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An Exelon/British Energy Company

Clinton Power Station
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RS-03-060

10 CFR 50.90

April 3, 2003

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

Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
NRC Docket No. 50-461

Subject: Request for License Amendment Related to Application of Alternative Source Term

References

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

(2) U. S. Nuclear Regulatory Commission Standard Review Plan 15.0 1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, July 2000

(3) Letter from J. M. Heffley (AmerGen Energy Company, LLC) to U. S. Nuclear Regulatory Commission, "Request for Amendment to Technical Specifications that Revise Plant System Requirements During Fuel Handling Based on Alternative Source Term," dated July 5, 2001

(4) Letter from K. R. Jury (Exelon Generation Company, LLC) to U. S Nuclear Regulatory Company, "Additional Information Supporting the License Amendment Request to Revise Plant System Requirements During Fuel Handling Based on Alternative Source Term," dated December 28, 2001

(5) Letter from K. R. Jury (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "Supplemental Information Supporting a License Amendment Request to Revise Plant System Requirements During Fuel Handling Based on Alternative Source Term," dated March 1, 2002

(6) Letter from U. S. Nuclear Regulatory Commission to O. D. Kingsley (Exelon Generation Company, LLC), "Clinton Power Station, Unit 1 – Issuance of Amendment (TAC No. MB2572)," dated April 3, 2002

A001

Pursuant to 10 CFR 50.90, "Application for amendment of license or construction permit.", AmerGen Energy Company (AmerGen), LLC hereby requests an amendment to Appendix A, Technical Specifications (TS), of Facility Operating License No. NPF-62 for Clinton Power Station (CPS). The proposed change is requested to support application of an alternative source term (AST) methodology, in accordance with 10 CFR 50.67, "Accident source term.", 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.

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 Reference 1. 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).

AmerGen has performed radiological consequence analyses of the four DBAs that result in offsite exposure to support a full-scope implementation of AST as described in Reference 1. The AST analyses for CPS were performed following the guidance in References 1 and 2. The AST fuel handling accident analysis was previously performed. Based on this analysis, AmerGen proposed changes to the CPS TS that would revise requirements that applied during the movement of irradiated fuel and during core alterations. This proposed amendment request was documented in References 3, 4, and 5. The NRC subsequently approved these changes as Amendment 147 in Reference 6.

AmerGen has since completed the remaining three AST analyses. The proposed changes to the current licensing basis for CPS 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 main steam isolation valve allowable leakage;
- TS and associated Bases revisions to decrease allowed feedwater isolation valve leakage to allow margin to be used for other release paths;
- TS and associated Bases revisions to delete requirements for the main steam isolation valve leakage control system;
- TS and associated Bases revisions to reflect requirements for availability of Standby Liquid Control (SLC) System in Mode 3 and use of the SLC System to buffer suppression pool pH to prevent iodine re-evolution during a postulated radiological release;
- TS and associated Bases revisions to reflect higher allowed charcoal adsorber penetrations in laboratory testing;
- TS Bases revision to reflect an increased allowed secondary containment drawdown time;

- TS Bases revision to identify additional containment leakage exclusions from L_a and exclusions from secondary containment bypass allowances;
- Additional allowance for filtered and unfiltered inleakage into the control room envelope; and
- Development of new offsite and control room atmospheric dispersion factors (χ/Qs) calculated using site-specific meteorology data collected between 2000 and 2002.

The amendment request is consistent with submittals associated with application of AST that have been previously provided to the NRC by the Carolina Power & Light (CP&L) Company for the Brunswick Steam Electric Plant Units 1 and 2 (Letter BSEP 01-0063, dated August 1, 2001), Entergy for the River Bend Station (Letter RBG-45930, dated April 24, 2002), and Exelon Generation Company, LLC for the Dresden and Quad Cities Stations (Letter RS-02-174, dated October 10, 2002).

This request is subdivided as follows.

1. Attachment 1 provides the notarized affidavit.
2. Attachment 2 provides our evaluation supporting the proposed changes.
3. Attachment 3 contains the copies of the marked up TS pages.
4. Attachment 4 provides the retyped TS pages and Bases pages for information only.
5. Attachment 5 consists of tables describing conformance with Regulatory Guide 1.183.

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

AmerGen requests approval of the proposed amendments by December 31, 2003. This will provide adequate time for the affected station documents to be revised using the appropriate change control mechanisms.

AmerGen is notifying the State of Illinois of this application for amendment by transmitting a copy of this letter and its attachments to the designated State Official.

If you have any questions or require additional information, please contact Mr. Timothy A. Byam at (630) 657-2804.

Respectfully,

Michael J. Pacilio
Site Vice President
Clinton Power Station

Attachments:

Attachment 1 Affidavit
Attachment 2 Evaluation of Proposed Changes
Attachment 3 Markup of Proposed Changes
Attachment 4 Retyped Pages for Technical Specification Changes and Bases Changes (for information only)
Attachment 5 Regulatory Guide 1.183 Comparison

cc: Regional Administrator – NRC Region III
NRC Project Manager, NRR – Clinton Power Station
NRC Senior Resident Inspector – Clinton Power Station
Office of Nuclear Facility Safety – Illinois Department of Nuclear Safety

ATTACHMENT 1
Affidavit

STATE OF ILLINOIS	)	
COUNTY OF DEWITT	)	
IN THE MATTER OF	)	
AMERGEN ENERGY COMPANY, LLC	)	Docket Number
CLINTON POWER STATION, UNIT 1	)	50-461

SUBJECT: Request for License Amendment Related to Application of Alternative Source Term

AFFIDAVIT

I affirm that the content of this transmittal is true and correct to the best of my knowledge, information and belief.

Michael J. Pacilio
Site Vice President
Clinton Power Station

Subscribed and sworn to before me, a Notary Public in and

for the State above named, this 3rd day of

April, 2003.

Jacqueline S. Matthias
Notary Public

"OFFICIAL SEAL"
Jacqueline S. Matthias
Notary Public, State of Illinois
My Commission Exp. 11/24/2005

Subject: Request for License Amendment Related to Application of Alternative Source Term

1.0 DESCRIPTION

Pursuant to 10 CFR 50.90, "Application for amendment of license or construction permit.", AmerGen Energy Company (AmerGen), LLC hereby requests an amendment to Appendix A, Technical Specifications (TS), of Facility Operating License No. NPF-62 for Clinton Power Station (CPS). The proposed change is requested to support application of an alternative source term (AST) methodology, in accordance with 10 CFR 50.67, "Accident source term.", with the exception that Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," (Reference 1) will continue to be used as the radiation dose basis for equipment qualification.

AmerGen has performed radiological consequence analyses of the four design basis accidents (DBAs) that result in offsite exposure to support a full-scope implementation of AST as described in Reference 2. The AST fuel handling accident analysis was previously performed. Based on the revised fuel handling accident analysis, AmerGen proposed changes to the CPS TS that would revise requirements that applied during the movement of irradiated fuel and during core alterations. This proposed amendment request was documented in References 3, 4, and 5. The NRC subsequently approved these changes as Amendment 147 in Reference 6. This submittal is based on the completion of the remaining three DBA analyses (i.e., loss of coolant accident, main steam line break, and control rod drop accident). The DBA analyses have been revised to define the impact of the new source term on doses to the public at the site boundary and to the operator in the control room. The AST analyses for CPS were performed following the guidance in References 2 and 7.

Based on the revised DBA analyses, AmerGen is proposing changes to the current licensing basis for CPS. The proposed TS changes are described in Section 2.0 of this Attachment. The marked-up TS pages are provided in Attachment 3. Revised TS pages reflecting these changes are provided in Attachment 4. A marked-up copy of the affected TS Bases is also included for informational purposes in Attachment 4.

AmerGen requests approval of the proposed amendments by December 31, 2003. This will provide adequate time for the affected station documents to be revised using the appropriate change control mechanisms.

2.0 PROPOSED CHANGE

2.1 TS Section 1.1, "Definitions"

The proposed change revises the definition of dose equivalent I-131 in TS Section 1.1 to replace the word "thyroid" with "inhalation CEDE." In addition, this change will replace the reference to TID-14844 with a reference to Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989. This change reflects the application of AST assumptions. The revised main steam line break (MSLB) accident analysis uses inhalation committed effective dose equivalent dose conversion factors from Federal

Guidance Report 11 for calculation of normalized I-131 dose equivalent activity. Therefore, the reference to TID-14844 is no longer appropriate and must be deleted.

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

TS Section 3.1.7 requires that two SLC subsystems shall be operable in Modes 1 and 2. The proposed change revises the applicability of TS Section 3.1.7 to add the requirement for the Limiting Condition for Operation (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 loss of coolant accident (LOCA) involving significant fission product release. In addition, the required actions for Condition C are being revised to add an additional requirement to be in Mode 4. The applicable sections of TS 3.1.7 Bases are also revised to reflect these changes.

2.3 TS Section 3.3.6.1, "Primary Containment and Drywell Isolation Instrumentation"

TS Section 3.3.6.1, Table 3.3.6.1-1 lists the applicability requirements for Primary Containment and Drywell 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 proposed revision to the SLC System applicability. The applicable sections of TS 3.3.6.1 Bases are also revised to reflect these changes.

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

TS Section 3.6.1.3 Surveillance Requirement (SR) 3.6.1.3.5 requires performance of leakage rate testing for each primary containment purge valve with resilient seals. The Bases for TS SR 3.6.1.3.5 are being revised to increase the allowed leakage per penetration from 1% to 2% of the maximum allowable primary containment leakage rate, L_a.

TS SR 3.6.1.3.9 requires verification of the total leakage rate through all four main steam lines at the peak calculated containment internal pressure. The proposed change revises TS SR 3.6.1.3.9 to increase the allowable limit for the combined leakage rate for all main steam isolation valve (MSIV) leakage paths from less than or equal to 112 standard cubic feet per hour (scfh) to less than or equal to 250 scfh when tested at greater than or equal to P_a (i.e., 9 psig). In addition, a new leakage rate limit of less than or equal to 100 scfh through each MSIV leakage path is also being added. The associated Bases for SR 3.6.1.3.9 are also revised to reflect the proposed changes to the surveillance requirement. The frequency for SR 3.6.1.3.9 is "[i]n accordance with the Primary Containment Leakage Rate Testing Program," and this frequency is not being changed. Application of AST supports the increase in MSIV leakage.

TS SR 3.6.1.3.11 requires verification of the combined leakage rate for both primary containment feedwater penetrations. The proposed change revises TS SR 3.6.1.3.11 to reduce the combined leakage rate for both primary containment feedwater penetrations from less than or equal to 3 gallons per minute (gpm) to less than or equal to 2 gpm when pressurized to greater than or equal to 1.1 P_a. The associated Bases are also revised to reflect the changes made to the surveillance requirement. As discussed in Section 4.1 of this Attachment, this change is consistent with the AST LOCA analysis assumptions and provides additional margin for other safety system parameters.

2.5 TS Section 3.6.1.8, "Main Steam Isolation Valve (MSIV) Leakage Control System (LCS)"

TS Section 3.6.1.8 requires that two MSIV LCS subsystems shall be operable in Modes 1, 2, and 3. The proposed change deletes this TS section since the MSIV LCS is no longer credited to mitigate the consequences of any design basis accident. The associated Bases are also deleted in accordance with this change.

2.6 TS Section 3.6.4.1, "Secondary Containment"

TS SR 3.6.4.1.4 requires verification that each Standby Gas Treatment System (SGTS) subsystem will draw down the secondary containment to ≥ 0.25 inch of vacuum water gauge within the time required. The proposed change to the associated Bases for TS SR 3.6.4.1.4 is required to indicate that the drawdown time limit is based on ensuring that the SGTS will draw down the secondary containment pressure to ≥ 0.25 inches of vacuum water gauge within the new required time of 12 minutes (i.e., 10 minutes from start of gap release which occurs 2 minutes after LOCA initiation) under LOCA conditions. This change is consistent with the AST analyses.

2.7 TS Section 5.5.7, "Ventilation Filter Testing Program (VFTP)"

TS Section 5.5.7 requires testing of engineered safety feature filter ventilation systems. TS Section 5.5.7.c provides methyl iodide penetration acceptance criteria and test conditions for a laboratory test of a sample of the charcoal adsorber. The proposed change revises Section 5.5.7.c to increase methyl iodide penetration acceptance criteria for the SGTS from 0.175% to 1.5%, for the Control Room ventilation makeup filter from 0.175% to 1.5% and for the Control Room ventilation recirculation filter from 6% to 15%. Application of AST supports increasing the methyl iodide penetration percentages.

In summary, the proposed changes to the current licensing basis for CPS that are justified by the AST analyses include the following.

- TS and associated Bases revisions to reflect implementation of AST assumptions;
- TS and associated Bases revisions to increase main steam isolation valve allowable leakage;

- TS and associated Bases revisions to decrease allowed feedwater isolation valve leakage to allow margin to be used for other release paths;
- TS and associated Bases revisions to delete requirements for the main steam isolation valve leakage control system;
- TS and associated Bases revisions to reflect requirements for availability of SLC System in Mode 3 and use of the SLC System to buffer suppression pool pH to prevent iodine re-evolution during a postulated radiological release;
- TS and associated Bases revisions to reflect higher allowed charcoal adsorber penetrations in laboratory testing;
- TS Bases revision to reflect an increased allowed secondary containment drawdown time;
- TS Bases revision to identify additional containment leakage exclusions from L_a and exclusions from secondary containment bypass allowances;
- Additional allowance for filtered and unfiltered inleakage into the control room envelope; and
- Development of new offsite and control room atmospheric dispersion factors (χ/Qs) calculated using site-specific meteorology data collected between 2000 and 2002.

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 acceptable application and implementation of AST is provided in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power reactors," (Reference 2). The use of AST changes only the regulatory assumptions regarding the analytical treatment of the design basis accidents (DBAs). AmerGen has performed radiological consequence analyses of the four 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 Reference 2. The AST analyses for CPS were performed following the guidance in References 2 and 7.

The proposed changes to the TS will allow CPS to apply the results of the plant-specific AST analyses using the guidance in Reference 2 and meeting the requirements of 10 CFR 50.67. Approval of this change will provide a realistic source term for CPS 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. Adopting the AST methodology may also support future evaluations and license amendments. The implementation of these changes would provide schedule flexibility during outages while maintaining safety margin.

In addition, this change will allow increasing MSIV leakage. Unplanned MSIV repairs are a significant contributor to increased outage duration and unplanned exposure during refueling outages. This leakage limit is associated with the TS for the operability of the primary containment isolation valves (i.e., TS Section 3.6.1.3). Revision to this TS will relax operational constraints during outage activities. This relaxation will result in a reduction in personnel radiation exposure due to valve maintenance being performed on the MSIVs. Under the AST assumptions proposed, the MSIV work can be strategically planned to maintain work ALARA and maximize the incremental benefit of work being performed in high dose areas.

Another potential benefit involves the elimination of the MSIV LCS. Elimination of the LCS will reduce radiation exposures to maintenance personnel and reduce outage durations.

