J.A. Hutton, PECO Energy Company, August 14, 2000, "Peach Bottom Atomic Power Station, Unit Nos. 2 and 3 - Issuance of Amendment Regarding Crediting of Containment Overpressure for Net Positive Suction Head Calculations For Emergency Core Cooling Pumps (TAC Nos. MA6291and MA6292)."). Explain the discrepancy in these two statements. Has the analysis and figure been updated to include the adder and approved for use? If the figure does include this adder then why was this not identified during the amendment review in 2000?

Updated Response:

The analyses provided during the amendment review in 2000 did not explicitly include a 2σ adder for decay heat, and additional information was provided to the staff to justify adequate conservatism in the MCPA calculation without the adder. Subsequent to that review, it was Exelon's intent to explicitly incorporate a 2σ adder for decay heat in future containment analyses. On May 24, 2001, GE issued SIL-636, and on June 6, 2001, SIL-636 revision 1, informing GE BWR owners of errors in the GE specific implementation of the ANSI/ANS 5.1-1979 decay heat standard. New decay heat values and uncertainties specific to PBAPS were regenerated using corrected GE procedures. Reanalysis of the containment response to a DBA-LOCA was performed by GE using their NRC approved SHEX computer code. This analysis included an explicit 2σ adder for decay heat. As described in the amendment review in 2000, it is this vendor containment analysis that was used as the basis for the revised utility generated MCPA calculation referred to in this submittal. With the explicit incorporation of the 2σ adder for decay heat, the additional justification of adequate conservatism in the MCPA calculation without the adder has been deleted.

January 16, 2004, NRC letter - RAI Question 11

Question:

The proposed revised UFSAR text identifies a change in methodology regarding how the containment leakage is addressed in the MCPA (minimum containment pressure available) analysis. (a) Provide the MCPA and containment overpressure license (COPL) calculation for staff review. (b) How is it different from the previously reviewed method described in PECO Energy Company Calculation PM-1013, "Minimum Containment Pressure Calculation", Revision 3, February 2000? (c) How are the MSIV and airlock leakages included in the calculation? (d) How are the leakages conservatively varied with the containment pressure assuming turbulent flow?

Updated Response:

(a) Exelon Nuclear Design Analysis PM-1013, "Minimum Containment Pressure Available", Revision 5 is available for review at Exelon Nuclear sites or at NRC offices.

(b)(c)(d) The previous containment leakage assumptions considered the 0.5% weight per day leakage as remaining constant over time, throughout the entire event. The constant nitrogen leak rate from the containment was (Equation 2 from Revision 3).

$$\dot{m} = \frac{144 * (P_l + P_{atm}) * L_a * V}{24 * 3600 * R_a * (T_l + T_0)}$$

where :

$\dot{m}$ = nitrogen leakage mass flow, lbm/sec

P_l = nitrogen leakage pressure, psig

P_{atm} = atmospheric pressure, psia

L_a = nitrogen volumetric leakage rate, % per day

R_a = nitrogen ideal gas constant, (ft-lbf)/(lbm-°R)

T_l = nitrogen leakage temperature, °F

T_0 = temperature conversion, °F to °R

Values were selected to maximize the leakage (i.e. dry, maximum P_l and minimum T_l). Only nitrogen is assumed to be expelled during the event.

The revised containment leakage methodology is determined based upon the proposed PBAPS Technical Specification limit of 0.70% weight per day for general containment leakage at the test pressure of 49.1 psig, plus 250 scfh to bound the proposed PBAPS Technical Specification limit of 204 scfh for total MSIV leakage at a test pressure of 25 psig, plus the current PBAPS Technical Specification limit of 9000 scc per minute for airlock seal leakage at a test pressure of 49.1 psig. MSIV and airlock leakages are converted to the equivalent percent weight per day as follows (Assumption 5.J from Revision 5):

$$L_{MSIV} = \frac{24 \cdot 144 \cdot Q_{MSIV} \cdot P_{atm}}{R_N \cdot (60 + T_0) \cdot M_{a_i}} \cdot \sqrt{\frac{\Delta P_{base}}{\Delta P_{test}}}$$

$$L_{airlock} = \frac{24 \cdot 60 \cdot 144 \cdot Q_{airlock} \cdot P_{atm}}{30.48^3 \cdot R_N \cdot (60 + T_0) \cdot M_{a_i}} \cdot \sqrt{\frac{\Delta P_{base}}{\Delta P_{test}}}$$

where:

L_{MSIV} = MSIV leak rate, % per day

$L_{airlock}$ = airlock leak rate, % per day

Q_{MSIV} = MSIV leak rate, scfh

$Q_{airlock}$ = airlock leak rate, sccm

P_{atm} = atmospheric pressure, psia

R_N = gas constant for Nitrogen

M_{a_i} = initial containment Nitrogen mass, lbm

ΔP_{base} = containment reference differential pressure, 49.10 psid

ΔP_{test} = test reference differential pressure, psid

T_0 = temperature conversion, °F to °R

Using the PBAPS Technical Specification leakage limits and the above expressions, L_{MSIV} and $L_{airlock}$ are calculated as 3.65% and 0.20% per day, respectively. Combining with the 0.70% per day general containment leakage, a total containment leak rate of 4.55% per day is estimated. Use of the assumed total containment leak rate of 4.55% per day, i.e. the maximum values, at the same time, is conservative.

Since these containment leakage values are stated at their respective test pressures, the leakage at any given time is assumed to be a function of containment pressure and the time-dependent non-condensible mass in the containment:

$$L(t) = M_a(t) \cdot \frac{(L_{containment} + L_{MSIV} + L_{airlock})}{24 \cdot 3600} \cdot \Delta t \cdot \sqrt{\frac{\Delta P_{DW}(t)}{\Delta P_{test}}}$$

where:

$L(t)$ = total leakage, lbm

$M_a(t)$ = containment Nitrogen mass, lbm

$L_{containment}$ = containment leak rate, % per day

L_{MSIV} = MSIV leak rate, % per day

$L_{airlock}$ = airlock leak rate, % per day

Δt = time step size, seconds

$\Delta P_{DW}(t)$ = containment differential pressure, psid

ΔP_{test} = test reference differential pressure, 49.10 psid

January 16, 2004, NRC letter - RAI Question 12

Question:

Previously, containment leakage was assumed to be constant at L_a = 0.5%/day throughout the event. The containment leakage has been increased to L_a = 0.7%/day for the first 24 hours, based on the proposed change to Technical Specification (TS) 5.5.12, for a peak post-accident containment pressure of 49.1 psig. This leakage is then reduced to 0.56 x L_a = 0.392%/day from 24 to 38 hours and then reduced to 0.50 x L_a = 0.350%/day, for 38 to 720 hours. In addition, MSIV leakage of 174 scfh is included (based on the proposed change to TS 3.6.1.3) in the MCPA calculation, with leakage measured at a test pressure of 25 psig. After 24 hours, the MSIV leak rate is reduced to 77.2%, then to 65.4% at 48 hours, to 59.0% at 72 hours, to 55.5% at 96 hours, and finally to 50% at 157 hours for the remainder of the event. Leakage from the personal airlock of 9,000 sccm, for a peak post-accident containment pressure of 49.1 psig, is also included in the proposed change to the MCPA calculation. (a) How are the leakages conservatively varied with the containment pressure assuming turbulent flow? (b) How does this evaluation differ from the MCPA and COPL calculation in RAI 2, which is only carried out to 12.5 hours? (c) Identify the TS which controls the allowable airlock leakage rate.

