Exelon Generation
4300 Winfield Road
Warrenville, IL 60555

www.exeloncorp.com

Exelon.
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10 CFR 50.90

RS-10-028
March 3, 2010

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

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

Subject: Request for Amendment to Technical Specification 3.1.7, "Standby Liquid Control (SLC) System"

References:

1) Letter from M. A. Satorius (U. S. NRC) to C. M. Crane (Exelon Generation Company, LLC), "Notice of Enforcement Discretion for Exelon Generation Company LLC Regarding Quad Cities Nuclear Power Station, Unit 1 (NOED 06-3-01)," dated October 18, 2006

2) Letter from M. A. Satorius (U. S. NRC) to C. M. Crane (Exelon Generation Company, LLC), "Notice of Enforcement Discretion for Exelon Generation Company LLC Regarding Dresden Nuclear Power Station, Unit 2 (NOED 07-3-01; TAC MD4044)," dated January 24, 2007

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (EGC) requests an amendment to Appendix A, Technical Specifications (TS) of Facility Operating License No. NPF-62 for Clinton Power Station (CPS), Unit 1.

The proposed amendment revises Technical Specification (TS) 3.1.7, "Standby Liquid Control (SLC) System," to extend the completion time (CT) for Condition B (i.e., "Two SLC subsystems inoperable") from eight hours to 72 hours.

In References 1 and 2, the NRC exercised discretion to not enforce compliance with the actions required in TS 3.1.7, Condition C for Quad Cities Nuclear Power Station, Unit 1 and Dresden Nuclear Power Station, Unit 2, respectively. These notices of enforcement discretion (NOEDs) provided a 72-hour extension to the 12-hour CT specified in Required Action C.1 (i.e., "Be in MODE 3"). This extension enabled each site to avoid a TS-required shutdown while implementing short-term repair and restoration activities for an emergent issue impacting SLC system operability. The purpose of this proposed license amendment request (LAR) is to adopt a permanent, risk-informed CT extension for CPS TS 3.1.7, Required Action B.1, thus minimizing the potential for thermal transients associated with placing CPS Unit 1 in Mode 3.

EGC has utilized the guidance in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," to develop the technical basis for this LAR. The EGC analysis demonstrates, with reasonable assurance, that the proposed LAR satisfies the risk acceptance guidelines in Regulatory Guide 1.174 and Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decision-making: Technical Specifications." The proposed LAR meets the intent of very small risk increases consistent with the NRC's Safety Goal Policy Statement.

EGC Probabilistic Risk Assessment (PRA) maintenance, update processes, and technical capability evaluations provide a robust basis for concluding that the PRA is suitable for use in risk-informed licensing actions. Additionally, a PRA technical adequacy evaluation was performed consistent with the requirements of Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 1.

This request is subdivided as follows:

- Attachment 1 provides a description and evaluation of the proposed changes.
- Attachment 2 provides a mark-up of the CPS TS page with the proposed change indicated.
- Attachment 3 provides the marked-up CPS TS Bases pages, with the proposed changes indicated. This attachment is provided for information only.
- Attachment 4 provides the risk assessment that supports the proposed TS change for CPS (i.e., RM Documentation CL-LAR-01, Revision 1).

The proposed amendment has been reviewed and approved by the CPS Plant Operations Review Committee and the Nuclear Safety Review Board in accordance with the requirements of the EGC Quality Assurance Program and procedures. EGC requests approval of the proposed amendment by March 3, 2011, with implementation within 60 days of issuance.

In accordance with 10 CFR 50.91, "Notice for public comment," EGC is notifying the State of Illinois of this application for amendment by transmitting a copy of this letter and its attachments to the designated State Official.

There are no regulatory commitments contained within this letter. If you have any questions or require additional information, please contact Mr. John L. Schrage at (630) 657-2821.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 3rd day of March 2010.

Respectfully,

Jeffrey L. Hansen
Manager - Licensing

Attachment 1:	Evaluation of Proposed Amendment
Attachment 2:	Proposed Markup of CPS Technical Specification 3.1.7
Attachment 3:	Proposed Markup of CPS Technical Specification Bases B 3.1.7
Attachment 4:	RM Documentation No. CL-LAR-01, Revision 1

ATTACHMENT 1
Evaluation of Proposed Amendment

1.0 DESCRIPTION

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (EGC) requests an amendment to Facility Operating License Nos. NPF-62 for Clinton Power Station (CPS) Unit 1. The proposed amendment changes Technical Specification (TS) 3.1.7, "Standby Liquid Control (SLC) System," by extending the Completion Time (CT) for two inoperable SLC subsystems from 8 hours to 72 hours.

CPS TS LCO 3.1.7 requires the operability of two SLC subsystems when the reactor is in Modes 1, 2, and 3. In Modes 1 and 2, the SLC system satisfies the requirements of 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants," and "10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion (GDC) 26, "Reactivity control system redundancy and capability." In Mode 3, the SLC system helps ensure that offsite doses remain within the limits of 10 CFR 50.67, "Accident source term" following a loss-of-coolant accident (LOCA) involving significant fission product releases.

TS 3.1.7, Condition B and the associated Required Action B.1 address the inoperability of both SLC subsystems. Specifically, Required Action B.1 requires restoration of one SLC subsystem to operable status, with a CT of eight hours. If Required Action B.1 cannot be satisfied within the CT, Condition C and associated Required Actions C.1 and C.2 require the reactor to be in Mode 3 within 12 hours and Mode 4 in 36 hours.

The current CT for Required Action B.1 is based on the low probability of a design basis accident or transient occurring, concurrent with the failure of the control rods to shut down the reactor. Consistent with this current basis, the proposed TS CT change is based upon a risk-informed assessment that evaluates the probability and consequences of transients, accidents, and severe accidents including the design basis accident and transients occurring concurrent with control rod insertion failure.

EGC has utilized the guidance in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," to develop the risk assessment for this proposed change. The EGC assessment demonstrates, with reasonable assurance, that the proposed license amendment satisfies the risk acceptance guidelines in Regulatory Guide 1.174 and Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decision-making: Technical Specifications." The proposed license amendment meets the intent of very small risk increases consistent with the NRC's Safety Goal Policy Statement.

In addition to evaluating the risk impact, EGC has evaluated the proposed change to determine whether the impact of the change is consistent with the intent of the defense-in-depth philosophy and the principle that sufficient safety margins are maintained (i.e., consistent with the requirements of RG 1.177, Section C, "Regulatory Position," paragraph 2.2, "Traditional Engineering Considerations").

EGC has also determined that the EGC Probabilistic Risk Assessment (PRA) maintenance, update processes, and technical capability evaluations provide a robust basis for concluding that the EGC PRA is suitable for use in risk-informed licensing actions. EGC conducted a PRA technical adequacy evaluation, consistent with the requirements of Regulatory Guide 1.200, "An

Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 1.

2.0 PROPOSED CHANGE

The proposed amendment revises the CT for CPS TS 3.1.7, Required Action B.1 from eight hours to 72 hours.

3.0 BACKGROUND

The SLC system is designed to provide the capability of bringing the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory to a subcritical condition with the reactor in the most reactive, xenon free state without taking credit for control rod movement. The SLC system satisfies the requirements of 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants."

CPS TS LCO 3.1.7 requires the operability of two SLC subsystems when the reactor is in Modes 1, 2, and 3. TS 3.1.7, Condition B and the associated Required Action B.1 address the inoperability of both SLC subsystems. Specifically, Required Action B.1 requires restoration of one SLC subsystem to operable status, with a CT of eight hours. If Required Action B.1 cannot be satisfied within the CT, Condition C and associated Required Actions C.1 and C.2 require the reactor to be in Mode 3 within 12 hours and Mode 4 in 36 hours.

In October 2006 and January 2007, EGC requested Notices of Enforcement Discretion (NOEDs) for Quad Cities Nuclear Power Station (QCNPS) Unit 1 and Dresden Nuclear Power Station (DNPS) Unit 2, respectively, to allow sufficient time for the repair of minor SLC system tank leaks. The NRC granted these NOEDs, allowing an additional 72 hours to the original 12-hour CT for TS 3.1.7, Required Action C.1 (i.e., "Be in MODE 3") for the emergent dual-train inoperability of the SLC systems (References 1 and 2).

The purpose of this proposed LAR is to adopt a permanent, risk-informed CT extension for CPS TS 3.1.7, Required Action B.1, thus minimizing the potential for thermal transients associated with placing CPS, Unit 1 in Mode 3. The integrity of the reactor vessel and other components of the primary system of a nuclear plant can be adversely affected by the number of thermal transients that they are subjected to during their lifetime. As each additional thermal transient can affect this integrity, it is prudent to avoid such transients.

4.0 TECHNICAL ANALYSIS

The proposed change is consistent with the principle that adequate defense-in-depth is maintained, that sufficient safety margins are maintained, and that increases in risk are very small and meet the acceptance guidelines in RG 1.174, RG 1.177, and the NRC's Safety Goal Policy Statement. This consistency is described below, as well as in Attachment 4.

4.1 System Description

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

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

The SLC system consists of a boron solution storage tank, two positive displacement pumps, two explosive valves, which are provided in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). The preferred flow path of the boron neutron absorber solution to the reactor vessel is by the High Pressure Core Spray (HPCS) System sparger. The SLC piping is connected to the HPCS System just downstream of the HPCS manual injection isolation valve. An alternate flow path to the reactor vessel is provided by the SLC sparger near the bottom of the core shroud. This flow path is normally locked out of service by the SLC manual injection valve.

The SLC system is manually initiated from the main control room, as directed by the emergency operating procedures, if the operator believes the reactor cannot be shut down, or kept shut down, with the control rods. The SLC system is used in the event that not enough control rods can be inserted to accomplish shutdown and cooldown in the normal manner. The SLC system injects borated water into the reactor core to compensate for all of the various reactivity effects that could occur during plant operation. To meet this objective, it is necessary to inject a quantity of boron that produces a concentration equivalent to at least 1000 ppm of natural boron in the reactor core at 68°F. This is accomplished by the use of enriched boron (i.e., greater than or equal to 30 atom% boron 10). To allow for potential leakage and imperfect mixing in the reactor system, an additional amount of boron equal to 25% of the amount cited above is added. The concentration versus volume limits are calculated such that the required concentration is achieved accounting for dilution in the RPV with normal water level and including the water volume in the residual heat removal shutdown cooling piping and in the recirculation loop piping.

4.2 Defense-in-Depth

The control rods are the primary reactivity control system for the reactor at CPS. In conjunction with the Reactor Protection System (RPS), the control rods provide the means for reliable control of reactivity changes to ensure that, under conditions of normal operation, including anticipated operational occurrences, specified acceptable fuel design limits are not exceeded. Operability of the control rods is governed by TS 3.1.3, "Control Rod OPERABILITY," and the control rods are demonstrated operable by the performance of TS Surveillance Requirements (SRs) 3.1.3.1 and 3.1.3.3 through

3.1.3.5. This Specification, along with TS 3.1.4, "Control Rod Scram Times," and TS 3.1.5, "Control Rod Scram Accumulators," ensure that the performance of the control rods in the event of a Design Basis Accident (DBA) or transient meets the assumptions used in the safety analyses.

Scram reliability is ensured by a number of design and operational features:

- An individual accumulator is provided for each control rod drive with sufficient stored energy to scram at any reactor pressure. The reactor vessel itself, at pressures above 600 psi, will supply the necessary force to insert a drive if its accumulator is unavailable.
- Each drive mechanism has its own scram valves and a dual solenoid scram pilot valve therefore only one drive can be affected if a scram valve fails to open. Both pilot valve solenoids must be deenergized to initiate a scram.
- The reactor protection system and the HCUs are designed so that the scram signal and mode of operation override all others.
- The collet assembly and index tube are designed so they will not restrain or prevent control rod insertion during scram
- The scram discharge volume is monitored for accumulated water and the reactor will scram before the volume is reduced to a point that could interfere with a scram.
- The alternate rod insertion (ARI) system provides an alternate means of exhausting the scram air header and closing the vent and drain valves of the scram discharge volume, thereby providing an additional reactor scram mechanism which is diverse, redundant and independent of the reactor protection system.
- In addition to the ARI system, the ATWS Recirculating Pump Trip (RPT) system provides an additional means for rapid power reduction. The ATWS-RPT system initiates a recirculation pump trip, adding negative reactivity, following events in which a scram does not, but should occur, to lessen the effects of an ATWS event.

As noted above, operability of the trip function of the control rods is demonstrated by specific SRs. For the control rod scram function to fail when a valid signal is sent, a diverse number of failures would have to occur in order in prevent the scram valves from opening.

Operability of the ATWS system (i.e., the ARI system and the ATWS RPT system) is governed by TS 3.3.4.2, "Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation," and is demonstrated operable by the performance of TS SRs 3.3.4.2.1 through 3.3.4.2.5.

The proposed change to the SLC CT does not affect the redundancy, independence, and diversity of the RPS, ARI, and the ATWS-RPT systems. These systems and instrumentation remain operable to mitigate the consequences of any previously analyzed accident. In addition to the TS requirements for control rod and ATWS system operability, the EGC Work Management and Maintenance Rule (i.e., 10 CFR 50.65(a)(4)) programs provide controls and assessments to minimize the probability of simultaneous outages of redundant trains and ensure system reliability. The proposed SLC CT extension does not involve any change to plant equipment or system design functions.

This proposed TS CT extension does not change the design function of the SLC system and does not affect the system's ability to perform its design function. As such, the proposed change complies with the defense-in-depth principles described in RG 1.174, paragraph 2.2.1.1 and RG 1.177, paragraph 2.2.1. These principles, and the impact of the proposed change on each, are described below.

- **A reasonable balance is preserved between prevention of core damage, prevention of containment failure, and consequence mitigation.**

  The proposed SLC CT extension does not affect the ability of SLC, or any system, to prevent core damage, prevent containment failure, or mitigate the consequences of an accident. The proposed change has only a very small impact on risk. The proposed change does not compensate for this risk impact with an assumption of improved containment integrity, nor does this proposed change degrade containment integrity and compensate with an assumption of improved core damage prevention.

- **Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided.**

  Plant design for both the primary (i.e., RPS and ARI/RPT) and alternate (i.e., SLC) reactivity control systems at CPS is robust. The proposed SLC CT extension does not require, nor rely upon programmatic activities to compensate for weaknesses in plant design. The four-channel RPS, in concert with the control rods, ensures reliable and automatic control of reactivity changes to ensure that fuel design limits are not exceeded. The scram system is designed so that the scram signal overrides all other operating signals. Upon loss of either instrument air or electrical power, the scram valves will fail open. Hence, failure of the valves' air system or electric system will produce, rather than prevent, a scram.

- **System redundancy, independence, and diversity are maintained commensurate with the expected frequency and consequences of challenges to the system.**

  The redundancy, independence, and diversity of the RPS, the control rods, and the control rod drive system are not affected during the extended 72-hour SLC CT. Entry into the dual-train SLC CT will be assessed and managed in accordance with the EGC Configuration Risk Management Program (CRMP).

  Additional redundancy for reactivity control is established by CPS procedures. These procedures describe the actions and criteria for manual addition of boron into the reactor coolant system (i.e., via the reactor water cleanup system), should RPS, the control rods, the control rod drive system, and the SLC be unable to perform the specifed design functions.

- **Defenses against potential common cause failures are maintained and the potential for introduction of new common cause failure mechanisms is assessed.**

  The extended SLC CT does not change the design function of the SLC system. Therefore, the proposed change does not affect existing common cause failure mechanisms. In addition, the operating environment and operating parameters for the SLC system, the RPS system, the control rods, and the control rod drive system remain constant; therefore, new common cause failures modes are not expected. Therefore, no new potential common cause failure mechanisms have been introduced by the proposed change.

- **Independence of barriers is not degraded.**

  The extended CT does not provide a mechanism that degrades the independence of fission product barriers, (i.e., fuel cladding, the reactor coolant system, or containment).

- **Defenses against human errors are maintained.**

  The risk assessment for the extended SLC CT does not credit, nor require new operator actions. Therefore, the proposed change does not impact defense-in-depth against human error.

4.3 Safety Margin Assessment

The proposed SLC CT extension does not involve a reduction in the margin of safety. The margin of safety is established through the design of the plant structures, systems, and components, the parameters within which the plant is operated, and the setpoints for the actuation of equipment relied upon to respond to an event. The proposed amendment does not modify the safety limits or setpoints at which protective actions are initiated. Since this proposed TS amendment does not change the SLC system design, but only extends a CT, safety margins are not challenged.

4.4 Risk Assessment

The CT is defined as part of the limiting condition for operation (LCO), and is intended to allow sufficient time to repair failed equipment while minimizing the risk associated with the loss of the component function. An extension of the CT increases the unavailability of a component due to the increased time the component is out-of-service for maintenance. The CT risk is reflected in the core damage frequency (CDF) and the large early release frequency (LERF) by adjusting the component unavailability due to maintenance.

The proposed CT extension for the dual-train inoperability of the CPS SLC system provides additional time to complete test and maintenance activities while at power, potentially reducing the number of forced outages related to compliance with the existing CT.

EGC completed a risk assessment for CPS using the full power internal events, Level 1 CDF model and the associated Level 2 LERF model. This risk assessment is provided in Attachment 4. The risk assessment was performed in accordance with the requirements in RG 1.174, RG 1.177, and RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk Informed Activities," Revision 1. The results of these risk assessments are discussed below.

4.4.1 Regulatory Standards

The RG 1.174 acceptance guidelines for a permanent TS change specify that the delta (Δ)CDF and the ΔLERF associated with the change should be less than specified acceptable values, which are dependent on the baseline CDF and LERF. These specified acceptable values are presented for two ranges of risk impacts, those described as "small changes" and those described as "very small changes". EGC utilized the acceptance guidelines for "very small changes" in the risk assessment for the proposed CPS TS change.

The RG 1.174 acceptance guidelines prescribe that the risk metrics of ΔCDF and ΔLERF be less than 1.0E-06/yr and 1.0E-07/yr, respectively, to establish a very small risk increase with no additional compensatory measures required. RG 1.174 also specifies guidelines for consideration of external events, and stipulates that external events can be evaluated in either a qualitative or quantitative manner.

RG 1.177 identifies a three-tiered approach for the evaluation of the risk associated with a proposed TS change.

- **Tier 1, PRA Capability and Insights**

  Tier 1 is an evaluation of the plant-specific risk associated with the proposed TS change, as shown by the change in CDF and incremental conditional core damage probability (ICCDP). Where applicable, containment performance should be evaluated on the basis of an analysis of LERF and incremental conditional large early release probability (ICLERP). The acceptance guidelines given in RG 1.177 for determining an acceptable TS change is that the ICCDP and the ICLERP associated with the change should be less than 5E-07 and 5E-08, respectively.

- **Tier 2, Avoidance of Risk Significant Plant Configuration**

  Tier 2 identifies and evaluates, with respect to defense-in-depth, any potential risk-significant plant equipment outage configurations associated with the proposed change. As such, procedures should provide reasonable assurance that risk-significant plant equipment outage configurations will not occur when equipment associated with the proposed TS change is out-of-service.

- **Tier 3, Risk-Informed Configuration Risk Management**

  Tier 3 provides for the establishment of an overall CRMP and confirmation that its insights are incorporated into the decision-making process before taking equipment out-of-service prior to or during the CT. Compared with Tier 2, Tier 3 provides additional coverage based on any additional risk significant configurations that may be encountered during maintenance scheduling over extended periods of plant operation. Tier 3 guidance can be satisfied by the Maintenance Rule (10 CFR 50.65(a)(4)), which requires a licensee to assess and manage the increase in risk that may result from activities such as surveillance, testing, and corrective and preventive maintenance.

RG 1.200, Revision 1 describes an acceptable approach for determining whether the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision-making for light-water reactors. This guidance is intended to be consistent with the NRC's PRA Policy Statement and more detailed guidance in RG 1.174.

RG 1.200, Revision 1 endorses Addendum B of the American Society of Mechanical Engineers (ASME) Standard RA-S-2002, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," Addenda RA-Sa-2003, and Addenda RA-Sb-2005, as applicable to full power internal event (FPIE) PRA models.

Since that time, the new ASME/American Nuclear Society (ANS) Standard RA-Sa-2009, "Addenda to RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," has been released. Although this standard is presently issued and endorsed by RG 1.200, Revision 2, neither of these documents adds further requirements that impact the results of the SLC CT risk assessment.

4.4.2 Tier 1: PRA Capability and Insights

As stated in RG 1.177, Tier 1 is an evaluation of the impact of the proposed TS change on CDF, ICCDP, and, when appropriate LERF and ICLERP considering PRA validity, and PRA insights and findings. Table 4.4.2-1 below provides the plant-specific risk associated with the proposed CPS TS change using the FPIE PRA models and based on the risk metrics of ΔCDF, ICCDP, ΔLERF, and ICLERP.