4.0 TECHNICAL ANALYSIS

The fission product release from the reactor core into containment is referred to as the "source term," and it is characterized by the composition and magnitude of the radioactive material, the chemical and physical properties of the material, and the timing of the release from the reactor core. Since the publication of Reference 1, significant advances have been made in understanding the timing, magnitude, and chemical form 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 8) 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 follows.

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, in addition to the changes approved in Amendment 147, constitutes 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 2.

AmerGen previously completed the radiological consequence analysis for the FHA as documented in References 3, 4, and 5. AmerGen has now performed radiological consequence analyses of the three remaining DBAs that result in offsite exposure (i.e., LOCA, CRDA, and MSLB). These analyses were performed to support full scope implementation of AST. The AST analyses have been performed in accordance with the guidance in References 2 and 7. The implementation consisted of the following steps.

- Identification of the AST based on plant-specific analysis of core fission product inventory,
- Identification of release pathways and mechanisms appropriate to each DBA, and the resulting atmospheric dispersion parameters,
- Calculation of the effects of applicable fission product removal mechanisms for LOCA such as natural deposition in containment and deposition in piping,
- Evaluation of suppression pool pH to ensure that the particulate iodine deposited into the suppression pool during a DBA LOCA does not re-evolve and become airborne as elemental iodine,
- Optimization of credited safety system parameters,
- Calculation of the resulting releases for the three remaining DBAs that could potentially result in control room and offsite doses (i.e., LOCA, MSLB, and CRDA), and
- Calculation of the exclusion area boundary (EAB), low population zone (LPZ) and control room personnel Total Effective Dose Equivalent (TEDE) doses.

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 2 for the three analyzed DBAs. The matrices addressing CPS compliance with each regulatory requirement as outlined in Reference 2 are provided in Attachment 5.

New AST calculations for the LOCA, CRDA and MSLB were prepared for the simulation of the radionuclide release, transport, removal, and dose estimates associated with the postulated accidents. The RADTRAD computer code (Reference 9) developed for and endorsed by the NRC for AST analyses was used in the calculations for LOCA and CRDA. The RADTRAD program is a radiological consequence analysis code used to estimate post-accident doses at plant offsite locations and in the control room. All of the non-LOCA analyses take no credit for SGTS operation, secondary containment isolation or control room air intake or recirculation filtration for the duration of the accident event.
Control room and offsite atmospheric dispersion factors (χ/Qs) developed for the LOCA-related releases from the SGTS/HVAC vent stack were utilized. Offsite χ/Qs were

calculated with the PAVAN computer code (Reference 10), using the guidance of Regulatory Guide 1.145 (Reference 11), and control room χ/Qs were calculated with the ARCON96 computer code (Reference 12). The PAVAN and ARCON96 codes calculate relative concentrations in plumes from nuclear power plants at offsite locations and control room air intakes, respectively. All of these codes have been used by the NRC in their safety reviews.

The ORIGEN 2 code (Reference 13) was used in the calculation of the reactor core fission products for RADTRAD analysis. The inventories were determined based on the licensed core power level following extended power uprate (EPU) of 3473 megawatts thermal (MWt) and further adjusted to 102% (3543 MWt) in support of the AST evaluations. To conservatively predict the fission product inventories, an irradiation time of 1095.75 days (i.e., three years at 100% power) was used.

The 10-meter and 60-meter meteorological data obtained from the CPS meteorological tower on wind speed and stability class for calendar years 2000, 2001 and 2002 were utilized in the analyses. The 2000-2002 data have recovery rates above 90% (i.e., greater than 90% of the hourly meteorological data is available). Comparisons of these data on a year-by-year basis indicate that there is a high degree of similarity and little variation from the long-term average. Therefore, the year 2000-2002 period is an adequately long period to represent the CPS site meteorological conditions for evaluating the consequences of postulated releases. The meteorological data was used to generate the appropriate χ/Q values using the ARCON96 computer code for the control room (Table 1), and the PAVAN code and Regulatory Guide 1.145 guidance for the EAB and LPZ boundaries (Table 2).

4.1 LOCA Analysis

The key inputs and assumptions used in the AST LOCA analysis are provided in Tables 3 through 6. These inputs and assumptions are grouped into three main categories (i.e., release, transport, and removal).

<u>LOCA Release Inputs</u>
The LOCA analysis assumes a total containment leakage rate at the limit of 0.65 percent of primary containment free volume per day. The primary containment leakage is then assumed to be reduced by 50 percent after 24 hours, based on the post-LOCA containment pressure history. All of this leakage is assumed to be an unfiltered release to the environment for an assumed 10-minute secondary containment drawdown time. The 10-minute drawdown time is a revision from the current 188 seconds as documented in the TS SR 3.6.4.1.4 bases. After 10 minutes, 92% of the primary containment leakage is considered to be filtered by the SGTS and the remaining 8% of this leakage is assumed to bypass the secondary containment and be released unfiltered to the environment.

An additional 10.98 cubic feet per minute (cfm) of air leakage from the feedwater isolation valve is assumed to be released unfiltered to the environment for a one-hour period until the FW piping is filled with ECCS water. The 10.98 cfm of air leakage corresponds to the 2 gpm of water leakage from the feedwater isolation valves as described in the proposed TS SR 3.6.1.3.11. After one hour, the 2 gpm water leakage unfiltered release is assumed. As described above, the 2 gpm feedwater isolation valve leakage limit is a reduction from the current limit of 3 gpm. This reduction is for the purpose of providing additional margins for other safety systems parameters.

In addition to the above assumed leakage, unfiltered releases from MSIV leakage at the proposed TS SR 3.6.1.3.9 rates to the environment are also assumed. This analysis takes no credit for a MSIV LCS. As discussed above, the proposed changes revise TS SR 3.6.1.3.9 to increase the allowable combined leakage rate for all MSIVs to less than or equal to 250 scfh and less than or equal to 100 scfh for each MSIV leakage path when tested at 9 psig. This constitutes an increase from the current total leakage limit of 112 scfh.

The dose consequences from leakage through the primary containment purge lines have been analyzed based on a leak rate of 0.02 L_a for each penetration. These penetrations are currently subject to leakage rate testing under TS SR 3.6.1.3.5, with a current leakage rate acceptance criterion of $\leq$ 0.01 L_a and therefore, the assumption of a leak rate of 0.02 L_a constitutes a change from the current licensing basis. Since a separate dose analysis has been performed for the primary containment purge lines, these penetrations no longer need to be considered in determining compliance with the secondary containment bypass leakage path SR 3.6.1.3.8 limit of $\leq$ 0.08 L_a, or the primary containment leakage rate acceptance criterion in TS 5.5.13 of $\leq$ 1.0 L_a.

The LOCA analysis assumes an Engineered Safety Feature (ESF) Systems leakage rate of 5 gpm outside of the primary containment into the secondary containment. From the secondary containment it is assumed to constitute an unfiltered release to the environment for the assumed 10-minute secondary containment drawdown time. This leak rate is conservatively assumed to begin at the onset of the accident and to continue throughout the 30-day duration of the postulated accident. This is a new release path for CPS, analyzed to comply with Reference 2 requirements This provides the basis for an acceptance criterion for the TS.

Figure 1 illustrates the LOCA release pathways, with its associated Table of leakage parameters. Figure 2 illustrates the radiological release and control room intake pathways utilized.

The Reference 2 accident isotopic release specification allows deposition of iodine in the suppression pool. Essentially all of the iodine is assumed to remain in solution as long as the pool pH is maintained at or above a level of 7. Operators, upon detection of symptoms indicating that severe accident conditions have occurred, are directed to manually initiate the SLC System. If an accident were to occur which would create the fuel damage conditions assumed in the analysis, it is reasonable to assume that manual initiation of the SLC system would be initiated promptly.

The analysis includes the assumptions that: (1) borated solution injection is initiated within 3 hours following the accident, (2) a minimum mass of

4246 pounds of sodium pentaborate is delivered to the RPV and the suppression pool, and (3) that the borated solution is well mixed within the suppression pool. The sodium pentaborate solution will be well mixed with the suppression pool water in less than 24 hours as a result of ECCS pumps reflooding the reactor vessel. As documented in Table 14, the analysis demonstrates that the buffering effect of the boron solution maintains the suppression pool pH above 7 for the 30-day duration of the postulated LOCA to prevent iodine re-evolution. Maintaining suppression pool pH at or above a level of 7, as an assumption in support of radiological consequence analysis, represents a change to the CPS design and licensing bases.

TS Section 3.1.7 will require revision to reflect the requirements for the SLC System availability in Mode 3 as well as Modes 1 and 2. Ensuring that the SLC system is operable in Mode 3 supports the basis for the assumption that the SLC System will be available for injection to maintain suppression pool pH above 7 in the event of a LOCA.

<u>LOCA Transport to the Control Room Inputs</u>
In the analysis, the accident activity was assumed to enter the control room unfiltered for the first 20 minutes of the LOCA at a nominal intake flow rate of 1/2 of the assumed filter makeup value required for control room pressurization (i.e., 1650 cfm). This intake value is based on historic design basis evaluations of worst-case control room ventilation system single failures. After an assumed 20 minute period for manual action, the full nominal intake flow rate of 3000 cfm + 10 percent is assumed, filtered at 97% efficiency for elemental and organic iodines and 99% for aerosols (the same efficiencies that are assumed for the SGTS), with an additional filtered inleakage flow of 650 cfm and an unfiltered inleakage flow of 600 cfm. A 70% efficiency of the recirculation filter is assumed. Figure 3 shows a schematic of the control room HVAC System.

These changes in assumptions will allow relaxation of TS Section 5.5.7.c laboratory charcoal adsorber methyl iodide penetration testing acceptance criteria. As described above, it is proposed that the acceptance criteria be revised to 1.5% for SGTS and Control Room Ventilation (CRV) Makeup Air Filters and 15% for the CRV Recirculation Filters. These revised acceptance criteria are based on the reduced credit taken for filter efficiency, and safety factors allowed per NRC Generic Letter 99-02, "Laboratory Testing of Nuclear-Grade Activated Charcoal."

Currently, filtered inleakage, defined in TS SR 3.7.3.5 as the "air inleakage rate of the negative portions of the Control Room Ventilation System" is analyzed at 650 cfm and controlled, in accordance with SR 3.7.3.5, to 95% of that value (i.e., 617.5 cfm) to allow for instrument uncertainties. Unfiltered inleakage is not currently covered by a surveillance requirement.

Table 7 indicates the standard breathing rates used for control room personnel dose assessments and for offsite personnel. Control room occupancy factors used are also included in Table 7.

<u>LOCA Removal Inputs</u>
The activity of elemental iodine and aerosols released from the core into the primary containment is reduced by deposition (i.e., plate-out) and settling utilizing the natural deposition values identified in the RADTRAD code. Primary containment leakage into the secondary containment is collected by the SGTS which exhausts the releases via filters, and reduces releases. The deposition removal mechanisms are characteristics of the AST methodology and represent a change in the plant design and licensing basis.

Main steam line pipe deposition was modeled using the Brockmann-Bixler model contained in the RADTRAD computer code, as applied to horizontal segments only and based on using the highest flows in the shortest steam lines (i.e., most rapid transport, least deposition). No credit is taken for holdup or plate-out in the main steam lines beyond the MSIVs. Additionally, no credit is taken for holdup and plate-out in the main condenser.

Primary containment deposition through the two 36-inch purge penetrations, representing 2% of the containment leak rate each, was also modeled using the Brockmann-Bixler model contained in the RADTRAD computer code, as applied to horizontal segments only.

<u>Results</u>
The radiological consequences of the DBA LOCA were analyzed with the RADTRAD code, using the inputs and assumptions discussed above. The postulated sources of activity in the control room include contributions from filtered intake, and filtered and unfiltered inleakage. Dose contributors include internal cloud immersion and inhalation, and gamma shine from sources outside the control room. Each of the pathways identified and shown on Figure 1 are included. The gamma shine dose from external sources was shown to be bounded by the 0.59 rem total contribution from the current TID-14844 based analyses. Tables 10 and 11 present the results of the LOCA radiological consequence analysis. As indicated, the control room, EAB, and LPZ calculated doses remain within the regulatory limits for implementation of AST.

4.2 CRDA Analysis

The key inputs and assumptions used in the AST CRDA analysis are included in Table 9. The design basis CRDA involves the rapid removal of a highest worth control rod resulting in a reactivity excursion that encompasses the consequences of any other postulated CRDA. The core performance analysis shows that the energy deposition that results from this event is inadequate to damage fuel. However, for the dose consequence analysis, it was assumed that 1200 of the fuel pins in the core were damaged, with melting occurring in 1 percent of the damaged rods. A conservative core average radial peaking factor of 1.70 was assumed in the analysis. For releases from the breached fuel, 10% of the core inventory of noble gases and iodines were assumed to be in the fuel gap. For releases attributed to fuel melting, 100% of the noble gases and 50% of the iodines were assumed to be released to the reactor coolant.

Two condenser release scenarios were considered, as described below. In both scenarios, no credit was taken for partitioning in the reactor pressure vessel or removal by the steam separators, or turbine building holdup or dilution. Both scenarios assume instantaneous mixing of the activity released from the fuel in the reactor coolant, 100% of the noble gases, 10% of the iodines and 1% of the remaining radionuclides released into the reactor coolant were assumed to reach the turbine and condenser. Both scenarios also assume 100% of the noble gases, 10% of the iodines and 1% of the particulate radionuclides reaching the turbine and condenser were available for release to the environment.

> Scenario 1: Unfiltered release from the turbine building via the HVAC vent stack at the rate of 1.0% of the condenser free volume per day for the duration of the accident.
>
> Scenario 2: Forced flow from the condenser by the steam jet air ejectors (SJAE), with release via the large charcoal delay beds of the Off-Gas (OG) System.

Forced flow from the condenser by mechanical vacuum pump operation was also considered; however, the Main Steam Line Radiation Monitoring System provides an effectively immediate isolation function for this pathway based on detection of the resulting high radiation levels in the main steam lines, so no releases are expected from this pathway.

The analysis takes no credit for control room operator action or filtration of the control room intake or recirculated air for the duration of the event. In addition, for Scenario 1, the maximum control room intake of normal unfiltered outside air was assumed to be 3300 cfm (including 10% uncertainty), plus an unfiltered inleakage of 2650 cfm. For Scenario 2, an effectively infinite control room unfiltered inleakage rate is assumed.

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 Appendix C of Regulatory Guide 1.183, as indicated in Attachment 5.

Results
The radiological consequences of the postulated CRDA are given in Table 12. As indicated, the control room, EAB, and LPZ calculated doses remain within regulatory limits after AST implementation.

4.3 MSLB Analysis

The key inputs and assumptions used in the AST MSLB analysis are included in Table 8. 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. Two activity release cases corresponding to the pre-accident spike and maximum equilibrium concentration allowed by TS of 4.0 μCi/gm and 0.2 μCi/gm dose equivalent I-131 respectively were assumed, with inhalation CEDE dose conversion factors from Federal Guidance Report 11. The external effective dose equivalent (EDE) dose conversion factors are from Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil," 1993. The released activity assumptions are consistent with the guidance provided in Appendix D of Regulatory Guide 1.183, as indicated in Attachment 5.

