

RS-04-174

10 CFR 50.90

December 17, 2004

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: Additional Information Supporting the Request for License Amendment Related to Application of Alternative Source Term

- References:
- (1) Letter from Michael J. Pacilio (AmerGen Energy Company, LLC) to U. S. NRC, "Request for License Amendment Related to Application of Alternative Source Term," dated April 3, 2003
  - (2) Letter from Keith R. Jury (AmerGen Energy Company, LLC) to U. S. NRC, "Additional Information Supporting the Request for License Amendment Related to Application of the Alternative Source Term," dated December 23, 2003

In Reference 1, AmerGen Energy Company (AmerGen), LLC requested an amendment to the facility operating license for Clinton Power Station (CPS), Unit 1. 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.

The NRC, in support of their review of the referenced amendment request, has requested additional information related to crediting the standby liquid control system for pH control of the suppression pool. A request for additional information concerning filter test criteria was also provided.

As documented in Reference 2, AmerGen is currently revising the associated LOCA analysis to address piping deposition and flashing fraction issues raised in other requests for information from the NRC. This reanalysis will also address the assumptions for filter efficiency and therefore needs to be considered in responding to the filter test criteria request referred to above. As a result, the attachment to this letter contains the requested information in support of the NRC review of the AmerGen credit taken for standby liquid control system in suppression


pool pH control. The request for additional information on the filter test criteria will be provided later as part of the response to the requests documented in Reference 2.

AmerGen has reviewed the information supporting a finding of no significant hazards consideration that was previously provided to the NRC in the referenced letter. The supplemental information provided in this submittal does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration.

If you have any questions concerning this letter, please contact Mr. Timothy A. Byam at (630) 657-2804.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 17<sup>th</sup> day of December 2004.

Respectfully,

A handwritten signature in black ink, reading "Keith R. Jury". The signature is written in a cursive, flowing style.

Keith R. Jury  
Director – Licensing and Regulatory Affairs  
AmerGen Energy Company, LLC

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cc: Regional Administrator – NRC Region III  
NRC Senior Resident Inspector – Clinton Power Station  
Illinois Emergency Management Agency – Division of Nuclear Safety

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*Clinton has proposed to credit control of the pH in the suppression pool following a LOCA by means of injecting sodium pentaborate into the reactor core with the standby liquid control (SLC) system. The SLC system design was not previously reviewed for this safety function (pH control post-LOCA). Licensees proposing such credit need to demonstrate that the SLC system is capable of performing the pH control safety function assumed in the AST LOCA dose analysis. The following questions are from a set of generic questions developed by the staff and which are being provided to all BWR licensees with pending AST license amendment requests. In responding to questions regarding the SLC system, please focus on the proposed pH control safety function. The reactivity control safety function is not in question. For example, the SLC system may be redundant with regard to the reactivity control safety function, but lacks redundancy for the proposed pH control safety function. If you believe that the information was previously submitted to support the license amendment request to implement AST, you may refer to where that information may be found in the documentation.*

#### **Request 1**

*Please identify whether the SLC system is classified as a safety-related system as defined in 10 CFR 50.2, and whether the system satisfies the regulatory requirements for such systems. If the SLC system is not classified as safety-related, please provide the information requested in Items 1.1 to 1.5 below to show that the SLC system is comparable to a system classified as safety-related. If any item is answered in the negative, please explain why the SLC system should be found acceptable for pH control agent injection.*

- 1.1 Is the SLC system provided with standby AC power supplemented by the emergency diesel generators?*
- 1.2 Is the SLC system seismically qualified in accordance with Regulatory Guide 1.29 and Appendix A to 10 CFR Part 100 (or equivalent used for original licensing)?*
- 1.3 Is the SLC system incorporated into the plant's ASME Code ISI and IST programs based upon the plant's code of record (10 CFR 50.55a)?*
- 1.4 Is the SLC system incorporated into the plant's Maintenance Rule program consistent with 10 CFR 50.65?*
- 1.5 Does the SLC system meet 10 CFR 50.49 and Appendix A to 10 CFR 50 (GDC-4, or equivalent used for original licensing)?*

#### **Response 1**

The classification of components in the Standby Liquid Control (SLC) system is summarized in Clinton Power Station (CPS) Updated Safety Analysis Report (USAR) Table 3.2-1, Item V. This table indicates the pumps, valves and piping upstream of the explosive valves, are classified as Safety Class 2. Safety Class 2 applies to those

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systems and components that are not Safety Class 1 but are necessary to accomplish specific safety functions as defined in USAR Section 3.2.3.2.1. The 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." In addition, the SLC system at CPS meets the following items.