Table 4.4.2-1 CPS Risk Assessment Summary Results				
Hazard	**ΔCDF**	**ICCDP**	**ΔLERF**	**ICLERP**
FPIE	2.9E-08/yr	2.9E-08	6.2E-09/yr	6.2E-09
Acceptance Guideline	<1.0E-06/yr	<5.0E-07	<1.0E-07/yr	<5.0E-08
External Events	(1)	(1)	(1)	(1)

(1) In accordance with RG 1.174, paragraph 2.2.5.5, "Comparisons with Acceptance Guidelines," EGC performed a qualitative assessment of external event risk associated with the proposed CPS SLC CT extension (i.e., as described below and in Appendix A of Attachment 4) to demonstrate that the changes in risk remain within the acceptance guidelines.

The base results of the risk assessment, as summarized in Table 4.4.2-1 above indicate that the ΔCDF, ICCDP, ΔLERF, and ICLERP risk metric values for the proposed change are below the acceptance guidelines as defined in RG 1.174 and RG 1.177. This analysis demonstrates that the proposed TS change satisfies the risk acceptance guidelines in RG 1.174 and RG 1.177, and therefore meets the intent of very small risk increases consistent with the NRC's Safety Goal Policy Statement.

As part of the risk assessments, EGC performed a sensitivity analysis to determine the maximum allowable CT prior to exceeding the "very small" acceptance criteria. For this sensitivity, ICCDP and ICLERP were set to their maximum allowable values in RG 1.177, and the CT_{NEW} allowable was calculated. ICLERP was determined to be the bounding parameter, and a CT_{NEW} value of 582 hours was calculated. This value represents significant margin, relative to the proposed CT extension.

The CPS risk assessment also includes a qualitative assessment of external event risks in accordance with RG 1.174, paragraph 2.2.5.5, "Comparisons with Acceptance Guidelines."

This qualitative external events assessment used the external event analyses in the 1995 CPS Individual Plant Examination of External Events (IPEEE).

The qualitative external events assessment is described in Appendix A of Attachment 4, and summarized below.

<u>Internal Fires</u>

The impact on the internal fires risk profile due to the proposed change was evaluated using the following information sources:

- NUREG/CR-6850 , "EPRI Report 1011989, 'Fire PRA Methodology for Nuclear Power Facilities'," September 2005

- CPS-PSA-021.06, "Clinton FPRA Summary and Quantification Report," Rev. 0, September 2008
- Boiling Water Reactor Owners' Group (BWROG), "Assessment of NRC Information Notice 2007-07," October 16, 2007 (i.e., Appendix C of Attachments 4 and 5)

The assessment concluded that a fire-induced ATWS is a non-significant contributor to the plant risk profile and thus does not impact the proposed SLC system CT.

Seismic

The impact on the seismic risk profile for CPS, due to the proposed change was evaluated using the following information sources:

- CPS Seismic Margins Assessment that was performed as part of the CPS IPEEE, and was consistent with the guidance in EPRI NP-6041, "A methodology for assessment of nuclear power plant seismic margin"
- NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," December 1990

The assessment concluded that the seismic hazard can be appropriately screened as a non-significant contributor to the risk assessment of the proposed change.

Other External Hazards

Other external event risks such as external floods, severe weather, high winds or tornados, transportation accidents, nearby facility accidents, turbine missiles, and other miscellaneous external hazards were also considered in the CPS IPEEE analysis. No significant quantitative contribution from these external events was identified by the CPS IPEEE evaluations. As such, other external hazards are appropriately screened as non-significant contributors to the risk assessment of the proposed CT.

Consistent with the ASME PRA Standard, quantitative parametric uncertainty analyses for both CDF and LERF were performed. The results of these analyses are summarized in Appendix B of Attachment 4.

An assessment of modeling uncertainties is also documented in Appendix B of Attachment 4. This assessment includes CPS-specific modeling uncertainty evaluations for the PRA Base Case and an examination of the specific cutsets that affect the change in the CDF risk metric associated with the change in the SLC CT extension. The results of the modeling uncertainty assessments do not change the conclusions of this risk assessment for the proposed SLC CT changes.

4.4.3 Tier 2, Avoidance of Risk Significant Plant Configurations

Tier 2 requires an examination of the need to impose additional restrictions when operating under the proposed CT in order to avoid risk-significant equipment outage configurations. Consistent with the guidance in Regulatory Position C.2.3 of RG 1.177, and as part of the CPS risk assessment (i.e., Attachment 4), EGC performed an evaluation of equipment according to its contribution to plant risk while the equipment covered by the proposed CT change is out of service for test or maintenance (i.e., site-specific modeling uncertainty evaluations for the PRA base case and an examination of the specific cutsets that affect the change in the CDF risk metric associated with the change in the SLC CT extension).

This evaluation is provided in Attachment 4, Appendix B, "Uncertainty Analysis," section B.2, "Model Uncertainties Associated with SLC System Out of Service." This evaluation indicates that the scram system hardware failure is the most important contributor to the ΔCDF assessment for the SLC system out-of-service case.

Entry into the dual-train SLC CT will be assessed and managed in accordance with the EGC CRMP. The CRMP will assess the emergent condition, including the impact of any additional out-of-service equipment. With both SLC subsystems unavailable, the CPS on-line risk would be depicted as "Orange," based on the deterministic assessment portion of the CRMP. In this condition, station procedures require senior management review and approval to remove equipment from service, as well as implementation of compensatory measures to reduce risk, including contingency plans.

4.4.4 Tier 3, Risk-Informed Configuration Risk Management

Tier 3 requires a proceduralized process to assess the risk associated with both planned and unplanned work activities. The objective of the third tier is to ensure that the risk impact of out-of-service equipment is evaluated prior to performing any maintenance activity. As stated in Section 2.3 of RG 1.177, "a viable program would be one that is able to uncover risk-significant plant equipment outage configurations in a timely manner during normal plant operation." The third-tier requirement is an extension of the second-tier requirement, but addresses the limitation of not being able to identify all possible risk-significant plant configurations in the Tier 2 evaluation.

EGC has developed and implemented a CRMP at CPS. The CRMP is governed by station procedures that ensure the risk impact of out-of-service equipment is appropriately evaluated prior to performing any maintenance activity. These procedures require an integrated review to uncover risk-significant plant equipment outage configurations in a timely manner both during the work management process and for emergent conditions during normal plant operation. Appropriate consideration is given to equipment unavailability, operational activities like testing or load dispatching, and weather conditions. CPS currently has the capability to perform a configuration dependent assessment of the overall impact on risk of proposed plant configurations prior to, and during, the performance of maintenance activities that remove equipment from service. Risk is re-assessed if an equipment failure/malfunction or emergent condition produces a plant configuration that has not been previously assessed.

For planned maintenance activities, an assessment of the overall risk of the activity on plant safety is currently performed prior to scheduled work. The assessment includes the following considerations.

- Maintenance activities that affect redundant and diverse structures, systems, and components (SSCs) that provide backup for the same function are minimized.
- The potential for planned activities to cause a plant transient are reviewed, and work on SSCs that are important in mitigating the transient are avoided.
- Work is not scheduled that is highly likely to exceed a TS or Operational Requirements Manual (ORM) Completion Time requiring a plant shutdown.
- For Maintenance Rule high risk significant SSCs, the impact of the planned activity on the unavailability performance criteria is evaluated.

A quantitative risk assessment is performed for those SSCs modeled in the CPS PRA model to ensure that the activity does not pose any unacceptable risk. This evaluation is performed using the impact on both CDF and LERF. The results of the risk assessment are classified by a color code based on the increased risk of the activity. As postulated risk for the activity increases, appropriate actions are required and implemented. Emergent work is reviewed by shift operations to ensure that the work does not invalidate the assumptions made during the work management process. EGC's PRA risk management procedure defines the requirements for ensuring that the PRA model used to evaluate on-line maintenance activities is an accurate model of the current plant design and operational characteristics.

Plant modifications and procedure changes are monitored, assessed, and dispositioned. Evaluation of changes in plant configuration or PRA model features are dispositioned by implementing PRA model changes or by the qualitative assessment of the impact of the change on the PRA assessment tool. Changes that have potential risk impact are recorded in an update requirements evaluations (URE) log for consideration in the next periodic PRA model update.

The reliability and availability of the SLC system, RPS, control rods, control rod drives and the ARI system are monitored under the Maintenance Rule Program. If the pre-established reliability or availability performance criteria is exceeded for an instrumentation component, that component is considered for 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," paragraph (a)(1) actions, requiring increased management attention and goal setting in order to restore performance (i.e., reliability and availability) to an acceptable level. The performance criteria are risk-informed, and therefore are a means to manage the overall risk profile of the plant. An accumulation of large core damage probabilities over time is precluded by the performance criteria.

Evaluation of changes in plant configuration or PRA model features are dispositioned by implementing PRA model changes or by qualitatively assessing the impact of the changes on the CRMP assessment tool. Procedures exist for the control and application of CRMP assessment tools.

4.4.5 Technical Adequacy and Quality of PRA Model

As stated in Section 1.0 above, RG 1.200, Revision 1 describes an acceptable approach for determining whether the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used as an input in regulatory decision-making.

With respect to the risk assessment for the proposed SLC CT extension, EGC has documented this determination of PRA quality in Attachment 4. Table 2-1 of Attachment 4 provides a "RG 1.200 Analysis Actions Roadmap." This roadmap cross references the required RG 1.200 actions to the applicable sections in the attachment that address the actions, which are summarized below.

- EGC employs a multi-faceted approach to establishing and maintaining the technical adequacy and plant fidelity of the PRA models for all operating EGC nuclear generation sites. This approach includes both a proceduralized PRA maintenance and update process, and the use of self-assessments and independent peer reviews.
- The EGC risk management process for maintaining and updating the PRA ensures that the PRA model remains an accurate reflection of the as-built and as-operated plants. This process is defined in the EGC Risk Management program, which consists of a governing procedure (i.e., ER-AA-600, "Risk Management") and subordinate Technical & Reference Material (T&RM) documents. EGC T&RM ER-AA-600-1015, "FPIE PRA Model Update" delineates the responsibilities and guidelines for updating the full power internal events PRA models at all operating EGC nuclear generation sites.
- The overall EGC Risk Management program, including ER-AA-600-1015, defines the process for implementing regularly scheduled and interim PRA model updates, for tracking issues identified as potentially affecting the PRA models (e.g., changes in the plant, errors or limitations identified in the model, industry operating experience), and for controlling the model and associated computer files.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

According to 10 CFR 50.92, "Issuance of amendment," paragraph (c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequence of an accident previously evaluated;

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

Exelon Generation Company, LLC (EGC) has evaluated the proposed changes to the Technical Specifications (TS) for Clinton Power Station (CPS), Unit 1 using the criteria in 10 CFR 50.92 and has determined that the proposed changes do not involve a significant hazards consideration. EGC is providing the following information to support a finding of no significant hazards consideration.

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

Response: No.

The proposed amendment revises Technical Specification (TS) 3.1.7, "Standby Liquid Control (SLC) System," to extend the completion time (CT) for Condition B (i.e., "Two SLC subsystems inoperable.") from eight hours to 72 hours.

The proposed change is based on a risk-informed evaluation performed in accordance with Regulatory Guides (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis," and RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decision-making: Technical Specifications."

The proposed amendment modifies an existing CT for a dual-train SLC system inoperability. The condition evaluated, the action requirements, and the associated CT do not impact any initiating conditions for any accident previously evaluated.

The proposed amendment does not increase postulated frequencies or the analyzed consequences of an Anticipated Transient Without Scram (ATWS). Requirements associated with 10 CFR 50.62 will continue to be met. In addition, the proposed amendment does not increase postulated frequencies or the analyzed consequences or a large-break loss-of-coolant accident for which the SLC system will be used for pH control. The extended CT provides additional time to implement actions in response to a dual-train SLC system inoperability, while also minimizing the risk associated with continued operation. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed amendment revises TS 3.1.7 to extend the CT for Condition B from eight hours to 72 hours. The proposed amendment does not involve any change to plant equipment or system design functions. This proposed TS amendment does not change the design function of the SLC system and does not affect the system's ability to perform its design function. The SLC system provides a method to bring the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory to a subcritical condition with the reactor in the most reactive xenon free state without taking credit for control rod movement. Required actions and surveillance requirements are sufficient to ensure that the SLC system functions are maintained. No new accident initiators are introduced by this amendment. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The proposed amendment revises TS 3.1.7 to extend the CT for Condition B from eight hours to 72 hours. The proposed amendment does not involve any change to plant equipment or system design functions. The margin of safety is established through the design of the plant structures, systems, and components, the parameters within which the plant is operated, and the setpoints for the actuation of equipment relied upon to respond to an event.

The proposed amendment does not modify the condition or point at which SLC is initiated, nor does it affect the system's ability to perform its design function. In addition, the proposed change complies with the intent of the defense-in-depth philosophy and the principle that sufficient safety margins are maintained, consistent with RG 1.177 requirements (i.e., Section C, "Regulatory Position," paragraph 2.2, "Traditional Engineering Considerations").

Based on the above analysis, EGC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants"

10 CFR 50.62 (c)(4) states that boiling water reactors are required to have a standby liquid control (SLC) system with the capability of injecting, into the reactor pressure vessel (RPV), a borated water solution with a flow rate, boron concentration, and boron-10 enrichment that would be necessary to ensure that the resulting reactivity control is at least equivalent to that resulting from injection of 86 gallons per minute of 13 weight percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside diameter reactor pressure vessel for a given core design. Furthermore, the SLC system and its injection location must be designed to perform its function in a reliable manner. The proposed change will not impact the ability of the CPS SLC system to ensure compliance with these requirements.

10 CFR 50.67, "Accident source term"

10 CFR 50.67.b(1) provided guidance to licensees with respect to revision of the licensee's current accident source term in design basis radiological consequence analyses. Specifically, the regulation states that in order to revise the accident source term, a licensee shall apply for a license amendment under 10 CFR 50.90 and that the application shall contain an evaluation of the consequences of applicable design basis accidents previously analyzed in the safety analysis report.

By letter dated April 3, 2003, AmerGen Energy Company, LLC (i.e., the CPS licensee at that time) requested an amendment to the CPS TS regarding the adoption of an alternate source term (AST) methodology. The NRC approved the requested license amendment by letter and safety evaluation (SE) dated September 11, 2006. As part of the proposed AST methodology, EGC will use the SLC system to inject sodium pentaborate into the RPV following a LOCA in order to maintain suppression pool pH above 7 (i.e., in order to ensure against re-evolution of elemental iodine).

As such, the SLC will be required to be operable in Mode 3 to ensure that offsite doses remain within the limits of 10 CFR 50.67, "Accident source term" following a loss-of-coolant accident (LOCA) involving significant fission product releases. However, additional redundancy for the addition of boron into the reactor coolant system is established by CPS procedures. The procedures describe the actions and criteria for manual addition of boron into the Reactor Core Isolation Cooling (RCIC) system tank, and the use of the RCIC system to inject the boron into the RPV, should RPS, the control rods, the control rod drive system, and the SLC be unable to perform the specifed design functions. Therefore, the proposed SLC CT extension will not impact the ability of CPS to comply with the requirements of 10 CFR 50.67.

10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion (GDC) 26, "Reactivity control system redundancy and capability"

GDC 26 requires the provision of two independent reactivity control systems of different design principles. While one of the systems shall use control rods, the second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. The proposed change will not impact the ability of the CPS SLC system to ensure compliance with this requirement.

Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis"

RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decision-making: Technical Specifications"

RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 1

Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," specifies risk-informed acceptance guidelines for a permanent TS change. These acceptance guidelines are presented for two ranges of risk impacts, those described as "small changes" and those described as "very small changes."

The RG 1.174 acceptance guidelines prescribe that the risk metrics of delta (Δ) CDF and ΔLERF be less than 1.0E-06/yr and 1.0E-07/yr, respectively, to establish a very small risk increase with no additional compensatory measures required. RG 1.174, paragraph 2.2.5.5, "Comparisons with Acceptance Guidelines," also specifies guidelines for consideration of external events, and stipulates that external events can be evaluated in either a qualitative or quantitative manner.

RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decision-making: Technical Specifications," identifies a three-tiered approach for the evaluation of the risk associated with a proposed TS change.

RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 1 describes an acceptable approach for determining whether the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision-making for light-water reactors.

The proposed change complies with the acceptance guidelines and requirements of RG 1.174, RG 1.177, and RG 1.200 to demonstrate a very small change in risk.

Regulatory Summary

Based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the NRC's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

EGC has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, "Standards for Protection Against Radiation." However, the proposed amendment does not involve: (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," paragraph (c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. Letter from M. A. Satorius (U. S. NRC) to C. M. Crane (Exelon Generation Company, LLC), "Notice of Enforcement Discretion for Exelon Generation Company LLC Regarding Quad Cities Nuclear Power Station, Unit 1 (NOED 06-3-01)," dated October 18, 2006
2. Letter from M. A. Satorius (U. S. NRC) to C. M. Crane (Exelon Generation Company, LLC), "Notice of Enforcement Discretion for Exelon Generation Company LLC Regarding Dresden Nuclear Power Station, Unit 2 (NOED 07-3-01; TAC MD4044)," dated January 24, 2007

ATTACHMENT 2

Proposed Markup of CPS Technical Specification 3.1.7

TS Page

3.1-20

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

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ACTIONS

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

SURVEILLANCE REQUIREMENTS

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

(continued)

Attachment 3

Proposed Markup of CPS Technical Specification Bases B 3.1.7

TS Bases Page

B 3.1-40

B 3.1-43a

BASES

ACTIONS

A.1 (continued)

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

B.1

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If both SLC subsystems are inoperable, at least one subsystem must be restored to OPERABLE status within ~~8~~ 72 hours. The allowed Completion Time of ~~8~~ 72 hours is considered acceptable, given the low probability of a DBA or transient occurring concurrent with the failure of the control rods to shut down the reactor (Ref. 10).

C.1 and C.2

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

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1, SR 3.1.7.2, and SR 3.1.7.3

SR 3.1.7.1 through SR 3.1.7.3 are 24 hour Surveillances, verifying certain characteristics of the SLC System (i.e., the volume and temperature of the borated solution in the storage tank, and temperature of the pump suction piping), thereby ensuring the SLC System OPERABILITY without disturbing normal plant operation. These Surveillances ensure the proper borated solution and temperature, including the temperature of the pump suction piping, are maintained. Maintaining a minimum specified borated solution temperature is important in ensuring that the boron remains in solution and does not precipitate out in the

(continued)

BASES

REFERENCES

1. 10 CFR 50.62.
2. USAR, Section 9.3.5.3.
3. Calculation IP-0-0012.
4. Calculation IP-0-0013.
5. Calculation IP-0-0014.
6. Calculation IP-0-0015.
7. Calculation IP-0-0016.
8. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants, Final Report," February 1, 1995.
9. 10 CFR 50.67, "Accident Source Terms."
10. RM Documentation No. CL-LAR-01, Revision 1, "Risk Assessment Input for Clinton Technical Specification Change for Standby Liquid Control System Completion Time from 8 hours to 72 hours," December 28, 2009

Attachment 4

RM Documentation No. CL-LAR-01, Revision 1

RM DOCUMENTATION NO. CL-LAR-01	REV: 1	PAGE NO. 1

STATION: Clinton

UNIT(S) AFFECTED: N/A

TITLE: Risk Assessment Input for Clinton Technical Specification Change for Standby Liquid Control System Completion Time from 8 hours to 72 hours

SUMMARY: This assessment is performed in support of the license amendment request (LAR) submittal to extend the Technical Specification 3.1.7, Condition B Completion Time (CT) for the Standby Liquid Control (SLC) System from 8 hours to 72 hours.

The risk assessment is performed in accordance with ER-AA-600-1046, Rev. 4, Risk Metrics – NOED and LAR

No UREs have been created as a result of this application.

[ ] Review required after periodic Update

[X] Internal RM Documentation **[ ] External RM Documentation**

Electronic Calculation Data Files:

<u>Method of Review:</u> [X] Detailed [ ] Alternate [ ] Review of External Document

This RM documentation supersedes: CL-LAR-01 Rev 0 in its entirety.

		Sign	Date
Prepared by:	L. K. Lee/R A Narain	/ [signature] [signature]	12/23/09 12/10/09
Reviewed by:	R A Hill	[signature]	12/10/09
Reviewed by:	G A Teagarden	[signature]	12/10/2009
Reviewed by:	V M Andersen (Model Owner)	[signature]	12/23/09
Approved by:	E T Burns	[signature]	12-28-09

TABLE OF CONTENTS

1.0 INTRODUCTION

1.1 PURPOSE

The purpose of this analysis is to assess the acceptability, from a risk perspective, of a change to the Clinton Technical Specification (TS) for the Standby Liquid Control (SLC) system to increase the Completion Time (CT), sometimes called the allowed outage time (AOT), from 8 hours to 72 hours when both SLC subsystems (i.e., both trains) are inoperable. An extension will provide flexibility during power operation in the performance of corrective maintenance, preventive maintenance, and surveillance testing of SLC system components that would cause the system to be inoperable. Consistent with the NRC's approach to risk-informed regulation, Exelon Generating Company (EGC) has identified a particular TS requirement that is very restrictive in its nature and, if relaxed, has a minimal impact on the safety of the plant. The Clinton analysis is consistent with similar analyses being conducted for all EGC Boiling Water Reactor (BWR) plants that currently have an 8 hour CT for the SLC system.