The analysis assumes an instantaneous ground level release. The released reactor coolant and steam is assumed to expand to a hemispheric volume at atmospheric pressure and temperature. This is consistent with an assumption of no credit for the Turbine Building. This hemisphere is then assumed to move at a speed of one 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 (Reference 14) methodology.

The radiological consequences following an MSLB accident were determined based on the following assumptions.

- Iodine and Noble Gas activity releases were based on the EPU accident radiological analysis.
- Instantaneous release from the break to the environment. No holdup in the Turbine Building or dilution by mixing with Turbine Building air volume is credited.
- Activity in the steam cloud was based on the total mass of water released from the break, not just the portion that flashes to steam. This assumption is conservative since it considers the maximum release of fission products.
- Fraction of liquid water contained in steam, which carries activity into the cloud, was 1.5%, as historically used.
- Flashing fraction of liquid water released was 40%. However, all activity in the water is assumed to be released.
- No credit for control room operator action or filtration of the control room intake or recirculated air for the duration of the event.

Results
The radiological consequences of the postulated MSLB are given in Table 13. As indicated, the control room, EAB, and LPZ calculated doses remain within regulatory limits after AST implementation.

4.4 Analyses Conclusions

As shown in Tables 10 through 13, the CPS accident radiological consequence analyses demonstrate that the post-accident offsite and control room doses can be maintained acceptably within regulatory limits following AST implementation.

4.5 Summary

Implementation of the AST as the plant radiological consequence analyses licensing basis requires a license amendment pursuant to the requirements of 10 CFR 50.67. The above described analyses demonstrate that the offsite and control room post-accident doses remain within the regulatory limits. Implementation of the AST provides the basis for several changes to the licensing and design bases for CPS. The principal changes affect primary containment and MSIV allowable leakage.

In the dose consequence analyses for the control room occupants, the assumed unfiltered leakage was increased to a value that continues to bound the measured data. Further evaluation of the analyses performed in support of the AST implementation support the conclusion that exposures to onsite and offsite receptors would not result in doses exceeding the values specified in 10 CFR 50.67.

4.6 Primary Containment Isolation

For Boiling Water Reactor (BWR) plants, NUREG 1465 acknowledges that the coolant activity release phase would last longer than for a Pressurized Water Reactor (PWR) and that more specific analyses could justify a longer duration. The BWR Owners Group (BWROG) performed a conservative analysis in General Electric Nuclear Energy report, NEDC-32963A, (Reference 15) to determine the minimum time to gap activity release for a generic BWR following a DBA LOCA with no emergency core cooling system (ECCS) injection. The analysis included sensitivity studies to determine the most limiting BWR design, fuel type, and core burnup. The NRC, in a letter to Grand Gulf Nuclear Station dated September 9, 1999, approved the BWROG analysis for use by all BWRs in support of plant-specific license amendments.

NEDC-32963A documents the results of an analysis performed to determine the minimum time to the onset of release of radioactive material from a perforated fuel assembly following a DBA LOCA at a generic BWR. NRC-approved computer codes were used to calculate the minimum duration of the coolant activity release phase described in NUREG-1465. The BWR coolant activity release phase, which represents the period from the time of the start of the accident until the initiation of a perforated fuel gap activity release, is calculated to last 121 seconds. Regulatory Guide 1.183 rounds this value to 2 minutes.

As described above, 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 appendix of Regulatory Guide 1.183 for the LOCA analysis. Approval of the AST LOCA analysis for CPS will provide a basis to allow the relaxation of the time required for the Primary Containment isolation valves (PCIVs) to isolate the containment in the event of a design basis accident. This will provide significant margin over the currently assumed values. This margin will be beneficial in optimizing the maintenance frequency for the subject valves while still providing assurance that the necessary functions can be performed within the required timeframes. Following NRC approval of the proposed amendment, AmerGen will process a change to the design basis documentation under 10 CFR 50.59, "Changes, tests, and experiments." to allow relaxation of the maximum time permitted for primary containment isolation.

Table 1: Control Room χ/Q Values

	STACK TO WEST INTAKE[1]	STACK TO EAST INTAKE[1]	STACK TO NORMAL INTAKE
	χ/Q (sec/m^3)	χ/Q (sec/m^3)	χ/Q (sec/m^3)
0-2 hour	9.45E-04	9.75E-04	1.54E-03
2-8 hour	7.58E-04	7.09E-04	1.09E-03
8-24 hour	3.28E-04	2.93E-04	4.67E-04
1-4 day	2.61E-04	2.13E-04	3.21E-04
4-30 day	1.85E-04	1.79E-04	2.64E-04

[1] To be divided by 4 for post-LOCA credit for most favorable intake of the dual intakes, based on manual control

Table 2: Offsite χ/Q (sec/m^3) Values

	STACK TO EAB (975 m)		STACK TO LPZ (4018 m)	
	χ/Q (sec/m^3)	Controlling Result	χ/Q (sec/m^3)	Controlling Result
0-2 hour	2.46E-04	0.5% max in SW sector	5.62E-05	0.5% max in SW sector
0-8 hour	1.19E-04	0.5% max in SW sector	2.48E-05	0.5% max in SW sector
8-24 hour	8.30E-05	0.5% max in SW sector	1.65E-05	0.5% max in SW sector
1-4 day	3.78E-05	0.5% max in SW sector	6.81E-06	0.5% max in SW sector
4-30 day	1.22E-05	0.5% max in SW sector	1.91E-06	0.5% max in SW sector

Table 3: Key Analysis Inputs and Assumptions				
Release Inputs - LOCA Radionuclide Source Term				
Input/Assumption	**Value**			
Core Fission Product Inventory	ORIGEN-2 Based Only the 60 nuclides considered by RADTRAD are utilized in the analysis			
Core Power Level	3543 MWt			
Fission Product Release Fractions for LOCA	**RG 1.183, 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
Fission Product Release Timing (Per RG 1.183, the release phases are modeled sequentially)	**RG 1.183, Table 4** **LOCA Release Phases for BWRs**			
	Phase	**Onset**	**Duration**	
	Gap Release	2 min*	0.5 hr	
	Early In-Vessel	0.5 hr	1.5 hr	

* For the RADTRAD analysis, this was represented as time equal to 0.

Table 4: Key LOCA Analysis Inputs and Assumptions	
Release Inputs - Primary and Secondary Containment Parameters	
Input/Assumption	**Value**
Radionuclide Release Pathways	See Figure 1
Drywell Free Volume	241,699 cubic feet
Containment Air Space Volume	1,512,341 cubic feet
Minimum Suppression Pool Volume	146,400 cubic feet
Primary Containment Leak Rate (SGTS Filtered and SC Bypass)	0.65% per day for first 24 hours (L_a) 0.325% per day thereafter
Total MSIV leak rate	250 scfh (100 scfh assumed for the two shortest lines) 125 scfh after 24 hours
FWIV leak rate (Each of Two Penetrations) Containment atmosphere: ECCS Water:	 10.98 cfm from 21.15 minutes to 1 hour 2 gpm from 1 hour to 24 hours 1 gpm after 24 hours
Secondary Containment (SC) Drawdown Time	10 minutes from start of gap release
Primary Containment Bypassing Secondary Containment	8.0% L_a for first day 4.0% L_a after first day
ECCS Systems Leakage into Secondary Containment Time of Unfiltered Release: Leak Rate: Flashing Fraction:	 10 minutes 5 gpm 1.36%
High Volume Purge Penetrations (101 and 102) Leak Rate	2.0% L_a / penetration for first day 1.0% L_a / penetration after first day

Table 5: Key LOCA Analysis Inputs and Assumptions	
Transport Inputs – Control Room Parameters	
Input/Assumption	**Value**
Control Room Filtered Intake and Recirculation Air Filtration Initiation Time (manual)	20 minutes During this period of no filtration and no CR pressurization, an inleakage of 1650 cfm is assumed, which is 1/2 of assumed filter makeup value required for CR pressurization.
Control Room Volume	324,000 cubic feet
Control Room Filtered Air Intake Flow Rate: Elemental and Organic Iodine Efficiencies: Aerosols Efficiencies:	 3000 + 10% = 3300 cfm 97% 99%
Control Room Filtered Recirculation Rate And Efficiency	61,000 – 10% = 54,900 cfm 70%
Control Room Filtered Inleakage Rate	650 cfm
Control Room Unfiltered Inleakage Rate	600 cfm
Control Room Filtered and Unfiltered Inleakage Control to maintain constant Iodine Protection Factor (IPF)	Allowed Unfiltered Inleakage [cfm] = 786.4 - 0.2868(measured Filtered Inleakage) and with 5% Instrument Uncertainty Allowed Unfiltered Inleakage [cfm] = 747 - 0.273(measured Filtered Inleakage)

Table 6: Key LOCA Analysis Inputs and Assumptions	
Removal Inputs	
Input/Assumption	**Value**
Containment Spray Removal Rates	Not Credited
Aerosol Natural Deposition Coefficients Used in the Containment	Credit is taken for natural deposition of aerosols based on equations for the Power's model in NUREG/CR 6189 and input directly into RADTRAD as natural deposition time dependent lambdas. No credit is assumed for natural deposition of elemental or organic iodine, or for suppression pool scrubbing.
Deposition/Plate-out (where credited)	Calculated for horizontal segments only using RADTRAD Brockmann-Bixler model.
Main Steam Line and Condenser Holdup Credit for MSIV Leakage	No credit is taken for holdup and plate-out downstream of the MSIVs or in the condenser since these components have not been evaluated for seismic ruggedness.
SGTS Filter Efficiencies - Elemental and Organic Iodine Aerosols	 97% 99%

Table 7: Personnel Dose Inputs	
Input/Assumption	**Value**
Onsite Breathing Rate	3.47E-04 m^3/sec
Offsite Breathing Rate	0-8 hours: 3.47E-04 m^3/sec 8-24 hours: 1.75E-04 m^3/sec 1-30 days: 2.32E-04 m^3/sec
Control Room Occupancy Factors	0-1 day: 1.0 1-4 days: 0.6 4-30 days: 0.4

Table 8: Key MSLB Accident Analysis Inputs and Assumptions	
Input/Assumption	**Value**
Break Discharge Mass Release	96,250 pounds, (42,500 as steam and 53,750 as liquid)
Pre-Accident Spike Iodine Concentration	4 μCi/gm I-131 equivalent
Maximum Equilibrium Iodine Concentration	0.2 μCi/gm I-131 equivalent
Turbine Building Holdup/ Control Room Filtration	No Credit Taken
Transport model for Control Room	Steam cloud moves past the Control Room intake at 1 m/sec

Table 9: Key CRDA Analysis Inputs and Assumptions	
Input/Assumption	**Value**
Core Damage	1,200 fuel rods failed
Percent of Damaged Fuel with Melt	1.0%
Radial Peaking Factor	1.7
Condenser Free Volume	175,000 cubic feet
Condenser Leak Rate	1% per day throughout entire release period
Release Period	24 hours
Control Room Filtration	Not utilized
Charcoal Delay Bed Noble Gas Delay for SJAE pathway	681.6 hours for Xe 31.8 hours for Kr

Table 10: LOCA Radiological Consequence Analysis– Dose Contributors			
Dose Contributor	**Control Room TEDE (rem)**	**EAB TEDE (rem)**	**LPZ TEDE (rem)**
Filtered Primary Containment (PC) Leakage	1.019	2.913	1.060
PC Leakage bypassing SC, with no piping deposition credit	0.934	4.978	0.965
MSIV Leakage, without LCS but with piping deposition credit	0.367	0.000	0.440
FWIV LCS Leakage of ECCS Water (unfiltered)	0.356	0.394	0.367
FWIV Air Leakage before fill with ECCS Water by LCS (unfiltered)	1.035	9.459	0.954
PC Leakage through purge penetrations 101 and 102, with piping deposition credit	0.009	0.000	0.011
ECCS Leakage in Secondary Containment (SC) (unfiltered for 10 minutes, SGTS filtered thereafter)	0.122	0.139	0.146
Gamma Shine to Control Room	0.590	-	-
Total Calculated Value	**4.43**	**17.88**	**3.94**

Table 11: LOCA Radiological Consequence Analysis - Totals*			
Location	Duration	TEDE (rem)	Regulatory Limit TEDE (rem)
Control Room	30 days	4.7	5
EAB	Maximum, 2 hours	19.5	25
LPZ	30 days	4.3	25

* Including margin of up to 10%

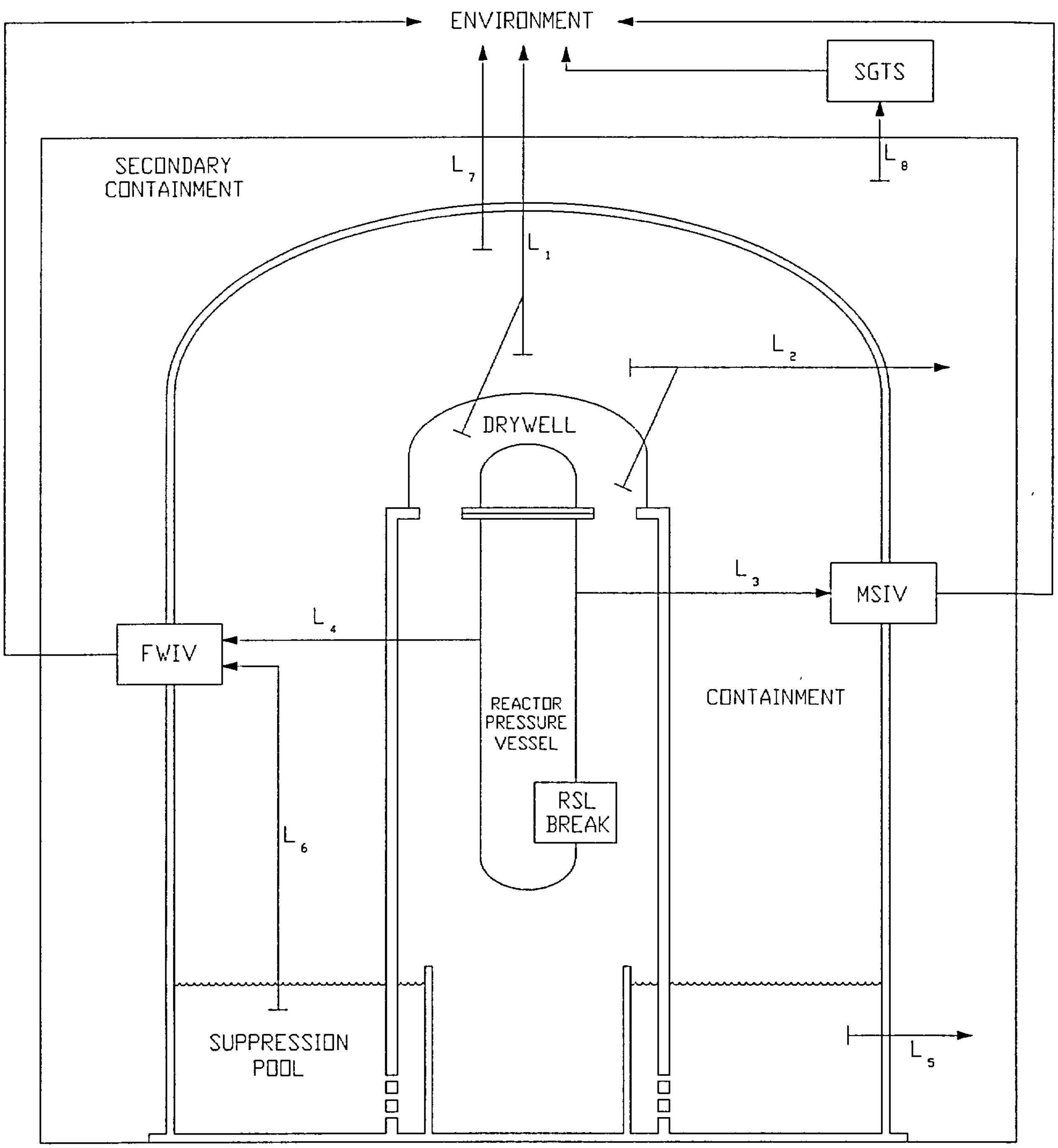
Table 12: CRDA Radiological Consequence Analysis			
Location	Duration	TEDE (rem)	Regulatory Limit TEDE (rem)
Control Room	30 days	Case 1: 0.43 Case 2: 0.25	5
EAB	Maximum, 2 hours	Case 1: 0.041 Case 2: 0.64	6.3
LPZ	30 days	Case 1: 0.016 Case 2: 0.15	6.3

