Maria Korsnick
Site Vice President

R.E. Ginna Nuclear Power Plant, LLC
1503 Lake Road
Ontario, New York 14519-9364
585.771.5200
585.771.3943 Fax
maria.korsnick@costellation.com

Constellation Energy
Generation Group

February 12, 2007

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: **R.E. Ginna Nuclear Power Plant**
Docket No. 50-244

Response to Request for Additional Information and Revised Mark-up Regarding Service Water Pump Operability Requirements for the R.E. Ginna Nuclear Power Plant

Reference 1: Letter to USNRC Document Control Desk from Mary Korsnick (Ginna), License Amendment Request: Change to Technical Specification 3.7.8, Service Water (SW) from an Electrical Train Based to a Pump Based Specification, dated September 29, 2006

Reference 2: Letter from Patrick Milano (NRC) to Mary Korsnick (Ginna), Request for Additional Information Regarding Service Water Pump Operability Requirements, R.E. Ginna Nuclear Power Plant, dated December 13, 2006 (TAC No. MD3118)

Reference 3: Letter to USNRC Document Control Desk from Mary Korsnick (Ginna), Addendum to September 29, 2006 Submittal RE: Change to Technical Specification 3.7.8, Service Water (SW) from an Electrical Train Based to a Pump Based Specification, dated December 7, 2006.

On September 29, 2006 R.E. Ginna Nuclear Power Plant, LLC (Ginna LLC) submitted a request to change Technical Specification (TS) 3.7.8, Service Water (SW) from an Electrical Train Based to a Pump Based Specification (Reference 1). On December 13, 2006 NRC issued a Request for Additional Information (RAI) regarding that submittal (Reference 2). Reference 3 contains information relevant to the RAI. Attachment 1 to this letter contains the Ginna LLC response to the RAI, Attachment 2 contains a new commitment and Attachment 3 contains a revised mark-up of the proposed change to TS 3.7.8. The revised mark-up provides consistency with other TS

A001

sections by clarifying the wording of Condition D to address more than three (3) SW pumps inoperable.

Should you have questions regarding the information in this submittal, please contact Mr. Robert Randall at (585) 771-5219 or Robert.Randall@constellation.com.

Very truly yours,

Mary G. Korsnick

Mary G. Korsnick

STATE OF NEW YORK :
: TO WIT:
COUNTY OF WAYNE :

I, Mary G. Korsnick, begin duly sworn, state that I am Vice President, R.E. Ginna Nuclear Power Plant, LLC (Ginna LLC), and that I am duly authorized to execute and file this request on behalf of Ginna LLC. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other Ginna LLC employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Mary G. Korsnick

Subscribed and sworn before me, a Notary Public in and for the State of New York and County of MONROE, this 12 day of February, 2007.

WITNESS my Hand and Notarial Seal:

Sharon L. Miller
Notary Public

SHARON L. MILLER
Notary Public, State of New York
Registration No. 01MI6017755
Monroe County
Commission Expires December 21, 2010

My Commission Expires:

12-21-10
Date

MK/MR

Attachments: (1) Response to Request for Additional Information
(2) List of Regulatory Commitments
(3) Proposed Technical Specification Changes (Revised Mark-up)

cc: S. J. Collins, NRC
D.V. Pickett, NRC
Resident Inspector, NRC (Ginna)
P.D. Eddy, NYSDPS
J. P. Spath, NYSERDA

Response to Request for Additional Information

To complete its review, the Nuclear Regulatory Commission (NRC) staff requested the following additional information:

Question 1 Regulatory Guide (RG) 1.177, "An Approach for Plant-Specific, Risk-Informed Decision making: Technical Specifications," indicates in Sections 2.1 and 2.2 that licensees should address compliance with current regulations and traditional engineering considerations, respectively, when requesting risk-informed changes to TS requirements. Provide this information consistent with the guidance provided in RG 1.177.