1.2 BACKGROUND

1.2.1 Technical Specification Changes

Since the mid-1980s, the NRC has been reviewing and granting improvements to TS that are based, at least in part, on probabilistic risk assessment (PRA) insights. In its final policy statement on TS improvements of July 22, 1993, the NRC stated that it: . . .

> *. . . expects that licensees, in preparing their Technical Specification related submittals, will utilize any plant-specific PSA or risk survey and any available literature on risk insights and PSAs. . . Similarly, the NRC staff will also employ risk insights and PSAs in evaluating Technical Specifications related submittals. Further, as a part of the Commission's ongoing program of improving Technical Specifications, it will continue to consider methods to make better use of risk and reliability information for defining future generic Technical Specification requirements.*

The NRC reiterated this point when it issued the revision to 10 CFR 50.36, "Technical Specifications," in July 1995. In August 1995, the NRC adopted a final policy statement on the use of PRA methods in nuclear regulatory activities that encouraged greater use of PRA to improve safety decision-making and regulatory efficiency. The PRA policy statement included the following points:

1. The use of PRA technology should be increased in all regulatory matters to the extent supported by the state of the art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.

2. PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state of the art, to reduce unnecessary conservatism associated with current regulatory requirements.

3. PRA evaluations in support of regulatory decisions should be as realistic as practicable and appropriate supporting data should be publicly available for review.

The movement of the NRC to more risk-informed regulation has led to the NRC identifying Regulatory Guides and associated processes by which licensees can submit changes to the plant design basis including Technical Specifications. Regulatory Guides 1.174 [Ref. 2] and 1.177 [Ref. 3] both provide processes to incorporate PRA input for decision makers regarding a Technical Specification modification.

Clinton, other EGC plants, and numerous other commercial nuclear plants in the industry have used these risk-informed guidelines to support both permanent and one-time CT extensions for EDGs and other systems.

1.2.2 Exelon SLC Experiences

In October 2006 (Quad Cities) and January 2007 (Dresden), EGC requested Notices of Enforcement Discretion (NOEDs) for SLC System Tank leaks allowing an additional 72 hours to the original 8-hour completion time required for a dual-train inoperability. These NOEDs were approved by the NRC. An extended CT would preempt the need for such NOEDs.

1.3 SLC TECHNICAL SPECIFICATIONS

The proposed TS change involves extending the completion time for TS 3.1.7 Condition B from 8 hours (current TS) to 72 hours (proposed TS). Condition B is the situation where both SLC subsystems are inoperable. Technical Specification requirements for other SLC conditions will remain unchanged. For Clinton the TS Condition B applies to Modes 1 and 2 for reactivity control. Consideration of TS applicability for Modes 1, 2, and 3 for pH control is not addressed in this report.

1.4 REGULATORY GUIDES

Three Regulatory Guides provide primary inputs to the evaluation of a Technical Specification change. Their relevance is discussed in this section.

1.4.1 Regulatory Guide 1.174

Regulatory Guide 1.174 [Ref. 2] specifies an approach and acceptance guidelines for use of PRA in risk informed activities. RG 1.174 outlines PRA related acceptance guidelines for use of PRA metrics of Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) for the evaluation of permanent TS changes. The guidelines given in RG 1.174 for determining what constitutes an acceptable permanent change specify that the ΔCDF and the ΔLERF associated with the change should be less than specified values, which are dependent on the baseline CDF and LERF, respectively. These specified values of ΔCDF and ΔLERF are given in RG 1.174 Figures 3 and 4, respectively. These values are presented for two ranges of risk impacts, those described as "small changes" and those described as "very small changes". The acceptance guidelines for "very small changes" are utilized in this risk assessment.

Based on the CL06C baseline internal events CDF of 5.6E-6/yr and LERF of 1.2E-7/yr for Clinton, the RG 1.174 acceptance guidelines prescribe that the risk metrics of ΔCDF and ΔLERF be less than 1.0E-06/yr and 1.0E-07/yr, respectively, to establish a very small risk increase with no additional compensatory measures required.

RG 1.174 also specifies guidelines for consideration of external events. External events can be evaluated in either a qualitative or quantitative manner.

1.4.2 Regulatory Guide 1.177

Regulatory Guide 1.174 [Ref. 2] specifies an approach and acceptance guidelines for the evaluation of plant licensing basis changes. RG 1.177 identifies a three-tiered approach for the evaluation of the risk associated with a proposed TS change as identified below:

- Tier 1 is an evaluation of the plant-specific risk associated with the proposed TS change, as shown by the change in core damage frequency (CDF) and incremental conditional core damage probability (ICCDP). Where applicable, containment performance should be evaluated on the basis of an analysis of large early release frequency (LERF) and incremental conditional large early release frequency (ICLERP). The acceptance guidelines given in RG 1.177 for determining an acceptable TS change is that the ICCDP and the ICLERP associated with the change should be less than 5E-07 and 5E-08, respectively.
- Tier 2 identifies and evaluates, with respect to defense-in-depth, any potential risk-significant plant equipment outage configurations associated with the proposed change. The licensee should provide reasonable assurance that risk-significant plant equipment outage configurations will not occur when equipment associated with the proposed TS change is out-of-service.

- Tier 3 provides for the establishment of an overall configuration risk management program (CRMP) and confirmation that its insights are incorporated into the decision-making process before taking equipment out-of-service prior to or during the CT. Compared with Tier 2, Tier 3 provides additional coverage based on any additional risk significant configurations that may be encountered during maintenance scheduling over extended periods of plant operation. Tier 3 guidance can be satisfied by the Maintenance Rule (10 CFR 50.65(a)(4)), which requires a licensee to assess and manage the increase in risk that may result from activities such as surveillance, testing, and corrective and preventive maintenance.

This risk analysis supports the Tier 1 element of RG 1.177, specifically the acceptance guidelines for ICCDP and ICLERP for permanent changes associated with changing a Technical Specification Completion Time. Other portions of the LAR submittal will address Tier 2 and Tier 3 elements.

1.4.3 Regulatory Guide 1.200, Revision 1

Regulatory Guide 1.200, Rev. 1 [Ref. 1], describes an acceptable approach for determining whether the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision-making for light-water reactors. This guidance is intended to be consistent with the NRC's PRA Policy Statement and more detailed guidance in Regulatory Guide 1.174.

It is noted that RG 1.200 Rev. 1 endorses Addendum B of the ASME PRA Standard [Ref. 5] applicable to full power internal event (FPIE) PRA models. Since that time, the new ASME/ANS Combined PRA Standard [Ref. 26] has been released. Although the Combined Standard is presently issued and endorsed by RG 1.200 Revision 2 [Ref. 27], neither of these document revisions impact this analysis.

1.4.4 Acceptance Criteria

Based on the guidance provided in Regulatory Guides 1.174 and 1.177, the following quantitative PRA related acceptance criteria are utilized in this risk analysis:

- ΔCDF < 1.0E-06/yr
- ΔLERF < 1.0E-07/yr
- ICCDP < 5.0E-07
- ICLERP < 5.0E-08

1.5 SCOPE

This section addresses the requirements of RG 1.200, Rev. 1 Section 3.2, which directs the licensee to define the treatment of the scope of risk contributors (i.e., internal initiating events, external initiating events, and modes of power operation at the time of the initiator). Discussion of these risk contributors are as follows:

- Full Power Internal Events (FPIE) – The Clinton CL06C PRA model used for this analysis includes a full range of internal initiating events (including internal flooding) for at-power configurations. The SLC system is credited in the PRA for criticality control. The FPIE model is further discussed in Section 1.6.
- Low Power Operation - The FPIE assessment is judged to adequately capture risk contributors associated with low power plant operations. The FPIE analysis assumes that the plant is at full power at the time of any internal events transient, manual shutdown, or accident initiating event. This analytic approach results in conservative accident progression timings and systemic success criteria compared to what may otherwise be applicable to an initiator occurring at low power. As such, low power risk impacts are not discussed further in this risk assessment.
- Shutdown / Refueling – In consideration of shutdown and refueling modes (i.e., Modes 3, 4, and 5), the SLC TS does not apply. As such, shutdown risk impacts are not discussed further in this risk assessment.
- Internal Fires - An interim fire PRA is available for Clinton. The Clinton Interim Fire PRA [Ref. 10], and a BWROG assessment [Ref. 19] are used to provide qualitative and semi-quantitative insights to the analysis (refer to Section 3.4.1).
- Seismic - Consistent with most sites, Clinton does not currently maintain a Seismic PRA. A qualitative assessment is performed in this analysis (refer to Section 3.4.2) based on insights from the Clinton IPEEE study [Ref. 11] and other industry studies.
- Other External Events - Other external event risks were assessed in the Clinton IPEEE study [Ref. 11] and found to be insignificant risk contributors (refer to Sections 3.4.3 and 3.4.4).

1.6 CLINTON PRA MODEL

This section addresses the requirements of Section 3.1 of RG 1.200, Rev. 1 which directs the licensee to identify the portions of the PRA used in the analysis.

The PRA analysis for the TS change uses the Clinton CL06C full power internal events Level 1 Core Damage Frequency (CDF) model and the associated Level 2 Large Early Release Frequency (LERF) model to calculate the risk metrics. This analysis is specific to the SLC system and therefore the SLC system fault tree model is the only portion of the CL06C PRA model modified for this risk application. The Clinton SLC system is a manually initiated system with two SLC pump required to meet the10 CFR 50.62 requirements for ATWS response. The PRA analysis involved identifying the system and components or maintenance activities modeled in the PRA which are most appropriate for use in setting both subsystems of SLC to be inoperable. As discussed later in Section 3.1, the model parameter 1SC-1A-1B----M-- "SBLC A AND SBLC B IN COINCIDENT MAINTENANCE," was selected as an appropriate parameter to adjust to make the entire SLC system unavailable in the PRA (to reflect SLC inoperable and entry into TS 3.1.7, Condition B).

No other aspect of the CL06C PRA model required adjustment for this risk application. The entire CL06C PRA model is quantified for this assessment using the "average maintenance" PRA model (i.e., no portions of the at-power internal events CL06C model were excluded or zeroed out of the quantification).

2.0 ANALYSIS ROADMAP AND REPORT ORGANIZATION

The analysis and documentation utilizes the guidance provided in RG 1.200, Revision 1 [Ref. 1]. Table 2-1 summarizes the RG 1.200 identified actions and the corresponding location of that analysis or information in this report.

Table 2-1

RG 1.200 ANALYSIS ACTIONS ROADMAP

RG 1.200 Actions	Report Section
1. Identify the parts of the PRA used to support the application	Section 3
1.a Systems, structures and components (SSCs), operational characteristics affected by the application, and how these are implemented in the PRA model	Section 3.2
1.b Acceptance criteria used for the application	Section 1.4.4
2. Identify the scope of risk contributors addressed by the PRA model. If not full scope (i.e., internal and external events), identify appropriate compensatory measures or provide bounding arguments to address the risk contributors not addressed by the model.	Section 1.5
3. Summarize the risk assessment methodology used to assess the risk of the application. Include how the PRA model was modified to appropriately model the risk impact of the change request.	Section 3
4. Demonstrate the technical adequacy of the PRA.	Section 4
4.a Identify plant changes (design or operational practices) that have been incorporated at the site, but are not yet in the PRA model and justify why the change does not impact the PRA results used to support the application.	Section 4.6.1
4.b Document that the parts of the PRA used in the decision are consistent with applicable standards endorsed by the RG (currently, in RG 1.200 Rev. 1. RG 1.200 Rev. 1 addresses the internal events ASME PRA standard). Provide justification to show that where specific requirements in the standard are not met, it will not unduly impact the results.	Section 4.6
4.c Document PRA peer review findings and observations that are applicable to the parts of the PRA required for the application, and for those that have not yet been addressed justify why the significant contributors would not be impacted.	Section 4.5
4.d Identify key assumptions and approximations relevant to the results used in the decision-making process.	Section 3.1

3.0 TIER 1 RISK ASSESSMENT

This section evaluates the plant-specific risk associated with the proposed TS change, based on the risk metrics of CDF, ICCDP, LERF, and ICLERP.

3.1 KEY ASSUMPTIONS

The following inputs and general assumptions are used in estimating the plant risk due to the proposed SLC System CT extension.

a. The SLC System CT is assumed to increase from its current duration of 8 hours to a proposed duration of 72 hours.

b. The base analysis in this risk assessment assumes one entry per year into the proposed CT. The duration of the proposed CT is assumed to be adequate for performing the majority of corrective maintenance, preventive maintenance, and surveillance testing on-line. An examination of SLC rolling unavailability for the past 24 months as of June 22, 2009 revealed that SLC Trains A and B were not both unavailable. Train A had unavailable 2.7 hours, and Train B had not been unavailable. Thus, any impact from extending the CT is assumed to be negligible, and it is conservatively assumed that the outage will not be entered more than once a year. Additionally, Configuration Risk Management at Clinton is governed by the Maintenance Rule (10 CFR 50.65(a)(4)). A sensitivity analysis of the risk associated with entering the CT was performed, and indicated that the SLC system could be taken out of service for up to 582 hours before the very small risk increase metrics of RG 1.174 and RG 1.177 are exceeded. This represents a significant margin compared to the proposed 72 hour CT. As stated above, the historical analysis of unavailability data shows that the SLC system does not exceed this ceiling value.

c. This risk assessment does not credit the averted risk due to a forced shutdown that would be required due to exceeding the existing CT.

3.2 INTERNAL EVENTS

The Clinton 2006C PRA model[1] [Ref. 4] was examined to determine which PRA basic event to modify to reflect the coincident unavailability of both SLC subsystems. The applicable basic event for the 2006C PRA model was identified as 1SC-1A-1B----M-- "SBLC A AND SBLC B IN COINCIDENT MAINTENANCE." This event is appropriate because it fails both SLC subsystems and no other equipment in the model.

Event 1SC-1A-1B----M-- was set to a binary logic value of "TRUE" (using a quantification flag file) and the entire CL06C model was requantified using the same PRA software codes and revisions as used for the base CL06C model [Ref. 4]. These configuration specific CDF and LERF values are used in conjunction with the base CL06C values to calculate the risk impacts of the proposed TS change.

The calculations of ΔCDF, ICCDP, ΔLERF and ICLERP for the CT change are determined as shown below.

The ΔCDF to be compared to the RG 1.174 acceptance guidelines is given by (as defined by [Ref. 21]):

$$\Delta CDF = CDF_{NEW} - CDF_{BASE} \qquad \text{[Equation 3-1]}$$

ΔCDF is the difference between the annual average CDF with the CT extended and the CDF with the current CT. The ΔCDF has units of "per reactor year."

In the above equation, CDF_{NEW} is equal to:

$$CDF_{NEW} = CT_{SLC\text{-}OOS} * CDF_{SLC\text{-}OOS} + [(1-CT_{SLC\text{-}OOS}) * CDF_{BASE}] \qquad \text{[Equation 3-2]}$$

Where:

$CDF_{SLC\text{-}OOS}$ = the annual average CDF calculated with both SLC subsystems out of service (1SC-1A-1B----M-- set to TRUE)

CDF_{BASE} = baseline annual average CDF with average unavailability for all equipment. This is the CDF result of the CL06C baseline PRA.

$CT^{H}_{SLC-OOS}$ = the new extended CT as an annual unavailability (i.e., 72 hours / 8760 hours/yr = 8.2E-03 yr)

[1] The CL06C baseline model used in the calculations contains the average maintenance associated with system trains.

$CT_{SLC\text{-}OOS}$ = the new extended CT as a probability (i.e., 72 hours / 8760 hours = 8.2E-03)

The ICCDP associated with the SLC System being out of service using the new CT is given by:

$$ICCDP^{(1)} = (CDF_{SLC\text{-}OOS} - CDF_{BASE}) \times CT^{H}_{SLC-OOS} \qquad \text{[Equation 3-3]}$$

Risk significance relative to ΔLERF and ICLERP[1] is determined using equations of the same form as noted above for ΔCDF and ICCDP.

The relevant input parameters for the base quantification of this risk analysis are summarized in Table 3.2-1. The corresponding base risk metric results for this risk analysis (based on quantification of the CL06C model and use of the above equations) are provided in Table 3.2-2.

Table 3.2-1

RISK ASSESSMENT INPUT PARAMETERS

Input Parameter	Value	Reference
CDF_{BASE}	5.6E-06/yr	CL06C PRA [Ref. 4]
$LERF_{BASE}$	1.2E-07/yr	CL06C PRA [Ref. 4]
$CT_{SLC\text{-}OOS}$	8.2E-03	One 72-hr TS 3.1.7 Condition B entry assumed per year (i.e., 72 hr/8760 hrs).

(1) ICCDP and ICLERP are probabilities, i.e. no units.

Table 3.2-2

RISK ASSESSMENT BASE RESULTS

Risk Metric	Value	Acceptance Guidelines
$CDF_{SLC\text{-}OOS}$	9.1E-6/yr	N/A
CDF_{NEW}	5.6E-6/yr	N/A
ΔCDF	2.9E-08/yr	<1.0E-06/yr
ICCDP	2.9E-08	<5.0E-07
$LERF_{SLC\text{-}OOS}$	8.7E-7/yr	N/A
$LERF_{NEW}$	1.3E-7/yr	N/A
ΔLERF	6.2E-09/yr	<1.0E-07/yr
ICLERP	6.2E-09	<5.0E-08

3.3 RESULTS COMPARISON TO ACCEPTANCE GUIDELINES

As can be seen from Table 3.2-2, the base results of the risk assessment indicate that the ΔCDF, ICCDP, ΔLERF, and ICLERP risk metric values are below the acceptance guidelines as defined in RG 1.174 and RG 1.177. In addition quantitative sensitivity cases for model uncertainties are provided in Appendix B.

This analysis demonstrates that the proposed TS change satisfies the risk acceptance guidelines in RG 1.174 and RG 1.177, and therefore meets the intent of very small risk increases consistent with the Commission's Safety Goal Policy Statement.

A sensitivity analysis was performed to determine the maximum allowable CT before exceeding the acceptance criteria for very small risk increases. For this sensitivity, ICCDP and ICLERP were set to their maximum allowable values in RG 1.177, and the CT_{NEW} allowable was calculated. ICLERP was determined to be the bounding parameter, and a CT_{NEW} of 582 hours was calculated. This represents a significant margin compared to the proposed 72 hour CT.

3.4 EXTERNAL EVENTS

A qualitative assessment of external event risks is provided. Further details are found in Appendix A.

3.4.1 Internal Fires

The impact on the internal fires risk profile due to the proposed CT is evaluated using the following information sources:

- NUREG/CR-6850 [Ref. 18]
- Clinton Interim FPRA [Ref. 10]
- BWROG Assessment of Fire-Induced Failure to Scram [Ref. 19]

The internal fires risk impact assessment is discussed in Appendix A.4. The assessment concluded that fire hazards can be appropriately screened as non-significant contributors to the risk assessment of the proposed SLC CT because of the low frequency of a fire coupled with a failure to scram.

3.4.2 Seismic

EGC does not currently maintain a seismic PRA for Clinton. The impact on the seismic risk profile due to the proposed CT is evaluated using the following information sources:

- CPS IPEEE [Ref. 11]
- NUREG-1150 [Ref. 23]

The seismic risk impact assessment is discussed in Appendix A.3. The assessment concluded that seismic risk can be appropriately screened as a non-significant contributor to the risk assessment of the proposed CT.

3.4.3 Other External Hazards

Other external event risks such as external floods, severe weather, high winds or tornados, transportation accidents, nearby facility accidents, turbine missiles, and other miscellaneous external hazards were also considered in the IPEEE analysis. The Clinton site characteristics and design meet all the applicable criteria of the NRC Standard Review Plan (SRP). No significant quantitative contribution from these external events was identified by IPEEE evaluations (refer to Appendix A.2).

As such, other external hazards are appropriately screened as non-significant contributors to the risk assessment of the proposed CT.

3.5 UNCERTAINTY ASSESSMENT

3.5.1 Parametric Uncertainty

Consistent with the ASME PRA Standard, quantitative parametric uncertainty analyses for both CDF and LERF are evaluated to determine if the point estimates calculated by the PRA model appropriately represent the mean. The results of these analyses are summarized in Appendix B.3.

The parametric uncertainty analysis shown in Appendix B.3 supports the use of the point estimate to represent the mean for the calculation of the changes in the risk metrics for the extended CT.

3.5.2 Modeling Uncertainty

An assessment of modeling uncertainty is documented in Sections B.1 and B.2. The results of these modeling uncertainty assessments are judged not to change the conclusions of this risk assessment for the proposed SLC CT change as they do not directly impact the SLC system or ATWS scenarios.

- Section B.1 provides the Clinton specific modeling uncertainty evaluations for the Base Case.
- Section B.2 provides an examination of the specific cutsets that affect the change in the CDF risk metric associated with the change in the SLC CT.

The results of these modeling uncertainty assessments do not change the conclusions of this risk assessment for the proposed SLC CT change.

