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SOUTHERN NUCLEAR
A SOUTHERN COMPANY

April 4, 2016

Docket Nos.: 50-321
50-366

NL-16-0426

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

Edwin I. Hatch Nuclear Plant Unit 1 and 2
Response to License Amendment Request Unacceptable with Opportunity to Supplement

Ladies and Gentlemen:

By letter dated August 11, 2015, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML115226A276), Southern Nuclear Operating Company (SNC) submitted a license amendment request (LAR) to revise Technical Specifications (TS) to adopt TSTF-500, Direct Current (DC) Electrical Rewrite, for the Edwin I. Hatch Nuclear Plant (HNP), Units 1 and 2.

By letter dated February 3, 2016, the Nuclear Regulatory Commission (NRC) provided the results of their acceptance review of SNC's request to amend the HNP TS. The NRC concluded that to make an assessment as to the acceptability of the LAR, more information was needed regarding the engineering evaluation performed as part of the request to increase the Completion Time (CT) on the station service batteries from 2 to 12 hours. Consequently, the NRC requested that SNC supplement the application to address the information requested in the Enclosure to their February 3, 2016 letter. SNC provided responses to the first five questions of the February 3 letter on March 16, 2016; these questions were related to the NRC acceptance review.

Questions 6 through 10 of the February 3 letter pertain to the technical review of the quantitative probabilistic risk assessment of SNC's RG 1.174 evaluation to increase the battery CT. The response to those questions are contained in the enclosure.

This letter contains no NRC commitments. If you have any questions, please contact Ken McElroy at (205) 992-7369.

Mr. M. D. Meier states he is Regulatory Affairs Vice President of Southern Nuclear Operating Company, is authorized to execute this oath on behalf of Southern Nuclear Operating Company and, to the best of his knowledge and belief, the facts set forth in this letter are true.

Respectfully submitted,

M. D. Meier
Vice President – Regulatory Affairs

MDM/OCV/

Sworn to and subscribed before me this 4th *day of* April *, 2016.*

Notary Public

My commission expires: 12/12/18

Enclosure: Response to Questions 6 through 10

cc: <u>Southern Nuclear Operating Company</u>
Mr. S. E. Kuczynski, Chairman, President & CEO
Mr. D. G. Bost, Executive Vice President & Chief Nuclear Officer
Mr. D. R. Vineyard, Vice President – Hatch
Mr. D. R. Madison, Vice President – Fleet Operations
Mr. B. J. Adams, Vice President – Engineering
Mr. C. R. Pierce, Regulatory Affairs Director
Mr. G. L. Johnson, Regulatory Affairs Manager - Hatch
RType: CHA02.004

<u>U. S. Nuclear Regulatory Commission</u>
Ms. C. Haney, Regional Administrator
Mr. M. D. Orenak, NRR Project Manager – Hatch
Mr. D. H. Hardage, Senior Resident Inspector – Hatch

<u>State of Georgia</u>
Mr. J. H. Turner, Director - Environmental Protection Division

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Edwin I. Hatch Nuclear Plant Units 1 and 2
Response to License Amendment Request Unacceptable with Opportunity to
Supplement

Enclosure

Response to Questions 6 through 10

Plant Hatch Station Service Batteries CT Extension from 2 hours to 12 hours

6. *Regulatory Position 2.3.4 of Regulatory Guide (RG) 1.177, "An approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," states that "Usually, data for outage times correspond to the current [completion time] CT, but not to the proposed CT. Different assumptions are made to estimate the outage time corresponding to the proposed CT. Assumptions concerning changes in maintenance practices under the extended CT regime should be discussed and their impact on the results of the analysis characterized."*

   a. *Discuss the assumption concerning changes in maintenance practices under the extended CT regime and characterize their impact on the results of the analysis.*

      SNC Response:

      It is not the intent of the Edwin I. Hatch Nuclear Plant (HNP) to change the current maintenance practice under the extended CT regime. The evaluation submitted with the LAR was performed assuming a total unavailability of 12 hours per year per battery.

   Regulatory Position 2.3.4 of RG 1.177 further states "[w]hen the risk impact of a CT change is evaluated, the yearly risk impact that is calculated takes into account the outage frequency."

   b. *Please state the battery outage frequency used for estimating the risk impact of the proposed change and explain the assumptions related to the outage frequency used in the calculations. Clarify whether the same assumption related to outage frequency is used in estimating the risk impact of both internal and external events.*

      SNC Response:

      The evaluation submitted with the LAR assumed a total unavailability of 12 hours per year per battery. However, there are no plans to change maintenance practices that would require the use of the additional 10 hours in the LCO CT, and there are also no plans to modify existing battery outage frequencies or add maintenance activities that would require more frequent entry into the LCO. The increase to the current CT for the station service batteries is intended to aid in emergent situations; it is not intended to change the preventative maintenance practices on the station service batteries.