Updated Response:

The leakage values stated in the question above are only used to support dose analysis and are governed by the guidance relative to containment leakage in Regulatory Guide 1.183. These leakage assumptions are not the same as those used in the determination of the MCPA in calculation PM-1013, which does not support, take input from or provide input to the dose calculations.

As described in the previous response to NRC RAI Question #11, the reference containment leakage is assumed to be 4.55% weight per day at a reference containment pressure of 49.1 psig. In addition to the 0.7% weight per day general containment leakage, this also includes the leakage from the MSIVs and personnel airlock (L_{MSIV} and $L_{airlock}$ are 3.65%, 0.20% weight per day, respectively). During the analysis of the MCPA, the % weight leakage is varied only as a function of the containment pressure:

$$L(t) = M_a(t) \cdot \frac{\left(L_{containment} + L_{MSIV} + L_{airlock}\right)}{24 \cdot 3600} \cdot \Delta t \cdot \sqrt{\frac{\Delta P_{DW}(t)}{\Delta P_{test}}}$$

$$
\begin{aligned}
\text{where}: \\
L(t) &= \text{total leakage, lbm} \\
M_a(t) &= \text{containment Nitrogen mass, lbm} \\
L_{containment} &= \text{containment leak rate, \% per day} \\
L_{MSIV} &= \text{MSIV leak rate, \% per day} \\
L_{airlock} &= \text{airlock leak rate, \% per day} \\
\Delta t &= \text{time step size, seconds} \\
\Delta P_{DW}(t) &= \text{containment differential pressure, psid} \\
\Delta P_{test} &= \text{test reference differential pressure, 49.10 psid}
\end{aligned}
$$

January 16, 2004, NRC letter - RAI Question 13

Question:

During the previous amendment review (Hutton, J. A., PECO Energy Company, to USNRC, "Peach Bottom Atomic Power Station, Units 2 and 3 Response to May 10, 2000, Telephone Questions Regarding PECO Energy License Amendment Request Related to Generic Letter 97-04," June 29, 2000) it was stated that the margin between the MCPA and COPL was set at 1 foot (0.42 psid), as agreed to on 11/18/98. The proposed amendment would decrease this margin to about 0.28 psid. (a) Provide a justification for reducing this agreed to margin. (b) Provide a comparison of the COPL value to the COPR (containment overpressure required) value for the RHR and core spray pumps for the most limiting event(s), including the margin to the COPL value before and after the proposed change to the MCPA/COPL calculation.

Response:

The COPL was established in reviews culminating in the previously mentioned NRC SER (Letter from B.C. Buckley, Sr., USNRC to J.A. Hutton, PECO Energy Company, August 14, 2000, "Peach Bottom Atomic Power Station, Unit Nos. 2 and 3 - Issuance of Amendment Regarding Crediting of Containment Overpressure for Net Positive Suction Head Calculations For Emergency Core Cooling Pumps (TAC Nos. MA6291 and MA6292)"). From that SER:

> *"The NRC staff performed confirmatory calculations of the RHR NPSH analysis. According to our calculations, the minimum margin between the COPL and the COPR or the RHR pumps is 0.88 psig. This occurred at the peak suppression pool temperature of 205.7ºF. This margin allows for minor design changes which could affect the COPR. This result is consistent with the licensee's calculations. Additionally, our calculations demonstrated that the minimum margin between the COPL and MCPR was approximately 0.42 psig (1 foot). Because of the way the COPL was defined, i.e., the COPL will be 1 foot less than the MCPA for a design basis LOCA, this minimum margin is maintained over the entire COPL curve."*

and:

"For the long term following a LOCA, the staff has approved the use of the containment overpressure depicted on UFSAR Figure 5.2.16 and provided in the table above for both the RHR and core spray pumps."

Since that time, some plant changes have been made which were not considered within the original intent of a minor design change. These changes included increases in the Technical Specification allowable river water temperature from 90°F to 92°F, correction of decay heat errors identified by GE in SIL-636 Rev.1, the formal incorporation of 2σ decay heat uncertainty in the containment calculations, and the currently proposed increases in MSIV and containment leakage as part of AST. These changes had the net effect of decreasing the MCPA. The change in MCPA methodology necessary to accommodate AST proposed leakages, and the potential impact to COPL is the very reason Exelon Nuclear has requested this NRC review.

(a) Although the COPL line was derived using a 1 foot margin to the MCPA, it is our understanding that, like the original Peach Bottom FSAR containment overpressure limit line, the NRC SER has established the COPL line itself, "depicted on UFSAR Figure 5.2.16" as the limit, rather than the maintaining of a specified margin to the MCPA. In that case, the proposed AST changes would have reduced the MCPA margin to the previous COPL fixed limit line from 1 foot of head (7.41-6.99=0.42 psid) to essentially zero (7.00-6.99=0.01 psid). Consequently, a new COPL limit needs to be proposed.

If instead, maintaining a 1-foot of head margin to the MCPA were maintained, adequate overpressure would still be available to satisfy the NPSHa requirements of the RHR pump durnig the design basis LOCA. A 1 foot of head margin to the new MCPA would produce a peak COPL of 6.59 psig, which still provides another foot of margin to the peak COPR for RHR of 6.14 psig (6.59-6.14=0.45 psid).

The COPL line that was proposed with the AST amendment request preserved the relative relationship between the COPR, COPL, and MCPA from the previous NRC review.

(b) The following table summarizes the COPL, MCPA, and COPR data provided in the attached charts.

	Peak MCPA	**Peak COPL**	**Peak COPR (RHR)**	**Peak COPR (CS)**
PM-1013 R3	7.41	6.99	6.11	4.83
PM-1013 R5	7.00	6.59	6.14	4.78

References:

1. Letter from Mr. J. P. Gallagher (Exelon Nuclear) to U.S. NRC dated April 23, 2004, "Supplement to the Request for License Amendments Related to Application of Alternative Source Term," dated July 14, 2003.

2. Letter from Mr. K. Jury (Exelon Nuclear) to U.S. NRC dated May 20, 2004, "Supplement to the Request for License Amendments Related to Application of Alternative Source Term," dated July 14, 2003.

In order to assess the primary containment response after the blowdown and to demonstrate the adequacy and redundancy of the core and containment spray cooling systems, an analysis has been made of the recirculation line break under various conditions of core and primary containment cooling.

The long-term pressure and temperature response of the primary containment has been analyzed for the following cooling conditions:

1. Two reactor core spray loops and both RHRS loops with four RHRS pumps, four heat exchangers, and four high-pressure service water pumps - with containment spray.

2. One reactor core spray loop and one RHRS loop with one RHRS pump, one heat exchanger, and one high-pressure service water pump per loop - with containment spray.

3. One reactor core spray loop and one RHRS loop with one RHRS pump, one heat exchanger, and one high-pressure service water pump - with no containment spray.

4. Same as Case 3 above, but also analyzed for operation at 3458 MWt.

These analyses are presented in Section 14.0, "Plant Safety Analysis."