- 1.1 The SLC system at CPS is powered by standby AC power. This includes the capability to supply power from the emergency diesel generators.
- 1.2 The SLC system components required for the alternative source term (AST) function are seismically designed and qualified in accordance with the requirements of Regulatory Guide (RG) 1.29, "Seismic Design Classification," and 10 CFR 100, "Reactor Site Criteria," Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants."
- 1.3 The SLC system is incorporated into the CPS American Society of Mechanical Engineers (ASME) inservice inspection (ISI) and inservice testing (IST) programs as required by 10 CFR 50.55a, "Codes and standards."
- 1.4 The SLC system is incorporated into the Maintenance Rule Program at CPS consistent with 10CFR50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants."
- 1.5 The SLC system has been evaluated against 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants." As stated in CPS USAR Section 9.3.5.3.1, the SLC system is designed for the expected environment in the containment and specifically for the compartment in which it is located as required in General Design Criterion (GDC) 4, "Environmental and dynamic effects design bases." In this compartment, the SLC system is not subject to the more dynamic conditions postulated in this criterion such as missiles, whipping pipes, and discharging fluids.

As described above, all electrical and instrument auxiliaries required to operate the SLC system are classified as Class 1E. In addition, all associated cables are Class 1E. USAR Section 3.2.3.4.3 defines Class 1E as the safety classification of electrical equipment and systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal, or otherwise are essential in preventing significant release of radioactive material to the environment. The CPS Equipment Qualification program evaluates all Class 1E electrical equipment that is located in a harsh environment, as defined in 10 CFR 50.49, "Environmental qualification of electric

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equipment important to safety for nuclear power plants,” for the environment that they are exposed to. This includes the Class 1E equipment that is part of the SLC system.

#### **Request 2**

*Please describe proposed changes to plant procedures that implement SLC sodium pentaborate injection as a pH control additive. In addition, please address Items 2.1 to 2.5 below in your response. If any item is answered in the negative, please explain why the SLC system should be found acceptable for pH control additive injection.*

- 2.1 *Are the SLC injection steps part of a safety-related plant procedure?*
- 2.2 *Are the entry conditions for the SLC injection procedure steps symptoms of imminent or actual core damage?*
- 2.3 *Does the instrumentation cited in the procedure entry conditions meet the quality requirements for a Type E variable as defined in RG 1.97 Tables 1 and 2?*
- 2.4 *Have plant personnel received initial and periodic refresher training in the SLC injection procedure?*
- 2.5 *Have other plant procedures (e.g., ERGs/SAGs) that call for termination of SLC as a reactivity control measure been appropriately revised to prevent blocking of SLC injection as pH control measure. (For example, the override before Step RC/Q-1, “If while executing the following steps:....It has been determined that the reactor will remain shutdown under all conditions without boron, terminate boron injection and...”)?*

#### **Response 2**

SLC injection is implemented under the direction of the Emergency Operating Procedure (EOP)/ Severe Accident Guideline (SAG) procedures and is performed by station off-normal procedure CPS 4411.10, “SLC Operations.” SLC injection is required under the following three conditions.

1. As an alternate injection source (i.e., to provide makeup volume to the reactor vessel) when Top of Active Fuel (TAF) cannot be maintained. This action is directed in EOP-1, “RPV Level Control,” EOP-1A, “ATWS RPV Level Control,” and EOP-2, “RPV Flooding.”
2. During an Anticipated Transient Without Scram (ATWS) when the reactor is at > 5% power, or when reactor power is < 5% and before suppression pool temperature exceeds the Boron Injection Temperature as directed by EOP-1A.
3. Upon entry into SAG-2, “RPV, Containment, Radioactivity Release Control.”