Case 1: Based on 1% of condenser free volume leakage per day.
Case 2: Based on SJAE activity release

Table 13: MSLB Accident Radiological Consequence Analysis				
		4.0 μCi/gm Dose Equivalent I-131 TEDE (rem)	0.2 μCi/gm Dose Equivalent I-131 TEDE (rem)	Regulatory Limit TEDE (rem)
Control Room	30-day integrated dose	1.4	0.069	5
EAB	Worst 2-hour integrated dose	0.65	0.033	25 (4.0 μCi/gm) 2.5 (0.2 μCi/gm)
LPZ	30-day integrated dose	0.18	0.0091	25 (4.0 μCi/gm) 2.5 (0.2 μCi/gm)

Table 14: Suppression Pool pH Summary	
Parameter	**Value**
SLC System sodium pentaborate inventory	4246 lb_m
Time sodium pentaborate injection into the RPV is initiated	3 hours
Suppression pool pH for 30 days following an accident	>7

Figure 1


ENVIRONMENT
SGTS
SECONDARY CONTAINMENT
L7
L8
L1
L2
DRYWELL
L3
MSIV
L4
FWIV
CONTAINMENT
REACTOR PRESSURE VESSEL
RSL BREAK
L6
SUPPRESSION POOL
L5

Leakage Rates and Secondary Containment Mixing Parameters		
Path	**Description**	**Parameters & Values**
L_1	Primary Containment Leakage Bypassing Secondary Containment to the Environment	Leak Rate: 0.08*L_a = 0.052%/day from 0 to 24 hours 0.08*L_a = 0.026%/day from 1 to 30 days
L_2	Primary Containment Leakage to Secondary Containment	Leak Rate: 0.92*L_a = 0.598%/day from 0 to 10 min (Unfiltered during drawdown period) 0.92*L_a = 0.598%/day from 10 min to 24 hrs (SGTS filtered) 0.92*L_a = 0.299%/day from 1 to 30 days (SGTS filtered)
L_3	MSIV Leakage to Environment	Leak Rate: 250 scfh for all main steam lines, 100 scfh for maximum for any one MS line
L_4	FWIV Containment Air Leakage to Environment	Leak Rate: 10.98 cfm total, for the one hour before FWLCS fills the lines
L_5	ECCS Leakage to Secondary Containment	Leak Rate: 5 gpm from 0 to 30 days
L_6	FWLCS Leakage of ECCS Liquid to the Environment	Leak Rate: 2 gpm from 0 to 1 days 1 gpm from 1 to 30 days
L_7	Purge Penetrations 101 and 102 Leakage to the Environment	Leak Rate (for each of two penetrations): 0.02*L_a = 0.013%/day from 0 to 1 day 0.02*L_a = 0.0065%/day from 1 to 30 days
L_8	Release of Secondary Containment Atmosphere through SGTS to the Environment	No Secondary Containment mixing credit Modeled as: Volume = 1 cu.ft. Outflow = 1000 cfm

Figure 2

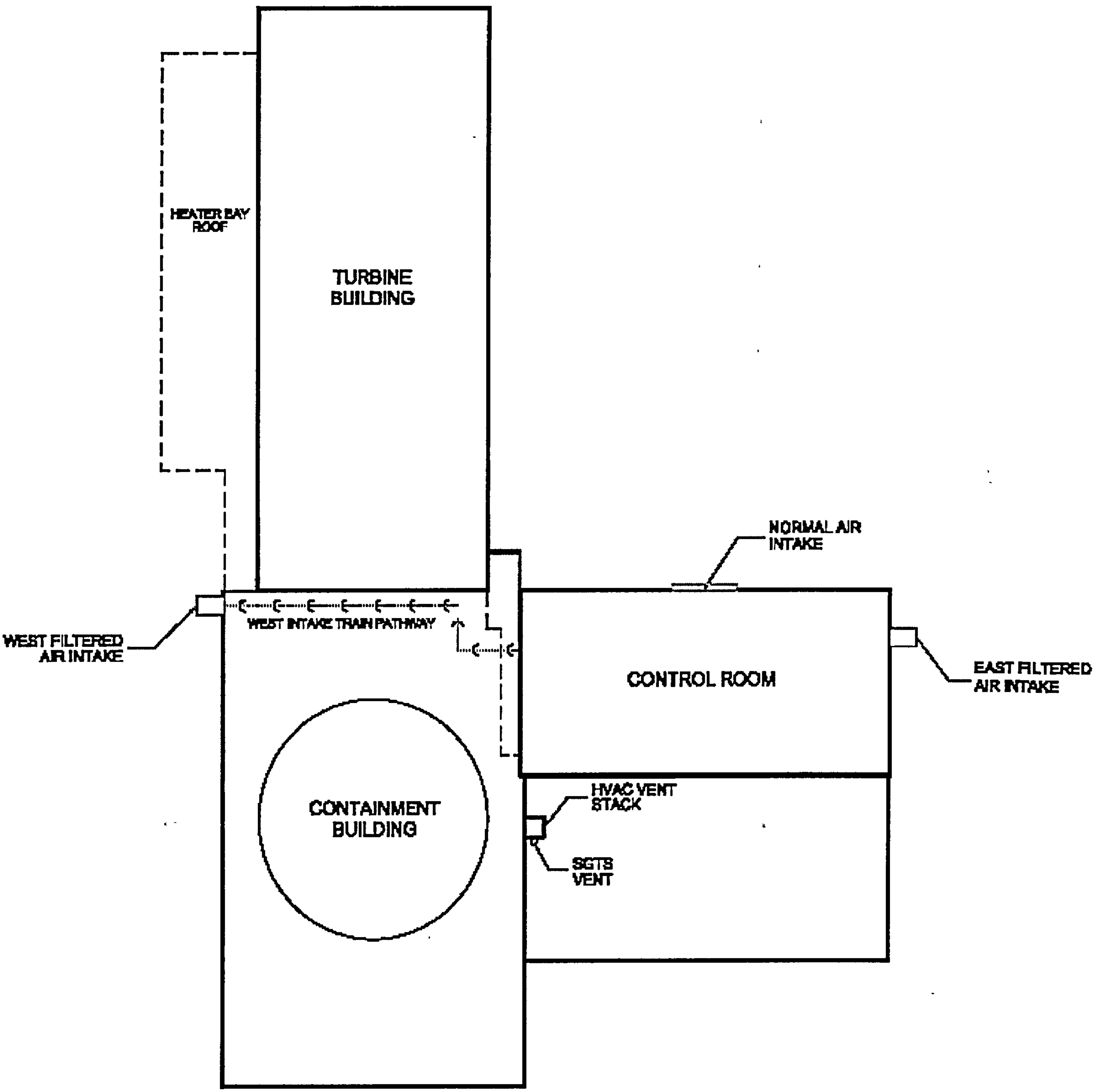
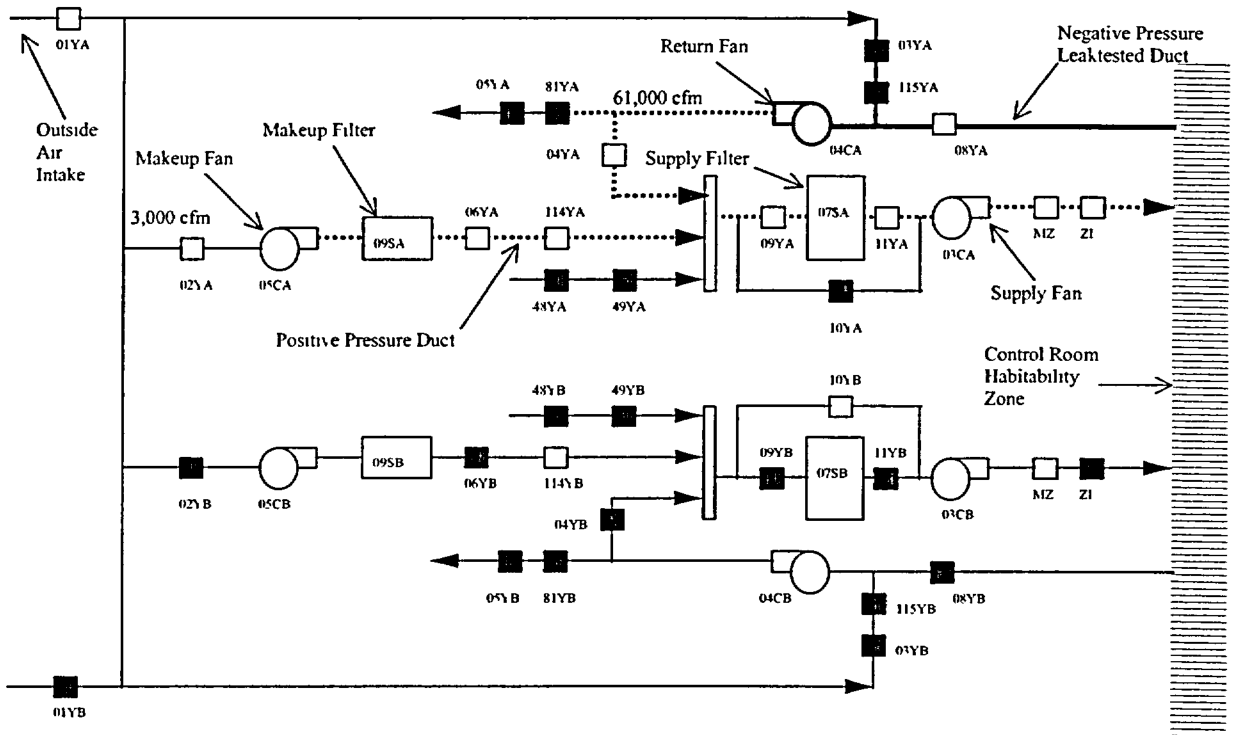

HEATER BAY ROOF
TURBINE BUILDING
NORMAL AIR INTAKE
WEST FILTERED AIR INTAKE
WEST INTAKE TRAIN PATHWAY
CONTROL ROOM
EAST FILTERED AIR INTAKE
CONTAINMENT BUILDING
HVAC VENT STACK
SGTS VENT

Figure 3. Simplified Control Room HVAC Diagram – 'A' Train High Rad Mode



5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

AmerGen Energy Company (AmerGen), LLC is requesting a revision to the Facility Operating License for Clinton Power Station (CPS), Unit 1. Specifically, we are 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 Consequences Analyses Using Alternative Source Terms."

AmerGen has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment.", as discussed below.

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed amendment implements alternative source term (AST) assumptions in revisions to the analyses of the following limiting design basis accidents at Clinton Power Station (CPS).

- Loss-of-Coolant Accident,
- Main Steam Line Break Accident, and
- Control Rod Drop Accident.

The AST does not require modification of the facility; rather, once the occurrence of an accident has been postulated the new source term is an input to evaluate the potential consequences. The implementation of the AST has been evaluated in revisions to the analyses of the limiting design basis accidents at CPS. Based upon the results of these analyses, it has been demonstrated that, with the requested changes, the dose consequences of these limiting events is 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 equipment affected by the revised operational conditions is not considered an initiator to any previously analyzed accident and therefore, inoperability of the equipment cannot increase the probability of any previously evaluated accident. The radiological consequences of the above design basis accidents have been evaluated with application of AST assumptions. The results conclude that the radiological consequences remain within applicable regulatory limits.

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

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The application of AST does not affect the design, functional performance or operation of the facility. Similarly, it does not affect the design or operation of any structures, systems or components involved in the mitigation of any accidents, nor does it affect the design or operation of any component in the facility such that new equipment failure modes are created.

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

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

Approval of the basis change from the original source term developed in accordance with Technical Information Document (TID) 14844 to a new AST, as described in Regulatory Guide 1.183, is requested. The results of the accident analyses revised in support of the proposed changes, and the requested Technical Specification changes, are subject to revised acceptance criteria. These analyses have been performed using conservative methodologies as specified in Regulatory Guide 1.183.

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 margin of safety is considered to be that provided by meeting the applicable regulatory limits. Relaxation of these Technical Specification requirements results in an increase in dose following certain design basis accidents. However, since the doses following these design basis accidents remain within the regulatory limits, there is not a significant reduction in a margin of safety. The changes continue to ensure that the doses at the exclusion area and low population zone boundaries, as well as the control room, are within the corresponding regulatory limits.

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

Conclusion

Based on the above, AmerGen concludes that the proposed amendments present 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 proposed changes have been evaluated to determine whether applicable regulations and requirements continue to be met. The analyses used the assumptions and guidance provided by Regulatory Guide 1.183. No exceptions to Regulatory Guide 1.183 assumptions were taken for the LOCA, CRDA, or MSLB analyses. CPS compliance with the guidance in Regulatory Guide 1.183 is summarized in the Table provided in Attachment 5. 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.

AmerGen has determined that the proposed changes do not require any exemptions or relief from regulatory requirements, other than the Technical Specifications, and do not affect conformance with any General Design Criteria (GDC) differently than described in the CPS Updated Safety Analysis Report (USAR).

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 to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

AmerGen Energy Company (AmerGen), LLC has evaluated the proposed changes against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21, "Criteria for and identification of licensing and regulatory actions requiring environmental assessments." AmerGen has determined that the proposed changes meet the criteria for a categorical exclusion as set forth in 10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review.", paragraph (c)(9), and as such, has determined that no irreversible consequences exist in accordance with 10 CFR 50.92, "Issuance of amendment.", paragraph (b). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," which changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20," Standards for Protection Against Radiation," or that changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria.

(i) The amendment involves no significant hazards consideration.