7. *Regulatory Position 2.3.2 of RG 1.177 states that the scope of the analysis to support risk-informed changes to Technical Specification (TS):*

> *...should include all hazard groups (i.e., internal events, internal flood, internal fires, seismic events, high winds, transportation events, and other external hazards) unless it can be shown that the contribution from specific hazard groups does not affect the decision.*

   a. *The licensee stated in the submittal that Edwin I. Hatch Nuclear Plant (HNP) does not have a Seismic probabilistic risk assessment (PRA) model. For estimating incremental conditional core damage probability (ICCDP) and incremental large early release probability (ICLERP) due to seismic and other external events, the licensee stated the ICCDP and ICLERP from those hazards were "conservatively" assumed to be the same as the internal events ICCDP and ICLERP. Please justify that this assumption is valid or provide a bounding assessment of seismic and other external hazards for estimating ICCDP and ICLERP and demonstrate that the updated results meet the acceptance guidelines.*

SNC Response:

The following information provides the bounding assessment of seismic and other external hazards. The results of the bounding assessment justify the use of Internal Events ICCDP and ICLERP to represent risk from the seismic hazard and other external events hazards.

Bounding assessment of seismic event:

Because a seismic PRA is not available for HNP, the risk was calculated using a bounding approach. The following information presents the analysis that bounds the potential seismic impact. The process for analyzing an unscreened external hazard without the use of a full PRA involves the following three steps:

1. Estimate Bounding CDF

2. Evaluate Potential Risk Increases Due to Out of Service Equipment

3. Qualitatively Evaluate Bounding LERF Contribution

Estimate Bounding CDF

The NRC recently published information on the estimates of the seismic risk levels for all plants in the Central and Eastern United States (CEUS) as part of Generic Issue 199 (Generic Issue 199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants," IN2010-18, September 2, 2010). Seismic hazards are a subject of considerable uncertainty. In order to address the changing state of knowledge on seismic hazards, the NRC Staff developed a technical analysis (Staff Report, "Implications of Updated Probabilistic Seismic Hazard Estimates In Central And Eastern United States On Existing Plants, Safety/Risk Assessment," ML100270639, August 2010) that computed conservative estimates of seismic

risk for all plants in the CEUS using estimates of the seismic risk levels developed as part of Generic Issue 199. The NRC Staff analysis used a variety of calculation approaches to compute a conservative estimate of the Seismic Core Damage Frequency (SCDF) using three different seismic hazard sources. The results of these analyses for the HNP site are presented in the Table below.

Estimates of Total Seismic Core Damage Frequency from Appendix D (ML100270756) of NRC Staff Report (ML100270639)

	Calculation Approach				
Hazard Source	**Maximum Spectral Result**	**Simple Average**	**IPEEE Weighted Average**	**Weakest Link Model**	**Highest Estimate**
1989 EPRI	6.3E-7	5.8E-7	5.8E-7	7.4E-7	7.4E-7
1994 LLNL	2.0E-5	7.7E-6	8.1E-6	1.7E-5	2.0E-5
2008 USGS	2.1E-6	1.50E-6	1.4E-6	2.2E-6	2.2E-6

These estimates span a fairly wide range, with the maximum value generated using the 1994 Lawrence Livermore National Lab (LLNL) hazard curve along with conservative estimates of the seismic fragility. Using these conservative analyses, the maximum total SCDF is computed to be 2E-5/yr. This value represents the convolution of the HNP seismic hazard curve with an assumed limiting plant fragility based on the high confidence of low probability of failure (HCLPF) of 0.3g, as reported in the HNP IPEEE. Such methods have been shown to provide a conservative estimate of SCDF. By adopting the maximum estimate generated by various methods, this provides a bounding estimate of the SCDF for use in this evaluation.

<u>Evaluate Potential Risk Increases Due to Out of Service Equipment</u>

The approach taken in the computation of SCDF by the NRC Staff (ML100270639) assumes that the SCDF can be based on the likelihood that a single seismic-induced failure leads to core damage. This approach is bounding and implicitly relies on the assumption that seismic-induced failures of equipment show a high degree of correlation (i.e., if one SSC fails, all similar SSCs will also fail). This assumption is conservative, but direct use of this assumption in evaluating the risk increase from out of service equipment could lead to an underestimation of the change in risk. However, if one were to assume no correlation at all in the seismic failures, then the seismic risk would be lower than the risk predicted by a fully correlated model, but the change in risk using the un-correlated model with a redundant piece of important equipment out of service would be equivalent to the level predicted by the correlated model.

If the industry accepted approach (ML100270639) of correlation is assumed, the conditional core damage frequency given a seismic event will remain unaltered whether equipment is out of service or not. Thus, the risk increase due to out of service equipment cannot be greater than the total SCDF estimated by the bounding method used in the NRC Staff report (ML100270639). That is, for the

HNP site, the delta SCDF from equipment out of service cannot be greater than 2E-5/yr.

Qualitatively Evaluate Bounding LERF Contribution

The current HNP internal events PRA includes a comprehensive treatment of LERF due to internally initiated events. The following table provides a summary of the results of the internal events analysis. These results show that the HNP containment is robust with respect to LERF contributors, except with the scenario initiated by a bypass event - i.e., Interfacing Systems Loss of Coolant Accident (ISLOCA) and break outside containment, and vessel rupture. All other scenarios, including Station Blackout (SBO) show a conditional probability of LERF of about 5%. For the purpose of this evaluation, assume a conditional probability of LERF of 0.1.

