September 19, 2005

Mr. Christopher M. Crane, President
  and Chief Executive Officer
AmerGen Energy Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: CLINTON POWER STATION, UNIT 1 - ISSUANCE OF AN AMENDMENT -
            RE: APPLICATION OF ALTERNATIVE SOURCE TERM METHODOLOGY
            (TAC NO. MB8365)

Dear Mr. Crane:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment
No. 167 to Facility Operating License No. NPF-62 for the Clinton Power Station, Unit 1, in
response to your application dated April 3, 2003, as supplemented December 23, 2003,
December 9 and 17, 2004, and March 30 and August 19, 2005. The amendment supports the
application of an alternative source term (AST) methodology, in accordance with Title 10 of the
Code of Federal Regulations (10 CFR), Section 50.67, "Accident Source Term," with the
exception that Technical Information Document (TID) 14844, "Calculation of Distance Factors
for Power and Test Reactor Sites," was used as the radiation dose basis for equipment
qualification at the Clinton facility. Previously, AmerGen had proposed a selective
implementation of the AST limited to the fuel handling accident during core alterations. The
implementation was approved in amendment 147 dated April 3, 2002.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the
Commission's next biweekly *Federal Register* notice.

                            Sincerely,

                            /RA/

                            Kahtan N. Jabbour, Senior Project Manager, Section 2
                            Project Directorate III
                            Division of Licensing Project Management
                            Office of Nuclear Reactor Regulation

Docket No. 50-461

Enclosures: 1. Amendment No. 167 to NPF-62
             2. Safety Evaluation

cc w/encls: See next page

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AmerGen Energy Company, LLC
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                              Office of Nuclear Reactor Regulation

Docket No. 50-461

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cc w/encls: See next page

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Clinton Power Station, Unit 1

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DOCKET NO. 50-461

CLINTON POWER STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 167
License No. NPF-62

1.   The U.S. Nuclear Regulatory Commission (the Commission) has found that:

A.   The application for amendment by AmerGen Energy Company, LLC (the licensee),
      dated April 3, 2003, as supplemented December 23, 2003, December 9 and 17,
      2004, and March 30, and August 19, 2005, complies with the standards and
      requirements of the Atomic Energy Act of 1954, as amended (the Act), and the
      Commission's rules and regulations set forth in 10 CFR Chapter I;

B.   The facility will operate in conformity with the application, the provisions of the Act,
      and the rules and regulations of the Commission;

C.   There is reasonable assurance (i) that the activities authorized by this amendment
      can be conducted without endangering the health and safety of the public, and
      (ii) that such activities will be conducted in compliance with the Commission's
      regulations;

D.   The issuance of this amendment will not be inimical to the common defense and
      security or to the health and safety of the public; and

E.   The issuance of this amendment is in accordance with 10 CFR Part 51 of the
      Commission's regulations and all applicable requirements have been satisfied.

2.   Accordingly, the license is amended by changes to the Technical Specifications
      contained in Appendix B, as indicated in the attachment to this license amendment, and
      paragraph 2.C.(2) of Facility Operating License No. NPF-62 is hereby amended to read
      as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental
Protection Plan contained in Appendix B, as revised through Amendment No. 167
 are hereby incorporated into this license. AmerGen Energy Company, LLC shall
operate the facility in accordance with the Technical Specifications and the
Environmental Protection Plan.

3.   This license amendment is effective as of its date of issuance and shall be implemented
within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Gene Y. Suh, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 19, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 167

FACILITY OPERATING LICENSE NO. NPF-62

DOCKET NO. 50-461

Replace the following pages of Appendix "A", Technical Specifications, with the attached revised pages. The revised pages are identified by an amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
1.0-2	1.0-2
3.1-20	3.1-20
3.3-59	3.3-59
3.6-19	3.6-19
3.6-19a	3.6-19a
3.6-26	3.6-26
3.6-27	3.6-27

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 167 TO FACILITY OPERATING LICENSE NO. NPF-62

AMERGEN ENERGY COMPANY, LLC

CLINTON POWER STATION, UNIT 1

DOCKET NO. 50-461

1.0   INTRODUCTION

By application dated April 3, 2003, as supplemented December 23, 2003, December 9 and 17, 2004, and March 30, and August 19, 2005, AmerGen Energy Company, LLC (AmerGen), the licensee, requested changes to the Technical Specifications (TSs) for Clinton Power Station (CPS), Unit 1. The proposed changes support the application of an alternative source term (AST) methodology, in accordance with Title 10, *Code of Federal Regulations* (10 CFR), Section 50.67, "Accident Source Term," with the exception that Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," dated March 23, 1962, was used as the radiation dose basis for equipment qualification at the Clinton facility.

The amendment modifies the CPS licensing and design basis as described in the updated final safety analysis report (UFSAR) to replace the current accident source term with an AST and to replace the previous whole body and thyroid accident dose guidelines with the total effective dose equivalent (TEDE) criteria of 10 CFR 50.67(b)(2). AmerGen has requested an application of the AST methodology, as described in RG 1.183, with the exception that TID 14844 was used as the radiation dose basis for CPS equipment qualification. In support of the present amendment request, AmerGen has performed radiological consequence analyses of the four design basis accidents (DBAs) that result in offsite exposure. These DBAs are the loss-of-coolant accident (LOCA), main steam line break (MSLB), control rod drop accident (CRDA), and the fuel handling accident (FHA). AmerGen had previously analyzed the FHA during core alterations in support of the selective implementation of the AST.   The implementation was approved in License Amendment (LA) 147, dated April 3, 2002. The present LA request does not affect the changes implemented in the previous amendment. Therefore, the FHA is not reanalyzed. Other proposed changes in the DBA dose analyses include an increase in the assumed control room envelope inleakage and development of new offsite and control room atmospheric relative concentrations (χ/Q).

The current LA request considers the remaining design basis accidents. AmerGen also proposed revisions to several TSs which are discussed in Section 3.1.5 of this SE. The supplemental letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original no significant hazards consideration determination as published in the *Federal Register* on September 2, 2003.

2.0   REGULATORY EVALUATION

In the past, power reactor licensees have typically used the U.S. Atomic Energy Commission TID 14844 as the basis for DBA analysis source terms. The power reactor siting regulation, which contains offsite dose limits in terms of whole body and thyroid dose, 10 CFR 100.11, Determination of Exclusion Area, Low Population Zone, and Population Center Distance, makes reference to TID-14844.

In December 1999, the U.S. Nuclear Regulatory Commission (NRC) issued a new regulation, 10 CFR 50.67, which provided a mechanism for licensed power reactors to replace the traditional accident source term used in their DBA analyses with an alternative source term. Section 50.67 of 10 CFR requires a licensee seeking to use an AST to apply for a license amendment and requires that the application contain an evaluation of the consequences of affected DBAs. Regulatory guidance for the implementation of these ASTs is provided in Regulatory Guide (RG) 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors. AmerGen's application of April 3, 2003, as supplemented, addresses these requirements in proposing to use the AST described in RG 1.183 as the DBA source term used to evaluate the radiological consequences of DBAs for Clinton Unit 1. As part of the implementation of the AST, the TEDE acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11 and 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 19, as the CPS licensing basis for the LOCA, the MSLB accident, and the CRDA.

Part 50 of 10 CFR, Appendix A, GDC 26, requires that each reactor have two independent reactivity control systems of a different design, while GDC 29 requires that the reactivity control system be capable of accomplishing its safety function in the event of anticipated operational occurrences.

This safety evaluation (SE) addresses the impact of the proposed changes on previously analyzed design basis accident radiological consequences and the acceptability of the revised analysis results. The regulatory requirements on which the staff based its acceptance are the accident dose criteria in 10 CFR 50.67, as supplemented in Regulatory Position 4.4 of RG 1.183 and 10 CFR Part 50 Appendix A, GDC 19, "Control Room." Except where the licensee has proposed a suitable alternative, the staff has utilized the regulatory guidance in the following documents in performing this review:

   a.   RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants"

   b.   RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"

   c.   RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants"

   d.   Standard Review Plan (SRP) Section 2.3.4, "Short-Term Diffusion Estimates for Accidental Atmospheric Releases"

e. SRP Section 6.4, "Control Room Habitability Systems" (with regard to control room meteorology)

f. SRP Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Term"

g. 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants"

h. 10 CFR 100, "Reactor Site Criteria"

i. NUREG-0933 Issue 187, "The Potential Impact Of Postulated Cesium Concentration on Equipment Qualification".   This document provides an evaluation of the impact of the AST on environmental qualification.

j. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants"

k. 10 CFR 50.36, "Technical Specifications," Section 50.36(c)(2)(ii)

The staff also considered relevant information in the CPS UFSAR and TSs.