As demonstrated in Section 5.1 of this attachment, the proposed changes do not involve a significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

As part of the implementation of the AST, the total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance.", and General Design Criterion (GDC) 19 of 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

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

Dose Results (rem)				
	EAB Doses and Limit		LPZ Doses and Limit	
Accident	CPS Dose (TEDE)	Limit (TEDE)	CPS Dose (TEDE)	Limit (TEDE)
Loss of Coolant Accident	19.5	25	4.3	25
Main Steam Line Break	0.65[1] 0.033[2]	25[1] 2.5[2]	0.18[1] 0.0091[2]	25[1] 2.5[2]
Control Rod Drop Accident	Case 1: 0.041 Case 2: 0.64	6.3	Case 1: 0.016 Case 2: 0.15	6.3

Notes: 1. Based on a pre-accident spike concentration of 4.0 μCi/gm dose equivalent I-131.
2. Based on a maximum equilibrium concentration of 0.2 μCi/gm dose equivalent I-131.
Case 1 is based on 1% of condenser free volume leakage per day.
Case 2 is based on steam jet air ejector activity release.

Based on the above, there will be no significant increase in the amounts of any effluents released offsite. These changes do not result in an increase in power level, do not increase the production, nor alter the flow path or method of disposal of radioactive waste or byproducts. Adoption of the alternative source term and Technical Specification changes which implement certain conservative assumptions in the alternative source term analyses will not result in modifications to the plant or changes in its operation that could significantly alter the type or amounts of effluents that may be released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

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

Control Room Dose Results (rem)		
Accident	CPS Dose (TEDE)	Limit (TEDE)
Loss of Coolant Accident	4.7	5
Main Steam Line Break	1.4[1] 0.069[2]	5
Control Rod Drop Accident	Case 1: 0.43 Case 2: 0.25	5

Notes: 1. Based on a pre-accident spike concentration of 4.0 μCi/gm dose equivalent I-131.
2. Based on a maximum equilibrium concentration of 0.2 μCi/gm dose equivalent I-131.
Case 1 is based on 1% of condenser free volume leakage per day.
Case 2 is based on steam jet air ejector activity release.

The alternative source term does not affect the design or operation of the facility; rather, once the occurrence of an accident has been postulated, the alternative source term is an input to evaluate the consequence. The implementation of the alternative source term has been evaluated in revisions to the analyses of the limiting design basis accidents at Clinton Power Station, Unit 1. These accidents include the control rod drop accident, loss of coolant 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 alternative source term (i.e., 10 CFR 50.67 and 10 CFR 50, Appendix A, General Design Criterion 19). Thus, there will be no significant increase in either individual or cumulative occupational radiation exposure.

7.0 REFERENCES

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

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

3. Letter from J. M. Heffley (AmerGen Energy Company, LLC) to U. S. Nuclear Regulatory Commission, "Request for Amendment to Technical Specifications

that Revise Plant System Requirements During Fuel Handling Based on Alternative Source Term," dated July 5, 2001

4. Letter from K. R. Jury (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Company, "Additional Information Supporting the License Amendment Request to Revise Plant System Requirements During Fuel Handling Based on Alternative Source Term," dated December 28, 2001

5. Letter from K. R. Jury (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "Supplemental Information Supporting a License Amendment Request to Revise Plant System Requirements During Fuel Handling Based on Alternative Source Term," dated March 1, 2002

6. Letter from U. S. Nuclear Regulatory Commission to O. D. Kingsley (Exelon Generation Company, LLC), "Clinton Power Station, Unit 1 – Issuance of Amendment (TAC No. MB2572)," dated April 3, 2002

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

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

9. RADTRAD Code, "A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," Version 3.03, developed by Sandia National Laboratories

10. PAVAN Code, "An Atmospheric Dispersion Program for Evaluating Design Bases Accidental Releases of Radioactive Materials from Nuclear Power Stations"

11. 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

12. ARCON96 Code, "Atmospheric Relative Concentrations in Building Wakes," developed by Pacific Northwest Laboratory

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

14. Regulatory (Safety) Guide 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors," 3/10/71

15. General Electric Nuclear Energy report, NEDC-32963A, "Prediction of the Onset of Fission Gas Release from Fuel in Generic BWR," March 2000

ATTACHMENT 3

MARKUP OF PROPOSED TECHNICAL SPECIFICATION PAGE CHANGES

Revised TS Pages

1.0-2
3.1-20
3.3-59
3.6-19
3.6-19a
3.6-26
3.6-27
5.0-12

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 of all devices in the channel required for channel OPERABILITY. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS: a. Movement of source range monitors, local power range monitors, intermediate 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. inhalation CEDE
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same inhalation CEDE ~~thyroid~~ dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The inhalation CEDE ~~thyroid~~ dose conversion factors used for this calculation shall be those listed in Insert 1 ~~Table III of TID 14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites."~~

(continued)

Clinton Power Station Technical Specification Insert

Insert 1 (Page 1.0-2):

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," ORNL, 1989.

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, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SLC subsystem inoperable.	A.1 Restore SLC subsystem to OPERABLE status.	7 days
B. Two SLC subsystems inoperable.	B.1 Restore one SLC subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify available volume of sodium pentaborate solution is within the limits of Figure 3.1.7-1.	24 hours

(continued)

Table 3.3.6.1-1 (page 5 of 6)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION F.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. RCIC System Isolation (continued)					
j. Drywell Pressure - High	1,2,3	2	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 1.88 psig
k. Manual Initiation	1,2,3	1	J	SR 3.3.6.1.6	NA
4. Reactor Water Cleanup (RWCU) System Isolation					
a. Differential Flow - High	1,2,3	2	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 66.1 gpm
b. Differential Flow-Timer	1,2,3	2	I	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 47 seconds
c. RWCU Heat Exchanger Equipment Room Temperature-High	1,2,3	2 per room	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 205°F
d. RWCU Pump Rooms Temperature-High	1,2,3	2 per room	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 202°F
e. Main Steam Line Tunnel Ambient Temperature-High	1,2,3	2	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 171°F
f. Reactor Vessel Water Level-Low Low, Level 2	1,2,3	4	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ -47.7 inches
	(c)	4	O	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ -47.7 inches
g. Standby Liquid Control System Initiation	1,2,3	2	L	SR 3.3.6.1.6	NA
h. Manual Initiation	1,2,3	2	J	SR 3.3.6.1.6	NA
	(c), (d)	2	N	SR 3.3.6.1.6	NA

(continued)

(c) During operations with a potential for draining the reactor vessel.
(d) During movement of recently irradiated fuel assemblies in the primary or secondary containment.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.8	------------------NOTE------------------ Only required to be met in MODES 1, 2, and 3. -- Verify the combined leakage rate for all secondary containment bypass leakage paths is $\leq 0.08\ L_a$ when pressurized to $\geq P_a$.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.9	------------------NOTE------------------ Only required to be met in MODES 1, 2, and 3. -- ~~Verify total leakage rate through all four main steam lines is $\leq$ 112 scfh when tested at $\geq P_a$.~~	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.10	------------------NOTE------------------ Only required to be met in MODES 1, 2, and 3. -- Verify combined leakage rate through hydrostatically tested lines that penetrated the primary containment is within limits.	In accordance with the Primary Containment Leakage Rate Testing Program

(continued)

Verify the leakage rate through each MSIV leakage path is $\leq$ 100 scfh when tested at $\geq$ Pa and the combined leakage rate for all MSIV leakage paths is $\leq$ 250 scfh when tested at $\geq$ Pa.

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.11	---------------NOTE-------------------- Only required to be met in MODES 1, 2, and 3. --------------------------------------- Verify that the combined leakage rate for both primary containment feedwater penetrations is ≤ ~~3~~ 2 gpm when pressurized to ≥ 1.1 P_a.	In accordance with the Primary Containment Leakage Rate Testing Program.
SR 3.6.1.3.12	Verify each instrumentation line excess flow check primary containment isolation valve actuates within the required range.	18 months

3.6 CONTAINMENT SYSTEMS

3.6.1.8 ~~Main Steam Isolation Valve (MSIV) Leakage Control System (LCS)~~ Deleted

LCO 3.6.1.8 Two MSIV LCS subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One MSIV LCS subsystem inoperable.	A.1 Restore MSIV LCS subsystem to OPERABLE status.	30 days
B. Two MSIV LCS subsystems inoperable.	B.1 Restore one MSIV LCS subsystem to OPERABLE status.	7 days
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.8.1	Operate each MSIV LCS blower $\geq$ 15 minutes.	31 days

(continued)

~~MSIV LCS~~
~~3.6.1.8~~

~~SURVEILLANCE REQUIREMENTS (continued)~~

SURVEILLANCE		FREQUENCY
~~SR 3.6.1.8.2~~	~~Verify electrical continuity of each inboard MSIV LCS subsystem heater element circuitry.~~	~~31 days~~
~~SR 3.6.1.8.3~~	~~Perform a system functional test of each MSIV LCS subsystem.~~	~~18 months~~

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5.5 Programs and Manuals (continued)

5.5.7 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in Regulatory Guide 1.52, Revision 2.

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980 at the system flowrate specified below ± 10%:

ESF Ventilation System	Flowrate
SGTS	4,000 cfm
Control Room Ventilation (CRV) Makeup Filter	3,000 cfm

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass less than specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980 at the system flowrate specified below ± 10%:

ESF Ventilation System	Flowrate	Penetration and Bypass
SGTS	4,000 cfm	0.05%
CRV Makeup Filter	3,000 cfm	0.05%
CRV Recirculation Filter	64,000 cfm	2%

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30 °C and a relative humidity of 70%:

ESF Ventilation System	Penetration
SGTS	~~0.175%~~ 1.5%
CRV Makeup Filter	~~0.175%~~ 1.5%
CRV Recirculation Filter	~~6%~~ 15%

(continued)

ATTACHMENT 4

RETYPED PAGES
FOR
TECHNICAL SPECIFICATION CHANGES
AND
BASES CHANGES (FOR INFORMATION ONLY)

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 of all devices in the channel required for channel OPERABILITY. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS: a. Movement of source range monitors, local power range monitors, intermediate 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 inhalation CEDE dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The inhalation CEDE dose conversion factors used for this calculation shall be those listed in Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," ORNL, 1989.

(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, 2 and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SLC subsystem inoperable.	A.1 Restore SLC subsystem to OPERABLE status.	7 days
B. Two SLC subsystems inoperable.	B.1 Restore one SLC subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify available volume of sodium pentaborate solution is within the limits of Figure 3.1.7-1.	24 hours

(continued)

Table 3.3.6.1-1 (page 5 of 6)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION F.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. RCIC System Isolation (continued)					
j. Drywell Pressure - High	1,2,3	2	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 1.88 psig
k. Manual Initiation	1,2,3	1	J	SR 3.3.6.1.6	NA
4. Reactor Water Cleanup (RWCU) System Isolation					
a. Differential Flow - High	1,2,3	2	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 66.1 gpm
b. Differential Flow-Timer	1,2,3	2	I	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 47 seconds
c. RWCU Heat Exchanger Equipment Room Temperature-High	1,2,3	2 per room	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 205°F
d. RWCU Pump Rooms Temperature-High	1,2,3	2 per room	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 202°F
e. Main Steam Line Tunnel Ambient Temperature-High	1,2,3	2	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 171°F
f. Reactor Vessel Water Level-Low Low, Level 2	1,2,3	4	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ -47.7 inches
	(c)	4	O	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ -47.7 inches
g. Standby Liquid Control System Initiation	1,2,3	2	L	SR 3.3.6.1.6	NA
h. Manual Initiation	1,2,3	2	J	SR 3.3.6.1.6	NA
	(c), (d)	2	N	SR 3.3.6.1.6	NA

(continued)

(c) During operations with a potential for draining the reactor vessel.
(d) During movement of recently irradiated fuel assemblies in the primary or secondary containment.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.8	------------------NOTE------------------ Only required to be met in MODES 1, 2, and 3. -- Verify the combined leakage rate for all secondary containment bypass leakage paths is $\leq 0.08\ L_a$ when pressurized to $\geq P_a$.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.9	------------------NOTE------------------ Only required to be met in MODES 1, 2, and 3. -- Verify the leakage rate through each MSIV leakage path is $\leq$ 100 scfh when tested at $\geq P_a$ and the combined leakage rate for all MSIV leakage paths is $\leq$ 250 scfh when tested at $\geq P_a$.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.10	------------------NOTE------------------ Only required to be met in MODES 1, 2, and 3. -- Verify combined leakage rate through hydrostatically tested lines that penetrated the primary containment is within limits.	In accordance with the Primary Containment Leakage Rate Testing Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.11	----------------NOTE---------------- Only required to be met in MODES 1, 2, and 3. ------------------------------------ Verify that the combined leakage rate for both primary containment feedwater penetrations is ≤ 2 gpm when pressurized to ≥ 1.1 P_a.	In accordance with the Primary Containment Leakage Rate Testing Program.
SR 3.6.1.3.12	Verify each instrumentation line excess flow check primary containment isolation valve actuates within the required range.	18 months

3.6 CONTAINMENT SYSTEMS

3.6.1.8 Deleted

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5.5 Programs and Manuals (continued)

5.5.7 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in Regulatory Guide 1.52, Revision 2.

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980 at the system flowrate specified below ± 10%:

ESF Ventilation System	Flowrate
SGTS	4,000 cfm
Control Room Ventilation (CRV) Makeup Filter	3,000 cfm

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass less than specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980 at the system flowrate specified below ± 10%:

ESF Ventilation System	Flowrate	Penetration and Bypass
SGTS	4,000 cfm	0.05%
CRV Makeup Filter	3,000 cfm	0.05%
CRV Recirculation Filter	64,000 cfm	2%

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30 °C and a relative humidity of 70%:

ESF Ventilation System	Penetration
SGTS	1.5%
CRV Makeup Filter	1.5%
CRV Recirculation Filter	15%

(continued)

(continued)

TABLE OF CONTENTS

(continued)

Clinton Power Station Bases Inserts

Insert A (Bases page B 3.1-38)

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 iodine will be retained in the suppression pool water (Ref. 8).

Insert B (Bases page B 3.1-39)

Following a LOCA, offsite doses from the accident will remain within 10 CFR 50.67, "Accident Source Term," limits (Ref. 9) provided sufficient iodine activity is retained in the suppression pool. Credit for iodine deposition in the suppression pool is allowed (Ref. 8) as long as suppression pool pH is maintained at or above 7. Alternative Source Term analyses credit the use of the SLC System for maintaining the pH of the suppression pool at or above 7.

Insert C (Bases page B 3.1-39)

In MODES 1, 2, and 3, the SLC System must be OPERABLE to ensure that offsite doses remain within 10 CFR 50.67 (Ref. 9) limits following a LOCA involving significant fission product releases. The SLC System is used to maintain suppression pool pH at or above 7 following a LOCA to ensure that iodine will be retained in the suppression pool water (Ref. 8).

Insert D (Bases page B 3.3-157)

Both channels are also required to be OPERABLE in MODES 1, 2, and 3, since the SLC System is also used to maintain suppression pool pH at or above 7 following a LOCA to ensure that iodine will be retained in the suppression pool water.