Seismic events would not be expected to induce containment bypass scenarios or break outside containment. Therefore, a bounding conditional large early release probability for seismic events (CLERPSeismic) is assumed to be 0.1. The incremental bounding large early release frequency from seismic events (ILERFSeismic) is then computed as:

ILERFSeismic = ICDFSeismic * CLERPSeismic = 2E-5 * 0.1 = 2E-6

Conditional Large Early Release Probability from Internal Event PRA

Initiating Event	Description	CDF (/yr)	LERF (/yr)	CLERP
Break outside containment	Break outside containment (FW lines, HPCI steam line, Main steam line, RCIC steam line, and RWCU line)	7.03E-07	7.03E-07	100.00%
%ISLOCA	ISLOCA Sequence	8.65E-09	8.65E-09	100.00%
%RPVRUPTURE	Rupture of reactor vessel	2.30E-09	2.28E-09	99.03%
%ATWSTT	ATWS following turbine trip event	1.05E-07	3.84E-08	36.72%
%ATWSSCRAM	Reactor scram with ATWS	1.41E-07	5.06E-08	35.89%
%ATWSMS	ATWS following MSIV closure / Loss of condenser vacuum event	1.59E-07	5.58E-08	35.03%
%ATWSFW	ATWS following loss of feedwater	1.11E-07	3.88E-08	34.97%
%LLOCA	Large break LOCA inside the drywell	2.89E-11	9.98E-12	34.54%
%ATWSLOSP	LOSP with ATWS	1.71E-08	5.86E-09	34.29%
%FL-LODC_SB	Flag for Loss of Station Service Battery '1B' DC Power Initiating Event	2.75E-08	3.83E-09	13.90%
%LOFW-CO	Loss of feedwater (due to loss of condensate)	3.26E-07	3.98E-08	12.21%
%FL-LOSUTD	Flag for initiating event caused by loss of startup transformer D	3.13E-07	2.88E-08	9.20%
%FL-LOMCHV	Flag for loss of MCR cooling initiating event	4.17E-07	2.76E-08	6.63%
%LOFW-FW	Loss of feedwater (not due to loss of condensate)	6.72E-08	4.17E-09	6.20%
%FL-&CCF-1R25S036037	Flag for Initiating Event Cause by Common Cause Loss of Essential Cabinets A and	1.07E-07	6.29E-09	5.89%
%FL-DISCH	Flag for Plant Service Water (PSW) discharge flow path failure initiating event	2.31E-08	1.10E-09	4.76%
%FL-INTAKE	Flag for initiating event caused by los of PSW due to intake plug in	3.44E-08	1.46E-09	4.24%
%FL-LOPSWTOTAL	Flag for complete loss of PSW initiator	5.82E-07	2.44E-08	4.20%
%FL-LOINSTAIR	Special initiator for loss of instrument air	2.29E-07	8.40E-09	3.67%
%IORV	Inadvertently opened SRV	2.93E-08	1.03E-09	3.53%
%SCRAM	Reactor scram initiating event	2.76E-07	8.60E-09	3.11%
%TTRIP	Turbine trip event	1.74E-07	5.21E-09	2.99%
%MLOCA	Medium break LOCA inside the drywell	1.53E-08	2.78E-10	1.82%
%LOCV	Loss of condenser vacuum	6.89E-07	9.19E-09	1.33%
%MSIVC	MSIV closure initiating event	2.00E-07	2.51E-09	1.26%
%SLOCA	Small break LOCA inside the drywell	1.59E-10	1.95E-12	1.23%
%LOSP	LOSP initiating event	1.19E-06	8.47E-09	0.71%

Conditional Large Early Release Probability from Internal Event PRA

Initiating Event	Description	CDF (/yr)	LERF (/yr)	CLERP
%ALOCA	LOCA – Spurious electrical SRV actuation and blowdown	1.54E-08	9.51E-11	0.62%
All other initiators	Flood, Loss of Instrument Bus, RPS Buses, etc.	1.92E-06	5.05E-08	2.63%
Total	**Total CDF/LERF**	**7.88E-6**	**1.14E-6**	**14.4%**

The above analysis provides the technical basis for addressing the seismic-induced core damage risk for HNP by reducing the ICDP/ILERP criteria to account for a bounding estimate of the configuration risks due to seismic events.

ICCDP = (12/(87600.93))*2E-05 = 2.95E-08

ICLERP = (12/(876009.3))*2E-06 = 2.95E-09

As demonstrated above, ICCDP and ICLERP for seismic hazard are slightly less than the Internal Events ICCDP (average for Units 1 and 2 is 4E-08) and ICLERP (average for Units 1 and 2 is 4E-09). Therefore, the assumption to use the Internal Events ICCDP and ICLERP to represent the seismic hazard is justified.

Bounding assessment of Other External Events:

The other external events are events other than seismic and fire. Risk from the other external events was evaluated in the IPEEE.

The overall methodology recommended by NUREG-1407 for analyzing plant risk due to high winds, floods, and transportation and other events is a progressive screening approach. The steps for this approach, which are described in section 5 of NUREG-1407, are summarized as follows:

1. Review plant-specific hazard data and licensing bases.
2. Identify significant plant changes that could affect the design conditions since the operating license issuance.
3. Determine whether the plant design meets current criteria as stated in the 1975 Standard Review Plan (SRP), NUREG-751087.
4. Determine whether the hazard frequency is acceptably low (if plant design does not conform to the 1975 SRP) (optional step).
5. Perform a bounding analysis if the hazard frequency is unacceptable (optional step).
6. Perform a probabilistic risk assessment if the bounding analysis is unacceptable (optional).