5.2.4.3.2 Minimum Containment Pressure Available

Emergency pumps that take suction from the suppression pool rely on some amount of containment pressure to provide for adequate net positive suction head (NPSH) at elevated suppression pool temperatures. The bounding event for containment overpressure required (COPR) is the design basis large break loss of coolant accident (LOCA).

Figure 5.2.16 provides the results of an analysis to determine the minimum containment pressure available (MCPA) following a LOCA. The suppression pool temperature used for this analysis is the design basis LOCA response of Figure 14.6.12A. Conservative assumptions and inputs used in the analysis for the suppression pool temperature response are discussed in Section 14.6.3. Additional assumptions and inputs used to determine the MCPA are listed below.

1. Offsite power is assumed lost at the time of the accident and is not restored for the duration of the event.

2. One of the onsite diesel-generators fails to start and remains out of service during the entire event.

3. The RHR heat exchanger performance and high pressure service water (HPSW) supply flow rate and temperature are consistent with the design basis LOCA analysis (Section 14.6.3).

4. Prior to the accident the maximum temperature of 145°F exists in the drywell together with 100% relative humidity. Temperature in the wetwell is also assumed at its maximum of 95°F and 100% humidity.

5. Minimum pre-accident containment pressure of 0 psig.

~~6. A containment gas leakage rate of 0.5% per day. This leakage is the maximum allowable containment leakage and is assumed to be constant throughout the event, even at low containment pressures. It is also assumed that only non-condensable gas leaks.~~

6. **Containment leakage is assumed consistent with the PBAPS Technical Specification limits. Specifically, containment leakage assumed includes (a) a general containment leakage rate of 0.7% per day at the peak post-accident containment pressure of 49.1 psig, and specific containment leakage rates through (b) the main steam isolation valves of ~~150%~~ 250 scfh at a test pressure of 25 psig, and (c) the personal airlock of 9,000 sccm at the peak post-accident containment pressure of 49.1 psig. The evaluated containment leakage rate is conservatively varied with containment pressure assuming turbulent flow. It is also assumed that only non-condensable gas leaks.**

7. At 10 minutes following the initiation of the event, pump flow rates are confirmed at their design flow rates and one loop of suppression pool cooling is initiated.

8. Suppression pool cooling return is via containment sprays. Spray effectiveness of 100% is assumed such that the containment atmosphere is saturated at the spray temperature.

~~Although t~~**The** ~~decay heat model used to generate the suppression pool temperature profile of Figure 14.6.12A did not includes a 2σ adder, other assumptions and input values ensure that the temperature profile of Figure 14.6.12A is conservative. This position has been reviewed and approved by the NRC in their letter dated August 14, 2000~~. **Subsequent to the August 14, 2000 NRC approval of the MCPA methodology, the suppression pool temperature profile of Figure 14.6.12A was reanalyzed using a decay heat model which explicitly includes a 2σ adder. The temperature profile of the revised analysis remains bounded by the temperature profile of Figure 14.6.12A.**

The MCPA analysis of Figure 5.2.16 begins at 10 minutes following initiation of the event. Although the containment pressure response of Figure 14.6.10A is a maximum containment pressure profile, without the use of containment sprays a minimum containment pressure profile would not be significantly less than the profile of Figure 14.6.10A.

The MCPA analysis of Figure 5.2.16 is evaluated until just after the suppression pool temperature reaches its peak and begins to decrease, about 12.5 hours. Beyond this time MCPA continues to decrease until it again becomes atmospheric.

Containment **Over**-Pressure License **(COPL)**

Because of the conservative assumptions and input values used in the MCPA analysis, use of the MCPA in NPSH analyses is conservative. However, the PBAPS licensing basis grants containment overpressure credit as follows:

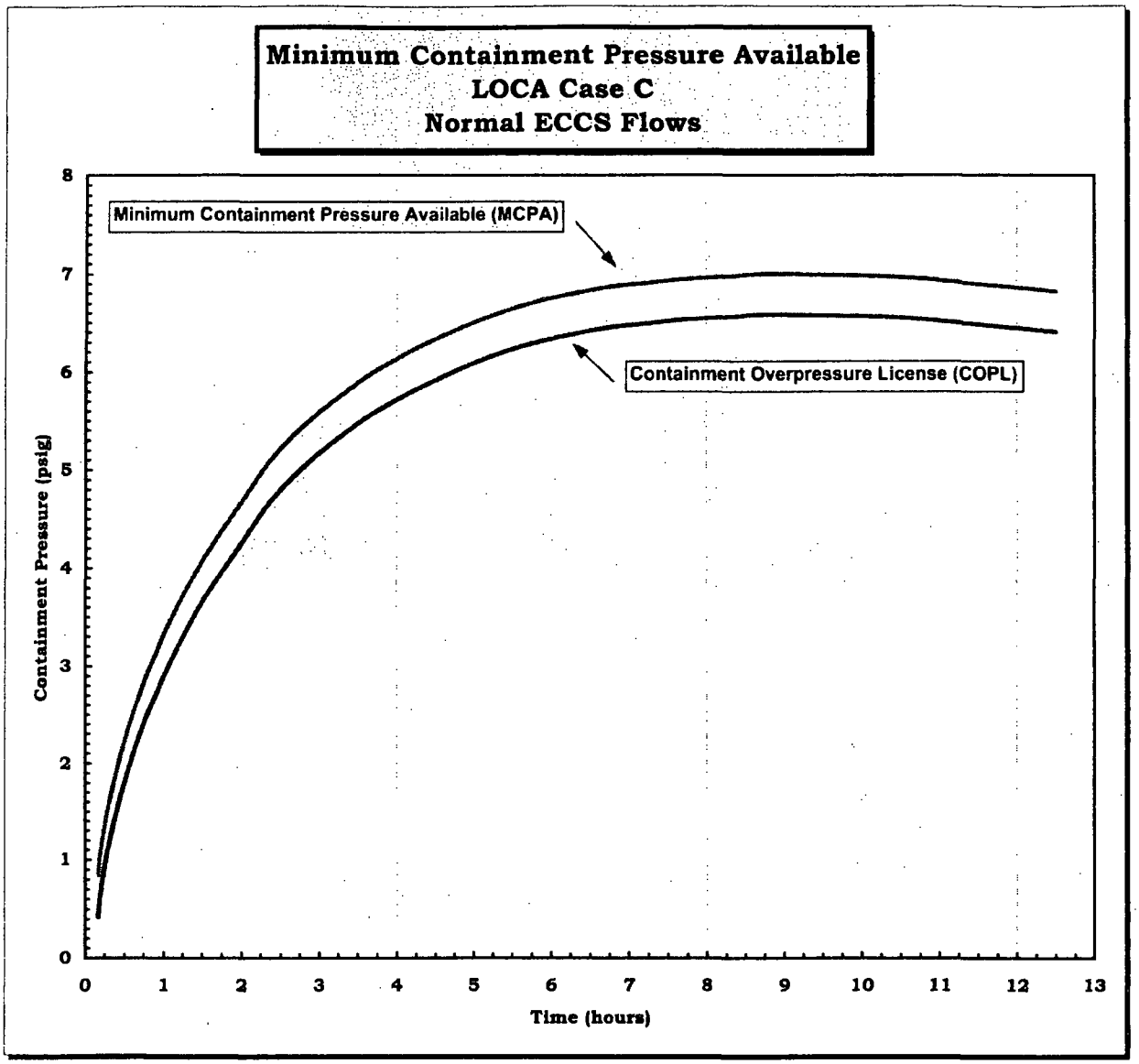
Time	COPL
0 to 10 minutes	2.25 psig
10 minutes to 12.5 hours	COPL of Figure 5.2.16
12.5 to 78 hours	Exponentially decreasing from Figure 5.2.16 to zero

For any design basis event, the maximum containment overpressure credit allowed is therefore the MCPA associated with that event, but not greater than the containment overpressure credit given in the table above.