Insert E (Bases page B 3.7-16)

This testing ensures that the inleakage through the negative pressure portion of the Control Room Ventilation System remains within the design basis accident analysis basis. This inleakage would be filtered by the Control Room Ventilation System recirculation filters. An additional allowance of 600 cfm of unfiltered inleakage is also considered in the design basis accident analysis.

Insert F (Bases page B 3.6-25)

Dose associated with leakage through the primary containment purge lines is considered to be in addition to that controlled as part of the primary containment leakage rate limit, L_a, and the 0.08 L_a limit for the other secondary containment bypass leakage paths.

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 (ATWS).

Insert A

The SLC System consists of a boron solution storage tank, two positive displacement pumps, two explosive valves, which 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 preferred flow path of the boron neutron absorber solution to the reactor vessel is by the High Pressure Core Spray (HPCS) System sparger. The SLC piping is connected to the HPCS System just downstream of the HPCS manual injection isolation valve. An alternate flow path to the reactor vessel is provided by the SLC sparger near the bottom of the core shroud. This flow path is normally locked out of service by the SLC manual injection valve.

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 not enough control rods can be inserted to accomplish shutdown and cooldown in the normal manner. The SLC System injects borated water into the reactor core to compensate for all of the various reactivity effects that could occur during plant operation. To meet this objective, it is necessary to inject a quantity of boron that produces a concentration of at least 660 ppm of natural boron in the reactor core 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 concentration versus volume limits in Figure 3.1.7-1 are calculated such that the required concentration is achieved accounting for dilution in the RPV with normal water level and including

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 storage tank level instrument zero. (The instrument zero is based on ensuring sufficient net positive suction head and includes additional margin to preclude air entrainment in the pump suction piping due to vortexing during two pump operation.)

Insert B

The SLC System satisfies the requirements of the NRC Policy Statement because operating experience and probabilistic risk assessment have generally shown it to be important to public health and safety.

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 containing an OPERABLE pump, an explosive valve and associated piping, valves, and instruments and controls to ensure an OPERABLE flow path.

APPLICABILITY

In MODES 1 and 2, shutdown capability is required. 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 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 during these conditions, when only a single control rod can be withdrawn.

Insert C

ACTIONS

A.1

If one SLC subsystem is inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE subsystem is adequate to perform the shutdown function. However, the overall reliability is reduced because a single failure in the

(continued)

BASES

ACTIONS

A.1 (continued)

remaining OPERABLE subsystem could result in reduced SLC System shutdown capability. The 7 day Completion Time is based on the availability of an OPERABLE subsystem capable of performing the intended SLC System function and the low probability of a Design Basis Accident (DBA) or severe transient occurring concurrent with the failure of the Control Rod Drive System to shut down the plant.

B.1

If both SLC subsystems are inoperable, at least one subsystem must be restored to OPERABLE status within 8 hours. The allowed Completion Time of 8 hours is considered acceptable, given the low probability of a DBA or transient occurring concurrent with the failure of the control rods to shut down the reactor.

C.1 and C.2

If any Required Action and associated Completion Time is not 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 MODE 3 within 12 hours and MODE 4 within 36 hours. The allowed Completion Times are ~~Time of 12 hours is~~ reasonable, based on operating experience, to reach ~~MODE 3~~ the required plant conditions 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 (i.e., the volume and temperature of the borated solution in the storage tank, and temperature of the pump suction piping), thereby ensuring the SLC System OPERABILITY without disturbing normal plant operation. These Surveillances ensure the proper borated solution 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

(continued)

BASES

REFERENCES

1. 10 CFR 50.62.
2. USAR, Section 9.3.5.3.
3. Calculation IP-0-0012.
4. Calculation IP-0-0013.
5. Calculation IP-0-0014.
6. Calculation IP-0-0015.
7. Calculation IP-0-0016.
8. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants, Final Report," February 1, 1995.
9. 10 CFR 50.67, "Accident Source Terms."

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4.f. Reactor Vessel Water Level-Low Low, Level 2 (continued)

containment. Thus, this Function is also required under those conditions in which a low reactor water level signal could be generated when secondary containment is required to be OPERABLE.

4.g. 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. 4). SLC System initiation signals are initiated from the two SLC pump start signals.

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.

Two channels (one from each pump) of 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 MODES are consistent with the Applicability for the SLC System (LCO 3.1.7).

Insert D

4.h. Manual Initiation

The Manual Initiation push button channels introduce signals into the RWCU System isolation logic that are redundant to the automatic protective instrumentation and provide manual isolation capability. There is no specific USAR safety analysis that takes credit for this Function. It is retained for the isolation function as required by the NRC in plant licensing basis.

There are two push buttons for the logic, one manual initiation push button per trip system. There is no Allowable Value for this Function, since the channels are mechanically actuated based solely on the position of the push buttons.

Two channels of the Manual Initiation Function are available and are required to be OPERABLE. This Function is also required to be OPERABLE during movement of recently

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.4 (continued)

in a time period less than or equal to that assumed in the safety analysis. The isolation time and Frequency of this SR are in accordance with the Inservice Testing Program.

With regard to isolation time values obtained pursuant to this SR, as read from plant indication instrumentation, the specified limit is considered to be a nominal value and therefore does not require compensation for instrument indication uncertainties (Ref. 8).

SR 3.6.1.3.5

for each penetration

0.02

For primary containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of the Primary Containment Leakage Rate Testing Program is required to ensure OPERABILITY. The acceptance criterion for this test is ≤ ~~0.01~~ L_a when pressurized to Pa, 9.0 psig. Since cycling these valves may introduce additional seal degradation (beyond that which occurs to a valve that has not been opened), this SR must be performed within 92 days after opening the valve. However, operating experience has demonstrated that if a valve with a resilient seal is not stroked during an operating cycle, significant increased leakage through the valve is not observed. Based on this observation, a normal Frequency in accordance with the Primary Containment Leakage Rate Testing Program was established.

The SR is modified by a Note stating that the primary containment purge valves are only required to meet leakage rate testing requirements in MODES 1, 2, and 3. If a LOCA inside primary containment occurs in these MODES, purge valve leakage must be minimized to ensure offsite radiological release is within limits. At other times when the purge valves are required to be capable of closing (e.g., during handling of recently irradiated fuel), pressurization concerns are not present and the purge valves are not required to meet any specific leakage criteria.

With regard to leakage rate values obtained pursuant to this SR, as read from plant indication instrumentation, the specified limit is considered to be a nominal value and therefore does not require compensation for instrument indication uncertainties (Ref. 9).

Insert F

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.8 (continued)

leakage through the isolation device. If both isolation valves in the penetration are closed, the actual leakage rate is the lesser leakage rate of the two valves. This method of quantifying maximum pathway leakage is only to be used for this SR.

The Frequency is consistent with the Primary Containment Leakage Rate Testing Program. This SR simply imposes additional acceptance criteria. Secondary containment bypass leakage is considered part of L_a.

A Note is added to this SR which states that these valves are only required to meet this leakage limit in MODES 1, 2 and 3. In the other conditions, the Reactor Coolant System is not pressurized and specific primary containment leakage limits are not required.

With regard to leakage rate values obtained pursuant to this SR, as read from plant indication instrumentation, the specified limit is considered to be a nominal value and therefore does not require compensation for instrument indication uncertainties (Ref. 9).

SR 3.6.1.3.9

The analyses in References 1, 2, and 3 are based on leakage that is less than the specified leakage rate. Combined leakage through all four main steamlines must be ≤ ~~112~~ 250 scfh when tested at P_a (9.0 psig). In addition, the leakage rate through any single main steam line must be ≤ 100 scfh when tested at Pa. The MSIV leakage rate must be verified to be in accordance with the assumptions of References 1, 2, and 3. A Note is added to this SR which states that these valves are only required to meet this leakage limit in MODES 1, 2, and 3. In the other conditions, the Reactor Coolant System is not pressurized and primary containment leakage limits are not required. The Frequency is required by the Primary Containment Leakage Rate Testing Program.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.11

This SR ensures that the combined leakage rate of the primary containment feedwater penetrations is less than the specified leakage rate. The leakage rate is based on water as the test medium since these penetrations are designed to 2 be sealed by the FWLCS. The ~~3~~ gpm leakage limit has been shown by testing and analysis to bound the condition following a DBA LOCA where, for a limited time, both air and water are postulated to leak through this pathway. The leakage rate of each primary containment feedwater penetration is assumed to be the maximum pathway leakage, i.e., the leakage through the worst of the two isolation valves [either 1B21-F032A(B) or 1B21-F065A(B)] in each penetration. This provides assurance that the assumptions in the radiological evaluations of References 1 and 2 are met (Ref. 15).

Dose associated with leakage (both air and water) through the primary containment feedwater penetrations is considered to be in addition to the dose associated with all other secondary containment bypass leakage paths.

The Frequency is in accordance with the Primary Containment Leakage Rate Testing Program.

A Note is added to this SR which states that the primary containment feedwater penetrations are only required to meet this leakage limit in Modes 1, 2, and 3. In other conditions, the Reactor Coolant System is not pressurized and specific primary containment leakage limits are not required.

SR 3.6.1.3.12

This SR requires a demonstration that each instrumentation line excess flow check valve (EFCV) which communicates to the reactor coolant pressure boundary (Ref. 16) is OPERABLE by verifying that the valve activates within the required flow range. For instrument lines connected to reactor coolant pressure boundary, the EFCVs serve as an additional flow restrictor to the orifices that are installed inside the drywell (Ref. 14). The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.8 ~~Main Steam Isolation Valve (MSIV) Leakage Control System (LCS)~~ Deleted

BASES

BACKGROUND

The MSIV LCS supplements the isolation function of the MSIVs by processing the fission products that could leak through the closed MSIVs after a Design Basis Accident (DBA) loss of coolant accident (LOCA).

The MSIV LCS consists of two independent subsystems: an inboard subsystem, which is connected between the inboard and outboard MSIVs; and an outboard subsystem, which is connected immediately downstream of the outboard MSIVs. Each subsystem is capable of processing leakage from MSIVs following a DBA LOCA. Each subsystem consists of blowers (one blower for the inboard subsystem and two blowers for the outboard subsystem), valves, piping, and heaters (for the inboard subsystem only). The four electric heaters in the inboard subsystem are provided to boil off any condensate prior to the gas mixture passing through the flow limiter.

Each subsystem operates in two process modes: depressurization and bleedoff. The depressurization process reduces the steam line pressure to within the operating capability of equipment used for the bleedoff mode. During bleedoff (long term leakage control), the blowers maintain a negative pressure in the main steam lines (Ref. 1). This ensures that leakage through the closed MSIVs is collected by the MSIV LCS. In the depressurization mode, the effluent is released to the main steam tunnel atmosphere. In the bleed off mode, the effluent is discharged to the Standby Gas Treatment (SGT) System.

The MSIV LCS is manually initiated approximately 20 minutes following a DBA LOCA (Ref. 1).

APPLICABLE SAFETY ANALYSES

The MSIV LCS mitigates the consequences of a DBA LOCA by ensuring that fission products that may leak from the closed MSIVs are diverted to the SGT System. The analyses in Reference 2 provide the evaluation of offsite dose consequences. The operation of the MSIV LCS prevents a release of untreated leakage for this type of event.

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BASES

APPLICABLE SAFETY ANALYSES (continued)	The MSIV LCS satisfies Criterion 3 of the NRC Policy Statement.
LCO	One MSIV LCS subsystem can provide the required processing of the MSIV leakage. To ensure that this capability is available, assuming worst case single failure, two MSIV LCS subsystems must be OPERABLE.
APPLICABILITY	In MODES 1, 2, and 3, a DBA could lead to a fission product release to primary containment. Therefore, MSIV LCS 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 MSIV LCS OPERABLE is not required in MODE 4 or 5 to ensure MSIV leakage is processed.
ACTIONS	A.1 With one MSIV LCS subsystem inoperable, the inoperable MSIV LCS subsystem must be restored to OPERABLE status within 30 days. In this Condition, the remaining OPERABLE MSIV LCS subsystem is adequate to perform the required leakage control function. However, the overall reliability is reduced because a single failure in the remaining subsystem could result in a total loss of MSIV leakage control function. The 30 day Completion Time is based on the redundant capability afforded by the remaining OPERABLE MSIV LCS subsystem and the low probability of a DBA LOCA occurring during this period. B.1 With two MSIV LCS subsystems inoperable, at least one subsystem must be restored to OPERABLE status within 7 days. The 7 day Completion Time is based on the low probability of the occurrence of a DBA LOCA.

(continued)

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BASES

ACTIONS (continued)

C.1 and C.2

If the MSIV LCS subsystem 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.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.8.1

Each MSIV LCS blower is operated for ≥ 15 minutes to verify OPERABILITY. The 31 day Frequency was developed considering the known reliability of the MSIV LCS blower and controls, the two subsystem redundancy, and the low probability of a significant degradation of the MSIV LCS subsystem occurring between surveillances and has been shown to be acceptable through operating experience.

With regard to operation time values obtained pursuant to this SR, as determined from plant indication instrumentation, the specified limit is considered to be a nominal value and therefore does not require compensation for instrument indication uncertainties (Ref. 3).

SR 3.6.1.8.2

The electrical continuity of each inboard MSIV LCS subsystem heater is verified by a resistance check, by verifying the rate of temperature increase meets specifications, or by verifying the current or wattage draw meets specifications. The 31 day Frequency is based on operating experience that has shown that these components usually pass this Surveillance when performed at this Frequency.

(continued)

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.8.3

A system functional test is performed to ensure that the MSIV LCS will operate through its operating sequence. This includes verifying that the automatic positioning of the valves and the operation of each interlock and timer are correct, that the blowers start and develop the required flow rate (i.e., ≥ 100 scfm for the inboard system and ≥ 200 scfm for the outboard system) and the necessary vacuum (i.e., ≥ 15 inches-water gauge), and the upstream heaters meet current (i.e., 7.4 to 9.2 amperes per phase) draw requirements (which may also be used to verify electrical continuity in SR 3.6.1.8.2). The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

With regard to flow rate, vacuum, and current values obtained pursuant to this SR, as read from plant indication instrumentation, the specified limit is considered to be a nominal value and therefore does not require compensation for instrument indication uncertainties (Ref. 4).

REFERENCES

1. USAR, Section 6.7.
2. USAR, Section 15.6.5.
3. Calculation IP-0-0069.
4. Calculation IP-0-0070.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.4 and SR 3.6.4.1.5

The SGT System exhausts the secondary containment atmosphere to the environment through appropriate treatment equipment. To ensure that all fission products are treated, SR 3.6.4.1.4 verifies that the SGT System will rapidly establish and maintain a pressure in the secondary containment that is less than the lowest postulated pressure external to the secondary containment boundary. This is confirmed by demonstrating that one SGT subsystem will draw down the secondary containment to ≥ 0.25 inches of vacuum water gauge within the time.