The first three steps are the starting point in the progressive screening process. If these steps conclude that the 1975 SRP criteria have been met, the optional steps are bypassed. Otherwise, one or more of the optional steps, which are a series of analyses that increase sequentially in level of detail may need to be performed. If the plant conforms to the 1975 SRP for each external event, the contribution to core damage from that external event is judged to be less than 1E-6 per year and the screening criteria is met. Likewise, the screening criterion for the optional steps is a core-damage contribution of 1E-6 per year (assuming the hazard frequency is less than 1E-5 per year and the conditional probability of core damage is 0.1, given the occurrence of the external event).

Summary of Evaluation:

The IPEEE evaluated risk from the following events: High winds, Floods, Transportation, Nearby Facility Accident, and Other. The full IPEEE report was submitted to NRC on January 26, 1996. The NRC safety evaluation report (dated October 23, 2000) concluded that HNP's IPEEE process was capable of identifying the most likely severe accidents and severe accident vulnerabilities, and therefore, the Hatch IPEEE has met the intent of Supplement 4 to GL 88-20.

The following information summarizes the conclusion from the IPEEE report (which was submitted to NRC on January 26, 1996) for each hazard.

<u>High Winds</u>:

The high wind evaluation concluded that the HNP structures are well designed to withstand the hazards associated with high winds and no potential vulnerability is identified. Therefore, it was concluded that the contribution to core damage

frequency from high winds is less than 1E-6 per year and the contribution to plant risk is insignificant.

As part of the evaluation of a performance deficiency with respect to two HNP LERs, Unit 1 LER 2008-004 and Unit 2 LER 2009-004, the Hatch tornado frequency was calculated using the methods in NUREG-4461, revision 2. The frequency was calculated at 3.35 E-06 per year. This is an order of magnitude less than the 4.09E-04 initiating event estimate used in the IPEEE. Since the IPEEE concluded that the contribution to core damage frequency from the higher estimate was very low, the reduction in frequency makes the contribution even lower.

Floods:

The Probable Maximum Flood (PMF) for the HNP site is estimated by HMR-51 as 24.8 inches of rainfall over a drainage area of 11,700 square miles. The expected Altamaha River flood level is 110 ft, which is the floor level of the intake structure (the diesel generator building and all other safety-related buildings are located well above this level at 129 ft). Should there also be coincidental high-wind-generated waves, as much as 3.25 ft would be added to the 110 ft of flood level. However, only two doors of the intake structure are located below this level, and they are placed at labyrinth offsets and are weather-stripped. Wave run-up, created by the coincidental high winds would cause leakage into the building, but the leakage would be mitigated by floor drains and valve pit submersible pumps.

The recurrence frequency of such a flood is estimated to be once every 1E+08 years, which makes the scenario unlikely. Moreover, this frequency is conservatively based upon the assumption that the Probable Maximum Precipitation (PMP) occurs at the same frequency as the Elba, Alabama storm.

Dam failures contribute only nominally to the flood level. Because of the long distance between the site and the nearest dams, any surges created by an instantaneous failure would be attenuated to a fraction of the original size. For the nearest, largest dam, the Sinclair Dam, the resultant wave from the dam's instantaneous failure would only add 100,000 ft^3/sec of flow. (The failure of the Sinclair Dam is not likely under the conditions of the PMF because of the standards with which it was designed; thus the scenario in which the dam fails coincident with the PMF would occur at a frequency that is much lower than the recurrence frequency of the PMF, making this scenario unlikely.)

Plant procedures provide measures for protecting all entrances to the intake structure, diesel building, reactor building, and turbine building if the Altamaha River is expected to crest at greater than or equal to 105 ft msl.

Other sources of flooding such as river obstruction and local intense rainfall do not contribute to any significant flooding events. River flooding caused by ice flows is very unlikely because river temperature historically has never dropped low enough. Local intense rainfall even at world-record level, would be sufficiently mitigated by the site's drainage system. Therefore, even if plant flooding from external causes were to lead directly to core damage, its frequency

is estimated to be less than the screening criteria, 1E-08 per year, which is insignificant.

Transportation and Nearby Facility Accident:

The following potential accidents involving transportation hazards were evaluated: Explosions, Flammable Vapor Clouds (Delayed Ignition), Toxic Chemicals, Fires, Collison With Intake Structure, and Liquid Spills.

The existing HNP design conforms to Standard Review Plan (SRP) criteria for transportation and nearby facility accidents. No significant changes were identified which impact the plant design. There are no potential vulnerabilities attributed to transportation or nearby facility accidents.

Other:

Pursuant to the guidelines of NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," June 1991, HNP was not required to address the following hazards:

- Lightning
- Severe temperature transients
- Severe weather storms
- External fires (forest fires, grass fires.)
- Extraterrestrial activities
- Volcanic activity

No other plant-unique external events were identified that pose any significant threat of a severe accident at HNP.

As described above, all other external hazards screen out. For the purpose of this evaluation, the ICCDP and ICLERP associated with the other external events being equal to internal events ICCDP and ICLERP is bounding even though the actual values for external events are significantly lower than those associated with internal events.

b. *The licensee stated in the submittal that HNP does not have a Fire PRA model that can be used for risk-informed applications. The licensee further stated that*

> *..it is conservatively assumed that fire risk contribution is three times as much as the internal events (including internal flooding) risk.*

The licensee stated that ICCDP and ICLERP for internal fire are "conservatively" assumed to be three times the internal events. Justify that this evaluation (i.e., ICCDP and ICLERP for internal fire are three times as much as the ICCDP and ICLERP for internal events) is bounding by summarizing an evaluation of the expected dominant fire scenarios and determining the impact of extended CT on those scenarios.