The pH control using SLC is credited in the AST analysis based on the actions performed in the SAGs, not the EOPs. The EOP actions provide transition into the SAGs, but are not relied upon for the pH control action.

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Based on the above, under all conditions that would require SLC to be utilized as part of the AST pH control strategy, SLC injection would have been implemented in accordance with the existing procedures. Additionally, CPS 4750.01, "SAMG Team – Technical Support Guidelines (TSG)," TAB 8, "Reactor Shutdown/Boron Injection," incorporated the following SLC post-LOCA AST insight discussion in accordance with References 1 and 2.

Entry into the SAGs requires SLC injection as part of the Alternative Source Term response strategy.

The SLC inventory which enters the suppression pool via the LOCA leak path will assist in maintaining Suppression Pool pH > 7.0, thereby minimizing the re-evolution of iodine from the suppression pool.

Injection of SLC is required within 3 hours of the LOCA event.

Proper implementation of the EOP/SAGs assures that sufficient SLC injection will occur within the first 30 minutes.

Operators, upon detection of symptoms indicating that severe accident conditions have occurred, are directed to manually initiate the SLC system. As documented in Attachment 2 to Reference 1, the AST LOCA analysis assumes that the borated solution injection is initiated within 3 hours following the accident. If an accident were to occur which would create the fuel damage conditions assumed in the analysis, the EOP/SAG procedures would require manual initiation of the SLC system within the assumed 3 hour time frame. Proper implementation of the EOP/SAGs assures that SLC system injection will occur in accordance with the analysis assumptions.

The following responses are provided to each of the specific requests (i.e., 2.1 through 2.5) above.

- 2.1 Procedures CPS 4411.10, "SLC Operations," and CPS 4750.01, in addition to the EOPs and SAGs are all safety related procedures that contain actions associated with SLC injection. These procedures are controlled in accordance with the criteria found in CPS 1005.09, "EOP/SAG Program," Technical Specification 5.4.1.b, USAR Section 13.5.2.1.3, "Emergency Operating Procedures," and USAR Table 13.5-5, "Off-Normal Procedures."
- 2.2 By definition of the purpose and function of the EOPs, entry conditions within the EOPs are not a symptom of imminent or actual core damage. Failure to accomplish such actions, however, may result in imminent or actual core damage. By definition of the purpose and function of the SAGs, entry conditions within the SAGs are a direct symptom of imminent or actual core damage. The suppression pool pH control (as part of the AST response strategy) using SLC is

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credited to the actions performed in the SAGs, not the EOPs. The EOP actions provide transition into the SAGs, but are not relied upon for the pH control action.

- 2.3 The drywell high pressure and reactor vessel water level instruments (i.e., LOCA signature parameters) meet the quality requirements for Type A or B variables as defined in RG 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident," Tables 1 and 2. Regulatory Positions 1.3 and 1.4 of RG 1.97 provide design and qualification criteria for the instrumentation used to measure the various variables listed in Tables 1 and 2. The criteria are separated into 3 separate groups or categories that provide a graded approach to requirements depending on the importance to safety of the measurement of a specific variable. Category 1 provides the most stringent requirements and is intended for key variables. Category 2 provides less stringent requirements and generally applies to instrumentation designated for indicating system operating status. Category 3 is intended to provide requirements that will ensure that high-quality off-the-shelf instrumentation is obtained and applies to backup and diagnostic instrumentation. As stated in RG 1.97 Table 1, Type A variables are Category 1 and for Type B, the key variables are Category 1 and the backup variables are generally Category 3. For Type E, the key variables are generally Category 2 and the backup variables are Category 3.

Therefore, because the instrumentation utilized for SLC injection meet the requirements for Type A and B variables as described above, this instrumentation actually meets more stringent quality requirements than Type E variables.