3.6 RISK SUMMARY

As discussed above and as summarized in Table 3.6-1, the FPIE quantitative evaluation results are well below the risk acceptance guidelines of RG 1.174 and RG 1.177. External events evaluations are discussed in Appendix A and do not change the results or conclusions of this risk assessment. As such, this risk evaluation demonstrates that the proposed TS change can be made with a very small risk increase.

Table 3.6-1

RISK ASSESSMENT SUMMARY RESULTS

Hazard	ΔCDF	ICCDP	ΔLERF	ICLERP
FPIE	2.9E-08/yr	2.9E-08	6.2E-09/yr	6.2E-09
Acceptance Criteria	<1.0E-06/yr	<5.0E-07	<1.0E-07/yr	<5.0E-08
Fire	(1)	(1)	(1)	(1)
Seismic	(1)	(1)	(1)	(1)

(1) Evaluated and determined not to change the conclusions of the FPIE risk analysis.

4.0 TECHNICAL ADEQUACY OF PRA MODEL

The 2006C update to the Clinton PRA model (CL06C) is the most recent evaluation of the risk profile at Clinton for FPIE challenges. The Clinton PRA modeling is highly detailed, including a wide variety of initiating events, modeled systems, operator actions, and common cause events. The PRA model quantification process used for the Clinton PRA is based on the event tree / fault tree methodology, which is a well-known methodology in the industry.

EGC employs a multi-faceted approach to establishing and maintaining the technical adequacy and plant fidelity of the PRA models for all operating EGC nuclear generation sites. This approach includes both a proceduralized PRA maintenance and update process, and the use of self-assessments and independent peer reviews. The following information describes this approach as it applies to the Clinton PRA.

4.1 PRA QUALITY OVERVIEW

The quality of the Clinton FPIE PRA is important in making risk-informed decisions. The importance of the PRA quality derives from NRC Policy Statements as implemented by RGs 1.174 and 1.177, rule making and oversight processes. These can be briefly summarized as follows using the words of the NRC Policy Statement (1995):

1. *"The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art...and supports the NRC's traditional defense-in-depth philosophy."*

2. *"PRA...should be used in regulatory matters...to reduce unnecessary conservatism..."*

3. *"PRA evaluations in support of regulatory decisions should be...realistic...and appropriate supporting data should be publicly available for reviews."*

4. *"The Commission's safety goals...and subsidiary numerical objectives are to be used with appropriate consideration of uncertainties in making regulatory judgments..."*

5. *"Implementation of the [PRA] policy statement will improve the regulatory process in three ways:*
   - *Foremost, through safety decision making enhanced by the use of PRA insights;*
   - *Through more efficient use of agency resources; and*
   - *Through a reduction in unnecessary burdens on licensees."*

PRA quality is an essential aspect of risk-informed regulatory decision making. In this context, PRA quality can be interpreted to have five essential elements:

- Scope (Section 4.2): The scope (i.e., completeness) of the FPIE PRA. The scope is interpreted to address the following aspects:
  - Challenges to plant operation (Initiating Events):
    - Internal Events (including Internal Floods)
    - External Hazards
    - Fires
  - Plant Operational states:
    - Full Power
    - Low Power
    - Shutdown
  - The metrics used in the quantification:
    - Level 1 PRA – CDF
    - Level 2 PRA – LERF
    - Level 3 PRA – Health Effects
- Fidelity (Section 4.3): The fidelity of the PRA to the as-built, as-operated plant.
- Standards (Section 4.4): ASME/ANS PRA Standard [Ref. 5] as endorsed by the NRC in Regulatory Guide 1.200 [Ref. 1].
- Peer Review (Section 4.5): An independent PRA peer review provides a method to examine the PRA process by a group of experts. In some cases, a PRA self-assessment using the available PRA Standards endorsed by the NRC can be used to replace or supplement this peer review.
- Appropriate Quality (Section 4.6): The quality of the PRA needs to be commensurate with its application. In other words, the needed quality is defined by the application requirements.

4.2 SCOPE

The Clinton PRA is a full power, internal events (FPIE) PRA that addresses both CDF and LERF. The quantitative insights from the FPIE PRA are directly applicable to the SLC CT Extension PRA application. This scope is judged to be adequate to support the SLC CT PRA application.

Because not all PRA standards are available to define the appropriate elements of PRA quality for all applications, the NRC has adopted a phased implementation approach. This phased approach uses available PRA tools and their quantitative results where standards are available and endorsed by the NRC. Where standards are not yet available or endorsed, this approach uses qualitative insights or bounding approaches as needed.

The quality assessment performed in this section confirms the adequacy of the FPIE PRA. This assessment does not address the risk implications associated with low power or shutdown operation or with external events (including fire).

4.3 FIDELITY: PRA MAINTENANCE AND UPDATE

The EGC risk management process for maintaining and updating the PRA ensures that the PRA model remains an accurate reflection of the as-built and as-operated plants. This process is defined in the EGC Risk Management program, which consists of a governing procedure (ER-AA-600, "Risk Management") and subordinate implementation procedures. EGC procedure ER-AA-600-1015, "FPIE PRA Model Update" delineates the responsibilities and guidelines for updating the full power internal events PRA models at all operating EGC nuclear generation sites. The overall EGC Risk Management program, including ER-AA-600-1015, defines the process for implementing regularly scheduled and interim PRA model updates, for tracking issues identified as potentially affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operating experience), and for controlling the model and associated computer files. To ensure that the current PRA model remains an accurate reflection of the as-built, as-operated plants, the following activities are routinely performed:

- Design changes and procedure changes are reviewed for their impact on the PRA model.
- New engineering calculations and revisions to existing calculations are reviewed for their impact on the PRA model.
- Maintenance unavailabilities are captured, and their impact on CDF is trended.
- Plant specific initiating event frequencies, failure rates, and maintenance unavailabilities are updated approximately every four years.

In addition to these activities, EGC risk management procedures provide the guidance for particular risk management and PRA quality and maintenance activities. This guidance includes:

- Documentation of the PRA model, PRA products, and bases documents.
- The approach for controlling electronic storage of Risk Management (RM) products including PRA update information, PRA models, and PRA applications.
- Guidelines for updating the full power, internal events PRA models for EGC nuclear generation sites.

- Guidance for use of quantitative and qualitative risk models in support of the On-Line Work Control Process Program for risk evaluations for maintenance tasks (corrective maintenance, preventive maintenance, minor maintenance, surveillance tests and modifications) on systems, structures, and components (SSCs) within the scope of the Maintenance Rule (10CFR50.65 (a)(4)).

In accordance with this guidance, regularly scheduled PRA model updates nominally occur on a four year cycle; shorter intervals may be required if plant changes, procedure enhancements, or model changes result in significant risk metric changes.

4.4 STANDARDS

The ASME PRA Standard [Ref. 5] provides the basis for assessing the adequacy of the Clinton PRA as endorsed by the NRC in RG 1.200, Rev. 1 [Ref. 1]. The predecessor to the ASME PRA Standard was NEI 00-02 which identified the critical internal events PRA elements and their attributes necessary for a quality PRA.

4.5 PEER REVIEW AND PRA SELF-ASSESSMENT

There are three principal ways of incorporating the necessary quality into the PRA in addition to the maintenance and update process. These are the following:

- A thorough and detailed investigation of open issues and the implementation of their resolution in the PRA. Table 4-1 includes the continuing investigations by EGC of plant modifications and changes that could influence the risk spectrum.
- A PRA Peer Review to allow independent reviewers from outside to examine the model and documentation. The ASME PRA Standard [Ref. 5] specifies that a PRA Peer Review be performed on the PRA.
- The use of the ASME PRA Standard to define the criteria to be used in establishing the quality of individual PRA elements.

Several assessments of technical capability have been made and continue to be planned for the Clinton PRA model. A chronological list of the assessments performed includes the following:

- An independent PRA peer review was conducted under the auspices of the BWR Owners Group (BWROG) in 2000, following the Industry PRA Peer Review process [Ref. 6]. This peer review included an assessment of the PRA model maintenance and update process.

- A self-assessment analysis was performed against Addenda B of the ASME PRA Standard and the draft of Revision 1 of Regulatory Guide 1.200 (DG-1161).
- During 2005 and 2006 the CPS PRA model results were evaluated in the BWROG PRA cross-comparisons study performed in support of implementation of the mitigating systems performance indicator (MSPI) process.
- A current industry peer review of the Clinton PRA was conducted in the fourth quarter of 2009. Results of this review are still being processed.

A summary of the disposition of the BWROG PRA Peer Review facts and observations (F&Os) for the Clinton PRA models was documented as part of the statement of PRA capability for MSPI in the Clinton MSPI Basis Document [Ref. 7]. As noted in that document, all five (5) of the significance level "A" F&Os have been resolved and eighty-nine (89) of the ninety-two (92) significance level "B" F&Os have been resolved. The remaining three (3) open significance level "B" F&Os are insignificant.

4.5.1 Self-Assessment Overview

A Self-Assessment of the 2003 CPS PRA was performed in support of the CPS 2006 PRA Update. This Gap Analysis was performed using Addenda B of the ASME PRA Standard (ASME RA-Sb-2005) and the draft of Revision 1 Regulatory Guide 1.200 (DG-1161). Potential gaps to Capability Category II of the Standard were identified and used to plan the Clinton 2006 PRA Update. Table 4-3 presents a discussion of the identified gaps and concludes that none impact this application.

4.5.2 PRA Peer Review Overview

Table 4-2 presents the open significant PRA Peer Review findings. PRAs can be used in applications despite not meeting all of the Supporting Requirements of the ASME/ANS PRA Standard. This is well recognized by the NRC and is explicitly stated in the ASME/ANS PRA Standard and RG 1.174. RG 1.174 states the following in Section 2.2.6:

> *There are, however, some applications that, because of the nature of the proposed change, have a limited impact on risk, and this is reflected in the impact on the elements of the risk model.*

The proposed SLC CT Extension PRA application may not require more than Capability Category I for some SRs. It is also acknowledged that for PRAs with SRs ranked as "Not Met," the PRA may be used for PRA applications but may require additional justification and support to allow their use. Finally, it is judged that no PRA has

Capability Category III for all of its SRs, nor is this currently expected as part of the NRC PRA Quality Program.

4.6 APPROPRIATE PRA QUALITY

The PRA is used within its limitations to augment the deterministic criteria for plant operation. This is confirmed by the PRA Peer Review and the PRA Self-Assessment. As indicated previously, RG 1.200 also requires that additional information be provided as part of the License Amendment Request (LAR) submittal to demonstrate the technical adequacy of the PRA model used for the risk assessment. Each of these items (plant changes not yet incorporated in to the PRA model, consistency with applicable PRA Standards, relevant peer review findings, and the identification of key assumptions) is discussed below.

4.6.1 Plant Changes Not Yet Incorporated into the PRA Model

A PRA updating requirements evaluation (URE) is EGC's PRA model update tracking database. These UREs are created for all issues that are identified with a potential to impact the PRA model. The URE database includes the identification of those plant changes that could impact the PRA model. A review of the current open items in the URE database associated with plant changes for Clinton as well as items related to SLC or ATWS modeling is summarized in Table 4-1 along with an assessment of the impact for this application.

The results of the assessment documented in Table 4-1 is that none of the plant changes have any measurable impact on the SLC CT extension request.

4.6.2 Consistency with Applicable PRA Standards

As indicated above, an independent peer review of the Clinton PRA was performed in 2000 following the review guidelines of NEI 00-02 (the predecessor to the ASME PRE Standard). All of the significance level "A" F&Os have been resolved and all but three (3) of the significance level "B" have been resolved. The three open significance level "B" F&Os from the peer review are summarized in Table 4-2 along with an assessment of the impact for this application. None of the three are found to impact this application.

4.6.3 Relevant PRA Peer Review Findings

As indicated above, a current industry peer review of the Clinton PRA is scheduled for the fourth quarter of 2009. However, a self-assessment against the PRA Standard and draft RG 1.200 was performed in support of the CPS 2006 PRA Update. Potential gaps to Capability Category II of the Standard were identified for treatment in the 2006 PRA Update. The identified gaps that were not closed by the 2006 PRA Update are summarized in Table 4-3 along with an assessment of the impact for this application.

Of the gaps identified and evaluated in Table 4-3, none have a measurable impact on the SLC CT extension request.

Table 4-1

IMPACT ON THE CLINTON PRA MODEL OF PLANT CHANGES SINCE THE LAST PRA UPDATE

URE Number	Plant change	Impact on the CL PRA	Impact on the Application
CL2005-002	This EC replaces the main feed breaker for DC MCC 1DC16E with a switch and fuses combination that will fit into the existing DC breaker cubicle. The normally closed breaker is modeled in the PRA, basic event DDC1E1ACBD (prob 2.4E-5), although the breaker has no modeled function other than to remain closed. This BE represents an inadvertent change of state for the breaker. A combination of switches and fuses could have a somewhat different failure rate, than a DC breaker. However because in both cases a passive function is modeled the failure rate should be very small.	Non-significant impact.	No Impact
CL2005-003	The RAT replacement project in which a single Reserve Auxiliary Transformer is being replaced with 3 RATs will also result in changes to the low side buswork. This is generally covered under EC 339047 "RAT Replacement Project-Transformers, Non-Seg Bus, Relay Panels, Power and Control Cabling for RAT Low Voltage Side". The PRA model needs to have the offsite power logic split to correspond to its RAT supply. This is covered by URE CL2005-001. Another feature of this change is that the fault protection scheme for the bus ductwork and for the new RATs is being modified. This is generally below the level of detail considered in PRA models and can be considered in the category of events that would cause inadvertent opening (or failure) of the common RAT feeds.	The plant change has yet to be implemented.	No Impact

Table 4-1

IMPACT ON THE CLINTON PRA MODEL OF PLANT CHANGES SINCE THE LAST PRA UPDATE

URE Number	Plant change	Impact on the CL PRA	Impact on the Application
CL2006-003	The 2006 CPS PRA update incorporated mitigation strategies from Rev. 1 of CPS 4303.01, "Loss of Ultimate Heat Sink". Subsequent to the 2006 PRA, CPS 4303.01 Rev 2 was issued and is enhanced to cover several situations that may be useful for addressing PRA accident sequences. Noteworthy topics include: Running RCIC without DC power, Starting Diesels without DC power for field flashing or air start solenoids, Containment Venting without AC power available, SRV operation with external DC power supplies and alternate DC power strategies. These mitigation strategies should be considered for their risk reduction benefit in a future PRA update.	Non-significant impact.	No Impact
CL2006-004	The EOP flowcharts CPS 4401.01, 4402.01, 4403.01 and 4405.01 have been revised from the previous revision 26.	Non-significant impact.	No Impact
CL2006-005	There have been a few changes in In Service Test (IST) Intervals. For example CPS 9054.02 RCIC Valve Operability has been modified to reduce the testing interval for 1E51F063 and 1E51F064 from quarterly to cold shutdown test intervals. This could have a minor impact on the data analysis for components. Note these two RCIC valves do not have fail to close or fail to open failure mechanisms in the PRA model.	Non-significant impact.	No Impact
CL2007-007	Power supply alternatives are being pursued for the hydrogen igniters. When these power supply modifications are instituted, the PRA models and documentation should be revised as appropriate to reflect the plant.	Plant hardware modifications not yet implemented.	No Impact
CL2007-009	EC364246 changed the logic for the low condenser vacuum trip of the Turbine Bypass Valves from a 1 out of 2 logic to a 2 out of 2 logic for tripping. This could result in a reduction in the spurious trip frequency of the Bypass Valves as represented by event 1MSPH-TBVLOG-F-- FALSE INDICATION OF COND. VACUUM LOCKS OUT TVBs.	Non-significant impact. Any impact would be reflected in plant-specific transient initiator frequencies.	No Impact

Table 4-1

IMPACT ON THE CLINTON PRA MODEL OF PLANT CHANGES SINCE THE LAST PRA UPDATE

URE Number	Plant change	Impact on the CL PRA	Impact on the Application
CL2007-016	EC 366903 changes out the Div 3 and 4 NSPS inverters with inverters make by a different manufacturer. They will function the same as the old to the extent that they have an inverter source (from the same DC bus) and the same AC transformer source and a static transfer switch that can swap between them. As a result the basic PRA logic and power dependencies remain the same. The internal design of the inverters is different so may have a somewhat different long term failure rate. Also because they are somewhat different they may not have all the same common cause failure mechanisms as the original inverters. The Div 1 and 2 NSPS inverters will not be replaced in the short term, but may also be replaced in coming years.	Non-significant impact.	No Impact
CL2008-002	CPS 3104.01 rev 25f allows plant operating conditions that are somewhat different from those modeled in the PRA: 1. This procedure allows 4 condensate pump operation. Because this provides additional NPSH margin to feedpump trips, the plant operators have pretty much elected to run 4 pumps all the time. Note one of the pumps (A or D depending upon which one is selected) will automatically trip upon a bus transfer from UAT to RAT (such as occurs on a normal plant trip). This prevents overloading the RAT. So reality is we run 4, we have 3 left immediately after a scram and we really only need 2 to support post scram operation of the MDRFP. I think the model presumes that there are only two CD and two CB pumps running and that others can be started; this modeling is conservative.	Non-significant impact. Any impact would be reflected in plant-specific transient initiator frequencies.	No Impact

Table 4-1

IMPACT ON THE CLINTON PRA MODEL OF PLANT CHANGES SINCE THE LAST PRA UPDATE

URE Number	Plant change	Impact on the CL PRA	Impact on the Application
	2. This procedure allows that plant to have two Steam Packing Exhauster Condensers in service which is more than the one assumed in the model. Again, the current modeling is conservative.		
CL2008-006	The ECCS door opening operator actions (i.e., for alternate room cooling) are being incorporated into annuciator procedure 5042.06 Rev. 030a. This action had previously been deleted from CPS 4303.01, so the procedural actions for a time had no home. The 5042.06 procedure steps should be capable of addressing typical SX failures in the model.	Non-significant impact.	No Impact
CL2008-010	Several changes have occurred with SA/IA that should improve its reliability: 1. Procedure 3214.01 now provides steps for bypassing SA dryers. 2. Procedure 3214.01 now provides steps for bypassing IA system isolations for particular ring header isolation valves. 3. EC 348731 removed the low CCW pressure trip for the SA compressors. This should reduce the loss of SA contribution from spurious pressure problems on the CCW system. We once tripped all SA compressors due to CCW flow problems.	Non-significant impact.	No Impact

Table 4-1

IMPACT ON THE CLINTON PRA MODEL OF PLANT CHANGES SINCE THE LAST PRA UPDATE

URE Number	Plant change	Impact on the CL PRA	Impact on the Application
	The model shows that we need CCW and we still do, just now the compressors will continue to run until they get into temperature problems, so now the compressors will run through some set of transient conditions that they previously would not. Although this is a real world improvement, it is below the level of resolution of the current model (accounted for in the loss of IA IE frequency calculation).		
CL2008-012	EC361430 changes the recirc pump trip logic such that the RR pumps will now trip to off instead of reducing to Slow Speed for some types of events, such as a turbine trip (and EC369429 evaluates the impact of this change on transient analysis). Since most of the PRA IEs involve a turbine trip, the ATWS RPT logic now may be backed up by the RR pump trip occurring with a turbine trip.	Non-significant impact.	No Impact

Table 4-1

IMPACT ON THE CLINTON PRA MODEL OF PLANT CHANGES SINCE THE LAST PRA UPDATE

URE Number	Plant change	Impact on the CL PRA	Impact on the Application
CL2008-013	Similar to EC 353793 (reference URE 2005-02) the 1F DC bus main feed breaker is being replaced, with a switch fuse combination. This may result in a slightly different failure rate than the original breaker. This change is expected to have a very minor change on calculated PRA results because inadvertent opening of passive electrical connections tend to have relatively low failure rates regardless of the device. This EC replaces the main feed breaker for DC MCC 1DC17E with a switch and fuses combination that will fit into the existing DC breaker cubicle. The normally closed breaker is modeled in the PRA, basic event 1DCCB-DC1F1A-U-- (prob 1.2E-5), although the breaker has no modeled function other than to remain closed. This BE represents an inadvertent change of state for the breaker. A combination of switches and fuses could have a somewhat different failure rate, than a DC breaker. However because in both cases a passive function is modeled the failure rate should be very small.	Non-significant impact.	No Impact
CL2009-003	Engineering Change 371020 addresses the in-plant changes related to the addition of Gas Circuit Breaker 4514 to the Clinton Switchyard. While Ameren owns the switchyard and new GCB 4514, the switchyard breakers are included in the CPS offsite power model. For completeness GCB 4514 should be added to this model and included in the Offsite Power Model system fault tree notebook. EC 371020 deletes some GCB control functions that do not belong to Exelon and use the existing cabling to provide indication for the new GCB 4514.	GCBs are largely suppressed from the cutset solution. In addition this switchyard breaker is unlikely to contribute significantly to LOOP or plant trip initiators.	LOOP events have minimal impact on ATWS scenarios.