Note: In a conference call on December 13, 2006 the NRC Staff requested that the response to this question be formatted to match the RG format. To aid the reader the text below in *italics* is taken directly from RG 1.177.

2.1 Compliance with Current Regulations:

In evaluating proposed changes to TS, the licensee must ensure that the current regulations, orders, and license conditions are met, consistent with Principle 1 of risk-informed regulation. The NRC regulations specific to TS are stated in 10 CFR 50.36, "Technical Specifications." Additional information with regard to the NRC's policies on TS is contained in the "Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors" (58 FR 39132) of July 22, 1993 (Ref. 9). These documents define the main elements of TS and provide criteria for items to be included in the TS. The final policy statement and the statement of considerations for 10 CFR 50.36 of July 19, 1995 (Ref. 10), also discuss the use of probabilistic approaches to improve TS. Regulations regarding application for and issuance of license amendments are found in 10 CFR 50.90, 50.91, and 50.92. In addition, the licensee should ensure that any discrepancies between the proposed TS change and licensee commitments are identified and considered in the evaluation.

Response:

On page 10 of Attachment (1) to Reference 1, Ginna stated that "...such activities will be conducted in compliance with the Commission's regulations..." To validate that statement the relevant regulation, 10CFR50.36, was reviewed and it was determined that the requested amendment will not violate the provisions of that regulation. Specifically, the change was focused on the Limiting Condition for Operation (LCO), with a minor wording change to a Surveillance Requirement (SR) for clarification. Neither of these changes will result in a non compliance with the associated sections of 10CFR50.36. In addition, a search of Ginna commitments indicated no regulatory commitments which conflict with the proposed amendment.

2.2 Traditional Engineering Considerations:

2.2.1 Defense in depth

The engineering evaluation conducted should determine whether the impact of the proposed TS change is consistent with the defense-in-depth philosophy. In this regard, the intent of the principle is to ensure that the philosophy of defense in depth is maintained, not to prevent changes in the way defense in depth is achieved. The defense-in-depth philosophy has traditionally been applied in reactor design and operation to provide multiple means to accomplish safety functions and prevent the release of radioactive material. It has been and continues to be an effective way to account for uncertainties in equipment and human performance. When a comprehensive risk analysis can be performed, it can be used to help determine the appropriate extent of defense in depth (e.g., balance among core damage prevention, containment failures, and consequence mitigation) to ensure protection of public health and safety. When a comprehensive risk analysis is not or cannot be performed, traditional defense-in-depth considerations should be used or maintained to account for uncertainties. The evaluation should consider the intent of the general design criteria, national standards, and engineering principles such as the single failure criterion. Further, the evaluation should consider the impact of the proposed TS change on barriers (both preventive and mitigative) to core damage, containment failure or bypass, and the balance among defense-in-depth attributes. As stated earlier, the licensee should select the engineering analysis techniques, whether quantitative or qualitative, traditional or probabilistic, appropriate to the proposed TS change.

The licensee should assess whether the proposed TS change meets the defense-in-depth principle. Defense in depth consists of a number of elements as summarized below. These elements can be used as guidelines for assessing defense in depth. Other equivalent acceptance guidelines may also be used.

Consistency with the defense-in-depth philosophy is maintained if:

> *A reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation is preserved, i.e., the proposed change in a TS has not significantly changed the balance among these principles of prevention and mitigation, to the extent that such balance is needed to meet the acceptance criteria of the specific design basis accidents and transients, consistent with 10 CFR 50.36. TS change requests should consider whether the anticipated operational changes associated with a TS change could introduce new accidents or transients or could increase the likelihood of an accident or transient (as is required by 10 CFR 50.92).*