- 2.4 Licensed operators have received initial training on the EOPs and SAGs, and will continue to receive periodic refresher training. Severe Accident Management Guideline (SAMG) team members, responsible for evaluation and implementation of SAGs as part of the Emergency Response Organization, have also received training on CPS 4750.01, TAB 8 insights.
- 2.5 At CPS, when the EOPs or SAGs direct initiation of the SLC System, the operator is referenced to operating procedure CPS 4411.10 for the specific steps to perform. The procedure directs the operator to inject the entire contents of the SLC storage tank whenever the reactor water level is below the TAF or when operation is directed by the SAGs regardless of the initiating event. This ensures that SLC injection will not be terminated during a LOCA.

The RC/Q-1 action applies to an ATWS event and does not assume a DBA LOCA (i.e., reactor water level less than TAF) has concurrently occurred. Since the EOP and SAG level control actions direct SLC injection when the reactor water level is below the TAF, there is no conflict between the ATWS exit action and the level control action. Additionally, entry into the SAGs requires SLC

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injection regardless of the initiating event. Therefore, no changes to the Emergency Procedure Guidelines or the EOP steps are required.

#### **Request 3**

*Please provide a description of the analysis assumptions, inputs, methods, and results that show that a sufficient quantity of sodium pentaborate can be injected to raise and maintain the suppression pool greater than pH 7 within 24 hours of the start of the event. (See also Position 2 of Appendix A to RG 1.183.) In your response, please discuss the adequacy of recirculation of suppression pool liquid via ECCS through the reactor vessel and the break location and back to the suppression pool in meeting the transport and mixing assumptions in the chemical analyses. Assume a large break LOCA.*

#### **Response 3**

In Reference 3, AmerGen responded to an NRC request for additional information regarding the control of suppression pool pH for the 30-day period after a large-break LOCA. A copy of CPS Calculation IP-M-0726 (i.e., Reference 2) was provided in the Reference 3 response. This calculation provided the design assumptions, methods, and results demonstrating that the suppression pool pH will be maintained above 7 throughout the 30-day period following a large-break LOCA.

The CPS EOP for reactor pressure vessel (RPV) control is entered at the onset of a LOCA based on low reactor water or high drywell pressure entry conditions. The Low Pressure Coolant Injection (LPCI), Low Pressure Core Spray (LPCS), and High Pressure Core Spray (HPCS) systems are among the preferred systems for maintaining reactor water level above the top of active fuel thereby ensuring adequate core cooling. As described above in Response 2, the EOP directs operators to manually initiate the SLC system as an alternate injection system when reactor water level cannot be maintained above the top of active fuel. In the event that these EOP actions are unable to maintain adequate core cooling, the EOP directs entry into the SAGs. Upon entry into the SAGs, the operator is directed to inject SLC regardless of the initiating event. Therefore, whenever there are symptoms of imminent core damage due to inadequate core cooling, operators are required by procedure to initiate SLC as well as ensure available emergency core cooling systems (i.e., LPCI, LPCS, and HPCS) are injecting.

The Emergency Core Cooling System (ECCS) takes water from the suppression pool and pumps it into the core region of the reactor vessel. The LPCI system is automatically actuated by low-water level in the reactor vessel or high pressure in the drywell and uses the three motor driven pumps to draw suction from the suppression pool and inject cooling water into the reactor core and accomplish cooling of the core by flooding. Each loop has its own suction and discharge piping and separate vessel nozzle that connects with the core shroud through the LPCI couplings to deliver flooding water near the top of the core. The LPCS system is an independent loop, which delivers water over the core at relatively low reactor pressures. Its primary purpose is to provide inventory makeup and spray cooling during large breaks in which the core is calculated to uncover. Following Automatic Depressurization System (ADS) initiation, LPCS provides inventory makeup following a small break. The LPCS injection piping enters



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the vessel, divides and enters the core shroud at two points near the top of the shroud. A semicircular sparger is attached to each outlet. Nozzles are spaced around the sparger to spray the water radially over the core and into the fuel assemblies. The HPCS system consists of a single motor-driven pump that takes suction from the Reactor Core Isolation Cooling (RCIC) storage tank and injects into the reactor vessel. After the HPCS injection piping enters the vessel, it divides and enters the shroud at two points near the top of the shroud similar to the LPCS system. A semicircular sparger is attached to each outlet. Nozzles are spaced around the spargers to spray the water radially over the core and into the fuel assemblies. In the event that the RCIC storage water supply becomes exhausted or is not available, automatic switchover to the suppression pool occurs for continuous operation of the HPCS system.