3.0   TECHNICAL EVALUATION

3.1   Accident Dose Calculations

The NRC staff has reviewed the technical analyses related to the radiological consequences of design basis accidents that were performed by AmerGen in support of this proposed license amendment. Information regarding these analyses was provided in the April 3, 2003, submittal and in supplemental letters dated December 23, 2003, December 9 and 17, 2004, and March 30 and August 19, 2005. The staff has reviewed the assumptions, inputs, and methods used by AmerGen to assess the impact of the proposed changes. The staff has performed independent calculations to confirm the conservatism of the AmerGen analyses. However, the findings of this SE are based on the descriptions of the analyses and other supporting information submitted by AmerGen.

In accordance with the guidance in RG 1.183, a licensee is not required to re-analyze all DBAs for the purpose of the application, just those affected by the proposed changes. However, on approval of this amendment, the AST and the TEDE criteria will become the licensing basis for all subsequently performed radiological consequence analyses intended to demonstrate compliance with 10 CFR Part 50 requirements, with exception to equipment qualification. This protocol is supported by staff evaluations that concluded that prior DBA analyses would remain bounding for the AST and the TEDE criteria and would not require updating. In keeping with this guidance, AmerGen performed an evaluation of previously analyzed DBAs to determine which, if any, were affected by the proposed amendment. AmerGen re-analyzed the radiological consequences of the following DBA events:

- LOCA

- MSLB

- CRDA

AmerGen had previously analyzed the FHA in support of its selective use of an AST. This analysis was approved in Amendment 147, dated April 3, 2002. The present amendment request does not affect that FHA analysis.

3.1.1 LOCA

The objective of analyzing the radiological consequences of a LOCA is to evaluate the performance of various plant safety systems intended to mitigate the postulated release of radioactive materials from the plant to the environment. AmerGen assumes an abrupt failure of a large reactor coolant pipe that results in a blowdown of the reactor coolant system (RCS) into the drywell. Emergency core cooling systems (ECCS) would function to make up lost water inventory and cool the core. DBA thermo-hydraulic analyses show that the ECCS can adequately cool the core and prevent significant fuel damage. Nonetheless, AmerGen conservatively assumes that the ECCS is not successful and that substantial core damage occurs.

3.1.1.1 LOCA Source Term

Fission products from the damaged fuel are released into RCS and then into the primary containment. It is anticipated that the initial release to the drywell will last 30 seconds and will release all of the radioactive materials dissolved or suspended in the RCS liquid at the start of the accident. Due to the postulated loss of core cooling, the fuel heats up, resulting in the release of fission products. The gap inventory release phase begins two minutes after the event starts and is assumed to continue for 30 minutes. As the core continues to degrade, the gap inventory release phase ends and the in-vessel release phase begins. This phase continues for 1.5 hours. Tables 1, 4, and 5 of RG 1.183 define the source term used for these two phases. The inventory in each release phase is released at a constant rate starting at the onset of the phase and continuing over the duration of the phase.

3.1.1.2 LOCA Fission Product Transport

The LOCA considered in this evaluation is a complete and instantaneous severance of one of the recirculation loops. The pipe break results in a blowdown of the reactor pressure vessel (RPV) liquid and steam to the drywell via the severed recirculation pipe. The resulting pressure buildup drives the mixture of steam, water, and other gases through the suppression pool water and into the primary containment. The suppression pool water condenses the steam and reduces the pressure. After the initial RPV blowdown, ECCS water injected into the RPV will spill into the drywell, transporting fission products to the suppression pool and then into the primary containment. In lieu of modeling this transport mechanistically, AmerGen has conservatively assumed that the fission product release from the fuel is homogeneously and instantaneously dispersed within the drywell free volume, with 3000 cubic feet per minute (cfm) of the drywell air flowing to the primary containment for the first 2 hours. Rapid mixing (10^8 cfm) between the drywell and primary containment is assumed after the first 2 hours for the duration

of the 30-day accident to model that the fission products are homogeneously distributed between the drywell and primary containment. The staff finds that the licensee's assumptions regarding drywell and containment mixing are consistent with assumptions previously found acceptable for a full implementation of an AST for the Perry Nuclear Power Plant, which has a Mark III containment such as CPS. AmerGen did not credit any reduction by suppression pool scrubbing for fission products transferred to the primary containment through the suppression pool.

AmerGen assumes that a portion of the fission products released from the RPV will plateout in the drywell and primary containment due to natural deposition processes. AmerGen models this deposition using the 10th-percentile values in the model described in the staff-accepted NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments" (i.e., the "Powers Model"). AmerGen did not assume natural deposition of elemental or organic forms of iodine in the drywell or containment. The licensee's assumptions on drywell/containment mixing and natural deposition processes are consistent with the guidance in RG 1.183.

The AST assumes that the iodine released to the containment consists of 95 percent cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic forms. The assumption of this iodine speciation is predicated on maintaining the containment sump water at pH 7.0 or higher. AmerGen proposes to use the SLC system to inject sodium pentaborate to the RPV, where it will mix with ECCS flow and spill over into the suppression pool. Sodium pentaborate, a base, will neutralize acids generated in the post-accident primary containment environment.

The staff has reviewed the quantity of sodium pentaborate available with respect to the quantity of acid producing debris and radiolytic acid production to confirm that adequate pH control exists, as discussed below in section 3.1.1.3.

3.1.1.3 Suppression Pool Post-LOCA pH

The staff has reviewed the methodology for maintaining the suppression pool pH above 7 for the 30-day period after a LOCA. Under normal operating conditions, there is no special requirement in boiling water reactors (BWRs) for controlling the suppression pool pH. The pH value will depend, therefore, on the chemical species dissolved in the suppression pool water. After an accident, these species could be released from the damaged core, generated in the radiation fields existing in the containment and drywell after a LOCA, or in the case of sodium pentaborate, manually injected from the standby liquid control system via the reactor vessel and a break in the primary coolant pressure boundary. The majority of the species introduced into the suppression pool are either acidic or basic and the resultant suppression pH will depend on their relative concentrations and on the buffering action of the sodium pentaborate added to the suppression pool water. According to NRC RG 1.183, "Alternative Radiological Source Terms Evaluating Design Basis Accidents At Nuclear Power Reactors," the analysis release duration for a LOCA is 30 days, and a pH greater than 7 will prevent iodine re-evolution.

3.1.1.3.1 Chemical Species Dissolved in Suppression Pool

A variety of acids and bases are produced in containment during an accident. The licensee identified the following chemicals which are introduced into the suppression pool in a post-LOCA environment: hydriodic acid, nitric acid, hydrochloric acid and cesium hydroxide. In addition to these chemical species, the suppression pool will contain dissolved carbon dioxide at the concentration remaining in equilibrium with its concentration in air. Also, beside the cesium hydroxide, there may be many basic chemicals, some of them produced from the interaction of molten core materials with concrete, but these were not included in the methodology. The licensee did not take credit for these materials because core damage is assumed to be arrested after the in-vessel release phase in accordance with SECY-94-302, "Source Term Related Technical and Licensing Issues Pertaining to Evolutionary and Passive Light Water reactor Designs."

Cesium and iodine are two fission products released from the damaged core and dissolved in the suppression pool water. Some of the dissolved cesium is in a form of cesium hydroxide and is a source of OH^- ions. Some of the dissolved iodine remains in a form of hydriodic acid, which is a strong acid providing H^+ ions.

The amount of chemicals released from the damaged fuel to the suppression pool used in the licensee's analysis was determined from the fractions of their total inventory in the core at the time of a LOCA. These fractions were specified in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," for the gap and early in-vessel release phases.

3.1.1.3.1.1 Hydriodic Acid

NUREG-1465 specifies that 5 percent of total iodine inventory is released during the gap release and an additional 25 percent during the early in-vessel release phase. Ninety-five percent of this iodine is in a form of cesium iodine. NUREG-1465 states that no more than 5 percent of the iodine existing in the reactor coolant system (RCS) will be composed of elemental iodine and hydriodic acid. The licensee made a very conservative assumption that the remaining iodine is in a form of hydriodic acid. The licensee assumed that all hydriodic acid is released at a constant rate and all of it is dissolved in the suppression pool water. The licensee calculated that 3.80×10^{-7} g-moles/liter of hydriodic acid are generated over the 30-day transient.

3.1.1.3.1.2 Nitric Acid

Nitric acid is produced by the irradiation of water and air. The licensee used the methodology described in NUREG/CR-5950, "Iodine Evolution and pH Control" and NUREG/CR-5732, "Iodine Chemical Forms in LWR Severe Accidents" to determine the amount of nitric acid in the suppression pool. This methodology considers the production of nitric acid to be proportional to the time-integrated radiation dose rate for gamma and beta radiation. The licensee calculated that 3.318×10^{-5} g-moles/liter of nitric acid are generated over the 30-day transient.