> <u>Response:</u>

> 10CFR50.36 states in part, "(2) *Limiting conditions for operation.* (i) Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a

nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met." The safety related function of the SW System is to provide cooling for safety related equipment, mitigate the containment response effects of a Main Steam Line Break (MSLB) and design basis Loss of Coolant Accident (LOCA), and provide long term containment and core cooling in the event of a LOCA. The most limiting phase of either accident for SW availability is the recirculation phase of the design basis Loss of Coolant Accident (LOCA). Two SW pumps are required for this phase to ensure appropriate SW flow to all components. Reference 1, Attachment 1, Table 3 demonstrates that the proposed change ensures that two SW pumps will be available for accident mitigation, using industry accepted single failure assumptions. Therefore the proposed change maintains defense in depth consistent with 10CFR50.36 requirements, without a shift in the balance between prevention and mitigation. 10CFR50.92 requirements are addressed in Reference 1, Attachment 1, Section 5, No Significant Hazard Determination.

Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided, e.g., use of high reliability estimates that are primarily based on optimistic program assumptions.

Response:

Reference 1, Attachment 2, Section 1.1.1.1 details the availability and reliability data assumed in the risk analysis. The information indicates that the proposed change in allowed outage time will not cause future unavailability to increase, largely because all planned maintenance is currently performed on line. SW pump reliability estimates are realistic, as they are based on generic industry data combined with actual plant data.

System redundancy, independence, and diversity are maintained commensurate with the expected frequency and consequences of challenges to the system, e.g., there are no risk outliers. The following items should be considered.

> *Whether there are appropriate restrictions in place to preclude simultaneous equipment outages that would erode the principles of redundancy and diversity,*
>
> Response:
>
> Reference 1, Attachment 1, Table 3 indicates the built in protection of redundancy due to restrictions relating to Diesel Generator availability, thereby protecting the redundancy of the system from a loss of power scenario. Ginna's Safety Function Determination Program required by TS 5.5.14 also ensures that there is no loss of safety function.
>
> Additionally, Ginna's work control process maintains redundancy through extensive work planning, scheduling and review. On line risk monitors are used in the planning and scheduling process to ensure that the overall plant risk is maintained within acceptable levels in accordance with 10CFR50.65. On-line work is scheduled, monitored and controlled such that key system functions are

protected by a Defense-in-Depth strategy and the plant is maintained within an acceptable risk envelope by incorporating results of risk management tools. On-line maintenance is scheduled and performed such that unavailability time is minimized.

Whether compensatory actions to be taken when entering the modified AOT for pre-planned maintenance are identified,

Response:

Compensatory measures are not required by the TS itself because the requirements are structured to maintain the necessary redundancy and diversity. However, as an added layer of conservatism, Ginna's scheduling and risk/work management procedures identify and implement compensatory measures where appropriate.

Whether voluntary removal of equipment from service during plant operation should not be scheduled when adverse weather conditions are predicted or at times when the plant may be subjected to other abnormal conditions

Response:

Ginna's scheduling and risk/work management procedures evaluate the potential impact of severe weather or other external conditions relative to the proposed scheduled maintenance. The impact is evaluated if such weather conditions are imminent or have a high probability of occurring during the planned out-of-service duration. Consideration is given to rescheduling or expediting the return to service of equipment important to safety.

Whether the impact of the TS change on the safety function should be taken into consideration. For example, what is the impact of a change in the AOT for the low-pressure safety injection system on the overall availability and reliability of the low-pressure injection function?

Response:

Per Reference 1, Attachment 2, Section 1.1.1.1, a review of the historical unavailability of the SW pumps for the four year data window used to develop the above data indicates that all but 78.3 hours (or 2.1%) of SW pump unavailability occurred while the plant was on line. Further, the entire 78.3 hours was unplanned maintenance, such that all planned maintenance of the SW pumps occurred on-line. This is due to the fact that the current Technical Specification (using the pre-uprate definition of train operability) allowed a single SW pump to be removed from service indefinitely, provided the other pump powered by the same electrical train was operable. Consequently, scheduled maintenance (as well as most corrective maintenance) was performed on-line. Since all planned maintenance was previously performed on-line, no increase in on-line SW pump unavailability is planned or anticipated.