The SLC System will pump sodium pentaborate solution from the SLC storage tank into the core region of the reactor vessel. The SLC system is composed of the SLC storage tank, the test water tank, the two positive displacement injection pumps, the two explosive valves, the two motor operated pump suction valves, and associated local valves and controls and is located in the containment. The flow path of the boron neutron absorber (i.e., sodium pentaborate) solution to the reactor vessel is by the HPCS sparger. The SLC piping is connected to the HPCS system just downstream of the HPCS manual injection isolation valve. This ECCS water and SLC sodium pentaborate solution will refill the Reactor Pressure Vessel (RPV) under post-LOCA conditions. The mixed ECCS water and SLC solution will spill out of the break into the pool formed in the drywell and will then spill over to the suppression pool.

To illustrate the adequacy of SLC solution mixing in the suppression pool, bounding calculations were completed assuming a maximum initial post-LOCA suppression pool liquid volume of 164,488 ft<sup>3</sup> and a Drywell fill volume of 33,495 ft<sup>3</sup>. The calculation assumes that initially two LPCI pumps are running at 4400 gpm (8800 gpm for two) and the HPCS pump will be running at 4900 gpm. As an alternative to HPCS pump utilization, the LPCS pump is also available, running at the same 4900 gpm, with no change in the results. After 30 minutes, the calculation assumes that the HPCS pump and one LPCI pump will be running. Based on a combined 13,700 gpm ECCS flow for the first 30 minutes and 9300 gpm flow after 30 minutes, the time to raise the amount of water in the drywell equivalent to the drywell fill volume is less than 18.5 minutes. At 9300 gpm, the time to turnover one suppression pool volume for mixing is calculated to take approximately 2 hours, and a three volume turnover can be achieved very conservatively in less than 7 hours post-LOCA.

Reference 2 shows that if no sodium pentaborate solution is injected, the suppression pool pH will fall below 7 at approximately 3.5 hours from the start of the design basis accident (DBA) for the worst-case beginning of cycle conditions. However, the analysis shows that if the entire SLC solution is not injected within 3.5 hours, the suppression pool pH is still shown to be controlled even if minimal amounts of sodium pentaborate are injected. For example, with only 5% of the sodium pentaborate solution injected, pH falls below 7 after 12 hours and with only 10% of the sodium pentaborate solution injected, pH falls below 7 after 24 hours. Clearly, injection time is flexible and can

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accommodate delays in drywell pool to suppression pool mixing. Analysis also shows that the suppression pool reaches 5% of required sodium pentaborate concentration within 18 minutes of the start of injection and 10 % of the required concentration within 27 minutes.

Based on the above injection and mixing considerations, sufficient pH control can be achieved as long as injection is initiated within a 3-hour period following LOCA indications of core damage resulting in significant radiological releases to the drywell. As provided in Response 2 above, CPS procedures assure such a timely initiation of SLC system injection.

#### **Request 4**

*Please show that the SLC system has suitable redundancy in components and features to assure that for onsite or offsite electric power operation its safety function of injecting sodium pentaborate for the purpose of suppression pool pH control can be accomplished assuming a single failure. For this purpose, the check valve is considered an active device since the check valve must open to inject sodium pentaborate. If the SLC system can not be considered redundant with respect to its active components, the licensee should implement one of the three options described below, providing the information specified for that option for staff review.*

4.1 Option 1 *Show acceptable quality and reliability of the non-redundant active components and/or compensatory actions in the event of failure of the non-redundant active components. If you choose this option, please provide the following information to justify the lack of redundancy of active components in the SLC system:*

- 4.1.1 *Identify the non-redundant active components in the SLC system and provide their make, manufacturer, and model number.*
- 4.1.2 *Provide the design-basis conditions for the component and the environmental and seismic conditions under which the component may be required to operate during a design-basis accident. Environmental conditions include design-basis pressure, temperature, relative humidity and radiation fields.*
- 4.1.3 *Indicate whether the component was purchased in accordance with Appendix B to 10 CFR Part 50. If the component was not purchased in accordance with Appendix B, provide information on the quality standards under which it was purchased.*
- 4.1.4 *Provide the performance history of the component both at the licensee's facility and in industry databases such as EPIX and NPRDS.*
- 4.1.5 *Provide a description of the component's inspection and testing program, including standards, frequency, and acceptance criteria.*