3.1.1.3.1.3 Hydrochloric Acid

Hydrochloric acid is generated in the post-LOCA environment by radiolytic decomposition of the Hypalon cable jacketing by beta and gamma radiation. The amount of hydrochloric acid generated is proportional to the radiation energy received and absorbed by the jacketing. The licensee assumed that the cables in a conduit or in totally enclosed raceways are completely protected from beta radiation and sufficiently shielded from gamma radiation to produce only a relative small amount of hydrogen chloride which would react with internal metal surfaces of the conduit before it could diffuse to the outside. The cables in cable trays are shielded by other cables and are only partially exposed to beta radiation. The licensee assumed, therefore, that for these cables, only half of the beta energy is effective in producing hydrogen chloride. The generation of hydrochloric acid by radiolytic decomposition of Hypalon cable jacketing was determined by the methodology described in NUREG/CR-5950 and NUREG-1801, "Post-Accident Gas Generation from Radiolysis of Organic Materials." The licensee calculated that 8.22×10^{-4} g-moles/liter of hydrochloric acid are generated over the 30-day transient.

3.1.1.3.1.4 Cesium Hydroxide

NUREG-1465 specifies that 5 percent of the total core cesium inventory is discharged to the suppression pool during the gap release phase and an additional 20 percent during the early in-vessel phase. In both cases cesium is released as cesium hydroxide and cesium iodine. The iodine discharged from the reactor coolant system will consist of at least 95 percent cesium iodide. The cesium that is not in the form of cesium iodide is assumed to exit the RCS in the form of cesium hydroxide and be deposited into the suppression pool. The licensee assumed that all cesium hydroxide is released at a constant rate and all of it is dissolved in the suppression pool water. The licensee calculated that 6.884×10^{-5} g-moles/liter of hydrochloric acid are generated over the 30-day transient.

3.1.1.3.1.5 Sodium Pentaborate

BWR plants use sodium pentaborate in their standby liquid control system to provide negative reactivity in case of a control rod failure. Although a solution of sodium pentaborate is injected into the reactor vessel, some of it will get into the suppression pool and dissolve in water producing boric acid and sodium borate.

$$Na_2B_{10}O_{16} + 15 H_2O \leftrightarrow 2 Na^+ + 2 H_2BO_3^- + 8 H_3BO_3$$
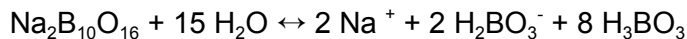
Since boric acid is a relatively weak acid and sodium hydroxide (formed by union of sodium ion hydroxyl ions) is a strong base, their solution has a buffering effect and will control pH in the suppression pool at values higher than 7.0. The operators are directed to manually initiate injection of the sodium pentaborate solution upon initiation of severe accident procedures. The NRC staff considers the buffering action of sodium pentaborate an important factor in enhancing the suppression pool pH control.

3.1.1.3.1.6 Summary

The staff has reviewed the licensee's methodology for calculating the amounts of acidic and basic chemical species dissolved in the suppression pool after a LOCA. Release of iodine and cesium from the damaged core were based on the information reported in NUREG-1465 and

generation of hydrochloric and nitric acid on the data in NUREG/CR-5960 and NUREG-1801. The NRC staff finds that the methodology developed by the licensee was based to a large extent on the best information available in this area, and therefore is acceptable.

3.1.1.3.2 Determination of Suppression Pool pH

The licensee has determined the change in pH of the suppression pool during the 30-day transient for two cases. The licensee assumed a very conservative value of 5.3 as the initial pool pH. In the first case, no buffering action of sodium pentaborate was assumed, but credit was taken for cesium hydroxide. The results indicate that initially, during gap and early in-vessel release of cesium hydroxide, pH will rise. It will reach a maximum of 8.4 and then, due to continuous generation of nitric acid and hydrochloric acids, it starts decreasing, eventually falling to a pH of 3.0 at 30 days after a LOCA. In the second case, buffering action of sodium pentaborate was assumed, but no credit was taken for the presence of cesium hydroxide. In this case the value of pH reached a maximum of 8.3 and stayed above eight for the duration of the 30-day transient. Hand calculations were performed to verify the resulting pH values after 30 days. The staff's calculations demonstrated the suppression pool pH would remain above eight for at least 30 days, consistent with the licensee's submittal. This illustrates the necessity of using sodium pentaborate in controlling suppression pool pH after a LOCA.

After an accident, the pH of the suppression pool depends on the amounts of acidic and basic chemical materials either released from the damaged core or generated in containment and drywell and subsequently dissolved in the suppression pool's water. The pH of the suppression pool water without addition of sodium pentaborate was determined by relative amounts of the acidic and basic chemicals dissolved in the pool's water. Addition of the sodium pentaborate solution produces a buffering action that causes an increase in pH, while the introduction of acidic chemicals causes a decrease in pH. The licensee has demonstrated that without sodium pentaborate it was not possible to maintain pH above 7.0 for the duration of the 30-day transient. However, the injection of 4246 lb of sodium pentaborate solution will produce sufficiently strong buffering action to ensure that pH will stay above 7.0 for the duration of the 30-day transient.

The NRC staff has reviewed the licensee's methodology and performed independent evaluation of the licensee's calculations. Based on this review, the staff concludes that the licensee's methodology would allow the post-LOCA suppression pool pH to stay above 7.0 for 30 days after a LOCA. Therefore, the staff finds that the licensee's analysis is acceptable.

In addition, the staff has reviewed the SLC system with respect to the newly proposed SLC system role in delivery of sodium pentaborate to the suppression pool for pH control. The control of pH in the suppression pool is required to mitigate the consequences of a design basis accident in which fuel is damaged. As such, the new role being assigned to the SLC is a safety-related role. The licensee stated that the SLC pumps, valves and piping upstream of the explosive valves are classified as Safety Class 2. Safety Class 2 applies to those systems and components that are not Safety Class 1 but are necessary to accomplish specific safety functions as defined in UFSAR Section 3.2.3.2.1. The SLC system piping downstream of the explosive valves is classified as Safety Class 1, which applies to components of the reactor coolant pressure boundary or core support structure whose failure could cause a loss of reactor coolant at a rate in excess of the normal makeup system. The electrical and instrument

auxiliaries that are necessary to operate the SLC system are classified as Class 1E. The entire SLC system is Seismic Category I and conforms to the quality assurance requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

The staff reviewed the licensee's submittals and the responses in the license's letter dated December 17, 2004, to a staff request for additional information on the use of the SLC system for the safety-related function. From the licensee's statements, the staff has concluded that the SLC system as designed and installed at CPS is a high quality safety class system that provides reasonable assurance that the sodium pentaborate will be injected into the core upon activation. Specifically:

a.   The SLC system components required for reactivity control and new suppression pool pH control functions are seismically qualified.

b.   The SLC system is provided with emergency power, with the capability to supply power from the emergency diesel generators.

c.   The SLC system is subject to American Society of Mechanical Engineers (ASME) Section XI, inservice inspection requirements as required by 10 CFR 50.55a, "Codes and Standards."

d.   The SLC system is within the scope of the 10 CFR 50.65 Maintenance Rule.

e.   Most components (pumps, squib valves, etc.) are redundant in parallel trains powered from different electrical busses. The exceptions are the containment isolation check valves. These valves are discussed below in the discussion of the staff's single failure review.

f.   SLC injection is implemented under the direction of Existing Emergency Operating Procedures (EOPs) and Severe Accident Guideline (SAG) procedures and is performed by station off-normal procedure CPS 4411.10, "SLC Operations." The procedure directs the operator to inject the entire contents of the SLC storage tank whenever the reactor water level is below the top of active fuel and when operation is directed by the SAGs regardless of the initiating event. This will ensure that SLC injection will not be terminated during a LOCA.

g.   Licensed operators have received initial training on the EOPs and SAGs and will continue to receive periodic training.

The staff has considered system components that could be subject to single failure. The licensee stated that the only non-redundant active components in the SLC system are the check valves (two in series) located on the SLC injection line. The check valves are one 3-inch and one 4-inch ASME Section III, Class 1 Anchor Darling valves that are the containment isolation valves for the SLC injection line. In the periodic inspections and testing of these valves, the CPS SLC check valves have no documented history of failure related to the valve opening function. The only failure identified during a review of the system performance history

was a body to bonnet gasket leak that was identified and repaired during the refueling outage in 1992. A review of operating plant experience and the Equipment Performance and Information Exchange System was performed and no failure data were identified for check valves failing to open. The Nuclear Plant Reliability Data System was reviewed and no failures related to check valve opening issues were identified. Although acknowledging that a single failure to open of one of the two check valves could prevent SLC injection, the staff has determined that the potential for failure is very low based on the check valve quality, as established by its procurement, periodic testing and inspection, and historical performance of the component. The staff finds that the use of a single penetration of the containment with the identified check valves, as described by the licensee, is acceptable for the purposes of SLC injection to control suppression pool pH after a LOCA.