Defenses against potential common cause failures are maintained and the potential for introduction of new common cause failure mechanisms is assessed, e.g., TS change requests should consider whether the anticipated operational changes associated with a change in an AOT or STI could introduce any new common cause failure modes not previously considered.

Response:

The method or mode of operation or the configuration of the SW system is not changed by this amendment. The system will continue to be operated, maintained and tested in the same manner as before. Therefore, no new common cause mechanisms are introduced. Additionally, Ginna is committing to an administrative requirement to evaluate for common cause in the event of a pump failure (Attachment 3).

Independence of physical barriers is not degraded, e.g., TS change requests should address a means of ensuring that the independence of barriers has not been degraded by the TS change (e.g., when changing TS for containment systems).

Response:

The relationship of the SW system to the individual barriers has will not change as a result of this amendment. The SW system will continue to support core cooling and containment cooling as before. Reference 1, Attachment 1 (specifically Table 3), demonstrate that adequate SW will be available to support the system design function.

Defenses against human errors are maintained, e.g., TS change requests should consider whether the anticipated operation changes associated with a change in an AOT or STI could change the expected operator response or introduce any new human errors not previously considered, such as the change from performing maintenance during shutdown to performing maintenance at power when different personnel and different activities may be involved.

Response:

Operator response is not expected to change during normal, abnormal or emergency operating conditions. Any minor procedure changes will be addressed, and appropriate training conducted, as part of the change implementation process. Also, and as stated above, normal pump maintenance is currently performed on-line, which is not expected to change as a result of the amendment. Therefore, defense against human errors is maintained.

The intent of the General Design Criteria in Appendix A to 10 CFR Part 50 is maintained.

Response:

The following General Design Criteria were reviewed to ensure the intent of each criteria is maintained, considering the proposed amendment:

IV. Fluid Systems

Criterion 34--Residual Heat Removal (RHR)

This system is indirectly supported by the SW system via the Component Cooling Water (CCW) System. The change will not impact the ability of the SW system to support the RHR system. The RHR system will continue to receive sufficient cooling to perform its function in both the accident mode or during normal operation, given industry accepted power and single failure assumptions.

Criterion 35--Emergency Core Cooling System (ECCS)

This system is supported by the SW system. The change will not impact the ability of the SW system to support the ECCS system. The ECCS system will continue to receive sufficient SW flow to perform its function in both the accident mode or during normal operation, assuming industry accepted power and single failure assumptions.

Criterion 38--Containment heat removal

This system, specifically the Containment Recirc Fan Coolers (CRFCs), is supported by the SW system. The change will not impact the ability of the SW system to support the CRFCs. Analysis shows that two SW pumps are required to support the all cooling loads during Loss of Coolant Accidents. The SW system will continue meet this requirement assuming industry accepted power and single failure assumptions.

Criterion 44--Cooling water

The ability of the SW system to transfer heat from structures, systems and components important to safety to the ultimate heat sink is not diminished by this change. Reference 1, Attachment 1, Table 3 demonstrates that the system maintains suitable redundancy (i.e., two pumps available) assuming off-site power is unavailable, and a single failure. System interconnections, leak detection and isolation capabilities are not affected by this change.

Criterion 45--Inspection of cooling water system

The ability to inspect, or frequency or method of inspection, of the SW system is not affected by this change.

Criterion 46--Testing of cooling water system

The ability to test, or frequency or method of testing, is not affected by this change.

2.2.2 Safety Margins

The engineering evaluation conducted should assess whether the impact of the proposed TS change is consistent with the principle that sufficient safety margins are maintained (Principle 3). An acceptable set of guidelines for making that assessment are summarized below. Other equivalent decision guidelines are acceptable.