SNC Response:

The following information provides justification for using a factor of three to estimate ICCDP and ICLERP for internal fire using ICCDP and ICLERP from the internal events.

HNP is developing a fire PRA model that can be used for risk-informed applications. A pre-generate cutset from the draft fire PRA model was used to gain insights related to change in risk. The insights were used to bound the fire risk.

Station Service Battery 1A

Revised Base Fire CDF	8.5317e-05
TSTF Case Fire CDF	1.1346e-04
Fire Delta CDF	2.814e-05 RAW = 1.33
ICCDP	4.15E-08

Station Service Battery 1B

Revised Base Fire CDF	8.5317e-05
TSTF Case Fire CDF	4.6982e-04
Fire Delta CDF	3.845e-04 RAW = 5.51
ICCDP	5.664E-07

For the internal events model, taking the 1A battery out of service has a Risk Achievement Worth (RAW) of 3.70 and taking the 1B battery out of service has a RAW of 5.29. This indicates that the change in risk in the fire PRA model is lower compared to the internal events risk for the 1A battery and is about the same as the change in risk for the internal events risk.

Based on the above results and given the uncertainty associated with the draft fire model, it is conservatively assumed in this assessment that the fire risk contribution is three times as much as the internal events (including internal flooding) risk.

c. *The methodology used by the licensee for estimating the change in core damage frequency (ΔCDF) and the change in large early release frequency (ΔLERF) from seismic events appears to only consider the change in risk due to seismically-induced loss of offsite power. Provide justification that this evaluation is a bounding evaluation that adequately estimates the seismic risk contribution for the proposed TS change even though it only consider seismically-induced loss of offsite power. Otherwise provide a quantitative assessment of seismic risk contribution for the proposed change.*

SNC Response:

The following information demonstrates that the use of delta CDF and delta LERF from LOSP initiator bounds the seismic risk from the other initiators like Large LOCA (LLOCA), Medium LOCA (MLOCA), Small LOCA (SLOCA), and Anticipated Transient Without Scram (ATWS).

Currently, HNP does not have a Seismic PRA model. To estimate the Seismic risk contribution, the following methodology was used:

The seismic risk evaluation approach is adopted from Risk Assessment of Operating Events (RASP) Handbook, Volume 2, Version 1.03. The approach used in the RASP handbook divides ground acceleration into three bins. The following information provides the three bins and expected consequences.

Bin 1: Earthquakes in Seismic Bin 1 are of interest because these earthquakes result in LOSP and demands on EDGs and sequencers. The internal events model can be used to calculate delta CDF and delta LERF - strip out initiator contribution from the IE CDF by dividing CDF with the initiator frequency and then multiplying the resulting number with seismic probability and Seismic bin frequency.

Bin 2: Earthquakes in Seismic Bin 2 are of interest because these earthquakes could result in small loss-of-coolant accident (SLOCA), large loss-of-coolant accident (LLOCA), LOOP, and/or structural failures. Consequently, EDGs and sequencers will be demanded. The internal events model can be used to calculate delta CDF and delta LERF - strip out initiator contribution from the IE CDF by dividing CDF with the initiator frequency and then multiplying the resulting number with seismic probability and Seismic bin frequency.

Bin 3: Large earthquakes damage major components, leading to core damage. In this scenario, it does not matter if batteries work or not. The limiting SSCs in large earthquakes are failure of Reactor Vessel, RCS piping, or three buildings (Reactor, Control, and Turbine).

The internal events (including internal flooding) PRA model was used to determine the seismic risk contribution. Because seismic LOSPs are not considered to be recoverable in the short or medium term, the internal events model was quantified by setting these events to True.

The delta CDF was obtained by using the following formula for each bin.

> Delta CDF = seismic initiating event bin frequency * conditional probability of seismic event causing initiator of interest *delta CCDP from the internal events PRA model

The total CDF due to seismic was obtained by adding Individual delta CDF obtained from each bin. A similar process was used to calculate delta LERF.

The above process was used for LOSP, LLOCA, MLOCA, SLOCA, and ATWS initiators. The total delta CDF is a summation of delta CDF from LOSP, LLOCA, MLOCA, SLOCA, and ATWS. The following two tables summarize results for U1 and U2.