3.1.1.4 LOCA Fission Products Release Pathways

In the licensee's analysis of the LOCA, released fission products enter the environment through five pathways:

- Leakage of primary containment atmosphere into the secondary containment and subsequent release to the environment via the SGTS with some bypassing the secondary containment

- Leakage of primary containment atmosphere via design leakage through MSIVs

- Leakage from ECCSs that recirculate suppression pool water outside of the primary containment

- Leakage of primary containment atmosphere and suppression pool water via design leakage through reactor feedwater isolation valves (FWIVs).

- Leakage of primary containment atmosphere via design leakage through primary containment purge system penetrations.

3.1.1.4.1 LOCA - Containment Leakage Pathway

The primary containment is projected to leak at its TS design leakage rate (L_a) of 0.65 percent of its contents by weight per day for the first 24 hours and is reduced to 0.413 weight percent per day for the remainder of the 30-day accident duration. The licensee estimated the reduction to 63.6 percent of the primary containment design leakage rate at 24 hours by correlating the reduction in containment leakage rate to the square root of the percentage reduction in primary containment pressure.

AmerGen assumed that 92 percent of the leakage from the primary containment collects in the free volume of the secondary containment and is subsequently released to the environment via ventilation system exhaust. Following a LOCA, the SGTS fans start and draw down the secondary containment to create a negative pressure in relation to the environment. This pressure differential ensures that leakage from the primary containment is collected and processed by the SGTS. SGTS exhaust is processed through filter media prior to release to the environment via the SGTS vent. AmerGen assumes that the requisite negative pressure will not be achieved for 12 minutes after the start of the event, consistent with the proposed

increase in the surveillance requirement draw down time. During the draw down period, the primary containment leakage is assumed released unfiltered to the environment. The remainder of the leakage from the primary containment (8 percent of L_a for the first day, 5.09 percent thereafter) is assumed to be released unfiltered to the environment for the duration of the LOCA. The staff finds that licensee's assumptions for the containment leakage and containment bypass release pathways are in accordance with the guidance in RG 1.183, and are therefore acceptable.

3.1.1.4.2 LOCA - Main Steam Isolation Valve Leakage

The four main steam lines, which penetrate the primary containment, are automatically isolated by the MSIVs in the event of a LOCA. There are two MSIVs on each steam line, one inside the drywell (i.e., inboard) and one outside the primary containment, (i.e., outboard). The MSIVs are functionally part of the primary containment boundary and design leakage through these valves provides a leakage path for fission products to bypass the secondary containment and enter the environment as a ground level release.

AmerGen conservatively assumes that the fission products released from the core are dispersed equally throughout the drywell. Following the initial blowdown of the RPV, the fuel heats up, fuel melt begins, and steaming in the RPV carries fission products to the drywell. When core cooling is restored, steam is rapidly generated in the core. This steam and the ECCS flow carry fission products from the core to the primary containment, resulting in well-mixed RPV dome and primary containment fission product concentrations. Once the rapid steaming stops, the contents of the primary containment atmosphere may flow through any opening including the postulated broken main steam line and may be released through the MSIVs. It should be noted that for this release pathway only the break was assumed in the main steam line in lieu of the recirculation line which provided additional conservatism.

AmerGen assumes that the outboard MSIVs fail to close on all four main steam lines with one line broken upstream of the inboard MSIV. AmerGen assumes a maximum MSIV leakage of 100 scfh in the broken line, one of the unbroken lines is assumed to leak at 100 scfh, and the other two lines are assumed not to leak. These leakrates are consistent with the proposed increase in the surveillance requirement MSIV leakage criterion. The leak rates decrease to 63.6 percent of the above after 24 hours, based on the decrease in containment pressure, as discussed above for the containment leakage pathway.

The licensee's analysis does not take credit for the MSIV leakage control system but does propose to take credit for aerosol and iodine removal in the main steam lines. The AmerGen iodine removal modeling assumes well-mixed control volumes. Only the volumes associated with horizontal runs of seismically qualified main steam line piping are included in the modeling of iodine aerosol deposition. AmerGen assumes two aerosol settling volumes (nodes) for the unbroken main steam line; one node between the RPV and the inboard MSIV, and the other node between the inboard MSIV and the turbine/auxiliary building (secondary containment) wall. The main steam line conservatively assumed to be broken does not have the volume between the RPV and inboard MSIV available for iodine removal, so only assumes one aerosol settling node. The licensee's main steam line modeling is conservative because it minimizes aerosol deposition credit.

AmerGen's modeling of aerosol settling is based on the methodology used by the NRC staff in its review of the implementation of an AST at the Perry Nuclear Power Plant. The aerosol settling model is described in a report, AEB-98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application Using the Revised (NUREG-1465) Source Term," which was written by the NRC Office of Nuclear Regulatory Research. AEB-98-03 gives a distribution of aerosol settling velocities that are estimated to apply in the main steam line piping. The model used in the Perry assessment assumed aerosol settling may occur in the main steam lines upstream of the outboard MSIV, at the median settling velocity given by the Monte Carlo analysis described in the AEB-98-03 report. In the Perry assessment, aerosol settling is assumed to occur in one settling volume between the inboard MSIV and the outboard MSIV for the main steam line which has been assumed broken inside the drywell. For the remaining main steam lines aerosol settling is assumed to occur in two settling volumes; one between the reactor vessel and the inboard MSIV, the other volume between the two closed MSIVs.

AmerGen's modeling of aerosol settling in the MSIV leakage pathway for CPS is different from that for Perry in that piping downstream from the outboard MSIV to the auxiliary building/turbine building wall is credited, giving a larger volume and area of piping for settling. The NRC staff had requested information from the licensee regarding the conservatism in its use of the same settling velocity in the entire piping volume credited, including a third settling volume consisting of piping downstream of the outboard MSIV. The staff's concern was that the removal through aerosol settling was overestimated by modeling three settling volumes with the same settling velocity in each, when the settling would be expected to be at a lesser rate for the later sections of piping considering that the larger and heavier aerosols would have already settled out of main steam line atmosphere in upstream sections of piping. AmerGen responded by changing the model to assume only two settling volumes in an unbroken main steam line and one settling volume for the broken main steam line. The other two main steam lines are not assumed to leak, since the entire TS total allowable leakage of 200 scfh is assumed to occur through two lines at the maximum allowable rate of 100 scfh each. Additionally, AmerGen calculated a weighted average for the aerosol settling velocity from the AEB-98-03 distribution, converted that average settling velocity to an effective aerosol filtration efficiency for each main steam line, and applied the applicable effective filtration efficiency to the leakage rate out of each main steam line. The settling area was assumed to be the projected horizontal area in the horizontal sections of the qualified main steam piping. AmerGen did not take credit for aerosol settling after 24 hours, to address the change in the aerosol distribution over time.

Additional assumptions in AmerGen's modeling of the MSIV leakage pathway include; no credit for deposition beyond third MSIV, no credit for holdup or plate-out in the main condenser, one outboard MSIV fails to close on each line, 100 scfh leakage in each of the two shortest MSLs, the pressure between closed MSIVs is assumed to be equal to containment pressure and the temperature is assumed to be normal steam line conditions, and the pressure downstream of outboard MSIVs (and inboard MSIV on faulted line) is assumed to be atmospheric and steam is at normal operating temperature.

The staff acknowledges that aerosol settling is expected to occur in the main steam line piping but, because of recent concerns with aerosol sampling used in AEB-98-03 and lack of further information, does not know how much deposition (i.e., settling velocity value) is appropriate. The licensee has used a model based on the methodology of AEB-98-03, but has applied additional conservatism to address the staff's comments on the applicability of the AEB-98-03 methodology to CPS as modeled. The staff has performed a sensitivity analysis to determine

the effect of the potential overestimation of the aerosol removal, and has found that the total LOCA dose from all pathways would continue to be acceptable even when the AEB-98-03 10th percentile settling velocity is assumed for the feedwater line aerosol settling. The 10th percentile aerosol settling velocity is a smaller value (and estimates less aerosol settling) than 90 percent of the calculated settling velocities in AEB-98-03. Use of the 10th percentile settling velocity is more conservative than use of the median settling velocity as noted as reasonable in AEB-98-03. Given this information, the staff finds the CPS main steam line aerosol settling model to be reasonable and appropriate.