Sufficient safety margins are maintained when:

> *Codes and standards (e.g., American Society of Mechanical Engineers (ASME), Institute of Electrical and Electronic Engineers (IEEE) or alternatives approved for use by the NRC are met, e.g., the proposed TS AOT or STI change is not in conflict with approved Codes and standards relevant to the subject system.*
>
> Response:
>
> A search was conducted of the Ginna UFSAR, the Ginna Service Water System Reliability Optimization Program (SWSROP), and NRC Generic Letter 89-13. The applicable code or standard potentially affected by this change is ANSI/ANS-58.9, Single Failure Criteria for Light Water Reactor Safety-Related Fluid Systems. Reference 1, Attachment 1, Table 3 demonstrates the different failure combinations and associated action statements. The summary for that section states, "The technical evaluation demonstrates that the required number of SW pumps will be available at all times given single failure criteria, or immediate action will be taken to place the plant in a safe mode." Therefore the safety margin relative to codes and standards is maintained.
>
> *Safety analysis acceptance criteria in the Final Safety Analysis Report (FSAR) are met, or proposed revisions provide sufficient margin to account for analysis and data uncertainties, e.g., the proposed TS AOT or STI change does not adversely affect any assumptions or inputs to the safety analysis, or, if such inputs are affected, justification is provided to ensure sufficient safety margin will continue to exist. For TS AOT changes, an assessment should be made of the effect on the FSAR acceptance criteria assuming the plant is in the AOT (i.e., the subject equipment is inoperable) and there are no additional failures. Such an assessment should result in the identification of all situations in which entry into the proposed AOT could result in failure to meet an intended safety function.*
>
> Response:
>
> This assessment was performed in Reference 1, Attachment 1. Table 3 demonstrates that the minimum number of SW pumps assumed to be operating in the accident analysis (2) is maintained for any combinations resulting in an AOT, assuming no additional failures. If not, immediate action is taken to place the plant in a safe mode. Therefore, the safety analysis criteria is maintained following this change.

Question 2 **In Section 1.1.7.4 of Attachment 2 of the application, credit is taken in the risk analysis for repair and recovery of an inoperable SW pump. The NRC staff does not consider it appropriate to assume recovery of inoperable equipment as part of the justification that the risk associated with that equipment being inoperable is small. Therefore, provide revised risk**

calculations that do not assume any credit for repair of inoperable SW pumps.

Response:

SW pump repair is not credited post-trip. The recoveries are only applied to the 'Loss of Service Water' initiating event fault tree model, not the post-initiator failure of SW pumps. It was conservatively considered that any shutdowns caused by exceeding the allowed out-of-service times would result in a loss of service water initiating event. To develop a realistic initiating event frequency, it was considered that a failed SW pump could potentially be repaired prior to the failure of a second SW pump. As these recoveries are considered pre-trip with no other initiating event in progress, all available plant resources could, and likely would, be focused on restoring one of the SW pumps to service, as opposed to a post-transient situation where resources would be focused on safely shutting down the plant and maintaining it in a safe condition. However, as noted in section 1.1.9.3, a sensitivity analysis of these non-recovery factors indicates that the overall results of the analysis have very little sensitivity to these events. Therefore, to eliminate any concerns regarding the appropriateness of these recovery events, they have been eliminated from the analysis (i.e., their failure probability has been set to 1.0), and the Incremental Conditional Core Damage Probability (ICCDP) values have been re-calculated.

The results of the updated ICCDP analysis, which also incorporates increased common cause failure probabilities (see RAI 5), an explicit seismic risk evaluation (see RAI 6), and the Incremental Conditional Large Early Release Probability (ICLERP) analysis requested in RAI 3 (which also accounts for the responses to RAI's 2, 5, and 6), are presented below.