U1 Seismic Delta CDF and Delta LERF									
	Seismic Bin	**Bin Acceleration**	**Bin Frequency (per year)**	**Delta CCDP from IE PRA Model**	**Seismic Probability**	**Seismic Delta CDF (per year)**	**Delta CLERP from IE PRA Model**	**Seismic Probability**	**Seismic Delta LERF (per year)**
LOSP	1 (0.05-0.3g)	0.122	5.87E-04	4.20E-08	4.81E-02	1.19E-12	4.04E-09	4.81E-02	1.14E-13
	2 (0.3-0.5g)	0.387	1.72E-05		6.81E-01	0.00E+00		6.81E-01	0.00E+00
	3 (>0.5g)	0.707	8.64E-06			0.00E+00			0.00E+00
ATWS	1 (0.05-0.3g)	0.122	5.87E-04	1.04E-08	7.73E-10	4.73E-21	9.21E-11	7.73E-10	4.18E-23
	2 (0.3-0.5g)	0.387	1.72E-05		9.53E-05	0.00E+00		9.53E-05	0.00E+00
	3 (>0.5g)	0.707	8.64E-06		5.77E-03	0.00E+00		5.77E-03	0.00E+00
LLCOA	1 (0.05-0.3g)	0.122	5.87E-04	0.00E+00	1.23E-08	0.00E+00	0.00E+00	1.23E-08	0.00E+00
	2 (0.3-0.5g)	0.387	1.72E-05		5.91E-04	0.00E+00		5.91E-04	0.00E+00
	3 (>0.5g)	0.707	8.64E-06		1.55E-02	0.00E+00		1.55E-02	0.00E+00
MLOCA	1 (0.05-0.3g)	0.122	5.87E-04	1.04E-08	1.00E-07	6.12E-19	1.01E-07	1.00E-07	5.93E-18
	2 (0.3-0.5g)	0.387	1.72E-05		4.00E-03	0.00E+00		4.00E-03	0.00E+00
	3 (>0.5g)	0.707	8.64E-06		4.00E-02	0.00E+00		4.00E-02	0.00E+00
SLOCA	1 (0.05-0.3g)	0.122	5.87E-04	1.04E-08	1.50E-05	9.18E-17	0.00E+00	1.50E-05	0.00E+00
	2 (0.3-0.5g)	0.387	1.72E-05		4.50E-02	0.00E+00		4.50E-02	0.00E+00
	3 (>0.5g)	0.707	8.64E-06		2.50E-01	0.00E+00		2.50E-01	0.00E+00
					Total Delta CDF	**1.19E-12**		**Total Delta LERF**	**1.14E-13**

U2 Seismic Delta CDF and Delta LERF									
	Seismic Bin	Bin Acceleration	Bin Frequency (per year)	Delta CCDP from IE PRA Model	Seismic Probability	Seismic Delta CDF (per year)	Delta CLERP from IE PRA Model	Seismic Probability	Seismic Delta LERF (per year)
LOSP	1 (0.05-0.3g)	0.122	5.87E-04	4.20E-08	4.81E-02	1.19E-12	6.43E-10	4.81E-02	1.82E-14
	2 (0.3-0.5g)	0.387	1.72E-05		6.81E-01	0.00E+00		6.81E-01	0.00E+00
	3 (>0.5g)	0.707	8.64E-06			0.00E+00			0.00E+00
ATWS	1 (0.05-0.3g)	0.122	5.87E-04	1.04E-08	7.73E-10	4.73E-21	9.67E-11	7.73E-10	4.39E-23
	2 (0.3-0.5g)	0.387	1.72E-05		9.53E-05	0.00E+00		9.53E-05	0.00E+00
	3 (>0.5g)	0.707	8.64E-06		5.77E-03	0.00E+00		5.77E-03	0.00E+00
LLCOA	1 (0.05-0.3g)	0.122	5.87E-04	0.00E+00	1.23E-08	0.00E+00	0.00E+00	1.23E-08	0.00E+00
	2 (0.3-0.5g)	0.387	1.72E-05		5.91E-04	0.00E+00		5.91E-04	0.00E+00
	3 (>0.5g)	0.707	8.64E-06		1.55E-02	0.00E+00		1.55E-02	0.00E+00
MLOCA	1 (0.05-0.3g)	0.122	5.87E-04	7.42E-06	1.00E-07	4.36E-16	1.01E-07	1.00E-07	5.93E-18
	2 (0.3-0.5g)	0.387	1.72E-05		4.00E-03	0.00E+00		4.00E-03	0.00E+00
	3 (>0.5g)	0.707	8.64E-06		4.00E-02	0.00E+00		4.00E-02	0.00E+00
SLOCA	1 (0.05-0.3g)	0.122	5.87E-04	0.00E+00	1.50E-05	0.00E+00	0.00E+00	1.50E-05	0.00E+00
	2 (0.3-0.5g)	0.387	1.72E-05		4.50E-02	0.00E+00		4.50E-02	0.00E+00
	3 (>0.5g)	0.707	8.64E-06		2.50E-01	0.00E+00		2.50E-01	0.00E+00
					Total Seismic Delta CDF	**1.19E-12**		**Total Seismic Delta LERF**	**1.82E-14**

Consideration for the Seismic Transient Events with Offsite Power Available:
The impact on ΔCDF and ΔLERF from seismic transient events with offsite power available is implicitly included as part of the internal events risk for these transients. The EPRI SPRA Implementation Guide states: "Seismic induced sequences with offsite power available may be subsumed within the internal events transient scenarios." Based on EPRI guidance and other Seismic PRAs (SPRA), transient initiating event scenarios with offsite power available are not significant contributors to seismic risk. Damaging earthquakes usually involve a loss of offsite power.