5.2.4.4 Primary Containment Capability

The pressure of the primary containment system depends on both the system temperatures and the amount of non-condensible gases. Thus, the capability of the system to house resulting gases from metal-water reaction varies with the rate and extent of the reaction.

Capability is defined as the maximum percent of fuel channels and fuel cladding material which can enter into metal-water reaction during a specified duration without the design pressure of the containment structure being exceeded. The analysis of the postulated LOCA discussed in Section 14.0, "Plant Safety Analysis," shows that the operation of either of the two core spray system loops will maintain continuity of core cooling such that the extent of the resultant metal-water reaction would be 0.1 percent or less. However, to evaluate the containment system design capability, various percentages of metal-water reaction were assumed to take place over various durations of time. This analysis presents an arbitrary method of measuring system capability without requiring prediction of the detailed events in a particular accident condition. The results are presented in Section 14.0, "Plant Safety Analysis."



PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 AND 3
UPDATED FINAL SAFETY ANALYSIS REPORT

MINIMUM CONTAINMENT PRESSURE AVAILABLE AND CONTAINMENT OVERPRESSURE LICENSE

FIGURE 5.2.16 Rev. ~~18~~ XX 04/~~01~~ XX

ATTACHMENT 7

PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 AND 3

Docket Nos. 50-277
50-278

Renewed License Nos. DPR-44
DPR-56

License Amendment Request
"PBAPS Alternative Source Term Implementation"

Regulatory Guide 1.183 - Conformance Tables

REGULATORY GUIDE 1.183 COMPARISON

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	PBAPS 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, including the BWR extended burnup library, was used to determine core inventory using the current licensed rated reactor thermal power of 3514 MWt. Source terms were evaluated at end-of-cycle and at beginning of cycle (100 effective full power days (EFPD) to achieve equilibrium) conditions and worst case inventory used for the selected isotopes. These values were then divided by the power to obtain activity in units of Ci/MWt. Consideration was also given to fuel enrichment in order to ensure bounding conditions. Accident analyses are based on a maximum power level of 3,528 MWt, which includes 0.38% margin for instrument uncertainty relative to the rated thermal power of 3,514 MWt after the 1.62% Thermal Power Optimization uprate. Source terms are based on a 2 year fuel cycle with a nominal 711 EFPD per cycle. (See Attachment 1, section 4.1.2 for details of sensitivity studies performed).
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	LOCA analysis uses core average inventory of all fuel assemblies. 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, consistent with PBAPS COLR.

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections

RG Section	RG Position	PBAPS Analysis	Comments
3.1	No adjustment to the fission product inventory should be made for events postulated to occur during power operations at less than full rated power or those postulated to occur at the beginning of core life. For events postulated to occur while the facility is shutdown, e.g., a fuel handling accident, radioactive decay from the time of shutdown may be modeled.	Conforms	No adjustments for less than full power fission product inventory are made in any analyses, including the FHA (decay from the time of shutdown is modeled). Full power is also assumed for the CRDA event although it is a low power event. No fuel damage is postulated during an MSLB.
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 1** **BWR 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 / 0.25 / 0.3 Alkali Metals: 0.05 / 0.20 / 0.25 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	Conforms	The fractions from Regulatory Position 3.1, Table 1 are used. Footnote 10 criteria are met.

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	PBAPS Analysis	Comments
	in this section may not be applicable to cores containing mixed oxide (MOX) fuel.		
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** **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.*	Conforms	Complies with Footnote 11. Peaking factor of 1.7 used for DBA events that do not involve the entire core.
3.3	Table 4 tabulates the onset and duration of each sequential release phase for DBA LOCAs at PWRs and BWRs. The specified onset is the time following the initiation of the accident (i.e., time = 0). The early in-vessel phase immediately follows the	Conforms	The BWR release durations from Table 4 are used. LOCA is modeled in a linear fashion. Non-LOCA accidents are modeled as instantaneous releases.

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections

RG Section	RG Position	PBAPS Analysis	Comments
	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** **PWRs** **BWRs** **Phase** **Onset** **Duration** **Onset** **Duration** Gap Release 30 sec 0.5 hr 2 min 0.5 hr Early In-Vessel 0.5 hr 1.3 hr 0.5 hr 1.5 hr		
3.3	For facilities licensed with leak-before-break methodology, the onset of the gap release phase may be assumed to be 10 minutes. A licensee may propose an alternative time for the onset of the gap release phase, based on facility-specific calculations using suitable analysis codes or on an accepted topical report shown to be applicable for the specific facility. In the absence of approved alternatives, the gap release phase onsets in Table 4 should be used.	Not Applicable	PBAPS does not use leak-before-break methodology for DBA analyses.
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.

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	PBAPS Analysis	Comments
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 is applied in the analyses.
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	Fuel damage assessment for CRDA and FHA are based on GESTAR standard analyses for GE14 fuel. No fuel damage is postulated during the MSLB.
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. Footnote 13: *The prior practice of basing inhalation exposure on only radioiodine and not including radioiodine in external exposure calculations is not consistent with the definition of*	Conforms	TEDE is calculated, with significant progeny included.

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	PBAPS Analysis	Comments
	TEDE and the characteristics of the revised source term.		
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. The factors in the column headed "effective" are used.
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 with either RG 1.183 or SRP 6.4	The LOCA analysis uses the given values from Section 4.1.3 of RG 1.183. The analyses for the FHA, CRDA and MSLB use the un-rounded values from SRP 6.4.
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. The factors in the column headed "effective" are used.
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	Conforms	The maximum 2-hour EAB dose was analyzed for all accidents. Conservatively, the maximum 2-hour period dose was determined by adding the maximum

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	PBAPS Analysis	Comments
	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.*		2-hour dose for each of the release pathways 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	The most limiting receptor at the outer boundary of the LPZ was evaluated.
4.1.7	No correction should be made for depletion of the effluent plume by deposition on the ground.	Conforms	No such corrections 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, 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	Conforms	All sources of radiation that would cause exposure to control room personnel were evaluated and can be found in the accident calculations. The post-LOCA radioactive release pathways that contribute to the CR TEDE dose are post-LOCA Containment Leakage, post-LOCA ESF Leakage, and Post-LOCA MSIV Leakage. The radioactivity from the above sources is assumed to be released into the atmosphere and transported to the CR air intake, where it may leak into the CR envelope or be filtered by the CR intake filtration system prior to being distributed in the