ICCDP	
	Single Pump Impacts
SW Pump OOS	14 Day AOT
A	1.006E-08
B	9.977E-09
C	1.006E-08
D	9.977E-09
	Two Pump Impacts
SW Pumps OOS	72 Hour AOT
A & B	5.416E-08
A & C	5.088E-08
A & D	5.416E-08
B & C	5.416E-08
B & D	5.088E-08
C & D	5.416E-08
ICLERP	
	Single Pump Impacts
SW Pump OOS	14 Day AOT
A	8.887E-11
B	8.882E-11
C	8.887E-11
D	8.822E-11
	Two Pump Impacts
SW Pumps OOS	72 Hour AOT
A & B	6.740E-10
A & C	6.329E-10
A & D	6.740E-10
B & C	6.740E-10
B & D	6.321E-10
C & D	6.740E-10

As can be seen by comparing these results to the original submittal results in Reference 1, all ICCDP values have increased. However, all are still well below the 5E-07 criteria. Similarly, all ICLERP values are well below the 5E-08 criteria.

Question 3 In Section 1.1.3 of Attachment 2 no specific incremental conditional large

early release probability (ICLERP) calculations were performed. The stated basis was that even if 100% of the sequences contributing to the incremental conditional core damage probability (ICCDP) became large early releases, all cases calculated would be at least a factor of 10 below the RG 1.177 criteria for ICLERP. This implicitly assumes that any increase in the large early release frequency (LERF) would only arise from sequences contributing to increases in core damage frequency (CDF), and not from existing core damage sequences, which would now become large early releases. Justify this assumption, or provide ICLERP analysis for the proposed changes.

Response:

An ICLERP analysis for the proposed change has been performed. The results of this ICLERP analysis, taking into account the removal of pre-trip SW pump recoveries (RAI 2), increased common cause failure probabilities (RAI 5), and an explicit seismic risk evaluation (RAI 6), are presented in the above response to RAI 2.

Question 4 **In Section 1.1.3 of Attachment 2, the licensee does not identify whether the evaluation of ICCDP was based on a zero maintenance model or on a average maintenance model. Identify the basis of the ICCDP and ICLERP calculations.**

Response:

The evaluation of ICCDP and ICLERP evaluation are based on an average maintenance model. That is, the unavailability of the SW pumps is explicitly addressed, while all other maintenance unavailability events have been set to their average values.

Question 5 **Section 1.1.3 of Attachment 2, the licensee has not provided separate risk calculations for planned vs. unplanned maintenance assuming a higher common-cause failure rate, consistent with RG 1.177, Appendix A, Section A.1.3.2. Provide these calculations of ICCDP and, if necessary (based on the response for RAI 2), of ICLERP.**

Response:

A revised analysis of ICCDP and an analysis of ICLERP have been performed to address this issue. For the updated analysis, increased common cause failure rates for the SW pumps failing to start were used, assuming that the unplanned maintenance, for either one or two pumps out of service, was caused by a failure or failures of a stand-by SW pump to start. The increased common cause failure rates used are the conditional common cause failure rates, given that one or two pumps have already failed. To address the potential for common cause failure of operating pumps, administrative controls will be put in place to require that if one SW pump is inoperable due to equipment failure, and a second SW pump fails before the first pump is returned to service, then an evaluation of possible common cause and determination of the operability of the remaining pumps will be performed within 24 hours of the second failure.

A sensitivity analysis was performed to demonstrate that the risk associated with two pumps failed with the remaining pumps potentially subject to the same common cause failure is acceptable for a 24 hour time period. For all combinations of two pumps out of service, the higher common-cause failure rates provide at least 33 hours of exposure before the ICCDP exceeds 5.0E-07 or the ICLERP exceeds 5E-08. Thus, even if the full 24 hours were used, and then the plant entered a TS required shutdown, the plant would be at hot shutdown within 30 hours, prior to the accumulated risk exceeding the allowed thresholds.

Based on the above requirement, the revised ICCDP analysis and the ICLERP analyses include increased common cause failure to run rates for a single SW pump out of service, or two SW pumps out of service where one is for planned maintenance. This is necessary since an explicit determination that no common cause failure mode exists is not required for these cases. The results of this revised analysis, incorporating the increased fail to start and fail to run common cause failure rates are included in the updated ICCDP and ICLERP values presented in the response to RAI 2.