Table 4-2

IMPACT OF OPEN SIGNIFICANT PRA PEER REVIEW FINDINGS FOR THE CLINTON PRA MODEL

Peer Review Element	FACTS & OBSERVATIONS (F&Os)			Impact on the Application
	ID	Priority	Summary	
TH-8	TH-8-1	B	Additional plant specific room heat-up calculations (or enhancements to existing calculations) should be performed to support modeling assumptions regarding room-cooling requirements. Areas specifically identified are Control Room, RCIC, LPCS, LPCI, and SWGR rooms.	Primarily a documentation issue. The CPS PRA references CPS specific room cooling calculations for RCIC, HPCS, and the shutdown service water (SX) rooms. Those rooms where specific room cooling calculations are not available are modeled appropriately. The LPCI, LPCS and EDG rooms are modeled as requiring room cooling. Room cooling for the main control room is not required due to the slowly evolving nature of the phenomena and the fact that the control room is continuously manned and doors could be opened. The AC switchgear room modeling does not require room cooling, based on the large size of these rooms. Room cooling modeling assumptions are assessed as a sensitivity study and do not change the conclusions of this risk application (see Appendix B).
HR-6	HR-6-1	B	All pre-initiator HEPs in the Clinton PSA model are based on screening estimates. For post-initiator screening HEPs with RAWs greater than 1.1, the HEPs were re-evaluated with more detailed calculations. For consistency sake, the pre-initiator HEP calculations should follow the same approach.	Non-significant impact. The current calculations are based on representative procedures/practices for similar pre-initiator HEPs. The current estimates are generally higher in error rates than would be obtained if various explicit recovery factors and testing frequencies were applied in specific HEP calculations for each pre-initiator. The impact on the model is non-significant, pre-initiator HEPs contribute approximately 2% to the CL06C CDF.
ST-4	ST-4-2	B	The documentation of the internal flooding analysis could be improved.	Documentation issue. No impact.

Table 4-3

IMPACT OF OPEN SELF-ASSESSMENT ITEMS FOR THE CLINTON PRA MODEL

#	Open Self-Assessment Recommendation	Applicable SRs	Importance to Application
1	Review initiating event precursors in identifying the initiating events to be modeled. A rigorous explicit assessment of all the events in NUREG-1275 could be pursued (if determined that this is the true intent of SR IE-A7); however, such an effort is judged not to provide much benefit to the CPS IE analysis.	IE-A7	No Impact - documentation item. Deferred: This documentation aspect has not been incorporated into the CPS PRA notebooks. This work was performed for another BWR plant (review of hundreds of events INPO SENs, SOERs, SERs, and NRC SECY letters on precursors) and no new initiating events were identified. It is expected that future industry studies will make provide this generic assessment.
2	Loss of switchgear room cooling assumptions should be supported by room cooling calculation.	IE-C4, AS-B3, SC-B2, SC-C1, SY-A17, SY-A19, SY-A20, SY-B7, SY-B8	Non-Significant. Loss of switchgear room cooling-induced ATWS scenarios are a non-significant contributor to ATWS CDF. Room cooling modeling assumptions are assessed as a sensitivity study and do not change the conclusions of this risk application (see Appendix B). Deferred: Room cooling calculations have not been performed at this time and are being considered in the next update.

Table 4-3

IMPACT OF OPEN SELF-ASSESSMENT ITEMS FOR THE CLINTON PRA MODEL

#	Open Self-Assessment Recommendation	Applicable SRs	Importance to Application
3	The following pre-initiator HEP documentation items should be considered: 1) A list of the PRA systems to consider for test and maintenance actions 2) Rules for identifying and screening test and maintenance actions from the PRA 3) A list of procedures reviewed, the potential test and maintenance actions associated with the procedures, and the disposition of the action (screened or evaluated). 4) Identify T&M activities that require realignment of the system outside its normal operational or stand by status. However, performing this task is judged not to have significant impact on the PRA model and results.	HR-A1, HR-A2, HR-A3, HR-C2, HR-C3	No Impact - documentation item. <u>Deferred</u>: The current methodology and documentation for identifying pre-initiator HEPs is judged adequate. Any additional documentation to conform to inferred requirements of pre-initiator identification SRs would be documentation enhancement and not result in increasing the number of pre-initiator HEPs included in the model. The CPS PRA includes over 100 pre-initiator HEPs in the model. Other BWRs that have attempted to rigorously follow these pre-initiator HEP SRs have resulted in explicitly modeling significantly less pre-initiator HEPs in the model.
4	Complete URE 2001-084 (concerning use of screening values for dominant pre-initiator HEPs). Although this will not significantly impact the HRA results, future PRA updates should include an assessment of the quality of plant written procedures and administrative controls as well as human-machine interface for both pre-initiator and post-initiator human actions. <u>Alternative</u>: Possible upgrade to the pre-initiator HRA to include specific quantifications for each pre-initiator HEP would be strict compliance with the standard. This is not considered necessary for most applications. It is recommended that CPS await further ASME clarification on this item before proceeding. This can be confirmed for each application in lieu of performing the quantifications.	HR-B1, HR-B2, HR-D1, HR-D2, HR-D3, HR-D4	Non-significant impact (pre-initiator HEPs contribute approximately 2% of CDF). <u>Deferred</u>: Future updates of the CPS PRA will consider explicit/specific pre-initiator HEP calculations. The current calculations are based on representative procedures/practices for similar pre-initiator HEPs. The current estimates are generally higher in error rates than would be obtained if various explicit recovery factors and testing frequencies were applied in specific HEP calculations for each pre-initiator. The impact on the model is non-significant, pre-initiator HEPs contribute approximately 2% to the CL06C CDF.

Table 4-3

IMPACT OF OPEN SELF-ASSESSMENT ITEMS FOR THE CLINTON PRA MODEL

#	Open Self-Assessment Recommendation	Applicable SRs	Importance to Application
5	Failure data development using surveillance test data should fulfill the requirements of DA-C10, and should be documented appropriately. Review surveillance test procedures and identify all failure modes that are fully tested by the procedures. Include data for the failure modes that are fully tested. The results of unplanned demands on equipment should also be accounted for.	DA-C10	Non-significant impact. Deferred: Current industry PRA efforts and PRA peer reviews are having difficulty understanding the full intent of this SR. Future updates of the CPS PRA will consider enhancement to the documentation and investigation of the plant failure data implied by this SR. The impact on the overall CDF or LERF values is judged to be non-significant.
6	As needed in maintenance unavailability determination, perform interviews of maintenance staff for equipment with incomplete or limited maintenance information and document appropriately.	DA-C12	Non-significant impact. Deferred: Future updates of the CPS PRA will consider performance of interviews of plant personnel to supplement maintenance unavailability estimates for equipment with limited maintenance information. Any refinements to maintenance unavailabilities are judged to result in a negligible impact on CDF (i.e., the dominant maintenance terms, by far, with respect to CDF are trains with good maintenance information - ECCS trains, RCIC, EDGs, SX).
7	The CPS internal flooding analysis and documentation should be updated to meet ASME Standard expectations.	SC-A6, SY-A4, IF Technical Element	Non-significant impact. Internal Flooding scenarios are a non-significant contributor to ATWS-induced CDF. In-Progress: An internal flooding update to the CPS CL06c model is currently in-progress. Update of the internal flooding analysis was deferred at the time of the 2006 update due to budget constraints.

Table 4-3

IMPACT OF OPEN SELF-ASSESSMENT ITEMS FOR THE CLINTON PRA MODEL

#	Open Self-Assessment Recommendation	Applicable SRs	Importance to Application
8	Identify significant basic events that contribute to the significant initiating events whose frequencies are quantified using fault tree methods. Otherwise, importance measures calculated and assessed to ensure results make logical sense.	QU-D5a	No Impact - documentation item. Significant events and cutsets to this risk application are identified in Appendix B of this report. Deferred: This documentation aspect has not been incorporated into the CPS PRA notebooks. Initiating event fault trees are not linked into the accident sequence models. Documentation of the importance of failures in initiating event fault trees in the base PRA notebooks is a documentation enhancement.
9	Strict reading of SR QU-F2 would indicate that the following enhancements to the documentation of the CPS PRA would need to be made to comply with the Standard: a) Incorporate an overview of the quantification process. b) Provide a list of human actions and equipment failures (significant basic events) that cause accidents to be non-dominant. c) Refer to the Disposition for SR QU-E4 regarding assumptions, sources of uncertainty and related sensitivity assessments. d) Bases for the elimination of mutually exclusive events from the model need to be added. e) Include cutsets segregated by accident sequence in the documentation. This is available but may not be needed in the formal documentation. This should await further ASME clarification before extensive resources are committed.	QU-F2	No Impact - documentation item. Significant events and cutsets to this risk application are identified in Appendix B of this report. Deferred: Not all items incorporated into the CPS 2006 PRA Update. Items (a) and (c) are incorporated into the quantification documentation of the CPS 2006 update. Items (b), (d) and (e) are documentation enhancements for the base PRA and are maintained for consideration for future updates.

4.7 GENERAL CONCLUSION REGARDING PRA CAPABILITY

The Clinton PRA maintenance and update processes and technical capability evaluations provide a robust basis for concluding that the PRA is suitable for use in risk-informed licensing actions, specifically in support of the requested extended CT for the SLC system.

Previously identified gaps to specific requirements in the ASME PRA Standard have been reviewed to determine which gaps might merit application-specific sensitivity studies in the presentation of the application results. No gaps were identified as needing specific sensitivity studies for this SLC CT extension request.

5.0 SUMMARY AND CONCLUSIONS

5.1 SCOPE INVESTIGATED

This analysis evaluates the acceptability, from a risk perspective, of a change to the Clinton TS for the SLC system to increase the CT from 8 hours to 72 hours when both SLC subsystems (i.e., both trains) are inoperable.

The analysis examines a range of risk contributors as follows:

- The Clinton FPIE PRA model is used to quantitatively address risk impacts.
- The FPIE assessment is judged to adequately capture risk contributors associated with low power plant operation
- The SLC TS only applies to Modes 1 and 2. Shutdown and refueling modes (Modes 3, 4 and 5) are not applicable to the SLC TS.
- The Interim Fire PRA model and other fire studies (e.g., NUREG/CR-6850) are used to provide qualitative and semi-quantitative insights, determining that fire hazards are negligible contributors.
- Seismic risk contributors are determined to be negligible based on qualitative insights from the Clinton IPEEE and the NUREG-1150 study.
- Other External Event risks were found to be negligible contributors based on the Clinton IPEEE.

5.2 PRA QUALITY

The PRA quality has been assessed and determined to be adequate for this risk application, as follows:

- Scope - The Clinton PRA modeling is highly detailed, including a wide variety of initiating events, modeled systems, operator actions, and common cause events. The PRA has the necessary scope to appropriately assess the pertinent risk contributors.

- Fidelity – The Clinton PRA model (CL06C) is the most recent evaluation of the risk profile at Clinton for FPIE challenges. The PRA reflects the as-built, as-operated plant.

- Standards – The PRA has been reviewed against the ASME PRA Standard [Ref. 5] and the PRA elements are shown to have the necessary attributes to assess risk for this application.

- Peer Review - The PRA received a Peer Review in 2000. Based on the Peer Review results and the incorporation of Peer Review comments, the PRA is found to have the necessary attributes to assess risk for this application.

- Appropriate Quality – The PRA quality is found to be commensurate with that needed to assess risk for this application.

5.3 QUANTITATIVE RESULTS VS. ACCEPTANCE GUIDELINES

As shown in Table 5.3-1 below, the base results of the risk assessment indicate that the ΔCDF, ICCDP, ΔLERF, and ICLERP risk metric values are below the acceptance guidelines as defined in the corresponding risk significance guidelines from RG 1.174 and RG 1.177.

This analysis demonstrates that the proposed TS change satisfies the risk acceptance guidelines in RG 1.174 and RG 1.177, and therefore meets the intent of very small risk increases consistent with the Commission's Safety Goal Policy Statement.

Table 5.3-1

RISK ASSESSMENT BASE RESULTS

Risk Metric	Value(1)	Acceptance Guidelines
ΔCDF	2.9E-08/yr	<1.0E-06/yr
ICCDP	2.9E-08	<5.0E-07
ΔLERF	6.2E-09/yr	<1.0E-07/yr
ICLERP	6.2E-09	<5.0E-08

5.4 CONCLUSIONS

This analysis demonstrates the acceptability, from a risk perspective, of a change to the Clinton TS for the SLC system to increase the CT from 8 hours to 72 hours when both SLC subsystems (i.e., both trains) are inoperable.

This analysis demonstrates that the proposed TS change satisfies the risk acceptance guidelines in RG 1.174 and RG 1.177. This meets the intent of very small risk increases consistent with the Commission's Safety Goal Policy Statement.

Additionally, a PRA technical adequacy evaluation was performed consistent with the requirements of RG 1.200, Revision 1. This included a process to identify potential key sources of model uncertainty and related assumptions associated with this application. This resulted in the identification of issues that could both decrease and increase the calculated risk metrics. None of these identified sources of uncertainty were significant enough to change the conclusions from the risk assessment results presented here.

6.0 REFERENCES

[1] RG 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk Informed Activities,” Revision 1, January 2007.

[2] RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1, November 2002.

[3] RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," August 1998.

[4] Exelon Risk Management Team, CPS-PRA-014, *Clinton Power Station Probabilistic Risk Assessment Quantification Notebook*, CL06C, Revision 4, March 2007.

[5] “Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications,” (ASME RA-S-2002), Addenda RA-Sa-2003, and Addenda RA-Sb-2005, December 2005.

[6] Boiling Water Reactors Owners’ Group, “BWROG PSA Peer Review Certification Implementation Guidelines,” Revision 3, January 1997.

[7] Clinton MSPI Basis Document, Rev. 3, December 12, 2008.

[8] Drouin, M. et al., “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making”, NUREG-1855, March 2009.

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[10] CPS-PSA-021.06, “Clinton FPRA Summary and Quantification Report”, Rev. 0, September 2008.

[11] AmerGen, “Clinton Power Station Individual Plant Examination for External Events,” September 1995

[12] “PRA Procedures Guide”, NUREG/CR-2300, September 1981.

[13] “Analysis of Core Damage Frequency: Peach Bottom, Unit 2, External Events,” NUREG/CR-4550, Volume 4, Revision 1, Part 3, Table 4.14, page 4-83.

[14] NUREG/CR-5042, “Evaluation of External Hazards to Nuclear Power Plants in the United States,” December 1987.

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[16] NUREG/CR-5500, "Reliability Study: General Electric Reactor Protection System, 1984-1995, Volume 3" May 1999.

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[18] NUREG/CR-6850, EPRI Report 1011989, "Fire PRA Methodology for Nuclear Power Facilities", September 2005.

[19] Gorman, Thomas, BWR Owners' Group (BWROG), "BWROG Assessment of IN 2007-07", 10/16/2007.

[20] Not Used.

[21] Exelon, ER-AA-600-1046, "Risk Metrics – NOED and LAR", Revision 4.

[22] Not Used.

[23] "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants", NUREG-1150, December 1990.

[24] Not Used.

[25] Electric Power Research Institute, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin", EPRI NP-6041, October 1988.

[26] ASME/ANS RA-Sa-2009, "Addenda to RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," February 2009.

[27] RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk Informed Activities," Revision 2, March 2009.

Appendix A

External Event Assessment

A.1 INTRODUCTION

This appendix discusses the external events assessment in support of the Clinton SLC system CT extension risk assessment. This appendix uses as the starting point of this assessment the external event work documented in the Clinton Individual Plant Examination of External Events (IPEEE) [Ref. A-1].

Because the effects of the SLC CT extension are evident only in the failure to scram (Anticipated Transients Without Scram (ATWS)) related sequences, the following examination of external events focuses on the ATWS accident sequence insights.

A.2 EXTERNAL EVENT SCREENING ASSESSMENT

The purpose of this portion of the assessment is to examine the spectrum of external event challenges to determine which external event hazards should be explicitly addressed as part of the Clinton SLC System CT extension risk assessment.

<u>Seismic</u>

There is no currently maintained quantitative Seismic PRA for Clinton. Section A.3 discusses seismic ATWS insights from the CPS IPEEE and NUREG-1150.

<u>Internal Fires</u>

This internal fire assessment is based on the Interim Clinton Fire PRA (FPRA) model developed in 2008 and generic assessments in NUREG/CR-6850 and the BWROG assessment of IN 2007-07. This assessment is discussed in Section A.4.

<u>Other External Hazards</u>

The Clinton plant design with respect to external flooding meets all the applicable criteria of the NRC Standard Review Plan (SRP). Core damage accidents induced by external flooding are negligible contributors to plant risk.

Other external event risks such as severe weather, high winds or tornados, transportation accidents, nearby facility accidents, turbine missiles, and other miscellaneous external hazards were also considered in the IPEEE analysis. The Clinton site characteristics and design meet all the applicable criteria of the NRC SRP. No significant quantitative contribution from these external events was identified by IPEEE evaluations. The compensatory actions and risk insights in this LAR are also judged applicable to qualitatively reduce the risk associated with these events.

<u>Conclusions of Screening Assessment</u>

Given the foregoing discussions, other external hazards are assessed to be insignificant contributors to plant risk. Explicit treatment of the "other" external hazards is not necessary for most PSA applications (including the SLC System CT extension risk assessment) and would not provide additional risk-informed insights for decision making.

Further information is presented in this appendix to justify the screening of Fire and Seismic hazards for the SLC CT extension application.

A.3 SEISMIC ASSESSMENT

There is no currently maintained quantitative Seismic PRA for Clinton. The following sections discuss seismic ATWS insights from the CPS IPEEE and NUREG-1150.

A.3.1 <u>Clinton Seismic IPEEE Overview</u>

Clinton performed a seismic margins assessment (SMA) as part of the IPEEE, following the guidance of EPRI NP-6041. [Ref. A-2] The SMA is a deterministic evaluation process that does not calculate risk on a probabilistic basis. No core damage frequency sequences were quantified as part of the seismic risk evaluation.

The conclusions of the Clinton seismic risk analysis are as follows: [Ref. A-1]

> *"No improvements to the plant were identified as a result of the Seismic Margins Assessment ... the plant was determined to be fully capable of attaining safe shutdown conditions after the Review Level Earthquake (RLE)."*

Based on a review of the Clinton IPEEE and the conclusions identified earlier in this assessment, the conclusions of the SMA are unaffected by the SLC CT extension. The SLC CT extension has no impact on the seismic qualifications of the SSCs.

A.3.2 <u>Peach Bottom NUREG-1150 Seismic Overview</u>

The NUREG/CR-4551 study completed an update of the NUREG-1150 severe accident analysis for five nuclear power plants, including the Peach Bottom Atomic Power Station. This analysis addressed both internal and external events, including seismic initiators. Peach Bottom utilized the Seismic Margins Analysis as part of the IPEEE. It is reasonably assumed that the seismic ATWS risk portion of the analysis is generically appropriate for all BWRs due to the similarity of CRD and SLC systems.

The NUREG/CR-4551 Peach Bottom seismic analysis screened seismic-induced ATWS accident sequences as non-significant contributors (<1%) to the plant seismic CDF. Based on the Peach Bottom results, it is judged that seismic-induced ATWS accident sequences are similarly non-significant contributors to Clinton plant seismic CDF.

A.3.3 Seismic Risk Impact Conclusion

Based on the preceding discussions, it is concluded that the risk of a seismically induced ATWS is non-significant and does not impact the decision-making for the proposed CPS SLC CT extension.

A.4 INTERNAL FIRES ASSESSMENT

This internal fire assessment is based on the Clinton Interim Fire PRA (FPRA) model developed in 2008 and generic assessments in NUREG/CR-6850 [Ref. A-3] and the BWROG assessment of IN 2007-07 [Ref. A-5].

A.4.1 NUREG/CR-6850 Screening

NUREG/CR-6850, Volume 2, Section 2.5.1 (page 2-7) [Ref. A-3] provides the following directions for selecting components and accident scenarios to be examined in an internal fire PRA:

> *"The types of sequences that could generally be eliminated from the PRA include the following...Sequences associated with events that, while it is possible that the fire could cause the event, a low-frequency argument can be justified. For example, it can often be easily demonstrated that anticipated transient without scram (ATWS) sequences do not need to be treated in the Fire PRA because fire-induced failures will almost certainly remove power from the control rods (resulting in a trip), rather than cause a "failure-to-scram" condition. Additionally, fire frequencies multiplied by the independent failure-to-scram probability can usually be argued to be small contributors to fire risk."*

As can be seen from the NUREG/CR-6850 excerpt above, fire-induced ATWS contributors are generally acknowledged as non-significant contributors to the fire risk profile.

A.4.2 Clinton Interim Fire PRA

The current Clinton FPRA [Ref. A-4] is an interim implementation of NUREG/CR-6850; that is, not all tasks identified in NUREG/CR-6850 are yet completely addressed or implemented due to the changing state-of-the-art of industry at the time of the 2007-2008 Clinton FPRA development.