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections

RG Section	RG Position	PBAPS Analysis	Comments
	buildup in recirculation filters.		CR envelope. The four major radioactive sources which contribute to the CR TEDE dose are post-LOCA airborne activity inside the CR, post-LOCA airborne cloud external to CR, post-LOCA containment shine to CR, and post-LOCA Main Control Room Emergency Ventilation (MCREV) filter shine.
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. This assumption does not result in non-conservative results for the control room.
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. Footnote 15: *The Iodine Protection Factor (IPF) methodology of Reference 22 may not be adequately conservative for all DBAs and control room arrangements since it models a steady-state control room condition. Since many analysis parameters change over the duration of the event, the IPF methodology should only be used with caution. The NRC computer codes HABIT (Ref. 23) and RADTRAD (Ref. 24) incorporate suitable methodologies.*	Conforms	This guidance is applied in the analyses through the use of the incorporated methodologies within the RADTRAD computer code. MicroShield is utilized for shielding analysis.
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	Exception to RG 1.52 taken regarding MCREV bypass per PBAPS current	Control Room intake filtration (MCREV) is credited in the accident analyses, as CR filtration conforms to Regulatory Guide 1.52 except for the allowable 1% bypass value. All credited equipment is qualified in accordance with the cited guidance. No credit is taken for SGTS HEPA or charcoal adsorber filtration in any accident analysis. (see Section 4.2.1.2 of

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	PBAPS Analysis	Comments
	Adsorption Units of Light-Water-Cooled Nuclear Power Plants" (Ref. 25), for guidance.	Licensing basis (TS 5.5.7).	Attachment 1 for MCREV requirements).
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	No Credit for the use of PPE or prophylactic drugs is 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. Footnote 16: *This occupancy is modeled in the X/Q values determined in reference 22 and should not be credited twice. The ARCON96 code (Ref. 26) does not incorporate these occupancy assumptions, making it necessary to apply this correction in the does calculations.*	Conforms with RG 1.183 or SRP 6.4	The cited occupancy factors and LOCA breathing rates from RG 1.183 are used. The SRP 6.4 breathing rates are used for the FHA, CRDA and MSLB. The ARCON96 code was used for X/Q calculations (occupancy factors are not applied in X/Q calculations).
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.

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	PBAPS Analysis	Comments
4.3	The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737 (Ref. 2). Design envelope source terms provided in NUREG-0737 should be updated for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of TEDE. Integrated radiation exposure of plant equipment should be determined using the guidance of Appendix I of this guide.	Exception	The Technical Support Center at PBAPS is in the Unit 1 Control Room (outside of the Protected Area Boundary). A review of the current TID-14844 based analysis indicates that it is unnecessary to reanalyze doses therein to assure accessibility. For other areas requiring plant personnel access, a qualitative assessment of the regulatory positions on source terms indicates that, with no new operator actions required, radiation exposures are bounded by those previously analyzed. (See Section 4.2.4 of Attachment 1 of this submittal.)
4.4	The radiological criteria for the EAB, the outer boundary of the LPZ, and for the control room are in 10 CFR 50.67. These criteria are stated for evaluating reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation, e.g., a large-break LOCA. The control room criterion applies to all accidents. For events with a higher probability of occurrence, postulated EAB and LPZ doses should not exceed the criteria tabulated in Table 6. The acceptance criteria for the various NUREG-0737 (Ref.2) items generally reference General Design Criteria 19 (GDC 19) from Appendix A to 10 CFR Part 50 or specify criteria derived from GDC-19. These criteria are generally specified in terms of whole body dose, or its equivalent to any body organ. For facilities applying for, or having received, approval for the use of an AST, the applicable criteria should be updated for consistency with the TEDE criterion in 10 CFR 50.67(b)(2)(iii).	Conforms	The criteria used for acceptance in measurement for the EAB, the outer boundary of the LPZ, and for the control room is the TEDE criterion in 10 CFR 50.67. The TSC at PBAPS is in the (decommissioned) Unit 1 Control Room. A review of the current TID-14844 based analysis indicates that it is unnecessary to reanalyze doses to ensure accessibility. For other areas requiring plant personnel access, a qualitative assessment of the regulatory positions on source terms indicates that, with no new operator actions required, radiation exposures would remain acceptable.
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	Conforms	These analyses were prepared following station procedures which conform to Appendix

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	PBAPS Analysis	Comments
	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.		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.	Exception taken regarding single failure criterion. Justification is provided in Section 4.2.3 of Attachment 1 of this document.	This submittal conforms with the RG position with the noted exception. Accident mitigation features that are credited within this analysis include the SLC system. The LOCA analysis takes credit for SLC System operation. The SLC System is safety-related, required to be operable by Technical Specifications, and is supplied with emergency power. The SLC System is manually initiated from the main control room, as directed by the emergency operating procedures. This is noted as an exception to Reg Guide 1.183 because the SLC system is not fully single failure proof. Credit for the use of the SLC system is justified in Attachment 1, Section 4.2.3.
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. Consideration is taken for parameter tolerances that are both conservative in one manner and non-conservative in others. In such cases, the most limiting dose parameter is considered in each manner.
5.1.4	The NRC staff considers the implementation of an AST to be a significant change to the design basis of the facility that is voluntarily initiated by the licensee. In order to issue a license amendment authorizing the use of an AST and the TEDE dose criteria, the NRC staff must make a current finding of compliance	Conforms	Analysis assumptions and methods were made per this guidance. Methods are compatible with AST's and the TEDE criteria.

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections

RG Section	RG Position	PBAPS Analysis	Comments
	with regulations applicable to the amendment. The characteristics of the ASTs and the revised dose calculational methodology may be incompatible with many of the analysis assumptions and methods currently reflected in the facility's design basis analyses. The NRC staff may find that new or unreviewed issues are created by a particular site-specific implementation of the AST, warranting review of staff positions approved subsequent to the initial issuance of the license. This is not considered a backfit as defined by 10 CFR 50.109, "Backfitting." However, prior design bases that are unrelated to the use of the AST, or are unaffected by the AST, may continue as the facility's design basis. Licensees should ensure that analysis assumptions and methods are compatible with the ASTs and the TEDE criteria.		
5.2	The appendices to this Regulatory Guide provide accident-specific assumptions that are acceptable to the staff for performing analyses that are required by 10 CFR 50.67. The DBAs addressed in these attachments were selected from accidents that may involve damage to irradiated fuel. This guide does not address DBAs with radiological consequences based on technical specification reactor or secondary coolant-specific activities only. The inclusion or exclusion of a particular DBA in this guide should not be interpreted as indicating that an analysis of that DBA is required or not required. Licenses should analyze the DBAs that are affected by the specific proposed applications of an AST.	Conforms	Full scope implementation of AST is recommended through analysis of the LOCA. However, all DBAs were re-analyzed, including the FHA, CRDA, and MSLB DBAs.
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	Conforms	New atmospheric dispersion values (X/Q) for the EAB, the LPZ, and the control room were developed, using five consecutive years of meteorological data. ARCON96 and PAVAN were used with these data to determine control room and EAB/LPZ atmospheric

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	PBAPS Analysis	Comments
	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". 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.		dispersion values. Control room X/Q from releases from the Main Stack were developed in conformance with guidance in RG 1.194. All new X/Q values utilized are included as part of this LAR. RG 1.23 wind speed categories were used for offsite X/Q analyses.