AmerGen also assumed deposition of elemental iodine in the main steam line piping. The licensee used the model described in a letter report dated March 26, 1991, by J. E. Cline, "MSIV Leakage Iodine Transport Analysis," which the staff has previously found acceptable as discussed in RG 1.183. The Cline report provides elemental iodine deposition velocities, resuspension rates and fixation rates. The deposition velocities were used in the well-mixed model formulation described above for use with AEB-98-03. Because elemental deposition is not gravity dependent, AmerGen assumed elemental iodine deposition occurs on the entire surface area of the horizontal and vertical piping. AmerGen evaluated the effects of resuspension, as described in the Cline report, and found the dose impacts to be small. Any resuspended iodine was modeled as organic iodine and assumed released instantly. No removal of organic iodine is credited.

Based on the above, the staff finds that AmerGen's modeling of the release by leakage through the MSIVs has been modeled conservatively and appropriately.

3.1.1.4.3 LOCA - Leakage from Emergency Core Cooling Systems

During the progression of a LOCA, some fission products released from the fuel will be carried to the suppression pool by spilling from the RCS and by natural processes such as deposition and plateout. Post-LOCA, the suppression pool is a source of water for the ECCSs. Since portions of these systems are located outside of the primary containment, leakage from these systems is evaluated as a potential radiation exposure pathway. For the purposes of assessing the consequences of leakage from the ECCS, AmerGen assumes that all of the radioiodines released from the fuel are instantaneously and homogeneously mixed in the suppression pool. Noble gases are assumed to remain in the containment atmosphere. Since aerosols and particulate radionuclides will not become airborne upon release from the ECCS, they are not included in the ECCS source term. This source term assumption is conservative for the LOCA analysis in that all of the radioiodine released from the fuel is assumed to be the source term in both the primary containment atmosphere leakage and the ECCS leakage. In a mechanistic treatment, the radioiodines in the primary containment atmosphere would relocate to the suppression pool occur over time.

The analysis considers the equivalent of 5 gpm unfiltered ECCS leakage starting at the onset of the LOCA, which is twice the maximum allowable leakage administrative limit for CPS. AmerGen assumes 10 percent of the iodine in the ECCS leakage becomes airborne and is available for release as 97 percent elemental and 3 percent organic iodine. No credit was assumed for hold-up and dilution in the secondary containment. As was assumed for the primary containment leakage pathway, the leakage enters the environment via the SGTS as a filtered elevated release, after the secondary containment draw down period. The release

continues for 30 days. The staff finds that licensee's assumptions for the ECCS leakage release are in accordance with the guidance in RG 1.183, and are therefore acceptable.

3.1.1.4.4 LOCA - Feedwater Isolation Valve Leakage Release Path

AmerGen assumes that feedwater penetrations provide a pathway for the fission products in the primary containment atmosphere to be released to the environment at an air leak rate equivalent to 2.0 gpm, total for two penetrations. AmerGen assumes that this leakage occurs for one hour, until the feedwater lines are filled with water. Since the source of this release is the primary containment atmosphere, this release is modeled as an additional secondary containment bypass pathway. However, AmerGen credits fission product deposition in the purge lines prior to release to the environment.

The licensee modeled aerosol and elemental iodine removal in the feedwater lines. The removal modeling was similar to that modeled for the MSIV leakage, although the AEB-98-03 median aerosol settling velocity was chosen because of the short time duration (1 hour). Only seismically qualified piping from the reactor vessel to the turbine/auxiliary building (secondary containment) wall was credited for iodine removal. AmerGen assumed two settling volumes, one node including the piping from the RPV to the inboard FWIV, the other node including the piping between the inboard FWIV and the secondary containment wall. The staff acknowledges that aerosol settling is expected to occur in the feedwater piping but because of recent concerns with aerosol sampling used in AEB-98-03 and lack of further information, does not know how much deposition (i.e., settling velocity value) is appropriate. The staff has performed a sensitivity analysis to determine the effect of the potential overestimation of the aerosol removal, and has found that the total LOCA dose from all pathways would continue to be acceptable even when the AEB-98-03 10th percentile settling velocity is assumed for the feedwater line aerosol settling. The 10th percentile aerosol settling velocity is a smaller value (and estimates less aerosol settling) than 90 percent of the calculated settling velocities in AEB-98-03. Given this information, the staff finds the CPS feedwater line aerosol settling model to be reasonable and appropriate.

AmerGen also assumed deposition of elemental iodine in the feedwater piping. The licensee used the model described in the Cline report dated March 26, 1991, which the staff has previously found acceptable as discussed in RG 1.183. The Cline report provides elemental iodine deposition velocities, resuspension rates and fixation rates. The deposition velocities were used in the well-mixed model formulation described above for use with AEB-98-03. Because elemental deposition is not gravity dependent, AmerGen assumed elemental iodine deposition occurs on the entire surface area of the horizontal and vertical piping. AmerGen evaluated the effects of resuspension, as described in the Cline report, and found the dose impacts to be small. Any resuspended iodine was modeled as organic iodine and assumed released instantly. No removal of organic iodine is credited. Given this information, the staff finds the CPS feedwater line elemental iodine deposition model to be reasonable and appropriate.

At one hour post-accident, it is assumed that the feedwater leakage control system mode of the residual heat removal (RHR) system fills the feedwater lines with water drawn from the suppression pool, thereby sealing the release pathway. The suppression pool becomes contaminated during a LOCA, therefore, FWIV leakage could carry contaminated water outside of the secondary containment. AmerGen assumes that the combined FWIV leakage is 2 gpm

for the remainder of the 30-day event. Since the source of this release is the suppression pool, this release is modeled as an additional ECCS leakage pathway. AmerGen assumes 10 percent of the iodine in the fluid leakage through the FWIVs becomes airborne and is available for release as 97 percent elemental and 3 percent organic iodine. The staff finds that licensee's assumptions for the FWIV leakage fluid release are in accordance with the guidance in RG 1.183, and are therefore acceptable.

3.1.1.4.5 LOCA - Primary Containment Purge Release Path

AmerGen assumes that the two primary containment purge line isolation leaks at a rate equivalent to 0.02 x L_a for each penetration for the first 24 hours. The leak rate decreases to 63.6 percent of the above after 24 hours, based on the decrease in containment pressure, as discussed above for the containment leakage pathway. This leakage provides a pathway for fission products in the primary containment to bypass the secondary containment and enter the environment. Since the source of this release is the primary containment atmosphere, this release is modeled as an additional secondary containment bypass pathway. However, AmerGen credits fission product deposition in the purge lines prior to release to the environment.

The aerosol and elemental iodine removal modeling was similar to that modeled for the MSIV leakage. Only seismically qualified piping from the reactor vessel to the turbine/auxiliary building (secondary containment) wall was credited for iodine removal. AmerGen assumed one settling volume node including the piping from the primary containment to the outboard isolation valve. The staff acknowledges that aerosol settling is expected to occur in the primary containment purge penetration piping but because of recent concerns with aerosol sampling used in AEB-98-03 and lack of further information, does not know how much deposition (i.e., settling velocity value) is appropriate. The licensee has used a model based on the methodology of AEB-98-03 expanded to draw leakage from a different source (primary containment vs. drywell), but has applied additional conservatism to address the staff's comments on the applicability of the AEB-98-03 methodology to CPS as modeled. AmerGen calculated a weighted average for the aerosol settling velocity from the AEB-98-03 distribution, converted that average settling velocity to an effective aerosol filtration efficiency for each primary containment purge penetration line, and applied the applicable effective filtration efficiency to the leakage rate out of each primary containment purge penetration line. The settling area was assumed to be the projected horizontal area in the horizontal sections of the qualified main steam piping. The staff has performed a sensitivity analysis to determine the effect of the potential overestimation of the aerosol removal, and has found that the total LOCA dose from all pathways would continue to be acceptable, even assuming that no credit is taken for aerosol settling in the primary containment purge penetration piping. Given this information, the staff finds the CPS primary containment purge penetration piping aerosol settling model to be reasonable and appropriate.

AmerGen also assumed deposition of elemental iodine in the primary containment purge penetration piping. The licensee used the model described in the Cline report dated March 26, 1991, which the staff has previously found acceptable as discussed in RG 1.183. The Cline report provides elemental iodine deposition velocities, resuspension rates and fixation rates. The deposition velocities were used in the well-mixed model formulation described above for use with AEB-98-03. Because elemental deposition is not gravity dependent, AmerGen assumed elemental iodine deposition occurs on the entire surface area of the

horizontal and vertical piping. AmerGen evaluated the effects of resuspension, as described in the Cline report, and found the dose impacts to be small. Any resuspended iodine was modeled as organic iodine and assumed released instantly. No removal of organic iodine is credited. Given this information, the staff finds the CPS primary containment purge penetration line elemental iodine deposition model to be reasonable and appropriate.