NUREG/CR-6850 task limitations and other precautions regarding the 2007-2008 FPRA upgrade for Clinton are as follows:

- Multiple Spurious Operation (MSO) Review (NUREG/CR-6850 Task 2) - MSOs are reviewed and considered; however, an expert panel is not used. At the time of the Clinton FPRA development the BWR Owners' Group was developing a generic list of MSOs to be considered. In future CPS FPRA updates this list will be reviewed and incorporated as necessary.
- Instrumentation Review (NUREG/CR-6850 Task 2) - The new requirements of NUREG/CR-6850 regarding the explicit identification and modeling of instrumentation required to support PRA credited operator actions is not addressed. The industry treatment for this task is still being developed.
- The Balance of Plant (NUREG/CR-6850 Task 2) - The BOP is not fully treated. BOP support system failure is conservatively assumed. Additional modeling could be conducted to reduce the fire CDF due to this assumption if time and funding is available in future updates.
- Large Early Release Frequency (LERF) (NUREG/CR-6850 Task 2) - LERF is not considered. LERF is expected to be addressed in future updates.
- Limited Analysis Iterations (NUREG/CR-6850 Task 9-12) - The process of conducting a FPRA is iterative, identifying conservative assumptions and high risk compartments and performing analyses to refine the assumptions and reduce those compartment risks. The ability to conduct iterations is limited based on resources. The scenarios developed for the 2008 Clinton FPRA may benefit from further refinement as necessary for application or for future updates.
- Multi-Compartment Review (NUREG/CR-6850 Task 11) - This subtask reviews the fire analysis compartment boundaries to ensure they are sufficiently robust to prevent the spread of fire between FPRA analysis compartments or that such propagations are adequately addressed by the developed scenarios. The design and plant layout of Clinton make fire propagation to multiple compartments unlikely compared to the fire risk in individual compartments.

- Seismic Fire Interactions (NUREG/CR-6850 Task 13) - This task reviews previous assessments to identify any specific interaction between suppression system and credited components or adverse impact of fire protection system interactions that should be accounted for in the FPRA.
- Uncertainty and Sensitivity Analysis (NUREG/CR-6850 Task 15) - This task explores the impacts of possible variation of input parameters used in the development of the model and the inputs to the analysis on the FPRA results. This task is not currently addressed because the industry is still developing an appropriate methodology.

Some limitations of these items are:

- Item 1(MSO), represents a source of additional fire CDF contribution (i.e., if the BWROG MSO list includes MSOs not addressed in this update).
- Item 2 (Instrumentation Review) represents a potential additional fire CDF contribution that cannot be estimated at this time since the methodology is not established.
- Items 3 (BOP) and 8 (Uncertainty) are potential sources of conservatism in the results.
- Item 4 (LERF) is a future scope issue not affecting the fire CDF model.
- Items 5 (Iterations) and 6 (Multi-compartment) represent modeling assumptions that should be reviewed with each FPRA application to determine their applicability and/or potential impact on the decision.
- Item 7 (Seismic) is a FPRA application completeness issue for which the methodology is not yet established.

Given the above, the 2008 Clinton FPRA model is judged to provide a meaningful representation of fire CDF contributors, and is appropriate for use in risk-informed decision-making, to the extent that these limitations are recognized and addressed in each application, as appropriate. The model is, however, “interim” due to the stated limitations.

Based on the interim CPS Fire PRA, fire-induced ATWS CDF is approximately 4E-8/yr (<1% of CPS FPRA CDF). This is approximately an order of magnitude lower than the CPS internal events ATWS CDF. As such, like NUREG/CR-6850, the CPS interim FPRA shows that fire-induced ATWS is a non-significant contributor to the plant risk profile and does not impact the decision-making of the proposed CPS SLC CT extension.

A.4.3 BWROG Position on Fire-Induced Failure to Scram

Fire scenarios that could threaten the function of the reactor protection system have been addressed in a BWROG assessment (refer to Appendix C) of NRC Information Notice 2007-07. [Ref. A-5] The assessment outlines the types of scenarios in which a fire could energize a circuit through a "hot short" that would compromise scram capabilities. The assessment also indicates that there are multiple actions that would have to occur in conjunction to the very specific fire scenarios for function to be lost.

The assessment concluded that these scenarios are of low-likelihood, low safety-significance, and have multiple layers of defense-in-depth which would either prevent the condition, or adequately mitigate it.

A.4.4 Fire Risk Impact Conclusion

Based on the preceding discussions, it is concluded that fire-induced ATWS is a non-significant contributor to the plant risk profile and thus does not impact the decision-making of the proposed CPS SLC CT extension.

A.5 REFERENCES

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[A-2] *A methodology for assessment of nuclear power plant seismic margin*, EPRI NP-6041, Palo Alto, CA: 2001.

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[A-4] CPS-PSA-021.06, "Clinton FPRA Summary and Quantification Report", Rev. 0, September 2008.

[A-5] Gorman, Thomas, BWR Owners Group (BWROG), "BWROG Assessment of IN 2007-07", 10/16/2007.

Appendix B

Uncertainty Analysis

This appendix evaluates uncertainties that could impact the SLC CT extension assessment. Section B.1 and B.2 evaluate model uncertainties. Section B.3 evaluates parametric uncertainty.

- Section B.1 provides Clinton specific modeling uncertainty evaluations for the base case.
- Section B.2 provides an examination of the specific cutsets that affect the change in the CDF risk metric associated with the change in the SLC CT.
- Section B.3 documents the parametric uncertainty analysis of the model used in this application.

B.1 MODEL UNCERTAINTIES SUMMARY

Postulated key modeling uncertainties are identified through a systematic structured process [Ref. B-1]. Table B-1 presents the candidate key modeling uncertainties for the CL06C model. The five modeling uncertainties that can be considered important model uncertainty are summarized in Table B-2 along with associated impacts on the CDF and LERF risk metrics.

It is noted that none of these five cases presented in Table B-2 evaluates modeling issues associated with the SLC system or ATWS sequences.

Table B-1

SUMMARY OF SENSITIVITY CASES TO IDENTIFY RISK METRIC CHANGES ASSOCIATED WITH CANDIDATE MODELING UNCERTAINTIES(5)

Source of Modeling Uncertainty	CDF Impact (/yr)(1)		LERF Impact (/yr)(2)	
	Upper Bound	Lower Bound	Upper Bound	Lower Bound
1A) Applicability of industry experience to environmentally influenced events (i.e., loss of service water, LOOP, etc.) – Loss of Service Water	5.66E-06	5.54E-06	1.21E-07	1.20E-07
1B) Applicability of industry experience to environmentally influenced events (i.e., loss of service water, LOOP, etc.) – Loss of Intake Structure	~6.2E-06(7)	(6)	~1.21E-07(7)	(6)
1C) Applicability of industry experience to environmentally influenced events (i.e., loss of service water, LOOP, etc.) – Severe and Extreme Weather Induced LOOP	7.78E-06	4.79E-06	1.52E-07	1.09E-07
2A) Treatment of Rare and Extremely Rare Events – Excessive LOCA	5.58E-06	5.57E-06	1.23E-07	1.19E-07
2B) Treatment of Rare and Extremely Rare Events – SW Flood in RB	5.90E-06	5.46E-06	1.22E-07	1.20E-07
3), 4), 6), 11), 17), 24) Beyond Design Basis Environment	8.57E-06	4.51E-06	1.46E-07	1.11E-07
5) and 8) Case A) Impact of LOOP/SBO conditions on allowable AC Recovery	6.59E-06	5.16E-06	1.20E-07	1.20E-07
5) and 8) Case B) Impact of LOOP/SBO conditions – DFP injection	5.57E-06	4.07E-06	1.20E-07	1.05E-07
7), 12), 18) Room Cooling Assumptions	1.77E-05	4.22E-06	1.31E-07	1.19E-07
9) & 15) Impact of venting on systems	5.92E-06	5.52E-06	1.20E-07	1.20E-07
10) Time Dependency failures due to environmental conditions	(3)	(3)	(3)	(3)
13) Recirc Pump Seal Leakage	(3)	(3)	(3)	(3)
14) Suppression Pool Strainer Performance	5.84E-06	5.48E-06	1.21E-07	1.20E-07
16) Treatment of Instrumentation required for operator action	6.77E-06(5)	~5.2E-06(7)	3.13E-07(5)	~1.0E-07(7)

Table B-1

SUMMARY OF SENSITIVITY CASES TO IDENTIFY RISK METRIC CHANGES ASSOCIATED WITH CANDIDATE MODELING UNCERTAINTIES[5]

Source of Modeling Uncertainty	CDF Impact (/yr)[1]		LERF Impact (/yr)[2]	
	Upper Bound	Lower Bound	Upper Bound	Lower Bound
19) Water Hammer Impact on System Performance (Failure Probability of Pipe Rupture)	[4]	[4]	[4]	[4]
20) Alternate Alignments	9.97E-06	[6]	1.50E-07	[6]
21) Procedural Changes	5.57E-06	[6]	1.20E-07	[6]
23) Flood Frequency Data	~8E-06[7]	~5E-06[7]	~1.5E-07[7]	~1.1E-07[7]
24) CST Inventory Capacity	7.20E-06	5.00E-06	1.44E-07	1.16E-07
25) Combined Sensitivity Case 1C and Case 5/8A for SBO related features	9.32E-06	[6]	1.52E-07	[6]
26) Dependent HEP Recovery file	1.52E-05[5]	[6]	1.98E-07[5]	[6]
27) Severe Accident Condition Impact on Vent Valves	6.30E-06	5.31E-06	1.24E-07	1.19E-07
28) Level 2 LERF as Affected by the Phenomenological Effects of Severe Accident Progression	n/a	n/a	1.82E-07*	~6E-08[7]

Notes to Table B-1;

(1) Compared with a CL06C base CDF of 5.57E-6/yr quantified with a 1E-11/yr truncation limit.
(2) Compared with a CL06C base LERF of 1.20E-07/yr quantified with a 1E-12/yr truncation limit.
(3) Subsumed by Case 5/8.
(4) Based on installed CPS system for suppression pool cooling, this candidate modeling uncertainty identified for other BWRs is considered not to be quantitatively significant and does not lead to a key modeling uncertainty.
(5) Most of the sensitivity results were produced by manipulating the cutset results file. These results were produced by re-quantifying the entire model.
(6) These lower bound cases not performed; interest is in the increase in CDF and LERF.
(7) Estimate for 2006C is based on sensitivity case results using the 2006B model.

Table B-2

IMPORTANT MODELING UNCERTAINTY CASES

Sensitivity Case	CDF Increase[1]	LERF Increase[2]
Sensitivity Cases 7, 12, 18: Room Cooling Requirements	3.18	1.09
Sensitivity Case 16: Instrumentation Effects	1.22	2.60
Sensitivity Case 24: RCIC, CY, and MC Tank Inventory Capacity	1.29[3]	1.19[3]
Sensitivity Case 26: Dependent HEP Recovery Treatment	2.72	1.65
Sensitivity Case 28: Level 2 Phenomenology	n/a	1.52[3]

Notes to Table B-2;

(1) Compared with a CL06C base CDF of 5.57E-6/yr quantified with a 1E-11/yr truncation limit.
(2) Compared with a CL06C base LERF of 1.20E-07/yr quantified with a 1E-12/yr truncation limit.
(3) These changes in the risk metric are below 2.0, but they are retained for identification to the decision-makers.

B.2 MODEL UNCERTAINTIES ASSOCIATED WITH SLC SYSTEM OUT OF SERVICE

To determine the relative importance of individual contributors for this SLC CT extension, the focus needs to be on the results of the CDF assessment for the SLC system out of service. To obtain insights regarding this change to the base case results, the first step is to take the out-of-service case cutsets and remove the base case cutsets. This is done in CAFTA through the delete term function of the cutest editor. The result of this process is cutsets that are unique to the SLC out-of-service case and do not appear in the base case. These cutsets can be used to determine information regarding significant accident sequences or cutsets that determine the change in risk metrics, i.e., drive the delta-CDF assessment.

Table B-3 presents the top ten cutsets for the delta-CDF assessment. Table B-4 presents the importance measures associated with the delta-CDF assessment.

Tables B-3 and B-4 show that the Scram system hardware failure is the most important contributor for the SLC system out-of-service case. The top ten cutsets are exclusively failures of the Scram system associated with various initiating events. The first nine represent single failures that lead to core damage. The tenth cutset is an electrical failure of the scram system coupled with operator failures to manually scram. Of the 76 events appearing in Table B-4, only 8 are basic events, with the rest being initiators. This is due to the fact that the cutsets associated with the SLC system out-of-service are again predominantly single failures of the Scram system leading to core damage.

It can be concluded that the SLC out-of-service case CDF is dominated by failures of the Scram system. The basic events used to model the Scram system failures are already considered in the base uncertainty assessment.

Similarly, the LERF results are dominated by failures of the Scram system for the SLC out-of-service case. The LERF results provide similar insights to the CDF results insights.

Because of the large potential impact of the mechanical failure to scram probability on the assessment of the risk metrics for this application, it is prudent to perform a sensitivity recognizing the uncertainty in the mechanical common cause failure to scram probability.

This sensitivity is performed by including the 95% upper bound on the common cause mechanical scram failure probability in both the base case and the case with the SLC system set to TRUE.

The results of the sensitivity case are shown in Table B-5.

Based on the results of the sensitivity analysis, it is found that the acceptance criteria are all met even for this extreme assumption regarding the common cause mechanical scram failure probability.

Table B-3

TOP TEN CUTSETS FOR CDF FOR THE SLC SYSTEM OUT OF SERVICE

#	Cutset Prob	Event Prob	Event	Description
1	2.84E-06	1.35E+00	%TT	TURBINE TRIP WITH BYPASS INITIATOR
		2.10E-06	1RPSYRPS-MECHFCC	SCRAM SYSTEM HARDWARE FAILURE
2	2.16E-07	1.03E-01	%TC	LOSS OF CONDENSER VACUUM INITIATOR
		2.10E-06	1RPSYRPS-MECHFCC	SCRAM SYSTEM HARDWARE FAILURE
3	8.90E-08	4.24E-02	%TI	INADVERTENT OPEN RELIEF VALVE INITIATOR
		2.10E-06	1RPSYRPS-MECHFCC	SCRAM SYSTEM HARDWARE FAILURE
4	8.42E-08	4.01E-02	%TM	MSIV CLOSURE INITIATOR
		2.10E-06	1RPSYRPS-MECHFCC	SCRAM SYSTEM HARDWARE FAILURE
5	5.63E-08	2.68E-02	%TF	LOSS OF FEEDWATER INITIATOR
		2.10E-06	1RPSYRPS-MECHFCC	SCRAM SYSTEM HARDWARE FAILURE
6	5.10E-08	2.43E-02	%FLOOD30	Flood in CW Area - CW Line Break
		2.10E-06	1RPSYRPS-MECHFCC	SCRAM SYSTEM HARDWARE FAILURE
7	4.89E-08	2.33E-02	%RAT	LOSS OF RESERVE AUX TRANSFORMER INITIATOR
		2.10E-06	1RPSYRPS-MECHFCC	SCRAM SYSTEM HARDWARE FAILURE
8	4.73E-08	2.25E-02	%TIA	LOSS OF INTRUMENT AIR INITIATOR
		2.10E-06	1RPSYRPS-MECHFCC	SCRAM SYSTEM HARDWARE FAILURE
9	3.40E-08	1.62E-02	%LOOP	LOSS OF OFFSITE POWER INITIATOR
		2.10E-06	1RPSYRPS-MECHFCC	SCRAM SYSTEM HARDWARE FAILURE
10	3.37E-08	1.35E+00	%TT	TURBINE TRIP WITH BYPASS INITIATOR
		1.35E-01	1RPOPMANSCRAMH--	OPERATOR FAILS TO MANUALLY SCRAM REACTOR
		3.70E-06	1RPSYRPS-ELECFCC	SCRAM SYSTEM INITIATION LOGIC FAILURE
		5.00E-02	CRH-ARI-FL	FAILURE OF THE AUTO ARI

Table B-4

BASIC EVENT IMPORTANCE MEASURES FOR CDF ASSESSMENT FOR SLC OUT OF SERVICE

Event Name	Probability	Fus Ves	Description
1RPSYRPS-MECHFCC	2.10E-06	9.88E-01	SCRAM SYSTEM HARDWARE FAILURE
%TT	1.35E+00	7.65E-01	TURBINE TRIP WITH BYPASS INITIATOR
%TC	1.03E-01	5.83E-02	LOSS OF CONDENSER VACUUM INITIATOR
%TI	4.24E-02	2.40E-02	INADVERTENT OPEN RELIEF VALVE INITIATOR
%TM	4.01E-02	2.27E-02	MSIV CLOSURE INITIATOR
%TF	2.68E-02	1.52E-02	LOSS OF FEEDWATER INITIATOR
%FLOOD30	2.43E-02	1.38E-02	Flood in CW Area - CW Line Break
%RAT	2.33E-02	1.32E-02	LOSS OF RESERVE AUX TRANSFORMER INITIATOR
%TIA	2.25E-02	1.27E-02	LOSS OF INTRUMENT AIR INITIATOR
1RPOPMANSCRAMH--	1.35E-01	1.17E-02	OPERATOR FAILS TO MANUALLY SCRAM REACTOR
1RPSYRPS-ELECFCC	3.70E-06	1.17E-02	SCRAM SYSTEM INITIATION LOGIC FAILURE
CRH-ARI-FL	5.00E-02	1.17E-02	FAILURE OF THE AUTO ARI
%LOOP	1.62E-02	9.18E-03	LOSS OF OFFSITE POWER INITIATOR
%FLOOD19A	1.48E-02	8.38E-03	Flood in Area D.1.10 - WO Line Break
%FLOOD11B	1.20E-02	6.80E-03	Flood in Area A.1.6 - Any Break
%FLOOD04D	1.00E-02	5.67E-03	Flood in Area A.1.10 - Other Line Break
%FLOOD03C	9.40E-03	5.32E-03	Flood in Area A.1.9 - Other Line Break
%FLOOD26B	6.72E-03	3.81E-03	Flood in Area F.1.7 - HP Line Break
%FLOOD19B	6.10E-03	3.45E-03	Flood in Area D.1.10 - SX-A Line Break
%FLOOD19C	6.10E-03	3.45E-03	Flood in Area D.1.10 - SX-B Line Break
%FLOOD12D	4.87E-03	2.76E-03	Flood in Area A.1.4 - Other Line Break
%TBCCW	4.14E-03	2.34E-03	LOSS OF TURBINE BUILDING CLOSED COOLING WATER INITIATOR
%FLOOD01C	3.63E-03	2.06E-03	Flood in Area A.2.1 - Other break
%S2-ST	3.55E-03	2.01E-03	INIT: SMALL BREAK LOCA - ABOVE CORE INSIDE DRYWELL
%FLOOD09B	2.80E-03	1.59E-03	Flood in Area A.2.2 - Other Line Break
%FLOOD22	2.80E-03	1.59E-03	Flood in Area D.5.8 - SX-B Line Break
%FLOOD23	2.80E-03	1.59E-03	Flood in Area D.5.16 - SX-A Line Break
%FLOOD05B	2.70E-03	1.53E-03	Flood in Area A.2.5 - Other Line Break
%FLOOD17	2.30E-03	1.30E-03	Flood in Area D.1.8 - Any Break

Table B-4

BASIC EVENT IMPORTANCE MEASURES FOR CDF ASSESSMENT FOR SLC OUT OF SERVICE

Event Name	Probability	Fus Ves	Description
%FLOOD18	2.30E-03	1.30E-03	Flood in Area D.1.9 - Any Break
%TRLA	2.24E-03	1.27E-03	MEDIUM RANGE RX WATER REFERENCE LEG A LINE BREAK
%TRLB	2.24E-03	1.27E-03	MEDIUM RANGE RX WATER REFERENCE LEG B LINE BREAK
%FLOOD31	2.00E-03	1.13E-03	Flood in Area T-1 - Condenser Line Break
%TAC4E	1.48E-03	8.38E-04	LOSS OF 6.9 KV BUS 1APO4E INITIATOR
%TAC5E	1.48E-03	8.38E-04	LOSS OF 6.9 KV BUS 1APO5E INITIATOR
%TAC6E	1.48E-03	8.38E-04	LOSS OF 4.16 KV BUS 1APO6E INITIATOR
%TAC8E	1.48E-03	8.38E-04	LOSS OF 4.16 KV BUS 1APO8E INITIATOR
%TSW	1.12E-03	6.34E-04	LOSS OF PLANT SERVICE WATER INITIATOR
%FLOOD32	1.00E-03	5.66E-04	Flood in Plant - WS Line Break
%FLOOD33	1.00E-03	5.66E-04	Flood in Plant - WO Line Break
%FLOOD34	1.00E-03	5.66E-04	Flood in Area T-1 - Other Line Break
%FLOOD04B	7.98E-04	4.52E-04	Flood in Area A.1.10 - RHRA Line Break
%FLOOD01A	7.08E-04	4.01E-04	Flood in Area A.2.1 - CC Line Break
%FLOOD15B	4.10E-04	2.32E-04	Flood in Area A.4.1 - Other Line Break
%FLOOD20B	2.70E-04	1.51E-04	Flood in Area D.4.10 - WO Line Break
%FLOOD08B	2.37E-04	1.33E-04	Flood in Area A.3.3 - WO Line Break
%FLOOD25	1.50E-04	8.40E-05	Flood in Area D.6.7 - Any Break
%FLOOD20D	1.08E-04	6.05E-05	Flood in Area D.4.10 - Other Line Break
%FLOOD20A	9.10E-05	5.09E-05	Flood in Area D.4.10 - CC Line Break
%FLOOD10	8.90E-05	4.98E-05	Flood in Area A.3.5 - Any Break
%FLOOD14	8.20E-05	4.59E-05	Flood in Area A.3.1 - Any Break
%FLOOD26A	7.96E-05	4.46E-05	Flood in Area F.1.7 - WS Line Break
%FLOOD12B	5.96E-05	3.34E-05	Flood in Area A.1.4 - WS Line Break
%FLOOD08E	5.41E-05	3.03E-05	Flood in Area A.3.3 - RHR Line Break
%FLOOD20C	5.04E-05	2.82E-05	Flood in Area D.4.10 - WS Line Break
%FLOOD08A	4.59E-05	2.57E-05	Flood in Area A.3.3 - FW Line Break
%FLOOD08D	3.50E-05	1.96E-05	Flood in Area A.3.3 - Other Line Break
%FLOOD02B	3.09E-05	1.73E-05	Flood in Area A.1.7 - RCIC Line Break