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	PBAPS 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	*Fission Product Inventory:* Core source terms are developed using ORIGEN-2.1 based methodology. *Release Fractions:* Release fractions are per Table 1 of RG 1.183, and are implemented by RADTRAD. *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 be assumed to be in particulate form.	Conforms	The stated distributions of iodine chemical forms are used. The post-LOCA suppression pool pH has 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 impact of SLC injection. Suppression pool pH remains above 7 for at least 30 days. The associated pH calculation is included as part of this LAR.
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	Conforms	For conservative analyses in evaluation of the containment leakage, and MSIV leakage

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
	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.		pathways, it is assumed that the radioactivity released from the fuel instantaneously and homogeneously mixes throughout the drywell air space. The suppression chamber free air volume is included after 2 hours based on expected flows between the drywell and the suppression chamber. Release into the drywell is terminated at the end of the 1.5 hours long LOCA release phase. For conservative assumptions in the ESF leakage pathway, it is assumed that all fission products released from the fuel to the containment instantaneously and homogeneously mix in the suppression pool water at the time of release from the core (See section 5.1 of this Table).
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	Credit is taken for natural deposition of aerosols within the drywell per the methodology of NUREG/CR-6189, and as implemented in RADTRAD. Powers aerosol decontamination factor is conservatively used with a 10% decontamination factor. No deterministically assumed initial plateout is credited.
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	Conforms	While containment sprays are a design feature at PBAPS, no credit is taken for airborne radioactivity removal by the containment spray

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
	in Chapter 6.5.2 of the SRP and NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays"1 (Ref. A-4). This simplified model is incorporated into the analysis code RADTRAD (Refs. A-1 to A-3). The evaluation of the containment sprays should address areas within the primary containment that are not covered by the spray drops. The mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building, provided that adequate flow exists between these regions, is assumed to be two turnovers of the unsprayed regions per hour, unless other rates are justified. The containment-building atmosphere may be considered a single, well-mixed volume if the spray covers at least 90% of the volume and if adequate mixing of unsprayed compartments can be shown. The SRP sets forth a maximum decontamination factor (DF) for elemental iodine based on the maximum iodine activity in the primary containment atmosphere when the sprays actuate, divided by the activity of iodine remaining at some time after decontamination. The SRP also states that the particulate iodine removal rate should be reduced by a factor of 10 when a DF of 50 is reached. The reduction in the removal rate is not required if the removal rate is based on the calculated time-dependent airborne aerosol mass. There is no specified maximum DF for aerosol removal by sprays. The maximum activity to be used in determining the DF is defined as the iodine activity in the columns labeled "Total" in Tables 1 and 2 of this guide multiplied by 0.05 for elemental iodine and by 0.95 for particulate iodine (i.e., aerosol treated as particulate in SRP methodology). Footnote 1: *This document describes statistical formulations with differing levels of uncertainty. The removal rate constants selected for use in design basis calculations should be those that will maximize the dose consequences. For BWRs, the simplified model should be used only if the release from the core is not directed through the suppression pool. Iodine removal in the suppression pool affects iodine species assumed by the model to be present initially.*		system in the LOCA AST reanalysis. Elemental iodine removal is credited utilizing drywell wetted surface area per SRP 6.5.2.

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
3.4	Reduction in airborne radioactivity in the containment by in-containment recirculation filter systems may be credited if these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. A-5 and A-6). The filter media loading caused by the increased aerosol release associated with the revised source term should be addressed.	Not Applicable	No in-containment recirculation filter systems exist at PBAPS.
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.	Conforms	No credit is taken for suppression pool scrubbing in the LOCA AST reanalysis. Analyses have been performed that determined that the suppression pool liquid pH is maintained greater than 7, and therefore, iodine re-evolution is not expected.
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	PBAPS does not 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	PBAPS uses a Mark I containment. Primary containment leakage is assumed as 0.7% of containment mass per day. The leak rate in the analysis is reduced to 50% after 38 hours in accordance with specific design analysis.

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	PBAPS 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.	Site Specific Conformance	The PBAPS containment is not routinely purged for pressure control during power operations. However, the PBAPS containment is purged in preparation for outages. Containment purging is currently limited to 90 hours per calendar year in accordance with UFSAR requirements. This submittal proposes to include dose requirements in the Technical Specifications in support of the proposed UFSAR changes regarding MCPA.
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	Leakage from primary containment is assumed to be released directly to the environment via the main stack after the assumed RB drawdown time. The point of release form the stack is greater than 2 and ½ times the height of any adjacent structure (i.e., Reactor Building). Therefore, credit for an elevated release through the main stack is justified. No credit is taken for ESF filters within the SGT system.
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	Ground level releases are assumed prior to the end of the drawdown period.
4.3	The effect of high wind speeds on the ability of the secondary	Conforms	The 1-hour average value wind

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
	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%).		speed exceeded only 5% of the time at PBAPS in the secondary containment vicinity is approximately 10 mph. This value is below the wind speed that would be required before the secondary containment pressures would be positive relative to outside air pressures at the downwind side of the reactor building. The ambient temperatures used are the 1-hour average value that is exceeded only 5% of the time.
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.	Conforms	No credit is taken for dilution/mixing in secondary containment.

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	PBAPS 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.	Conforms	Primary containment leakage that bypasses the secondary containment has been evaluated. No primary containment leakage, with the exception of MSIV leakage, has been identified which bypasses the secondary containment. Only the MSIV pathway leak rates are incorporated into the Technical Specifications.
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).	N/A	SGTS HEPA and charcoal adsorber filters are not credited for secondary containment release in the evaluation of analyzed accidents for onsite and offsite dose consequences.
5	ESF systems that recirculate sump water outside of the primary containment are assumed to leak during their intended operation. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components. This release source may also include leakage through valves isolating interfacing systems (Ref. A-7). The radiological consequences from the postulated leakage should be analyzed and combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. The following assumptions are acceptable for evaluating the consequences of leakage from ESF components outside the primary containment for BWRs and PWRs.	Conforms	ESF systems leakage is considered and radiological consequences evaluated using an expected leakage of 5.0 gal/min. ESF leakage radiological pathway is calculated and combined with other fission product release paths for determination of total consequential LOCA dose.
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	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 theTorus at

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
	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.		the time of release from the core. No credit for mechanistic transport is assumed.
5.2	The leakage should be taken as two times the sum of the simultaneous leakage from all components in the ESF recirculation systems above which the technical specifications, or licensee commitments to item III.D.1.1 of NUREG-0737 (Ref. A-8), would require declaring such systems inoperable. The leakage should be assumed to start at the earliest time the recirculation flow occurs in these systems and end at the latest time the releases from these systems are terminated. Consideration should also be given to design leakage through valves isolating ESF recirculation systems from tanks vented to atmosphere, e.g., emergency core cooling system (ECCS) pump min-flow return to the refueling water storage tank.	Conforms	Although PBAPS does not have a TS or licensee commitment for a leakage limit for declaring systems inoperable, an expected leakage of 5.0 gpm for the sum of the simultaneous leakage from all components in the ESF recirculation systems is multiplied by 2 in the analysis. This leak path is assumed to start at time 0, since certain ESF systems take suction immediately from the suppression pool, and terminate at the end of the credible 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 ESF 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	Not Applicable	The temperature of the leakage does not exceed 212°F.