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- 4.1.6 *Indicate potential compensating actions that could be taken within an acceptable time period to address the failure of the component. An example of a compensating action might be the ability to jumper a switch in the control room to overcome its failure. In your response please consider the availability of compensating actions and the likelihood of successful injection of the sodium pentaborate when non-redundant active components fail to perform their intended functions.*
- 4.2 *Option 2 Provide for an alternate success path for injecting chemicals into the suppression pool. If you chose this option, please provide the following information.*
- 4.2.1 *Provide a description of the alternative injection path, its capabilities for performing the pH control function, and its quality characteristics.*
- 4.2.2 *Do the components which make up the alternative path meet the same quality characteristics required of the SLC system as described in Items 1.1 to 1.5, 2 and 3 above?*
- 4.2.3 *Does the alternate injection path require actions to be taken in areas outside the control room? How accessible will these areas be? What additional personnel would be required?*
- 4.3 *Option 3 Show that 10 CFR 50.67 dose criteria are met even if pH is not controlled. If you chose this option, demonstrate through analyses that the projected accident doses will continue to meet the criteria of 10 CFR 50.67 assuming that the suppression pool pH is not controlled. The dissolution of CsI and its re-evolution from the suppression pool as elemental iodine must be evaluated by a suitably conservative methodology. The analysis of iodine speciation should be provided for staff review. The analysis documentation should include a detailed description and justification of the analysis assumptions, inputs, methods, and results. The resulting iodine speciation should be incorporated into the dose analyses. The calculation may take credit for the mitigating capabilities of other equipment, for example the standby gas treatment system (SGTS), if such equipment would be available. A description of the dose analysis assumptions, inputs, methods, and results should be provided. Licensees proposing this approach should recognize that this option will incur longer staff review times and will likely involve fee-billable support from national laboratories.*

#### **Response 4**

The CPS SLC system can be considered redundant with respect to its active components, except as outlined below. This limited lack of redundancy is offset as described in the justifications provided. Therefore, Options 2 and 3 are not applicable to CPS and the following information is provided in accordance with Option 1 above.

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- 4.1.1 The only non-redundant active components in the SLC system are the check valves (i.e. two in series) located on the SLC injection line. Details regarding these components are listed below:

Valve Number	Description	Manufacturer	Model Number
1C41-F006	3", ASME Section III, Class 1	Anchor Darling	W8121407
1C41-F336	4", ASME Section III, Class 1	Anchor Darling	W8221539

- 4.1.2 The environmental data for the SLC system check valves are listed in the following table. These components are seismically qualified.

Valve Number	1C41-F006		1C41-F336	
Location	Containment, Zone H-26		Drywell, Zone H-27	
	<b>Design Max</b>	<b>Qualified to</b>	<b>Design Max</b>	<b>Qualified to</b>
Temperature (°F)	185	345 Max	339.97	345
Pressure (psig)	15	30	30	30
Humidity (%)	100	Steam	Steam	Steam
Radiation (rads)	2.0E08	2.0E08	2.0E08	2.0E08
Seismic Qualification Package SQ-CL073	Maximum Credible Earthquake		Maximum Credible Earthquake	

- 4.1.3 The SLC system components were purchased in accordance with 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."
- 4.1.4 In accordance with the Inservice Test program, these check valves are inspected and tested during scheduled refueling outages. In addition, Technical Specifications require verification of flow through the SLC system from the pump to the reactor pressure vessel at least once every 18 months on a staggered test basis. This demonstrates the operability of the integrated system.

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 (EPIX) was performed and no failure data was identified for check valves failing

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to open. The Nuclear Plant Reliability Data System (NPRDS) was reviewed and no failures related to check valve opening issues were identified.