- From earthquake experience, of the "important to safety" equipment at nuclear power plants, the most fragile components are those associated with offsite power. The large columns of ceramic insulators in the switchyard, and the incoming transformers, have been among the first components to fail due to a seismic event.
- Modern nuclear power plants are designed to withstand low acceleration level seismic events, and the balance of plant systems (such as condensate and feedwater) are generally well-anchored. Even if a plant transient is caused by a low-level seismic event, it is extremely unlikely that safety systems would be damaged, and it is very likely that the non-safety balance of plant systems could be available to provide mitigation.
- The frequency of a seismic event that could cause a transient with offsite power remaining available (such as loss of feedwater or main condenser) is about two orders of magnitude less than the corresponding internal events PRA transient frequency. Since offsite power is available and the safety systems are not damaged (because of the low acceleration), the transient event can be subsumed within the internal events transient scenarios.

Thus, based on EPRI guidance and other SPRAs, transient initiating event scenarios with offsite power available are not significant contributors to seismic risk.

The results show that the risk contribution from other initiators (LLOCA, MLOCA, SLOCA, and ATWS) is very small compared to the LOSP contribution. Therefore, use of LOSP contribution is bounding, and its use in calculating delta CDF and delta LERF is justified.

d. *Confirm that the 2008 United States Geological Survey (USGS) hazard curves were used in performing the seismic risk evaluation in response to part (c) above. Otherwise, discuss how the hazard curves used in the submittal or the bounding assessment in response to part (c) yield results that are more conservative than the results obtained from 2008 USGS hazard curves.*

SNC Response:

When evaluating seismic risk, the 2014 seismic hazard curves were used. The 2014 hazard (as documented in the SNC calculation, SCNH-13-093) for HNP was found to be acceptable (ML15097A424) by the NRC staff. For this reason, the 2014 seismic hazard curve is used to perform the seismic risk assessment. The following approach was used to select frequency for each bin. The approach is conservative as the resulting frequencies are significantly higher than the frequencies mentioned in the RASP Handbook, Volume 2, Version 1.03, that were taken from the NUREG 1488, which is dated April 1994.

The SNC calculation, SCNH-13-093 (Revision 1.0), provides 100-point hazard curves for the mean and five fractile levels for seven Peak Ground Acceleration (100, 25, 10.0, 5.0, 2.5, 1.0, and 0.5 Hz) spectral frequency in Tables 6a through 6g. For the purpose of this evaluation (i.e., TSTF-500 Battery AOT Extension Evaluation), the highest mean fractile level from each PGA spectral frequency was used to simplify the calculation.

PGA (g) \ Hz	100	25	10	5	2.5	1	0.5	Maximum Value (AEP) Per Year
~ 0.05 g	7.782E-04	9.445E-04	1.877E-03	2.201E-03	2.355E-03	**4.183E-03**	1.616E-03	4.183E-03
~ 0.30 g	7.254E-06	2.788E-05	1.050E-04	**1.201E-04**	1.186E-04	4.779E-05	7.835E-06	1.201E-04
~ 0.5 g	1.108E-06	3.838E-06	**2.205E-05**	2.170E-05	2.170E-05	8.790E-06	8.790E-06	2.205E-05
~1.0 g	1.158E-07	3.891E-07	2.736E-06	2.711E-06	**2.829E-06**	5.708E-07	5.708E-07	2.829E-06

PGA = Peak Ground Acceleration

Hz= Hertz
AEP = Annual Exceedance Probability

The resulting bin frequencies are as follows:

Bin 1 (0.05-0.3g): 4.183E-3 – 1.201E-04 = 4.06E-03

Bin 2 (0.3-0.5g): 1.201E-04 – 2.205E-05 = 9.81E-05

Bin 3 (> 0.5g): 2.205E-05 – 2.829E-06 = 1.92E-05

e. *The licensee estimated "the bounding" ΔCDF due to internal fires by multiplying ΔCDF from the internal events by three. Justify that the estimated ΔCDF is bounding or provide a bounding assessment of ΔCDF due to internal fire.*

SNC Response:

The following information provides justification for using a factor of three to estimate ΔCDF for internal fire using ΔCDF from the internal events PRA model.

HNP is developing a fire PRA model that can be used for risk-informed applications. A pre-generate cutset from the draft Fire PRA model was used to gain insights related to change in risk.

Revised Base Fire CDF	8.5317e-05
TSTF Case Fire CDF	8.5644e-05
Fire Delta CDF	3.27e-07(0.4% of revised base fire CDF)

When comparing the change in fire risk with the change in internal events risk, the following observation is made. The internal events Delta CDF is 0.9% ((7.05e-08/7.88E-06) * 100) of the modified base internal risk. This indicates that the change in risk in the draft fire model is about half of the change to internal risk.

Based on the above results and given the uncertainty associated with the draft fire PRA model, it is conservatively assumed in this assessment that fire risk contribution is three times as much as the internal events (including internal flooding) risk.

f. *For estimating ΔCDF and ΔLERF from other external events (other than seismic and fire), the licensee has only discussed the risk contribution from tornadoes and the tornado analysis is limited to risk of tornado-induced loss of offsite power. Justify that the reported results provide a conservative or bounding assessment of the impact of all other external events on this application by summarizing the hazards, the design basis scenarios that mitigate the hazards, the compliance with the design basis, and the potential impact of the extended CT on these scenarios. Alternatively, provide a qualitative bounding analyses of other external hazards and report the results, or provide a justification that those hazards do not impact this application.*

SNC Response:

See response to RAI 7c above. A qualitative bounding assessment has been provided.