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
	pressure; h_{f2} is the enthalpy of liquid at saturation conditions (14.7 psia, 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	The temperature of the leakage is less than 212°F. Therefore, an airborne release fraction of 10% of total iodine activity in leaked fluid is used. Although the Suppression Pool water pH is maintained above 7 for the entire 30 days of the accident dose assessment period, no additional credit for iodine retained in liquid is assumed.
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).	Exception to RG 1.52 taken regarding MCREV bypass per PBAPS current Licensing basis (TS 5.5.7).	The radioiodine available for release to the environment is assumed as 97% elemental and 3% organic. No credit is taken for SGT system filtration. (see Section 4.3.23.1 of Attachment 1 for MCREV requirements).
6	For BWRs, the main steam isolation valves (MSIVs) have design leakage that may result in a radioactivity release. The radiological consequences from postulated MSIV leakage should be analyzed and combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA.	Conforms	The proposed design basis MSIV leakage is 204 scfh for all steam lines and 116 scfh for any one line when tested at ≥ 25 psig. The consequences of MSIV leakage are analyzed and totaled to the other fission product release paths for total dose consequences from a LOCA.
6.1	For the purpose of this analysis, the activity available for release via MSIV leakage should be assumed to be that activity determined to be in	Conforms	The activity available for release via MSIV leakage is the radioactivity

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
	the drywell for evaluating containment leakage (see Regulatory Position 3). No credit should be assumed for activity reduction by the steam separators or by iodine partitioning in the reactor vessel.		released from the fuel that has instantaneously and homogeneously mixed throughout the drywell. The Torus free air volume is included after 2 hours based on expected flows between the drywell and the suppression chamber. No credit is taken for steam separators or iodine partitioning within the reactor vessel.
6.2	All the MSIVs should be assumed to leak at the maximum leak rate above which the technical specifications would require declaring the MSIVs inoperable. The leakage should be assumed to continue for the duration of the accident. Postulated leakage may be reduced after the first 24 hours, if supported by site-specific analyses, to a value not less than 50% of the maximum leak rate.	Conforms	MSIV leakage assumed in this accident analysis is 204 scfh for all steam lines and 116 scfh for any one line when tested at ≥ 25 psig. The leakage is assumed to continue for the duration of the accident and is reduced to 50% after 38 hours in accordance with specific design analysis.
6.3	Reduction of the amount of released radioactivity by deposition and plateout on steam system piping upstream of the outboard MSIVs may be credited, but the amount of reduction in concentration allowed will be evaluated on an individual case basis. Generally, the model should be based on the assumption of well-mixed volumes, but other models such as slug flow may be used if justified.	Conforms	Deposition and plateout on steam system piping upstream of the outboard MSIVs is credited only for the assumed intact line. Modeling is per the AEB 98-03 well-mixed volume approach.
6.4	In the absence of collection and treatment of releases by ESFs such as the MSIV leakage control system, or as described in paragraph 6.5 below, the MSIV leakage should be assumed to be released to the environment as an unprocessed, ground- level release. Holdup and dilution in the turbine building should not be assumed.	Conforms	MSIV leakage release is assumed to be a ground-level release without credit for holdup or dilution in the condenser or turbine building.
6.5	A reduction in MSIV releases that is due to holdup and deposition in main steam piping downstream of the MSIVs and in the main condenser, including the treatment of air ejector effluent by offgas systems, may be credited if the components and piping systems used in	Conforms	Deposition and plateout on steam system piping is credited downstream of the outboard MSIVs. All piping credited is seismically

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
	the release path are capable of performing their safety function during and following a safe shutdown earthquake (SSE). The amount of reduction allowed will be evaluated on an individual case basis. References A-9 and A-10 provide guidance on acceptable models.		analyzed to assure the piping wall integrity during and after a seismic (safe shutdown earthquake [SSE]) event. No credit is taken for holdup and deposition in the condenser or off-gas system.
7.0	The radiological consequences from post-LOCA primary containment purging as a combustible gas or pressure control measure should be analyzed. If the installed containment purging capabilities are maintained for purposes of severe accident management and are not credited in any design basis analysis, radiological consequences need not be evaluated. If the primary containment purging is required within 30 days of the LOCA, the results of this analysis should be combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. Reduction in the amount of radioactive material released via ESF filter systems may be taken into account provided that these systems meet the guidance in Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).	Conforms	Containment purging as a combustible gas or pressure control measure is not required nor credited in any design basis analysis for 30 days following a design basis LOCA at PBAPS. A new TS limit of 90 hours per calendar year is proposed in this request in order to limit the potential for another release path through purging.

Table C: Conformance with Regulatory Guide 1.183 Appendix B (Fuel Handling Accident)			
RG Section	RG Position	PBAPS 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 Section 3.1 of Table A
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 potential failed fuel rods is based on a generic evaluation of PBAPS fuel, with heavy mast. Damage due to a fuel assembly drop into the reactor vessel bounds a drop in the spent fuel pool. This is due to the 33-foot drop in the vessel as opposed to the less than 4-foot fall height in the pool.
1.2	The fission product release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. All the gap activity in the damaged rods is assumed to be instantaneously released. Radionuclides that should be considered include xenons, kryptons, halogens, cesiums, and rubidiums.	Conforms	Gap activity assumed is per this guidance.
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	The chemical form of iodine is treated in accordance with this guidance.
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	Conforms	The water depth above damaged fuel is analyzed at 23 feet. Although the actual water coverage over damaged fuel in the reactor vessel is 52 feet, no further credit is applied for the additional (i.e., >23 feet) water depth in

Table C: Conformance with Regulatory Guide 1.183 Appendix B (Fuel Handling Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
	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).		accordance with regulatory guidance. An overall DF of 200 is used. For a drop over the spent fuel pool, coverage over the "striking" fuel assembly as it lies across the top of the "struck" fuel is 21.5 feet. With less than 23 feet of water coverage in the spent fuel pool, the reduction in DF is offset by the reduction in the amount of fuel damage due to a much shorter fall height over the spent fuel pool as compared to that in the reactor vessel. Therefore, 23 feet of water coverage is used since a drop into the vessel is the most limiting case.
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.
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 (21) should be determined and accounted for in the radioactivity release analyses.	Exception to RG 1.52 taken regarding MCREV bypass per PBAPS current Licensing basis (TS	No credit is taken for the Standby Gas Treatment System or its elevated release.