Question 6 **Section 1.1.5 of Attachment 2, the Ginna probabilistic safety assessment (PSA) model covers both internal and external events, including fire and external flooding. However, the licensee has not provided any quality information for the fire and external flooding models, nor provided any disposition of other external events which are not quantitatively addressed by the Ginna PSA model. Since the potential risk increase due to the proposed changes is due to loss-of-coolant accidents (LOCAs) of size equivalent to 2-inches diameter or greater (Section 1.1.7.2), confirm that the external events included in the quantitative risk assessment model (i.e., fire and external floods) do not potentially induce LOCAs greater than 2 inches in diameter requiring two SW pumps for mitigation. Otherwise, provide quality information for the external events probabilistic risk assessment model used to support the risk evaluation. In either case, the licensee should disposition seismic risk and other external events risk for this application.**

Response:

Although external floods are addressed in the Ginna PSA model, they do not have the capability to cause a LOCA event of 2" or greater. In order for a LOCA of 2" or greater to be caused by a transient initiator, both power operated relief valves (PORVs) must be challenged due to the RCS pressure excursion caused by the plant trip, and then both PORVs must fail to reseat. Only certain plant trip events create a pressure excursion in the reactor coolant system (RCS) which challenges the PORVs (e.g., large loss of electrical load, complete loss of feedwater, locked RCP rotor, etc.). An external flooding event at Ginna is a slowly developing event (i.e., over several hours) resulting from rising levels in Lake Ontario, or in Deer Creek which runs by the plant. By procedure, prior to water levels rising to the point that plant equipment is impacted, a controlled load reduction is required. The plant should then be at or near hot shutdown conditions prior to any potential equipment impact from the flood. Thus, a plant trip caused by an external flooding event would not challenge the PORVs. Even if the plant was at or near 100% power and operators manually tripped the plant due to imminent impact of rising water on plant equipment, the PORVs would not be challenged since secondary side heat removal capability is more than adequate to remove decay heat levels, preventing primary side heat-up. Thus, for external flooding events a PORV challenge is precluded, and a LOCA of 2" or greater will not result.

Ginna fire events can potentially induce a LOCA greater than 2 inches in diameter. The fire events can result in a reactor trip which challenges two PORVs. If both PORVs fail to reseat, a LOCA of equivalent size of > 2" will result. No peer review has been performed on the fire portion of the Ginna PSA, in part due to the fact that there is no approved standard for fire PRA quality. However, a significant internal review of the fire modeling has taken place. Any time that the Ginna PSA is updated, the model is quantified, and the resulting cutsets are examined for reasonableness. Further, since the fire PSA uses the existing top logic from the internal events PRA, which has been subjected to a Peer Review, once the fire impacts have been incorporated into the model correctly, the validity of the top logic is ensured. Specifically for this submittal, the cutset files were reviewed to ensure that the expected scenario (the transient induced PORV LOCAs described above) appeared. Additionally, fires have a very minor impact on this submittal. For both the one pump and two pumps out of service cases, fire events contribute less than 1% of the overall ICCDP and ICLERP. Thus, the quality of the fire portion of the PSA is considered adequate for the purposes of this submittal.

Ginna Station does not have a seismic PSA. However, to address this issue, a seismic analysis has been performed, similar to the one provided in Ginna's Extended Power Uprate (EPU) submittal, to assess the risk from a seismic event as a result of the AOT extension. As the delta risk calculations are only applicable to LOCAs greater than 2", the additional fragility of the RCS is considered. As with the EPU evaluation, a seismically induced loss of off-site power (LOOP) (with no recovery [i.e. 24 hr LOOP duration]) is calculated to occur at a frequency of 1E-4 per yr. Further, the RCS is considered a seismically rugged structure with a High-Confidence-Low-Probability-of-Failure (HCLPF) of 0.3 g. This equates to a conditional probability of 2.8% that a LOCA occurs given a seismically induced LOOP. This LOCA frequency is parsed over the spectrum of LOCA break sizes. The contribution of the seismic to the SW license amendment request is approximately 9% for the single pump out of service cases, and approximately 17% for the two pumps out of service cases. Given this relatively small contribution, the very conservative modeling used, and the significant margin available below the ICCDP and ICLERP thresholds, this level of detail is considered adequate.