12 minutes (i.e., 10 minutes from start of gap release which occurs 2 minutes after LOCA initiation)

Specifically, the required drawdown time limit is based on ensuring that the SGT system will draw down the secondary containment pressure to ≥ 0.25 inches of vacuum water gauge within ~~188 seconds~~ under LOCA conditions. Typically, however, the conditions under which drawdown testing is performed pursuant to SR 3.6.4.1.4 are different than those assumed for LOCA conditions. For this reason, and because test results are dependent on or influenced by certain plant and/or atmospheric conditions that may be in effect at the time testing is performed, it is necessary to adjust the test acceptance criteria (i.e., the required drawdown time) to account for such test conditions. Conditions or factors that may impact the test results include wind speed, whether the turbine building ventilation system is running, and whether the containment equipment hatch is open (when the test is performed during plant shutdown/outage conditions). The acceptance criteria for the drawdown test are thus based on a computer model (Ref. 6), verified by actual performance of drawdown tests, in which the drawdown time determined for accident conditions is adjusted to account for performance of the test during normal but certain plant conditions. The test acceptance criteria are specified in the applicable plant test procedure(s). Since the drawdown time is dependent upon secondary containment integrity, the drawdown requirement cannot be met if the secondary containment boundary is not intact.

SR 3.6.4.1.5 demonstrates that each SGT subsystem can maintain ≥ 0.25 inches of vacuum water gauge for 1 hour at a flow rate ≤ 4400 acfm. The 1-hour test period allows secondary containment to be in thermal equilibrium at steady state conditions. Therefore, the tests required per SR 3.6.4.1.4 and SR 3.6.4.1.5 are performed to ensure secondary containment boundary integrity. Since these SRs are secondary containment tests, they need not be performed with each SGT subsystem and an inoperable SGT subsystem does not result in this SR being not met. The SGT subsystems are tested on a STAGGERED TEST BASIS, however, to ensure that in addition to the requirements of LCO 3.6.4.3, either SGT subsystem will perform this test. Operating experience has

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.3.5

This SR verifies the integrity of the negative pressure portions of the Control Room Ventilation System ductwork located outside the main control room habitability boundary between fan OVC04CA(B) and isolation dampers OVC03YA(B) inclusive and fire dampers OVC042YA(E), OVC042YB(F), OVC042YC(G), and OVC042YD(H). In addition, the integrity of the recirculation filter housing flexible connection to fan OVC03A(B) must be verified. ~~This testing ensures the unfiltered inleakage (scfm) into the main control room habitability boundary is within the analysis assumptions.~~ Insert E Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, this Frequency is concluded to be acceptable from a reliability standpoint.

With regard to ~~unfiltered~~ inleakage values obtained pursuant to this SR, as read from plant indication instrumentation, the specified limit is not considered to be a nominal value with respect to instrument uncertainties. This requires additional margin to be added to the limit to compensate for instrument uncertainties, for implementation in the associated plant procedures (Ref. 12).

SR 3.7.3.6

This SR verifies the integrity of the control room enclosure and the assumed inleakage rates of potentially contaminated air. The control room positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper function of the Control Room Ventilation System. During the high radiation mode of operation, the Control Room Ventilation System is designed to slightly pressurize the control room to $\geq 1/8$ inches water gauge positive pressure with respect to adjacent areas to prevent unfiltered inleakage. The Control Room Ventilation System is designed to maintain this positive pressure at a flow rate of $\leq$ 3000 scfm to the control room in the high radiation mode. The Frequency of 18 months on a STAGGERED TEST BASIS is consistent with industry practice and other filtration system SRs.

With regard to control room positive pressure and flow rate values obtained pursuant to this SR, as read from plant indication instrumentation, the specified limit is not considered to be a nominal value with respect to instrument uncertainties. This requires additional margin to be added to the limit to compensate for instrument uncertainties, for implementation in the associated plant procedures (Ref. 13).

ATTACHMENT 5

REGULATORY GUIDE 1.183 COMPARISON

Regulatory Guide 1.183 Comparison and Compliance Matrix

Table I: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	**RG Position**	**CPS Analysis**	**Comments**
3.1	The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty. The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 or ORIGEN-ARP. Core inventory factors (Ci/MWt) provided in TID-14844 and used in some analysis computer codes were derived for low burnup, low enrichment fuel and should not be used with higher burnup and higher enrichment fuels.	Conforms	ORIGEN 2.1 based methodology was used to determine core inventory. Power level used was 3543 MWt to account for two percent uncertainty (3473 x 1.02 = 3543). Fission product inventory is based on an average core burnup of 1095.75 effective full power days.
3.1	For the DBA LOCA, all fuel assemblies in the core are assumed to be affected and the core average inventory should be used. For DBA events that do not involve the entire core, the fission product inventory of each of the damaged fuel rods is determined by dividing the total core inventory by the number of fuel rods in the core. To account for differences in power level across the core, radial peaking factors from the facility's core operating limits report (COLR) or Technical Specifications should be applied in determining the inventory of the damaged rods.	Conforms	Peaking factors of 1.7 are used for DBA events that do not involve the entire core, with fission product inventories for damaged fuel rods determined by dividing the total core inventory by the number of fuel rods in the core.

Table I: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	**RG Position**	**CPS 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	
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	Conforms	The fractions from Table 1 are used.

<table>
<tr><th colspan="4">Table I: Conformance with Regulatory Guide (RG) 1.183 Main Sections</th></tr>
<tr><th>RG Section</th><th>RG Position</th><th>CPS Analysis</th><th>Comments</th></tr>
<tr><td>3.2</td><td>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
<table><tr><th>Group</th><th>Fraction</th></tr><tr><td>I-131</td><td>0.08</td></tr><tr><td>Kr-85</td><td>0.10</td></tr><tr><td>Other Noble Gases</td><td>0.05</td></tr><tr><td>Other Halogens</td><td>0.05</td></tr><tr><td>Alkali Metals</td><td>0.12</td></tr></table></td><td>Conforms</td><td>Complies with Note 11 of Table 3.

Peaking factors of 1.7 are used for DBA events that do not involve the entire core.</td></tr>
<tr><td>3.3</td><td>Table 4 tabulates the onset and duration of each sequential release phase for DBA LOCAs at PWRs and BWRs. The specified onset is the time following the initiation of the accident (i.e., time = 0). The early in-vessel phase immediately follows the gap release phase. The activity released from the core during each release phase should be modeled as increasing in a linear fashion over the duration of the phase. For non-LOCA DBAs, in which fuel damage is projected, the release from the fuel gap and the fuel pellet should be assumed to occur instantaneously with the onset of the projected damage.

Table 4
LOCA Release Phases
<table><tr><th></th><th colspan="2">PWRs</th><th colspan="2">BWRs</th></tr><tr><th>Phase</th><th>Onset</th><th>Duration</th><th>Onset</th><th>Duration</th></tr><tr><td>Gap Release</td><td>30 sec</td><td>0.5 hr</td><td>2 min</td><td>0.5 hr</td></tr><tr><td>Early In-Vessel</td><td>0.5 hr</td><td>1.3 hr</td><td>0.5 hr</td><td>1.5 hr</td></tr></table></td><td>Conforms</td><td>The BWR durations from Table 4 are used.

LOCA is modeled in a linear fashion.

Non-LOCA is modeled as an instantaneous release.</td></tr>
</table>

Table I: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	CPS Analysis	Comments
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	CPS 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.
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	Conforms	

Table I: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	CPS Analysis	Comments
	fractions. The accident-specific appendices to this regulatory guide provide additional details.		
3.6	The amount of fuel damage caused by non-LOCA design basis events should be analyzed to determine, for the case resulting in the highest radioactivity release, the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and the fraction of fuel elements for which the fuel clad is breached. Although the NRC staff has traditionally relied upon the departure from nucleate boiling ratio (DNBR) as a fuel damage criterion, licensees may propose other methods to the NRC staff, such as those based upon enthalpy deposition, for estimating fuel damage for the purpose of establishing radioactivity releases.	Conforms	The current design basis for fuel damage from a CRDA is not impacted by application of AST and is, therefore, unchanged. The fuel damage assumptions are more conservative than those of the CPS Updated Safety Analysis Report.
4.1.1	The dose calculations should determine the TEDE. TEDE is the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure. The calculation of these two components of the TEDE should consider all radionuclides, including progeny from the decay of parent radionuclides, that are significant with regard to dose consequences and the released radioactivity.	Conforms	TEDE is calculated, with significant progeny included.
4.1.2	The exposure-to-CEDE factors for inhalation of radioactive material should be derived from the data provided in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers". 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", 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.
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	Conforms	The analysis uses RADTRAD default values (three significant figures), which corresponds to the values in

Table I: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	CPS Analysis	Comments
	accident, the rate should be assumed to be 2.3 x 10^{-4} cubic meters per second.		Section 4.1.3 of RG 1.183.
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", 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.
4.1.5	The TEDE should be determined for the most limiting person at the EAB. The maximum EAB TEDE for any two-hour period following the start of the radioactivity release should be determined and used in determining compliance with the dose criteria in 10 CFR 50.67. The maximum two-hour TEDE should be determined by calculating the postulated dose for a series of small time increments and performing a "sliding" sum over the increments for successive two-hour periods. The maximum TEDE obtained is submitted. The time increments should appropriately reflect the progression of the accident to capture the peak dose interval between the start of the event and the end of radioactivity release (see also Table 6).	Conforms	
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	
4.1.7	No correction should be made for depletion of the effluent plume by deposition on the ground.	Conforms	

Table I: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	**RG Position**	**CPS Analysis**	**Comments**
4.2.1	The TEDE analysis should consider all sources of radiation that will cause exposure to control room personnel. The applicable sources will vary from facility to facility, but typically will include: • Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility, • Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope, • Radiation shine from the external radioactive plume released from the facility, • Radiation shine from radioactive material in the reactor containment, • Radiation shine from radioactive material in systems and components inside or external to the control room envelope, e.g., radioactive material buildup in recirculation filters.	Conforms	The principal source of dose within the control room is due to airborne activity. The dose estimates from post LOCA sources external to the control room were based on TID-14844 source terms. The associated analyses were evaluated to confirm that they would bound an analysis that would use AST source terms and assumptions.
4.2.2	The radioactive material releases and radiation levels used in the control room dose analysis should be determined using the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values, unless these assumptions would result in non-conservative results for the control room.	Conforms	The source term, transport, and release methodology is the same for both the control room and offsite locations.
4.2.3	The models used to transport radioactive material into and through the control room, and the shielding models used to determine radiation dose rates from external sources, should be structured to provide suitably conservative estimates of the exposure to control room personnel.	Conforms	
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	Conforms	Pressurization and intake filtration are credited in the LOCA accident analysis. No

Table I: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	**RG Position**	**CPS Analysis**	**Comments**
	filtration. Refer to Section 6.5.1, "ESF Atmospheric Cleanup System," of the SRP and Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post-accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants", for guidance.		credit is taken in the MSLB and CRDA accident analyses.
4.2.5	Credit should generally not be taken for the use of personal protective equipment or prophylactic drugs. Deviations may be considered on a case-by-case basis.	Conforms	Such credits are not taken.
4.2.6	The dose receptor for these analyses is the hypothetical maximum exposed individual who is present in the control room for 100% of the time during the first 24 hours after the event, 60% of the time between 1 and 4 days, and 40% of the time from 4 days to 30 days. For the duration of the event, the breathing rate of this individual should be assumed to be 3.5×10^{-4} cubic meters per second.	Conforms	Breathing rate based on RADTRAD default values (three significant figures), which corresponds to the values in Section 4.2.6 of RG 1.183.
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. $$DDE_{finite} = \frac{DDE_{\infty} V^{0.338}}{1173}$$	Conforms	The equation given is utilized for finite cloud correction when calculating external doses due to the airborne activity inside the control room.
4.3	The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737. Design envelope source terms provided in NUREG-0737 should be updated for	Conforms	The Technical Support Center at CPS is served by the same HVAC system as the control room, and therefore plant

Table I: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	**RG Position**	**CPS Analysis**	**Comments**
	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.		personnel located there will have similar radiation exposures. 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 be lower than currently reported.
5.1.1	The evaluations required by 10 CFR 50.67 are re-analyses of the design basis safety analyses and evaluations required by 10 CFR 50.34; they are considered to be a significant input to the evaluations required by 10 CFR 50.92 or 10 CFR 50.59. These analyses should be prepared, reviewed, and maintained in accordance with quality assurance programs that comply with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.	Conforms	
5.1.2	Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The single active component failure that results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences.	Conforms	The analyses take credit for SLC System operation. The SLC System is safety-related, required to be operable by technical specifications, and powered by emergency power. The SLC System is manually initiated from the main control room, as directed by the emergency

Table I: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	**RG Position**	**CPS Analysis**	**Comments**
			operating procedures.
5.1.3	The numeric values that are chosen as inputs to the analyses required by 10 CFR 50.67 should be selected with the objective of determining a conservative postulated dose. In some instances, a particular parameter may be conservative in one portion of an analysis but be nonconservative in another portion of the same analysis.	Conforms	
5.1.4	Licensees should ensure that analysis assumptions and methods are compatible with the AST and the TEDE criteria.	Conforms	
5.3	Atmospheric dispersion values (χ/Q) for the EAB, the LPZ, and the control room that were approved by the staff during initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified by this guide. Methodologies that have been used for determining χ/Q values are documented in Regulatory Guides 1.3 and 1.4, Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and the paper, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19". The NRC computer code PAVAN implements Regulatory Guide 1.145 and its use is acceptable to the NRC staff. The methodology of the NRC computer code ARCON96 is generally acceptable to the NRC staff for use in determining control room χ/Q values.	Conforms	New atmospheric dispersion values (χ/Q) for the EAB, the LPZ, and the control room were developed, using meteorology data for the years 2000 through 2002. ARCON96 and PAVAN were used with these data to determine control room and EAB/LPZ atmospheric dispersion values, respectively.