3.1.1.5 LOCA Doses

AmerGen evaluated the maximum 2-hour TEDE to an individual located at the exclusion area boundary (EAB) and the 30-day TEDE to an individual at the outer boundary of the low population zone (LPZ). The LOCA doses calculated for each of the pathways discussed above were added together for the total LOCA dose estimate. The licensee's resulting offsite doses are less than the 10 CFR 50.67 criteria.

AmerGen evaluated the dose to the operators in the control room. The control room at CPS can be characterized as zone isolation with filtered makeup and filtered recirculation. There are two separate outside air intakes separated by about 375 feet east and west of the plant. During normal operation, the unfiltered makeup flow is 3300 cfm. An alarm on one or both of the radiation monitors in the outside air intakes automatically causes the normal makeup flow paths to isolate, one of the two filtered makeup fans to start, and the recirculation flow is routed through the recirculation charcoal filters. Although the control room isolation is automatic, AmerGen assumes a 20-minute delay in control room pressurization. This time period and the assumed 1650 cfm unfiltered intake during this period are based on the results of historical design basis accidents. Following this initial 20-minute period, AmerGen assumes that the filtered makeup flow is 2700 cfm, the filtered recirculation flow is 54,900 cfm and the filtered inleakage is 650 cfm. Although the control room is designed to be pressurized after this initial 20-minute period, AmerGen assumes that unfiltered inleakage into the control room is 144 cfm with the maximum allowable filtered inleakage of 650 cfm. AmerGen used filter efficiencies consistent with the current DBA analyses, and accounted for the TS surveillance filter system bypass criteria for the control room ventilation systems.

The licensee calculated the total control room intake filter loadings for the AST and determined that the loadings are less than the filter loadings calculated previously based on the LOCA source term from Technical Information Document TID 14844, "Calculation of Distance Factors for Power and Test Reactor Sites." The licensee has determined that therefore the gamma shine dose from sources outside the control room for the AST would be bounded by the current analysis gamma shine dose calculated using the TID14844 source term. The licensee added the conservative TID14844-based gamma shine dose value to the control room dose calculated for the release pathways discussed above. The staff finds this formulation to be conservative and acceptable.

The staff finds that the licensee's LOCA analysis assumptions and methodology are consistent with the guidance of RG 1.183. In addition, the staff has performed independent calculations and confirmed the licensee's results. Therefore, the staff finds that the analysis assumptions presented in Table 1, and the EAB, LPZ, and control room doses estimated by AmerGen for the LOCA listed in Table 2, are acceptable.

3.1.2   Main Steam Line Break

The accident considered is the complete severance of a main steam line outside the primary containment. No fuel damage is projected to occur. The MSIVs are expected to isolate the leak within 5.5 seconds. However, AmerGen assumed an instantaneous ground level release with no holdup or mixing in the turbine building. AmerGen assumes that the fission product activity in the release cloud is based on a bounding total mass of primary coolant released from the break.

The MSLB analysis is performed for two activity release cases, based on the maximum equilibrium and pre-accident iodine spike concentrations of 0.2 μCi/gm and 4.0 μCi/gm dose equivalent I-131, respectively. AmerGen proposed redefining the term "dose equivalent I-131" to be based in the inhalation CEDE rather than the thyroid dose previously specified. AmerGen stated that this change was necessary since their proposed full implementation of the AST would replace the previous whole body and thyroid dose guidelines with TEDE criteria. The staff finds this proposed change acceptable since CEDE is encompassed within the definition of TEDE and is generally the more limiting constituent of TEDE.

AmerGen also included cesium activity, as cesium iodide, in the release. RG 1.183, Appendix D, Regulatory Position 4.4 states that for a boiling water reactor (BWR) MSLB 95 percent of the iodine species released from the main steam line should be assumed to be cesium iodide. Therefore, the licensee assumed that one atom of cesium accompanies each of the iodine atoms for 95 percent of the total iodine release. The licensee determined the cesium isotopic abundance (ratio of each isotope to the total cesium release) for longer lived or stable isotopes, such as Cs-133, Cs-134, Cs-135, and Cs-137, based on source terms developed for pH control at BWRs. The isotopic abundance for shorter lived isotopes such as Cs-136 and Cs-138 was based on the source term in the American National Standards Institute/American Nuclear Society standard ANSI/ANS-18.1-1999, "Radioactive Source Term for Normal Operation of Light Water Reactors." The staff finds this modeling of the cesium isotopic abundance and release to be reasonable.

For the MSLB, AmerGen did not use the atmospheric dispersion factors listed in Table 1 for the offsite dose analysis, but instead used the release dispersion assumptions given in RG 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors." The licensee used RG 1.5 and the CPS specific distances of 975 meters to the EAB and 4018 meters to the LPZ to result in EAB and LPZ χ/Q values of 3.68E-4 and 1.02E-4 sec/m^3, respectively. The licensee assumed that the released steam and water expands to a hemispheric volume at atmospheric pressure and temperature. This cloud is assumed to move at a speed of one meter per second downwind past the control room intake. In the control room dose calculation, AmerGen credited only the dilution provided by the thermo-hydraulic expansion. No additional credit was taken for the meteorological dispersion, buoyant rise, or radioactive decay. In following this approach, the steam expansion establishes the concentration of fission products in the cloud and the duration of the expanded concentration which is present at the control room intake. AmerGen did not credit operation of the control room emergency filtration systems.

The staff finds that the licensee's MSLB analysis assumptions and methodology are consistent with the guidance of RG 1.183. In addition, the staff has performed independent calculations and confirmed the licensee's results. Therefore, the staff finds that the assumptions presented

in Table 1, and the EAB, LPZ, and control room doses estimated by AmerGen for the MSLB listed in Table 2, are acceptable.

3.1.3 Control Rod Drop Accident

The CRDA analysis postulates a sequence of mechanical failures that results in the rapid removal (i.e., drop) of a control rod, upon which a reactor trip will occur. Localized damage to fuel cladding is expected to occur, resulting in a breach of the fuel cladding. The temperature of a small fraction of the fuel in the breached rods will be sufficient to cause localized melting.

The fuel rod fission product inventory is based on long term reactor operation at 102 percent of the rated thermal power. It is assumed that 10 percent of the inventory of noble gases and 10 percent of the inventory of iodine are located in the fuel rod gap. Since the power level is not equal across the core, AmerGen multiplies the fuel rod fission product inventory by a radial peaking factor of 1.7 to conservatively maximize the fission product release. AmerGen has projected 1200 fuel rods would be breached by the event and, 1.0 percent of these damaged rods would exceed the threshold for melting. AmerGen assumes that all of the gap activity in the breached fuel rods is released to the reactor vessel. AmerGen assumes that 100 percent of the noble gases and 50 percent of the iodine in the melted fuel are released to the reactor vessel.

The analysis assumes that the fission products released from the damaged fuel are instantaneously transported to the main condenser. It is assumed that 100 percent of the noble gases and 10 percent of the iodine released reach the main condenser due to plateout in the RPV and main steam lines. Of the iodine that enters the main condenser, 90 percent plates out in the main condenser. The MSIVs are assumed to remain open for the duration of the event. These assumptions are consistent with the guidance of RG 1.183.

AmerGen considers two scenarios for the release from the main condenser. In the first, the unfiltered release is from the turbine building via the heating, ventilation and air conditioning (HVAC) vent stack at a rate of 1 percent of the condenser free volume per day. In the second scenario, the condenser is evacuated by the steam jet air ejectors (SJAE) with the release directed through the large charcoal delay beds of the off-gas system. AmerGen did not consider a release from the mechanical vacuum pumps since these pumps are automatically isolated by the main steam line radiation monitoring system.

AmerGen did not credit intake mitigation by the control room habitability systems. For the first case, the maximum unfiltered flow rate of 3300 cfm plus an unfiltered inleakage of 2650 cfm was assumed. For the second case, an effectively infinite unfiltered inleakage was assumed.

The staff finds that the licensee's CRDA analysis assumptions and methodology are consistent with the guidance of RG 1.183. In addition, the staff has performed independent calculations and confirmed the AmerGen results. Therefore, the staff finds that the assumptions presented in Table 1, and the EAB, LPZ, and control room doses estimated by AmerGen for the CRDA listed in Table 2, are acceptable.

3.2 Atmospheric Dispersion

AmerGen recalculated the atmospheric dispersion factors (χ/Q values) for releases from the SGTS/HVAC vent stack for each of the three control room intakes and the EAB and LPZ. AmerGen used hourly meteorological observations obtained from the site meteorological tower, collected for the years 2000, 2001, and 2002. AmerGen used the NRC-sponsored ARCON96 computer code to calculate control room χ/Q values and used the NRC-sponsored PAVAN computer code to calculate the EAB and LPZ χ/Q values. The resulting atmospheric dispersion factors represent a change from those values used in the current CPS UFSAR analyses. To aid in the staff's review, the licensee has provided the hourly data in the ARCON96 format, and also submitted the input data files for PAVAN and ARCON96.