Table B-4

BASIC EVENT IMPORTANCE MEASURES FOR CDF ASSESSMENT FOR SLC OUT OF SERVICE

Event Name	Probability	Fus Ves	Description
%FLOOD12C	2.23E-05	1.25E-05	Flood in Area A.1.4 - RHRB line Break
%FLOOD09A	2.20E-05	1.23E-05	Flood in Area A.2.2 - WO Line Break
%FLOOD16	2.20E-05	1.23E-05	Flood in Area A.4.6 - Any Break
%FLOOD08C	1.86E-05	1.04E-05	Flood in Area A.3.3 - RCIC Line Break
%FLOOD15A	1.50E-05	8.40E-06	Flood in Area A.4.1 - SX-B Line Break
%FLOOD07A	7.50E-06	4.20E-06	Flood in Area A.2.3 - WO Line Break
%FLOOD07B	7.50E-06	4.20E-06	Flood in Area A.2.3 - Other Line Break
%FLOOD03A	7.45E-06	4.17E-06	Flood in Area A.1.9 - RCIC Line Break
%FLOOD04A	7.45E-06	4.17E-06	Flood in Area A.1.10 - RCIC Line Break
%FLOOD05A	7.45E-06	4.17E-06	Flood in Area A.2.5 - WO Line Break
%FLOOD13A	7.45E-06	4.17E-06	Flood in Area A.1.1 - WS Line Break
%FLOOD27A	7.45E-06	4.17E-06	Flood in Area H.1.1 - WS Line Break
%FLOOD28A	7.45E-06	4.17E-06	Flood in Area H.1.2 - WS Line Break
%FLOOD29A	7.45E-06	4.17E-06	Flood in Area H.1.3 - WS Line Break
1CTSYLRGPCFLLR--	2.00E-01	2.67E-06	CONT. CATASTROPHIC FAILURE MODE
1CVPH-SMALLD-F--	1.00E+00	2.67E-06	SMALL DIA VENTS ASSESSED AS UNSUCCESSFUL (4411.06 PROC SECT 2.3 & 2.4)
1CVPH-TEMPF--F--	1.00E-02	2.67E-06	IN CONTAINMENT MOV/AOV FAILS CLOSED DUE TO ENVIRONM. STRESS (LEVEL 1)
1--RX-SPC-VACH--	5.00E-07	2.67E-06	DEP HEP: OP FAILURE OF SPC (LATE) AND VACUUM PUMPS

Table B-5
RISK ASSESSMENT SENSITIVITY RESULTS

Risk Metric	Value	Acceptance Guidelines	Reference
ΔCDF	1.1E-7/yr	<1.0E-06/yr	RG 1.174
ICCDP	1.1E-7	<5.0E-07	RG 1.177
ΔLERF	2.4E-8/yr	<1.0E-07/yr	RG 1.174
ICLERP	2.4E-8	<5.0E-08	RG 1.177

B.3 PARAMETRIC UNCERTAINTY

Consistent with the ASME PRA Standard [Ref. B-2], quantitative parametric uncertainty analyses for both CDF and LERF have been performed using an EPRI method [Ref. B-1] and are summarized in this section. The results of the uncertainty analysis for the proposed CT are compared with the results of the uncertainty analysis performed for the 2006C PRA Update.

The parametric uncertainty analyses are performed using Monte Carlo simulation. The analysis is performed using the EPRI R&R workstation UNCERT software.

B.3.1 Core Damage Frequency Parametric Uncertainty Distribution

The resulting uncertainty distribution for the proposed CT configuration (i.e., $CDF_{SLC-OOS}$) calculated by UNCERT Version 2.3a for CDF is shown in Figure B-1. It summarizes:

- Distribution statistics (e.g., mean, error factor, etc.)
- Probability density chart of the CDF

The approximate error factor (or range factor) for the proposed CT is 2.5, as compared to the error factor of 2.0 for the CL06C base model.

One of the critical aspects of the parametric uncertainty assessments is the desire to ensure that the point estimate calculation performed with the base PRA model (i.e., using CAFTA) produces a point estimate result that is not too dissimilar from the true mean calculation when the correlation effect is accounted for.

Table B-6 provides this comparison for the proposed CT model case (i.e., $CDF_{SLC-OOS}$):

Table B-6

PARAMETER UNCERTAINTY COMPARISON FOR CDF

CDF Parameter	$CDF_{SLC-OOS}$ Result	Code
Point Estimate	9.1E-6/yr	CAFTA
Uncertainty Mean	9.1E-6/yr	UNCERT
Δ	ε	-

The propagated uncertainty mean for $CDF_{SLC-OOS}$ is the same as the $CDF_{SLC-OOS}$ point estimate calculation. If the $CDF_{SLC-OOS}$ propagated uncertainty mean instead of the CL06C CDF_{BASE} propagated uncertainty mean were used to calculate the risk metrics, the results would not differ from those presented in Table 5.3-1.

B.3.2 Large Early Release Frequency (LERF) Parametric Uncertainty Distribution

The same process as used for CDF is also used for LERF. The resulting uncertainty distribution calculated by UNCERT Version 2.3a for LERF is shown in Figure B-2. The figure summarizes the following:

- Distribution statistics (e.g., mean, error factor, etc.)
- Probability density chart of the LERF

The approximate error factor (or range factor) for the proposed CT for the LERF uncertainty distribution is 2.0 (calculated using SQR(95%/5%)), as compared to the error factor of 3.1 for the CL06C base model.

Table B-7 provides a comparison of the PRA LERF point estimate and the propagated uncertainty mean for the proposed CT case (i.e., $LERF_{SLC-OOS}$):

Table B-7

PARAMETER UNCERTAINTY COMPARISON FOR LERF

LERF Parameter	$LERF_{SLC-OOS}$ Result	Code
Point Estimate	8.7E-7/yr	CAFTA
Uncertainty Mean	8.8E-7/yr	UNCERT
Δ	1E-8/yr (1.1%)	-

If the $LERF_{SLC-OOS}$ propagated uncertainty mean (8.8E-7/yr) and the CL06C $LERF_{BASE}$ propagated uncertainty mean (1.2E-7/yr) are used to calculate the risk metrics, the results would change in the second decimal place compared to the results shown in Table 5.3-1 (i.e., non-significant change).

Figure B-1

CDF PARAMETRIC UNCERTAINTY DISTRIBUTION
FOR THE PROPOSED COMPLETION TIME

UNCERT 2.3a
COREDAMAGE.CUT
CL206C-UNCERT.BE
Samples 50,000
Random Seed Auto

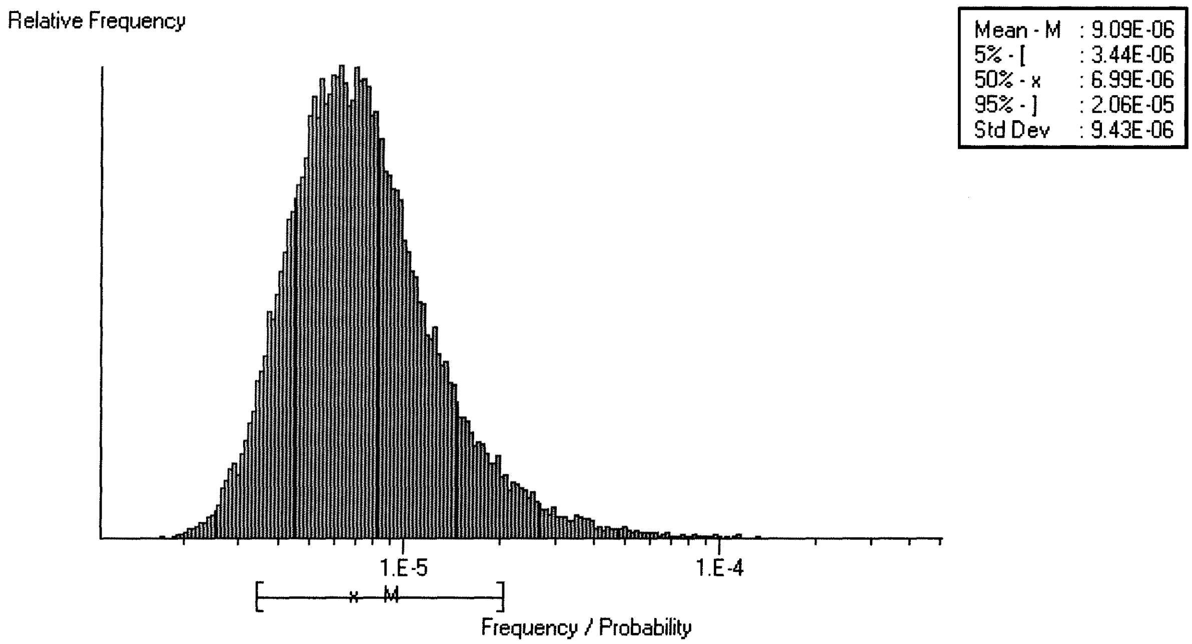
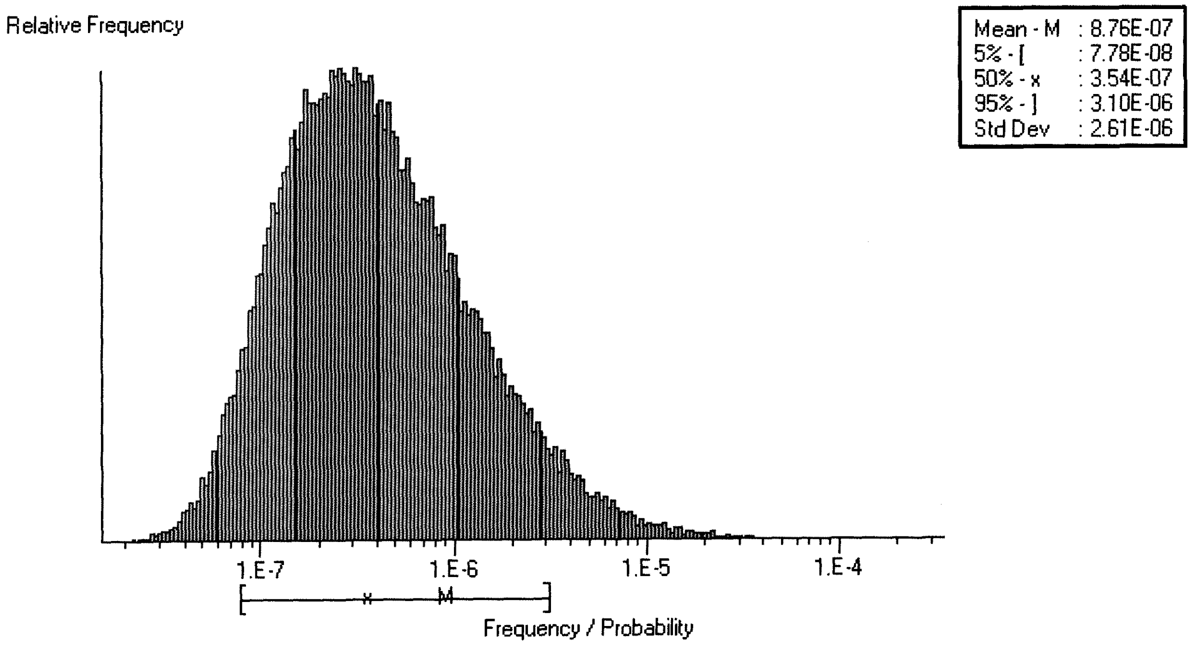


Figure B-2

LERF PARAMETRIC UNCERTAINTY DISTRIBUTION
FOR THE PROPOSED COMPLETION TIME

UNCERT 2.3a
LERF-TOT.CUT
CL06C-UNCERT.BE
Samples 50,000
Random Seed Auto



B.5 REFERENCES

[B-1] *Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments*, EPRI Report 1016737, Palo Alto, CA: 2008.

[B-2] ASME/ANS RA-Sa-2009, "Addenda to RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," February 2009.

Appendix C

BWROG Assessment of NRC Information Notice 2007-07

The BWROG assessment of NRC Information Notice 2007-07 is provided in this appendix. This assessment discusses the low-likelihood scenario of fire-induced failure to scram. Refer to Section A.4.3 of this risk assessment.

BWROG Assessment of NRC Information Notice 2007-07

1.0) Summary:

This assessment addresses the condition described by the NRC in NRC Information Notice 2007-07 and in the inspection report referenced therein.

The overall assessment of the condition described in NRC Information Notice 2007-07 by the BWROG is that it represents a condition with a low likelihood of occurrence, with low safety significance and with multiple layers of defense-in-depth currently in place each with the capability to either prevent the condition from occurring or to effectively mitigate the effects of the occurrence without consequence.

It is the position of the BWROG that all BWRs should have a manual operator action tied to their post-fire safe shutdown procedures instructing the operator to implement the requirements of EO-113 should the fire impact the ability to scram. This manual operator action should be endorsed by the NRC for use in both III.G.1 and 2 areas, as well as, III.G.3 and III.L areas. The evaluation provided in this paper and the limited likelihood of occurrence of the condition are considered to be sufficient justification for concluding that this manual operator action is both feasible and reliable.

It is recommended that each BWR review this assessment and assure that their plant specific conditions are consistent with the measures described herein. As a minimum, each licensee should assure that the EOP action to implement the requirements of EO-113 is linked to their post-fire safe shutdown procedures.

2.0) Description of Issue:

NRC Information Notice 2007-07 postulates a condition where two (2) hot shorts could result in the failure of one of four control rods groups to insert during a manual scram from the Control Room. The IN further postulates that with the reactor in this condition the operator rapidly depressurizes the reactor and re-floods the reactor with cold water using a low pressure system. The IN further states:

> "By design, the negative reactivity, added by all four rod groups during a scram, provides adequate shutdown margin to offset the positive void and temperature reactivity [that] would have been added to the vessel [during such a shutdown sequence]".

3.0) Scram System Design Description:

Typically, the Reactor Protection System (RPS) for a BWR consists of two (2) Trip Systems (A and B), each containing two Trip Channels (A1, A2, B1, B2) of sensors and logic. The four channels contain automatic scram logic for the monitored parameters listed below, each of which has at least one input to each of the logic channels:

- Scram Discharge Volume Water Level

BWROG Assessment of NRC Information Notice 2007-07

- Main Steam Line Isolation Valve Position
- Turbine Stop Valve Position
- Turbine Control Valve Fast Closure
- Reactor Vessel Water Level
- Main Steam line Radiation
- Neutron Monitoring System
- Primary Containment Pressure
- Reactor Vessel Pressure

The RPS automatic trip logic requires at least one channel in each trip system to be tripped in order to cause a scram. This is referred to as one-out-of-two-taken-twice trip logic.

The two RPS Trip Systems are independently powered from their respective RPS Buses. The trip channels (A1, A2, B1, B2) associated with each Trip System (A, B) operate the automatic scram Trip Logic Relays (K14 A-H). The RPS auto scram logic string is sometimes referred to as "trip actuator" or "actuation" logic because the output of the logic is what actually causes the control rods to scram by de-energizing the pilot scram solenoid valves.

The RPS circuits are a fail-safe design in that the circuits are normally energized, and the loss of power, including the loss of offsite power, will initiate the scram.

Once the scram has occurred, re-energization of the RPS logic will not, in and of itself, cause the control rod movement necessary to re-establish reactor criticality.

4.0) <u>Evaluation:</u>

The evaluation performed is divided into two sections. The first section performs a circuit analysis of the scram circuitry. This portion of the evaluation examines the scram circuitry in an effort to determine the set of hot shorts that, should they occur, have the potential to prevent one or more rod groups from inserting. The first section also addresses the significance of the postulated condition and the features currently in place with the capability to prevent or mitigate the effects of the condition. The second section addresses the implications for Appendix R Compliance given the required circuit design for this important safety system and given the potential ramifications of the hot shorts postulated in the first section.

4.1) <u>Circuit Analysis:</u>

Figures 1 through 4 attached to this paper shows portions of the scram circuitry for a typical BWR. Three (3) separate cases involving up to two hot shorts are discussed in this paper.

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Case I: (Refer to Figure 1)

Case I attempted to identify the condition described in IN 2007-07. IN 2007-07 concluded that two (2) hot shorts were required to prevent a single rod group from scramming.

The BWROG, however, was unable to identify any circuitry where two (2) fire-induced hot shorts would prevent one of four scram rod groups from inserting.

The BWROG identified that a single hot short in either of the divisionalized trip logics can prevent the scram of a single rod group. This finding is different than the conclusion in IN 2007-07. The finding of the BWROG assessment is a direct consequence of the 1 out of 2 taken twice logic used in the design for the scram function.

The single hot short with the potential for preventing the scramming of a single rod group could occur in either the Trip System A or B Relay Panel. [Refer to Figure 1 attached for a description of the location of the subject hot short, labeled as "Hot Short 1".] The hot short must occur prior to the operator scramming the reactor. The location of the hot short shown in Figure 1 would be either in one of the Trip System Relay Panels or in a raceway carrying the circuit from the Trip System Relay Panel to the Scram Pilot Solenoid Valves. (Note: For some licensees, the relay panels are located in separate relay rooms outside of the main control room.)

For the hot short in this case to affect the reactivity function, it must remain in effect until such time when the operator depressurizes the reactor and begins re-flooding with a low pressure system. The Emergency Operating Procedures for a BWR instruct the operator not to depressurize the reactor until reactor level reaches the top of active fuel. In a typical BWR, it will take approximately 20 to 25 minutes of boil-off for reactor level to decrease to the top of active fuel. Industry and NRC cable fire testing have shown that hot shorts last for only a few minutes prior to shorting to ground. [EPRI Testing determined the maximum duration of a hot short was 11.3 minutes. CAROLFIRE Testing determined that the maximum duration of a hot short was 7.6 minutes.]

Therefore, it appears unlikely that the required hot short could last for a sufficient amount of time that the impacted control rod group would fail to insert prior to the time when the EOPs directed the operator to depressurize the reactor.

Case II: (Refer to Figure 2)

Case II is one of two cases identified where two (2) fire-induced hot shorts could prevent a full scram. (Note: No conditions were identified where two (2) fire-induced hot shorts were required to prevent a single rod group from scramming.)

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Refer to Figure 2 attached for the case where two (2) fire-induced hot shorts could prevent a full scram.

This case postulates a condition where two hot shorts just below the manual scam switches for two trip channels can prevent a full scram. The postulated hot shorts could occur in either the main control room operating bench board or in a raceway carrying the trip circuit to one of the Trip System Relay Panels. The hot short will keep the K15 relays from de-energizing and this will subsequently keep the K14 relays energized. By keeping the K14 relays energized, as shown in Figure 1, none of the rod groups will de-energize and none will insert. Figure 2 shows the location of the two individual hot shorts. One affects the K15B relay and one affects the K15D relay. The K15 relays are de-energized by actuating the manual scram switches in the Control Room on the main control board. Keeping the K15 relays energized by the hot shots shown in Figure 2, will keep the K14 relays energized, as shown in Figures 3. Keeping the K14 relays energized, as shown in Figure 3, will prevent rod group insertion, as shown in Figure 1.

For this case, however, there are numerous other inputs into the scram logic that can override the effects of the hot short affecting the K15 relays. Refer to Figures 3 and 4 for the additional input signals to the scram function. For example, as shown on Figure 4, closure of the MSIVs or reactor level reaching the +13" level will override the effects of the hot shorts affecting the K15 relays and result in a de-energization of the K14 relays and full rod insertion.

Therefore, it appears unlikely that the required hot shorts, even if they were to co-exist, could prevent the scram and cause the reactivity transient described in the IN. This is true because the effect of the hot short would be overriddened by the reduction in reactor level that would be necessary before the operator would take the action to depressurize the reactor prior to making up with a low pressure system.

<u>Case III:</u> (Refer to Figure 3) (Limited to the Trip System Relay Panels)

Case III is similar to Case II. Hot shorts are postulated in the locations shown in Figure 3, the K14 relays will again remain energized. The energization of the K14 relays will prevent the scram for all rod groups.