Table 2: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	**RG Position**	**CPS 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 R.G. 1.183, and are implemented by RADTRAD. ***Timing of Release Phases:*** Release Phases are per Table 4 of R.G. 1.183, and are implemented by RADTRAD. ***Radionuclide Composition:*** Radionuclide grouping is per Table 5 of R.G. 1.183, as implemented in RADTRAD. ***Chemical Form:*** Treatment of release chemical form is per R.G. 1.183, Section 3.5.
2	If the sump or suppression pool pH is controlled at values of 7 or greater, the chemical form of radioiodine released to the containment should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine species, including those from iodine re-evolution, for sump or suppression pool pH values less than 7 will be evaluated on a case-by-case basis. Evaluations of pH should consider the effect of acids and bases created during the LOCA event, e.g., radiolysis products. With the exception of elemental and organic iodine and noble gases, fission products should	Conforms	The stated distributions of iodine chemical forms are used. 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

Table 2: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	**RG Position**	**CPS Analysis**	**Comments**
	be assumed to be in particulate form.		fission product releases, and the impact of SLCS injection. Suppression pool pH remains above 7 for at least 30 days.
3.1	The radioactivity released from the fuel should be assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment in PWRs or the drywell in BWRs as it is released. This distribution should be adjusted if there are internal compartments that have limited ventilation exchange. The suppression pool free air volume may be included provided there is a mechanism to ensure mixing between the drywell to the wetwell. The release into the containment or drywell should be assumed to terminate at the end of the early in-vessel phase.	Conforms	See Item 3.7 below.
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 and in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments". The latter model is incorporated into the analysis code RADTRAD.	Conforms	Credit is taken for natural deposition per the methodology of NUREG/CR-6189, as implemented in RADTRAD. 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 may be credited. Acceptable models for the removal of iodine and aerosols are described in Chapter 6.5.2 of the SRP and NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays". This simplified model is incorporated into the analysis code RADTRAD. The evaluation of the containment sprays should address areas within the primary containment that are not covered by the spray drops. The	Not Applicable	While containment sprays are a design feature that is available at CPS, no credit is taken for aerosol removal by them in the LOCA AST reanalysis. This is because of their limited benefit, and their unavailability during Mode 3 when portions of upper pools

Table 2: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	CPS Analysis	Comments
	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).		have been drained to facilitate preparations for refueling outage operations.
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. 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 CPS.
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	Conforms	No credit is taken for suppression pool scrubbing in the LOCA AST reanalysis. As indicated for Item 1 above, analyses have been

Table 2: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	**RG Position**	**CPS Analysis**	**Comments**
	through the pool, and the potential for any bypass of the suppression pool. Analyses should consider iodine re-evolution if the suppression pool liquid pH is not maintained greater than 7.		performed that determined that the suppression pool liquid pH is maintained greater than 7, and that, 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.	Not Applicable	CPS does not have ice condensers.
3.7	The primary containment (i.e., drywell for Mark I and II containment designs) should be assumed to leak at the peak pressure technical specification leak rate for the first 24 hours. For PWRs, the leak rate may be reduced after the first 24 hours to 50% of the technical specification leak rate. For BWRs, leakage may be reduced after the first 24 hours, if supported by plant configuration and analyses, to a value not less than 50% of the technical specification leak rate. Leakage from subatmospheric containments is assumed to terminate when the containment is brought to and maintained at a subatmospheric condition as defined by technical specifications. For BWRs with Mark III containments, the leakage from the drywell into the primary containment should be based on the steaming rate of the heated reactor core, with no credit for core debris relocation. This leakage should be assumed during the two-hour period between the initial blowdown and termination of the fuel radioactivity release (gap and early in-vessel release phases). After two hours, the radioactivity is assumed to be uniformly distributed throughout the drywell and the primary containment.	Conforms	No credit is taken for suppression pool scrubbing in the LOCA AST reanalysis. As indicated for Item 1 above, analyses have been performed that determined that the suppression pool liquid pH is maintained greater than 7, and that, therefore, iodine re-evolution is not expected. CPS uses a Mark III containment, and leakage from the drywell into primary containment is based on the conservative 3000 cfm flow steaming flow simulation for the first two hour period applied for other BWR Mark IIIs. Rapid mixing is

Table 2: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	**RG Position**	**CPS Analysis**	**Comments**
			considered thereafter to provide the uniform distribution required. For the 1 hour period required to fill FW piping by the FWLCS, the above assumptions are excessively conservative. Recirculation Suction Line Break containment pressure analyses show that flow from containment to the drywell can occur early after initial blowdown to replace the drywell air largely removed during blowdown. For this pathway a well mixed drywell-containment atmosphere is assumed. For conservatism the historically assumed suppression pool scrubbing credit is no longer assumed.
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,	Conforms	CPS has a continuous purge system and a high-volume purge system used on an as-needed (i.e., non-routine) basis. The impact of a LOCA while the containment is being continuously purged has been conservatively analyzed and discussed in USAR Section

Table 2: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	**RG Position**	**CPS Analysis**	**Comments**
	the release fractions associated with the gap release and early in-vessel phases should be considered as applicable.		9.4.6.3.2.h, with assumptions consistent with this guidance. The results indicate trivial doses (less than 0.001 rem) which can be ignored by comparison to the other LOCA doses calculated.
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	Secondary Containment filtered release credit is taken at 10 minutes after the start of gap release. Gap release begins at ~ 2 minutes after LOCA initiation. Therefore, 12 minutes is available for achieving a negative pressure of 1/4" W.G. For EAB and LPZ doses, ground level releases are assumed. For Control Room doses, releases are based on zero-velocity vent release assumptions, yielding ground level release equivalent dispersion factors.
4.2	Leakage from the primary containment is assumed to be released directly to the environment as a ground-level release during any period in which the secondary containment does not have a negative pressure as defined in Technical Specifications.	Conforms	For EAB and LPZ doses, ground level releases are assumed. For Control Room doses, releases are based on zero-velocity vent release assumptions.

Table 2: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	**RG Position**	**CPS Analysis**	**Comments**
4.3	The effect of high wind speeds on the ability of the secondary containment to maintain a negative pressure should be evaluated on an individual case basis. The wind speed to be assumed is the 1-hour average value that is exceeded only 5% of the total number of hours in the data set. Ambient temperatures used in these assessments should be the 1-hour average value that is exceeded only 5% or 95% of the total numbers of hours in the data set, whichever is conservative for the intended use (e.g., if high temperatures are limiting, use those exceeded only 5%).	Conforms	The potential for high wind speeds impacting the ability of secondary containment to maintain negative pressures for wind speeds has been previously evaluated as discussed in USAR Section 6.5.1.1.1, where it is shown that bypass of SGTS would not occur for wind speeds up to approximately 30 miles per hour. Inspection of the data set used in the development of the AST χ/Qs shows that 30 mph is exceeded less than 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.
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	Conforms	Bypass leakage has been analyzed at 8% of L_a Additionally, the penetration 101 and 102 purge supply

Table 2: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	**RG Position**	**CPS Analysis**	**Comments**
	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.		and exhaust penetrations are analyzed separately with an additional 2% of L_a each, with deposition of radioactivity analyzed using RADTRAD Brockmann-Bixler methodology. Since doses through the 101 and 102 penetrations are analyzed separately, they need no longer be considered as among the penetrations controlled under the 8% of L_a bypass leakage limit. Release of MSIV leakage at CPS has previously been based on the use of the MSIVLCS to assure filtration by the SGTS. This system is no longer credited. MSIV leakage will have a separate technical specification limit of 250 scfh total leakage with not more that 100 scfh per line. The dose consequences for releases through this pathway (with piping deposition credit) are separately calculated. Since doses from MSIV leakage are analyzed separately, they

Table 2: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	CPS Analysis	Comments
			need not be considered as among the penetrations controlled under the 8% of L_a bypass leakage limit. Feedwater piping deposition has also been evaluated for the 1 hour period before the lines are filled using the FWLCS. As discussed above, piping deposition credit is determined using the Brockmann-Bixler routines available in RADTRAD. Delay in transit through these piping system is also credited, based on volume divided by standard pressure flow rates. The resulting delay values are 3 hours for MSIVs, 12 hours for purge penetrations, and 20 minutes for feedwater penetrations.
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 and Generic Letter 99-02.	Conforms	SGTS filters meet these criteria and are therefore credited at an efficiency of 99% for all iodine chemical forms.
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)	Conforms	With the exception of noble gases, all the fission products

Table 2: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	CPS Analysis	Comments
	should be assumed to instantaneously and homogeneously mix in the primary containment sump water (in PWRs) or suppression pool (in BWRs) at the time of release from the core. In lieu of this deterministic approach, suitably conservative mechanistic models for the transport of airborne activity in containment to the sump water may be used. Note that many of the parameters that make spray and deposition models conservative with regard to containment airborne leakage are nonconservative with regard to the buildup of sump activity.		released from the fuel to the containment are assumed to instantaneously and homogeneously mix in the suppression pool at the time of release from the core.
5.2	The leakage should be taken as two times the sum of the simultaneous leakage from all components in the ESF recirculation systems above which the technical specifications, or licensee commitments to item III.D.1.1 of NUREG-0737, would require declaring such systems inoperable. The leakage should be assumed to start at the earliest time the recirculation flow occurs in these systems and end at the latest time the releases from these systems are terminated. Consideration should also be given to design leakage through valves isolating ESF recirculation systems from tanks vented to atmosphere, e.g., emergency core cooling system (ECCS) pump miniflow return to the refueling water storage tank.	Conforms	The design basis 5 gpm leak rate is more than 2 times the acceptance criteria for the sum of the simultaneous leakage from all components in the ESF recirculation systems as addressed in the Program committed to in TS 5.5.2 "Primary Coolant Sources Outside Containment". Since certain ECCS systems take suction immediately from the suppression pool, this leak path is assumed to start at time 0. Leakage to atmospheric tanks are credible only for lines connecting from ECCS pump discharges to such a tank, because of relative

Table 2: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	CPS Analysis	Comments
			elevations. The sole leakage paths to a tank vented to atmosphere meeting this condition are the HPCS / RCIC test lines that return to the RCIC Storage Tank (RST). These lines are isolated by two normally closed valves, one air operated and one motor operated. The RST contents are demineralized water, which leaked ECCS fluid would quickly turn basic. Therefore, minimal elemental iodine is expected, and as a result, negligible iodine volatilization.
5.3	With the exception of iodine, all radioactive materials in the recirculating liquid should be assumed to be retained in the liquid phase.	Conforms	With the exception of iodine, all radioactive materials in ECCS liquids are assumed to be retained in the liquid phase.
5.4	If the temperature of the leakage exceeds 212°F, the fraction of total iodine in the liquid that becomes airborne should be assumed equal to the fraction of the leakage that flashes to vapor. This flash fraction, FF, should be determined using a constant enthalpy, h, process, based on the maximum time-dependent temperature of the sump water circulating outside the containment:	Not Applicable	The temperature of the leakage does not exceed 212°F.

Table 2: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	**RG Position**	**CPS Analysis**	**Comments**
	$$FF = \frac{h_{f1} - h_{f2}}{h_{fg}}$$ *Where:* h_{f1} is the enthalpy of liquid at system design temperature and pressure; h_{f2} is the enthalpy of liquid at saturation conditions (14.7 psia, 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	An airborne release fraction of 1.36% is used. Suppression water pH is maintained above 7 for the entire 30 days accident dose assessment period. Under these conditions virtually none of the iodine will be in elemental form, and organic iodine formation will be inhibited. Because of the subcooled condition no flashing is expected. Nevertheless, the current design basis value, derived based on ORNL-TM-2412 methodology for iodine partition factor determination, is used.
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 and Generic Letter 99-02.	Conforms	The credited SGTS and Control Room intake charcoal and HEPA filters meet the requirements of R.G. 1.52 and Generic Letter 99-02. These are credited at 97%

Table 2: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	**RG Position**	**CPS Analysis**	**Comments**
			efficiency for elemental and organic iodines. Aerosol removal efficiencies are assumed to be 99% based on the HEPA/charcoal combination. Control Room recirc filters are credited at 70% filter efficiency for all iodine species and for all aerosols.
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 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.	Conforms	Because the CPS MSIVLCS will not be credited in this accident analysis, MSIV leakage will be considered an unfiltered radioactivity release pathway, with piping deposition credit, and the radiological consequences of such a release are analyzed. The radioactivity release from the fuel is assumed to instantaneously and homogeneously mix throughout the drywell air space. Mixing of this activity into the containment air space is as discussed under Item 3.7 above.

Table 2: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	**RG Position**	**CPS Analysis**	**Comments**
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 250 scfh for all steam lines and 100 scfh for any one line. A reduction in leakage of 50% is assumed at 24 hours, based on expected containment pressures at that time.
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	Modeling is per RADTRAD Brockmann-Bixler approach, with a conservatively derived transport delay credit of 3 hours.
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	Since MSIVLCS is no longer credited, no ESFs are assumed to be available to collect or treat MSIV leakage. Releases are assumed to be from the combined exhaust stack, 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 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.	Conforms	Main steam piping downstream of the MSIVs is credited for piping that is capable of performing their safety function during and following an SSE. No credit is taken for holdup and

Table 2: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	**RG Position**	**CPS Analysis**	**Comments**
	Regulatory Guide 1.187 References A-9 and A-10 provide guidance on acceptable models.		deposition in piping downstream of this, or in the condenser. The modeling is per the RADTRAD Brockmann-Bixler approach.
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 and Generic Letter 99-02.	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 CPS. Also see the Regulatory Guide Section 3.8 discussion in this Table.

Table 3: Conformance with Regulatory Guide 1.183 Appendix C (Control Rod Drop Accident)			
RG Section	**RG Position**	**CPS 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: Note that a more conservative value than that used in the current CPS USAR is used for the number of fuel rods breached (1200 vs. 770) and the fraction of the breached rods melting (1% vs. 0.77%). Other release assumptions (10% of the core inventory of the noble gases and iodines is in the fuel gap, and 100% of the noble gases and 50% of the iodines contained in that fraction are released to the reactor coolant for fuel melting) are in agreement with USAR section 15.4.9.5.1.1. Also, a conservative peaking factor of 1.7 is used in agreement with the AST Calculation for the Fuel Handling Accident. 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 the guide) is

Table 3: Conformance with Regulatory Guide 1.183 Appendix C (Control Rod Drop Accident)			
RG Section	**RG Position**	**CPS Analysis**	**Comments**
			assumed, based on Table 3 in Regulatory Position 3 of the guide. Radionuclide grouping is per Table 5 in Regulatory Position 3 of the guide, as implemented in RADTRAD.
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 dose equivalent (DE) I-131) allowed by the Technical Specifications.	Conforms	Substantial fuel damage is postulated.
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	
3.2	Credit should not be assumed for partitioning in the pressure vessel or for removal by the steam separators.	Conforms	
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 remaining radionuclides are assumed to reach the turbine and condensers.	Conforms	
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	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 vent release assumptions. Radioactive decay during holdup in the

Table 3: Conformance with Regulatory Guide 1.183 Appendix C (Control Rod Drop Accident)			
RG Section	**RG Position**	**CPS Analysis**	**Comments**
	the turbine and condenser may be assumed.		condenser is assumed.
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	
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	Control Room filter efficiencies for all iodine chemical species are the same; therefore, variation in species has no effect.
Foot-note 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	Projected fuel damage is the limiting case.
Foot-note 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	Forced flow paths of mechanical vacuum pumps and steam jet air ejectors are considered, and the most limiting path is determined.

Table 4: Conformance with Regulatory Guide 1.183 Appendix D (Main Steam Line Break)			
RG Section	**RG Position**	**CPS 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, release estimate based on 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.	Conforms	TS SR 3.6.1.3.7 verifies the isolation time of each MSIV is between 3 and 5 seconds, well within the 5.5 seconds up to full MSIV closure assumed for the release period. TS LCO 3.4.7 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.

Table 4: Conformance with Regulatory Guide 1.183 Appendix D (Main Steam Line Break)			
RG Section	RG Position	CPS Analysis	Comments
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	
4.1	The main steam line isolation valves (MSIV) should be assumed to close in the maximum time allowed by technical specifications.	Conforms	See Item 2 above.
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	
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	
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.	Not Applicable	No filtration is credited, so the iodine species are irrelevant.