For CPS, the SGTS/HVAC stack is the only accident release point and does not meet the definition of an elevated release point per RG 1.194 and RG 1.145 (i.e., the SGTS/HVAC stack is less than 2½ times the height of adjacent structures). For control room χ/Q values, the licensee modeled the SGTS/HVAC stack as a point source using the ARCON96 vent release option with zero vertical exit velocity and zero stack flow. Wind speed and wind direction data from the site's 33-foot (10-meter) and 197-foot (60-meter) meteorological tower levels were provided as input to ARCON96 and stability class was calculated using temperature difference measurements between the 197-foot and 33-foot levels. For EAB and LPZ χ/Q values, the licensee provided a joint frequency distribution derived from 33-foot wind data as input to PAVAN.

The staff has performed a quality review of the licensee's provided onsite hourly meteorological data using the methodology described in NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data." Further review was performed using computer spreadsheets. The staff has considered the wind speed, wind direction, and stability class frequency distributions for each year alone and also in comparison to the other years. The wind speed and stability class frequency distributions showed reasonable consistency from year to year and to the CPS UFSAR distributions. The wind direction data for 2000 and 2001 appear to have a higher degree of consistency as compared to the 2002 wind direction data. However, when the wind direction data from all three years are compared on a single graph, the 2003 data appear reasonable.

The CPS UFSAR states that the meteorological measurement program is designed to comply with the recommendations of RG 1.23, "Onsite Meteorological Programs." The licensee's data recovery rate was 87.1 percent for 2000, 92.8 percent for 2001, and 93.3 percent for 2002. Although the data recovery for 2000 is less than the RG 1.23 goal of 90 percent, the staff concludes that the quality of the data is reasonable and appropriate, based on the consistency of the data for each year as compared to the earlier collected UFSAR data.

The control room and offsite χ/Q values calculated by AmerGen are given in Table 1. Because the control room operators can manually select either the west or east control room emergency ventilation intake, the licensee used the most favorable intake value divided by four for the time period after the intake selection. This reduction in the control room χ/Q values is in accordance with SRP Section 6.4 and RG 1.194 guidance on appropriate estimation of atmospheric dispersion for control room habitability analyses and is therefore acceptable. The licensee modeled unfiltered inleakage into the control using the control room normal intake χ/Q values with no factor of four dual-intake-based reduction.

In summary, the staff has reviewed the available information relative to the onsite meteorological measurements program, the hourly 2000 through 2002 data, and the ARCON96 and PAVAN meteorological data input files provided by the licensee. On the basis of this review, staff concludes that these data provide an acceptable basis for making estimates of atmospheric dispersion estimates for design basis accident dose assessments.

3.3   TS Changes

3.3.1   TS Section 1.1, "Definitions"

AmerGen proposed to revise the definition of dose-equivalent I-131 in TS Section 1.1. The change replaces the thyroid dose quantity with CEDE from inhalation. This proposed change is consistent with the TEDE basis of the radiological consequence analyses and provides an improved correlation between the TS specific activity LCO (e.g., where this definition is used) and the projected offsite and control room doses. The staff finds this proposed change acceptable.

3.3.2   TS 3.1.7, "Standby Liquid Control (SLC) System"

The specific changes proposed to TS 3.1.7 are the extension of applicability to Mode 3 and an additional required action and completion time for Action C to be in Mode 4 in 36 hours. The extension of applicability to Mode 3 ensures the capability of injecting SLC during hot standby.

These changes implement the AST methodology regarding the use of SLC to buffer the suppression pool following a LOCA involving fuel damage. The licensee has proposed using the SLC system to inject pH control agent as a means of minimizing iodine re-evolution from the suppression pool during LOCAs. This is a new design function for this system. The proposed change adds a LOCA pH control function requirement which is not needed for the anticipated transients without scram function of the SLC. Additionally, this addition does not affect the shutdown requirements based on the SLC system availability. Increasing the LCO action and response time is appropriate for this post accident pH control action. On the basis of the staff's review of the use of the SLC system for post-accident suppression pool pH control as discussed in Section 3.1.1.3 of this SE, and because the shutdown requirements continue to meet GDCs 26 and 29, the staff finds the proposed changes to TS 3.1.7 acceptable.

3.3.3   TS 3.3.6.1, "Primary Containment and Drywell Isolation Instrumentation"

The proposed change would revise TS 3.3.6.1-1 to extend from Modes 1 and 2 to Modes 1, 2, and 3, the SLC system initiation function to isolate the reactor water cleanup system. The licensee has proposed using the SLC system to inject pH control agent as a means of minimizing iodine re-evolution from the suppression pool during LOCAs. This is a new design function for this system. The staff finds that the proposed change is consistent with the application of the SLC system in its safety-related pH control mode, and is therefore acceptable.

3.3.4   TS 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)"

The licensee proposed to increase the allowed leakage for the primary containment purge valves in the bases for Surveillance Requirement (SR) 3.6.1.3.5 from 1 percent to 2 percent of the maximum allowable primary containment leakage rate, L_a.

Release from these valves has historically been treated as part of the secondary containment bypass allowance of $0.08 \times L_a$. With the AST dose analyses, the licensee has now analyzed the dose from leakage through the primary containment purge valves separately from the secondary containment bypass leakage. The staff finds that the licensee's dose analyses assumptions properly use the proposed TS allowable leakage rates for the primary containment purge valves. The analysis results show that CPS meets the offsite and control room regulatory dose criteria for operation with the proposed TS surveillance changes. Therefore, the staff finds the proposed change to SR 3.6.1.3.5 is acceptable.

The licensee proposed to change the surveillance in SR 3.6.1.3.9 from:

> "Verify total leakage rate through all four main steam lines is $\leq$ 112 scfh when tested at $\geq P_a$."

to the following:

> "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 $\leq$ 200 scfh when tested at $\geq P_a$."

The licensee states that the change will allow an increased allowable MSIV leakage. This increased allowable leakage criterion is desirable because unplanned MSIV repairs are a significant contribution to increased outage duration and unplanned exposure during refueling outages. In the original submittal the licensee had asked for a combined leakage for all MSIV leak paths $\leq$ 250 scfh. In the August 19, 2005, letter, the value was changed to $\leq$ 200 scfh to be consistent with the assumptions used in the design basis analysis. The staff has reviewed the information in the licensee's submittal, as supplemented, and finds that the $\leq$ 200 scfh value is consistent with the design basis analysis and is acceptable as the appropriate surveillance requirement in SR 3.6.1.3.9.

The licensee also has proposed to decrease the combined allowable leakage rate for both feedwater containment penetrations specified in SR 3.6.1.3.11 from less than or equal to 3 gpm to less than or equal to 2 gpm when pressurized to greater or equal to 1.1 P_a. The licensee

proposes to revise the TS and associated bases to decrease the feedwater isolation valve leakage from 3 gpm to 2 gpm, in order to allow analysis margin to be used for other LOCA release path calculations. The proposed change is conservative in that a lesser amount of FWIV leakage is allowed. The use of the margin for other release paths has been reviewed by the staff and has been determined to meet the requirements of design basis analyses. As such, the staff finds the proposed change to SR 3.6.1.3.11 is acceptable.

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

The licensee proposed to delete this TS in its entirety because operability of the system is not assumed in any DBA analysis. Therefore, this system does not meet any of the 4 criteria in 10 CFR 50.36 that determines which system requires a TS LCO. The current licensing basis for CPS requires the two MSIV LCS sub systems to be operable during Modes 1, 2, and 3, to satisfy 10 CFR 50.36(c)(2)(ii)(c). The proposed change deletes this TS section since under the AST, the MSIV LCS would no longer be credited for accident mitigation. The associated TS bases would also be deleted, consistent with this change. The staff has reviewed the licensee's submittal and confirmed that the MSIV LCS was not credited in the AST dose analyses to reduce the dose consequences of the LOCA, which is the most limiting design basis accident for the MSIV LCS. The results of the licensee's analysis indicate that the calculated doses remain within the regulatory limits without taking credit for the MSIV LCS. Further, the licensee did not take any credit for hold-up and plate-out downstream of the MSIVs or in the main condenser since these components have not been evaluated for seismic ruggedness. Therefore, the staff finds that the licensee's proposed change to delete TS Section 3.6.1.8 and its associated bases is acceptable.

3.3.6   TS 3.6.4.1, "Secondary Containment"

The licensee proposed to increase the allowed secondary containment draw down time from 188 seconds to 12 minutes in the Bases of SR 3.6.4.1.4. The staff has reviewed the licensee's submittal and confirmed that the drawdown time of 12 minutes is appropriately modeled in the AST dose analyses. The results of the licensee's analysis indicate that the calculated doses remain within the regulatory limits, assuming the longer drawdown time. Therefore, the staff finds that the licensee's proposed change to the Bases of SR 3.6.4.1.4 is acceptable.