For this case to occur, the fire must sufficiently damage two separate circuits and the fire induced damage must occur on each circuit simultaneously. Industry and NRC cable fire testing have shown that hot shorts last for only a few minutes prior to shorting to ground. [EPRI Testing determined the maximum duration of a hot short was 11.3 minutes. CAROLFIRE Testing determined that the maximum duration of a hot short was 7.6 minutes.]

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> Therefore, it appears unlikely that the required hot shorts would co-exist given that the time required for fire damage to the individual cables and fire propagation between relay compartments to occur.

For all of the cases discussed above, regardless of the number of fire-induced hot shorts postulated, the required hot short configuration must occur prior to the operator scramming the unit. For those configurations requiring more than a single hot short, the two hot shorts must exist coincidentally.

The hot short configurations must remain in effect until such time when the operator depressurizes the reactor and begins re-flooding with a low pressure system. The Emergency Operating Procedures for a BWR instruct the operator not to depressurize the reactor until reactor level reaches the top of active fuel.

Additionally, the scenario described in the IN represents a condition more severe than many BWRs would experience due to the availability of additional safe shutdown system capability. Many BWRs also have high pressure systems available for alternative shutdown at their remote shutdown panel. For a BWR with a high pressure system safe shutdown capability, the time available prior to the need to reduce pressure reactor pressure for injection with either a low pressure system or for shutdown cooling would be extended by a number of hours.

Finally, operators for all BWRs are trained on the use of the Emergency Operating Procedures. EO-113 for each BWR provides clear direction to the to either remove RPS power or the vent the SCRAM air header to achieve a full scram.

4.2) <u>Implications for Appendix R Compliance:</u>

For all plants the main operating bench board is in the main control room. At some plants, the relay panels are located in the main control room. In other plants the relay panels are located in a relay room separate from the main control room. For these latter set of plants, some classify the relay room as III.G.3 areas and some classify the relay room as III.G.1 and 2 areas.

This issue, therefore, has implications for redundant safe shutdown under Appendix R Section III.G.1 and 2 and for alternative and dedicated safe shutdown under the requirements of Appendix R Section III.G.3 and III.L.

With respect to Case I, it is clear that none of the methods available under III.G.2 would be effective in preventing the condition. Protection of the subject circuits with a 3 hour fire rated barrier, with a one hour fire rated barrier with automatic suppression and detection or by separation of 20 feet with automatic suppression and detection and no intervening combustibles, would not prevent the occurrence of this event. Additionally, even if the relay panels for each of the four channels are located in separate control/relay room in separate fire areas, the condition could still occur and 3-hour fire rated barriers

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for each of these postulated fire areas would be ineffective in preventing the occurrence of the condition. The condition postulated in Case I can only be mitigated by the use of a manual operator action consistent with the manual operator actions currently invoked under Emergency Operating Procedure, EO-113.

The conditions described for Cases II and III are similar. Neither of these cases represents a condition that is prevented by the type of redundant train separation invoked under Appendix R, since the postulated hot shorts occur within a single division.

Therefore, the provision of Appendix R cannot be used to address the conditions described in this paper. Re-design of the scram circuitry is not a viable option without compromising the design function of this important safety function. In addition to the features of the RPS system described above, the Alternate Rod Insertion (ARI) system (vents SCRAM air header), Backup Scram Solenoids (vents SCRAM air header), and Standby Liquid Control (SLC) system (inserts sodium pentaborate) provide additional redundant means to achieve reactor shutdown. For areas such as the main Control Room and the Relay Rooms, however, similar fire-induced impacts could be postulated.

This paper has highlighted one example of an area where verbatim compliance with the requirements of Appendix R is insufficient in preventing fire induced damage from potentially impacting safe shutdown. The BWROG believes that this case and, potentially, other like it are the reason why from the initial issuance of Appendix R that certain conditions were considered to be initial boundary conditions for the Appendix R Post-Fire Safe Shutdown Analysis. Assuming that the reactor is scrammed was one of those initial boundary conditions given for the Post-Fire Safe Shutdown Analysis. NRC Generic letter 86-10 in the Response to Question 3.8.4, Control Room Fire Considerations, endorsed the assumption of a reactor trip prior to evacuating the Control Room. Based on this and on the fail-safe nature of the reactor protection system, many licensees assumed and the NRC accepted that a reactor trip was an initial boundary condition for the start of the post-fire safe shutdown analysis, i.e. the plant is scrammed prior to the scram circuitry being damaged by the fire.

Although the BWROG believes that the prior industry position related to the scram is correct and its use provides for a safe plant design, the BWROG also recognizes that fires have some limited potential to impact the scram capability. As a precaution, it is the position of the BWROG that all BWRs should have a manual operator action tied to their post-fire safe shutdown procedures instructing the operator to implement the requirements of EO-113 should the fire impact the ability to scram. This manual operator action should be endorsed by the NRC for use in both III.G.1 and III.G.2 areas, as well as, III.G.3 and III.L areas. The evaluation provided in this paper and the limited likelihood of occurrence of the condition are considered to be sufficient justification for the feasibility and reliability of this manual operator action.

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5.0) Risk Assessment:

Given the unlikely set of circumstances required for this condition to occur and to remain in effect until such time that it could pose a beyond design basis concern to the reactor, the risk associated with this issue is judged to be low.

6.0) Safety Assessment:

Given the fact that there are multiple barriers (circuit failure characteristics, design features, procedural guidance and rigorous operator training) in place to prevent the occurrence of this condition, the safety significance of this issue is also judged to be very low.

7.0) Conclusions and Recommendations:

This assessment addresses the condition described by the NRC in NRC Information Notice 2007-07 and in the inspection report referenced therein.

The overall assessment of the condition described in NRC Information Notice 2007-07 by the BWROG is that it represents a condition with a low likelihood of occurrence, with low safety significance and with multiple layers of defense-in-depth currently in place each with the capability to either prevent the condition from occurring or to effectively mitigate the effects of the occurrence without consequence.

It is the position of the BWROG that all BWRs should have a manual operator action tied to their post-fire safe shutdown procedures instructing the operator to implement the requirements of EO-113 should the fire impact the ability to scram. This manual operator action should be endorsed by the NRC for use in both III.G.1 and 2 areas, as well as, III.G.3 and III.L areas. The evaluation provided in this paper and the limited likelihood of occurrence of the condition are considered to be sufficient justification for concluding that this manual operator action is both feasible and reliable.

It is recommended that each BWR review this assessment and assure that their plant specific conditions are consistent with the measures described herein. As a minimum, each licensee should assure that the EOP action to implement the requirements of EO-113 is linked to their post-fire safe shutdown procedures.

Prepared by: Thomas A. Gorman **Date:** 10/16/2007
Thomas A. Gorman, PE, SFPE

Reviewed by: Gary Birmingham **Date:** 11/13/2007
Gary S. Birmingham

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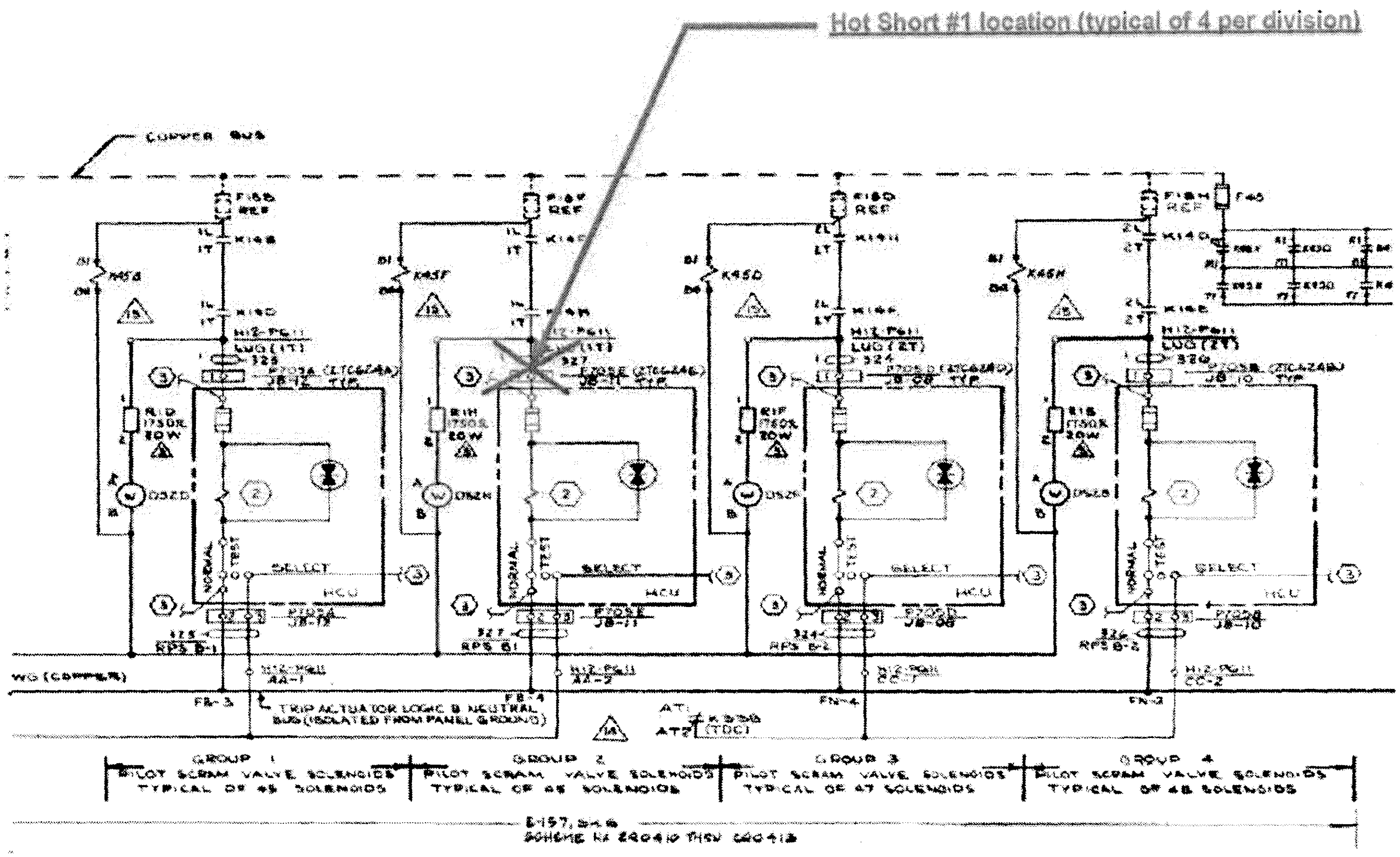

Figure 1 - Relay Panel Circuitry Controlling Individual Rod Groups (typical of two Trip Systems)

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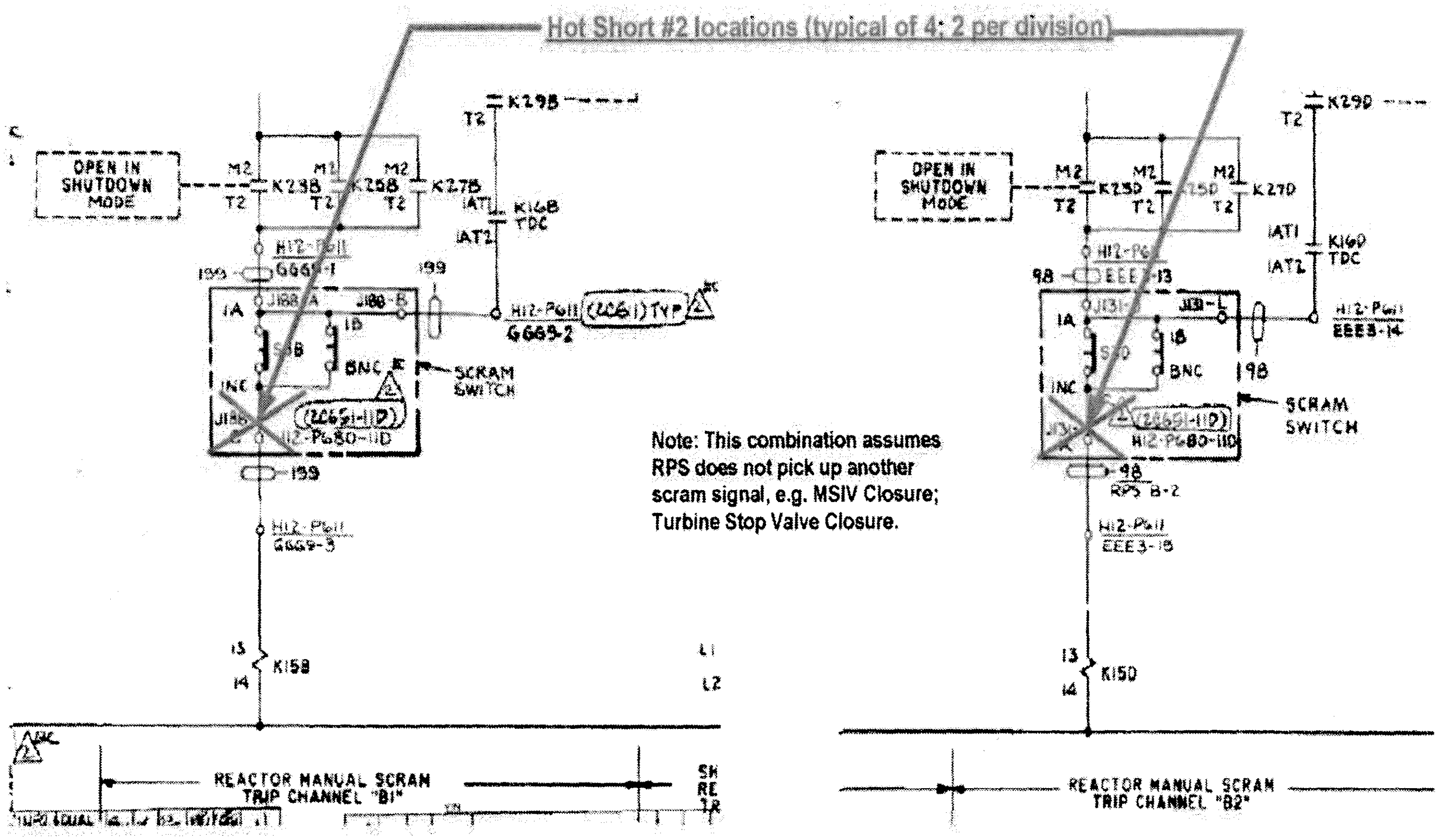

Figure 2 - Manual Scram Circuitry - Typical of two Trip Systems

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Refer to Figure 4 for the remaining set of contacts that affect the automatic scram function

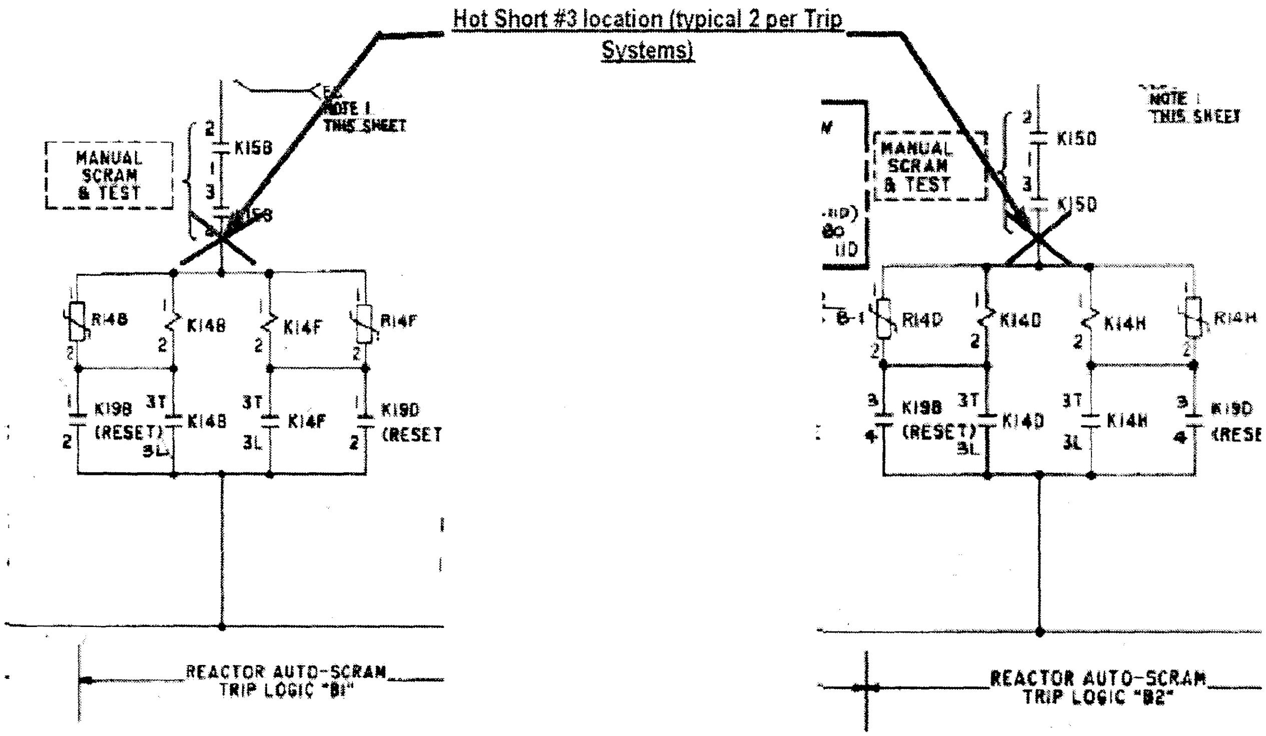

Figure 3 - Reactor Auto-Scram Circuitry - Typical of four Trip Channels in two Trip Systems

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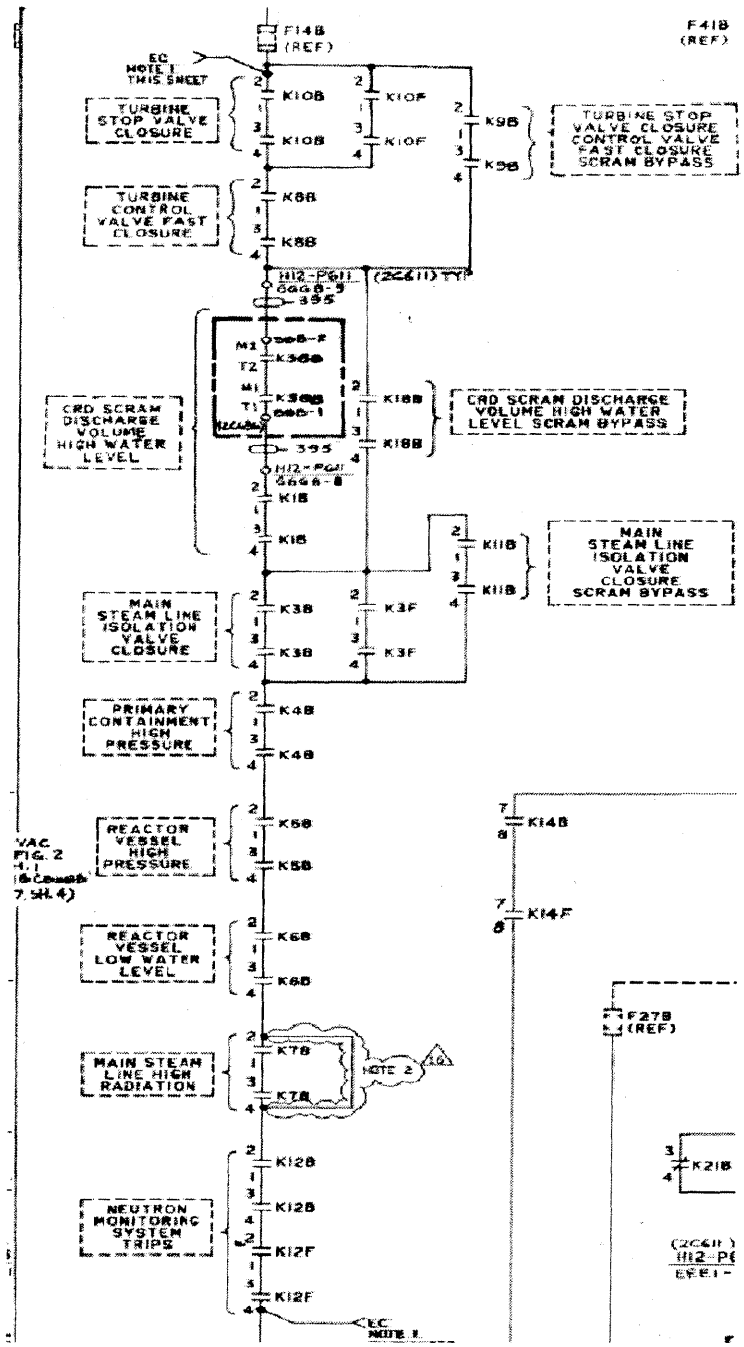

Figure 4- Balance of Auto-Scram Circuitry - (typical of 4 Trip Channels)