The results of this seismic analysis are included in the updated ICCDP and ICLERP values presented in the response to RAI 2.

Question 7 **In Section 1.1.2 of Attachment 2, it states that the Ginna PSA model, revision 6.2, has a calculated internal and external event CDF of less than 1E-4 per year, and therefore justifies the application of RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," criteria for risk metrics. Provide the specific value of the CDF and LERF of the baseline model.**

Response:

The baseline values for CDF and LERF in revision 6.2 of the Ginna PSA model are 8.82E-05/yr and 4.13E-06/yr, respectively. This includes internal events (including floods), fires, shutdown risk, external floods, and the seismic evaluation discussed in the response to question 6, above. Of these values, 18% of CDF and 14% of LERF is shutdown risk.

Question 8 **In Section 1.1.10.1 of Attachment 2 it states that 33 of 35 B-level facts and observations (F&Os) from the licensee's 2002 peer review have been addressed. However, the NRC staff review of the detailed information on F&Os identified 36 B-level F&Os. It is not apparent which of the B-level F&Os have not yet been completed. Therefore, specifically identify the F&Os not yet resolved and provide a disposition for each of these F&Os relevant to this application.**

Response:

This information was provided in Reference 3.

Question 9 **In Section 1.3 of Attachment 2, the licensee identifies Sections 1.3.4 and 1.3.5 to address the risk of level 2 and external events in configuration risk management program. These sections are missing from the submittal. Provide this missing information.**

Response:

The missing sections were provided in Reference 3.

Attachment (2)

List of Regulatory Commitments

Attachment (2)

List of Regulatory Commitments

The following table identifies those actions committed to by R.E. Ginna Nuclear Power Plant, LLC in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments. Please direct questions regarding these commitments to Robert Randall at (585) 771-5219, or Robert.Randall@constellation.com.

REGULATORY COMMITMENT	DUE DATE
Administrative controls will be put in place to require that if one SW pump is inoperable due to equipment failure, and a second SW pump fails before the first pump is returned to service, an evaluation of possible common cause and determination of the operability of the remaining pumps will be performed within 24 hours of the second failure.	Upon implementation of the TS amendment

Attachment (3)

Proposed Technical Specification Changes (Revised Mark-up)

3.7 PLANT SYSTEMS

3.7.8 Service Water (SW) System

LCO 3.7.8 Four ~~Two~~ SW pumps ~~trains~~ and the SW loop header shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SW pump ~~train~~ inoperable.	A.1 Restore SW pump ~~train~~ to OPERABLE status.	~~72 hours~~ 14 days
B. Two SW pumps inoperable	B.1 Restore SW pump(s) to OPERABLE status	72 hours
~~B.~~ C Required Action and associated Completion Time of Condition A or B not met.	~~B.1~~ C.1 Be in MODE 3. AND ~~B.2~~ C.2 Be in MODE 5.	6 hours 36 hours
~~C.~~ D ~~Two~~ Three or more SW pumps ~~trains~~ or loop header inoperable.	~~C.1~~ D.1 - NOTE - Enter applicable conditions and Required Actions of LCO 3.7.7, "CCW System," for the component cooling water heat exchanger(s) made inoperable by SW. Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.8.1	Verify screenhouse bay water level and temperature are within limits.	24 hours
SR 3.7.8.2	- NOTE - Isolation of SW flow to individual components does not render the SW loop header inoperable. Verify each SW manual, power operated, and automatic valve in the SW ~~train~~ flow path and loop header that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.8.3	Verify all SW loop header cross-tie valves are locked in the correct position.	31 days
SR 3.7.8.4	Verify each SW automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.7.8.5	Verify each SW pump starts automatically on an actual or simulated actuation signal.	24 months