3.3.7   TS 5.5.7 c. "Ventilation Filter Test Program"

In the August 19, 2005, letter, the licensee withdrew its request to change engineered safety feature ventilation system penetration criteria. There is no change to this TS. The current licensing basis is to be maintained.

3.4   Environmental Qualification

Although TID-14844 initially was used only for siting evaluations, the TID source term has been used in other design basis applications, such as environmental qualification (EQ) of equipment under 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants." Also, the TID is referenced in 10 CFR Part 100.11 as a source of further guidance on the equipment qualification analyses.

Regulatory Guide 1.183 states that licensees may use either the AST or the TID 14844 assumptions for performing the required EQ analyses. AmerGen has elected to retain the TID assumptions for performing the required EQ analyses.

Based on the above, the staff finds that the licensee's position to retain the TID 14844 assumptions for performing the required EQ analyses is acceptable.

3.5   Summary

As described above, the NRC staff has reviewed the assumptions, inputs, and methods used by AmerGen to assess the radiological impacts of the application of an AST methodology (with the exception that TID 14844 was used as the radiation dose basis for equipment qualification at the Clinton facility), and the changes to the TS requirements for the SLC system, primary containment isolation valves, MSIV leakage control system, and secondary containment at Clinton. The staff finds that AmerGen used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.0 above. The staff has compared the doses estimated by AmerGen to the applicable criteria identified in Section 2.0. The NRC staff finds that there is reasonable assurance that the licensee's estimates of the EAB, LPZ, and control room doses will continue to comply with these criteria. Therefore, the proposed TS changes are acceptable with regard to the radiological consequences of postulated design basis accidents.

This licensing action is considered to be an application of an AST methodology (with the exception that TID 14844 was used as the radiation dose basis for equipment qualification). With this approval, the previous accident source term in the CPS design basis is superceded by the AST proposed by AmerGen. The previous offsite and control room accident dose criteria expressed in terms of whole body, thyroid and skin doses are superceded by the TEDE criteria of 10 CFR 50.67 or fractions thereof, as defined in RG 1.183. All future radiological analyses performed to demonstrate compliance with regulatory requirements shall address all characteristics of the AST and the TEDE criteria as described in the CPS design basis.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes a surveillance requirement. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (68 FR 52234). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 <u>CONCLUSION</u>

The staff has concluded, based on the considerations discussed above, that: (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.

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Attachments: 1. Table 1
2. Table 2

Date: September 19, 2006

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Table 1

CPS Accident Analysis Parameters

General
</div>

Reactor power (3473 x 1.02), MWt	3543
Core Inventory	RADTRAD-Generated
Dose conversion factors	FGR11/FGR12
Breathing rate, offsite, m³/s	
0-8 hours	3.47E-4
8-24 hours	1.75E-4
>24 hours	2.32E-4
Breathing rate, control room, m³/s	3.47E-4
Control room normal intake flow, cfm	3000 ± 10%
Control room unfiltered infiltration, cfm	
First 20 minutes	1650
Remainder	144
Control room filtered infiltration, cfm	650
Control room filtered pressurization, cfm	2700
Control room filtered recirculation, cfm	54,900
Control room volume, ft³	324,000
Control room charcoal filter efficiency, %	99
Control room recirculation system	
Charcoal filter efficiency, %	70
Filter bypass, %	2
Control room occupancy factor	
0-24 hrs	1.0
1-4 days	0.6
4-30 days	0.4
SGTS Filter Efficiency, %	99

Offsite χ/Q values, sec/m³

Time Period	EAB	LPZ
0 - 2 hours	2.46E-4	5.62E-5
0 - 8 hours	--	2.48E-5
8 - 24 hours	--	1.65E-5
24 - 96 hours	--	6.81E-6
96 - 720 hours	--	1.91E-6

Control Room χ/Q values, sec/m³

Time Period	West Intake	East Intake	Normal Intake
0 - 2 hours	9.45E-4	9.75E-4	1.54E-3
2 - 8 hours	7.58E-4	7.09E-4	1.09E-3
8 - 24 hours	3.28E-4	2.93E-4	4.67E-4
24 - 96 hours	2.61E-4	2.13E-4	3.21E-4
96 - 720 hours	1.85E-5	1.79E-4	2.64E-4

<div align="right">ATTACHMENTS</div>

Loss of Coolant Accident (LOCA)

Containment Leakage

Onset of gap release phase, min	2.0
Core release fractions and timing	RG 1.183, Table 1
Iodine species distribution	
Elemental	0.95
Organic	0.0485
Particulate	0.0015
Primary containment volume, ft^3	
Drywell	241,699
Containment air space	1,512,341
Primary containment leakrate, %/day	
0- 24 hours (L_a)	0.65
Greater than 24 hours ($0.636*L_a$)	0.413
Secondary containment bypass	
0- 24 hours	$0.08*L_a$
Greater than 24 hours	$0.0509*L_a$
SGTS drawdown time, min	12
(10 minutes from start of gap release)	
Drywell natural deposition	
Particulate	Powers model, 10th-percentile
Elemental	None
Control room isolation delay, minutes	20

MSIV Leakage

Activity same as containment leakage case

MSIV TS leak rate at 9 psig, scfh*	
One line	100
Total	200
*reduced to 63.6% of value after 24 hours	

Aerosol and elemental iodine removal in piping (see text)	Credited
Weighted average of AEB-98-03 settling velocity used for aerosol deposition	
Cline model used for elemental iodine removal	
Release via Turbine Building HVAC Exhaust	

ECCS Leakage

Iodine species fraction	
Particulate/aerosol	0
Elemental	97
Organic	3
Suppression pool liquid volume, ft^3	146,400
Estimated total leakage, gpm	5
Flashing fraction for iodine	0.1
SGTS charcoal filtration start delay, min	10

Feedwater Isolation Valve Leakage (total for two penetrations)

Containment atmosphere leakage (0 to 1 hour), cfm	10.98
Aerosol and elemental iodine removal in piping (see text)	Credited
ECCS water leakage (1 hour to 30 days), gpm	2.0
Flashing fraction for iodine	0.1

Primary Containment Purge Penetrations Leakage

Total leakage from primary containment	
0 - 24 hours	$0.02*L_a$
Greater than 24 hours	$0.01272*L_a$
Aerosol and elemental iodine removal in piping (see text)	Credited

Control Rod Drop Accident (CRDA)

Core radial peaking factor	1.7
Fraction of core inventory in fuel rod gap	
Noble gases	0.1
Iodine	0.1
Failed rods	1200
Fraction of failed rods that reach melt	0.01
Melted fuel release fraction to vessel	
Noble gases	1.0
Iodine	0.5
Fraction of activity released to vessel that enters main condenser	
Noble gases	1.0
Iodine	0.1
Remaining nuclides	0.01
Fraction of activity released from main condenser	
Noble gases	1.0
Iodine	0.1
Remaining nuclides	0.01
Control room isolation	Not credited

Case 1 - Release from main condenser to HVAC vent stack

Main condenser (plus LP turbine) free volume, ft^3	175,000
Release rate from main condenser, %/day	1.0
Release duration, hours	24

Case 2 - Release from SJAE to Off-gas

Main condenser (plus LP turbine) free volume, ft^3	175,000
Release rate from main condenser, %/day	1.0
Release duration, hours	24
Off-gas charcoal delay bed, hours	
Xenons	681.6
Kryptons	31.8

Main Steam Line Break

Reactor coolant activity, µCi/gm CEDE equivalent I-131

Normal	0.2
Spike	4.0
Mass release, lbm	140,000
Release flash fraction	0.40

(All activity in water is assumed to be released)

Release duration (instantaneous puff), sec	5.5
Iodine species	100% elemental
Control room isolation	Not credited

Table 2

CPS Licensee Calculated Doses

TEDE (rem)

Event	0-2 hr EAB	30-day LPZ	30-day Control Room
Loss-of-Coolant Accident	17.11	7.28	4.70
RG 1.183 Dose Criterion	*25*	*25*	*5*
Main Steam Line Break			
Equilibrium Activity	2.47E-2	6.88E-3	8.31E-2
RG 1.183 Dose Criterion	*2.5*	*2.5*	*5*
Pre-incident Iodine Spike	0.490	0.136	1.66
RG 1.183 Dose Criterion	*25*	*25*	*5*
Control Rod Drop Accident*			
1: Condenser Leakage	0.041	0.016	0.43
2: SJAE Release	0.64	0.15	0.25
RG 1.183 Dose Criterion	*6.3*	*6.3*	*5*

* Two separate cases