NUREG/CR-5944, "A Characterization of Check Valve Degradation and Failure Experience in the Nuclear Power Industry," was reviewed and it was determined that larger valves were found to be more likely to degrade and to degrade significantly than smaller valves. Valves that are greater than 10 inches and used in normally operating systems were twice as likely to fail as the valve population as a whole. Valves used in service water, main steam, feedwater and diesel starting systems were found to be two or more times as likely to fail when compared to the valves used in other systems. BWR plants had a higher overall relative failure rate, however, this was clearly the result of the fact that BWRs are better structured to detect failures programmatically. Since CPS requires valve inspection and testing as described above, as well as ensuring that proper monitoring and programmatic controls are in place for reliable operation of the SLC check valves, the stuck closed failure of the common SLC system discharge check valves is highly unlikely.

- 4.1.5 The check valves listed in response to request 4.1.1 above are tested in accordance with CPS Technical Specifications (TS) and IST program requirements. These valves have an open function to support injection of sodium pentaborate and a close function for primary containment isolation.

TS Surveillance Requirement (SR) 3.1.7.8 requires verification of flow through one SLC subsystem from a pump into the reactor pressure vessel every 18 months on a staggered test basis. This surveillance ensures that there is a functioning flow path from the boron solution storage tank to the reactor pressure vessel. During this test, one of the subsystems, including an explosive valve, is initiated, and it is verified that a flow path from the pump to the reactor pressure vessel is available. This testing necessitates replacement of the explosive charge in the shear plug valves. Both complete flow paths are tested every 36 months. This test verifies the flow path to the reactor pressure vessel and particularly the proper operation of the check valves in the drywell. This SR is implemented by procedure CPS 9015.02, "Standby Liquid Control Injection Operability."

In addition to the above test, TS SR 3.1.7.9 requires verification that all piping between the SLC storage tank and pump suction is unblocked every 18 months and once within 24 hours after the pump suction piping temperature is restored to greater than or equal to 70°F. This test demonstrates that all piping between the boron solution storage tank and the suction inlet to the injection pumps is unblocked thus ensuring that there is a functioning flow path for injecting the sodium pentaborate solution. This test is also implemented by procedure CPS 9015.02.

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The SLC check valves 1C41-F006 and 1C41-F336 are tested every refueling outage as part of the CPS IST program. This testing includes an exercise open and an exercise close test and is included as part of the testing performed under procedure CPS 9015.02.

In summary, the required TS and IST program testing provide assurance of a high degree of system reliability and confidence that the system injection function would perform satisfactorily if called upon following a design basis LOCA. As noted above, performance of all valves has been exemplary with no failures identified that would preclude injection.

- 4.1.6 As noted above, the only non-redundant active components in the SLC system are the two in-series primary containment isolation check valves in the SLC injection line. Failure of either of these isolation check valves to open would prevent SLC injection via the normal injection pathway. However, AmerGen believes that compensating actions are not warranted due to the reliability of the non-redundant components in the SLC system. This can be illustrated, for example, using data from the latest revision of the CPS Probabilistic Risk Analysis (PRA). The probability for failure of a SLC discharge check valve to open is estimated at  $2.0E-4$ /demand in the CPS PRA. This failure rate is calculated based on a Bayesian statistical update of an industry generic failure rate for check valves with CPS-specific experience in accordance with standard and expected PRA industry techniques as described in Supporting Requirement DA-D1 of the ASME PRA Standard. The probability of either of the two in-series check valves failing open on demand is  $4E-4$ . A review of the SLC system fault tree indicates that failure of the non-redundant check valves to open represents less than 6% of the total SLC system failure probability.

#### **References**

1. Letter from Michael J. Pacilio (AmerGen Energy Company, LLC) to U. S. NRC, "Request for License Amendment Related to Application of Alternative Source Term," dated April 3, 2003 (RS-03-060)
2. Calculation IP-M-0726, "Suppression Pool pH Calculation for Alternative Source Term"
3. Letter from Keith R. Jury (AmerGen Energy Company, LLC) to U. S. NRC, "Additional Information Supporting the Request for License Amendment Related to Application of the Alternative Source Term," dated December 23, 2003 (RS-03-239)