Table C: Conformance with Regulatory Guide 1.183 Appendix B (Fuel Handling Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
		5.5.7).	
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	A two-hour release to the environment is assumed after an instantaneous release from the fuel to the fuel pool / reactor building
5.1	If the containment is isolated during fuel handling operations, no radiological consequences need to be analyzed.	Not Applicable	Secondary Containment isolation is not credited.
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 Secondary Containment isolation is not credited.
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.*	Exception: A closure time of 60 minutes is proposed. Site-specific controls to be implement-ed	It is assumed that all radioactive material released from the damaged fuel escapes to the environment within the 2-hour time period. Although secondary containment closure will be accomplished within a 1-hour time period (instead of the 30 minutes indicated in RG 1.183), no credit is taken for this closure. Administrative controls will be in place associated with closure of doors and penetrations. (See Attachment 4 of this LAR for the regulatory commitment).
5.4	A reduction in the amount of radioactive material released from the containment by ESF filter systems may be taken into account provided	Exception to RG 1.52	No credit is being taken for filtration of release from the reactor building via

Table C: Conformance with Regulatory Guide 1.183 Appendix B (Fuel Handling Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
	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.	taken regarding MCREV bypass per PBAPS current Licensing basis (TS 5.5.7).	SGT.
5.5	Credit for dilution or mixing of the activity released from the reactor cavity by natural or forced convection inside the containment may be considered on a case-by-case basis. Such credit is generally limited to 50% of the containment free volume. This evaluation should consider the magnitude of the containment volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the reactor cavity, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the reactor cavity and the exhaust plenums.	Not Applicable	A two-hour release to the environment is assumed after an instantaneous release from the fuel to the fuel pool / reactor building.

Table D: Conformance with Regulatory Guide 1.183 Appendix C (Control Rod Drop Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
1	Assumptions acceptable to the NRC staff regarding core inventory are provided in Regulatory Position 3 of this guide. For the rod drop 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 on the assumption that 100% of the noble gases and 50% of the iodines contained in that fraction are released to the reactor coolant.	Conforms	Breached / melted fuel rods and release fractions have been updated to reflect PBAPS fuel and RG 1.183 guidance.
2	If no or minimal fuel damage is postulated for the limiting event, the released activity should be the maximum coolant activity (typically 4 μCi/gm DE I-131) allowed by the technical specifications. *Footnote 1: 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 the 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	Fuel damage is postulated as the limiting case.
3.1	The activity released from the fuel from either the gap or from fuel pellets is assumed to be instantaneously mixed in the reactor coolant within the pressure vessel.	Conforms	Instantaneous mixing is assumed per this guidance.
3.2	Credit should not be assumed for partitioning in the pressure vessel or for removal by the steam separators.	Conforms	No partitioning is assumed.
3.3	Of the activity released from the reactor coolant within the pressure vessel, 100% of the noble gases, 10% of the iodine, and 1% of the	Conforms	Released activity assumed is per this guidance.

Table D: Conformance with Regulatory Guide 1.183 Appendix C (Control Rod Drop Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
	remaining radionuclides are assumed to reach the turbine and condensers.		
3.4	Of the activity that reaches the turbine and condenser, 100% of the noble gases, 10% of the iodine, and 1% of the particulate radionuclides are available for release to the environment. The turbine and condensers leak to the atmosphere as a ground- level release at a rate of 1% per day for a period of 24 hours, at which time the leakage is assumed to terminate. No credit should be assumed for dilution or holdup within the turbine building. Radioactive decay during holdup in the turbine and condenser may be assumed. *Footnote 2: If there are forced flow paths from the turbine or condenser, such as unisolated motor vacuum pumps or unprocessed air ejectors, the leakage rate should be assumed to be the flow rate associated with the most limiting of these paths. Credit for collection and processing of releases, such as by off gas or standby gas treatment, will be considered on a case-by-case basis.*	Conforms	The condenser leak rate of 1% per day for a period of 24 hours is assumed. All releases are assumed to be at ground level and based on zero-velocity Reactor Building / Turbine Building ventilation release assumptions. Radioactive decay during holdup in the condenser is assumed. Upon detection of high radiation levels by the Main Steam Line Radiation Monitor system, the MSIVs close and the mechanical vacuum pump trips. Therefore, forced flow path is not applicable to PBAPS.
3.5	In lieu of the transport assumptions provided in paragraphs 3.2 through 3.4 above, a more mechanistic analysis may be used on a case-by-case basis. Such analyses account for the quantity of contaminated steam carried from the pressure vessel to the turbine and condensers based on a review of the minimum transport time from the pressure vessel to the first main steam isolation (MSIV) and considers MSIV closure time.	Not Applicable	Paragraphs 3.2 through 3.4 are used in the analysis.
3.6	The iodine species released from the reactor coolant within the pressure vessel should be assumed to be 95% CsI as an aerosol, 4.85% elemental, and 0.15% organic. The release from the turbine and condenser should be assumed to be 97% elemental and 3% organic.	Conforms	No credit for SGTS or MCREV filters is taken, and therefore variation in iodine species has no effect.

Table E: Conformance with Regulatory Guide 1.183 Appendix D (Main Steam Line Break)			
RG Section	RG Position	PBAPS 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 guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.	Not Applicable	No fuel damage is postulated. The release is estimated based on Technical Specification coolant activity.
2	If no or minimal fuel damage is postulated for the limiting event, the released activity should be the maximum coolant activity allowed by technical specification. The iodine concentration in the primary coolant is assumed to correspond to the following two cases in the nuclear steam supply system vendor's standard technical specifications. Footnote: 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 PBAPS licensing basis does not assume fuel damage associated with the MSLB event. Technical Specification LCO 3.4.6 limits the reactor coolant Dose Equivalent (DE) I-131 specific activity to 0.2 µCi/gm, with action to isolate all main steam lines if the reactor coolant DE I-131 specific activity exceeds 4.0 µCi/gm during Power Operation or Startup.
2.1	The concentration that is the maximum value (typically 4.0 µCi/gm DE I-131) permitted and corresponds to the conditions of an assumed pre-accident spike, and	Conforms	See Item 2 above.
2.2	The concentration that is the maximum equilibrium value (typically 0.2 µCi/gm DE I-131) permitted for continued full power operation.	Conforms	See Item 2 above.
3	The activity released from the fuel should be assumed to mix instantaneously and homogeneously in the reactor coolant. Noble gases should be assumed to enter the steam phase instantaneously.	Conforms	Mixing is per this guidance.
4.1	The main steam line isolation valves (MSIV) should be assumed to close in the maximum time allowed by technical specifications.	Conforms	Technical Specification SR 3.6.1.3.9 verifies the isolation time of each MSIV is between 3 and 5 seconds, well within the 10.5 seconds up to full MSIV closure assumed for the

Table E: Conformance with Regulatory Guide 1.183 Appendix D (Main Steam Line Break)			
RG Section	RG Position	PBAPS Analysis	Comments
			release period.
4.2	The total mass of coolant released should be assumed to be that amount in the steam line and connecting lines at the time of the break plus the amount that passes through the valves prior to closure.	Conforms	Mass of coolant released is per this guidance.
4.3	All the radioactivity in the released coolant should be assumed to be released to the atmosphere instantaneously as a ground-level release. No credit should be assumed for plateout, holdup, or dilution within facility buildings.	Conforms	This guidance was used in the analysis.
4.4	The iodine species released from the main steam line should be assumed to be 95% CsI as an aerosol, 4.85% elemental, and 0.15% organic.	Conforms	The iodine species assumed is per this guidance. However, no filtration is credited for the MSLB, so the iodine species are not relevant.

ATTACHMENT 8

PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 AND 3

Docket Nos. 50-277
50-278

Renewed License Nos. DPR-44
DPR-56

License Amendment Request
"PBAPS Alternative Source Term Implementation"

Compact Disk Containing AST Calculations