8. *As described in RG 1.177, the objective of Tier 2 evaluation is to ensure that appropriate restrictions on dominant risk-significant configurations associated with the change are in place. Regulatory Position 2.3 of RG 1.177 states that "an effective way to perform such an assessment is to evaluate equipment according to its contribution to plant risk (or safety) while the equipment covered by the proposed CT change is out of service." Regulatory Position 2.4 of RG 1.177 provides TS acceptance guidelines specific to permanent CT changes, which states "[t]he licensee has demonstrated that there are appropriate restrictions on dominant risk-significant configurations associated with the change". Regulatory position 4 of RG 1.177 states that documentation to support risk-informed TS change requests should include, among other items, a tabulation of the outage configurations that could threaten the integrity of the safety functions of the subject equipment and that are, or will be, prohibited by TS or plant procedures.*

   Licensee stated in Enclosure 2 that "[a]voidance of Risk Significant Plant Configurations is accomplished by the Plant Hatch Maintenance Scheduling and Risk Assessment process." The licensee further stated that "[n]o restrictions are proposed concerning the use of the 12-hour AOT on the station service battery."

   a. *Summarize the dominant risk scenarios and associated plant characteristics that cause the risk associated with degraded direct current (DC) sources to be insensitive to all other plant configurations, such that risk significant plant configurations would never be created.*

      SNC Response to (a)

      A review of cutsets from the Internal Events (including Internal Flooding) PRA model indicates that the risk associated with degraded direct current (DC) sources is sensitive to other plant configurations and that risk significant plant configurations would be created. However, these risk significant plant configurations are managed using existing processes, including a living configuration risk management program (CRMP), the Technical Specifications Loss of Safety Function Determination program, and the Protected Train program.

      As stated in R.G. 1.177, a tier 2 evaluation should include identification of risk-significant combinations to be avoided. Table 1 below is an importance report generated from the computerized CRMP and lists the top ten risk significant components, when each of the Unit 1 and Unit 2 batteries is removed from service. The term 'risk significant' is based on RG 1.174 limits for both instantaneous and integrated risk. An instantaneous configuration specific CDF of 1.0e-03 or LERF of 1.0e-04 results in a RED risk color and is prohibited by the plant CRMP process. This equates to a CDF RAW of 199 and an LERF RAW of 97. Integrated risk of greater than 1.0e-06 (CDF) or 1e-07 (LERF) during the 12 hour

allowed out of service time would result in a YELLOW color and require risk management actions. This equates to an instantaneous RAW of 142 for CDF and 67.6 for LERF. The importance report shows the change in risk of the component by displaying the ratio of the RAW prior to the configuration change and the RAW after the configuration change.

Because the station batteries do not power the emergency diesel controls, the station DC system can be recovered during LOSP conditions by manual actions to reclose the circuit breakers supplying the station battery chargers. This is reflected by the top events shown in the Table 1 importance reports, which are the DC breaker between the station chargers and the DC switchgear, and the disconnect switches that allow placement of the alternate charger into service. All of the other significant items listed are in the opposite train and would result in a possible Loss of Safety Function. This would result in a prohibited Technical Specification condition.

For these reasons, blanket restrictions are not proposed concerning the use of the 12-hour AOT on the station service battery. The use of the Tier 3 CRMP process, which is performed real-time as the configurations develop, is more appropriate for controlling risk significant conditions.

b. Recent risk-informed applications for extending completion times have shown that the internal fires could result in dominant risk-significant configurations. As HNP does not have a fire PRA, explain how potential risk-significant configurations initiated by internal fires are identified to implement appropriate restrictions on activities, such as maintenance activities involving unavailability of fire protection equipment (detection, suppression or fire barriers), hot work, or introduction of transient combustible materials.

SNC Response to (b):

For configuration risk management of both internal events and fire initiators, SNC implements the guidance of NUMARC 93-01, endorsed by Regulatory Guide 1.160. In April 2011, Nuclear Energy Institute (NEI) issued revision 4A of NUMARC 93-01. This revision provided additional guidance on consideration of internal fires for risk assessments performed in accordance with 10 CFR 50.65, Paragraph (a)(4). The guidance includes methods licensees may use to identify equipment which is important to mitigation of core damage risk from fire initiators, approaches to developing and implementing appropriate risk management actions, and tools for effective implementation of the guidance.

The approach chosen at Hatch is to evaluate the configuration risk

associated with internal fire initiators through the use of a fire risk software tool. This is an add-in to the existing on-line CRMP program and combines configuration information such as flags and basic events with a fire specific pre-generated cutset file. This provides a fire initiator specific risk indicator separate from the risk indicator for internal event risk. If the risk indicator indicates a condition other than low risk, the Operator or Work Week Manager (depending on whether risk is being calculated in real-time or during planning) is alerted with a color change. This is similar to the CRMP process described in 8(a) above for internal events CDF and LERF.

Risk management actions are developed, in part, based on the results of an additional Importance Calculator software tool. The output of the tool for internal event combinations is attached, but it is also capable of evaluating the risk for fire initiators and identifying fire zones that may increase in significance.

Existing fire protection program procedures require mandatory compensatory actions if fire protection features are degraded, independent of maintenance configurations, and the a(4) process described in NUMARC 93-01 recognizes that these are adequate to address those features. The additional risk assessment actions to identify important components and zones address the risk associated with removal of the component from service.

TABLE 1 – Important Components

Importance Calculator Results – Unit 1
Plant Alignment

Item	Description
FL-1E11D002A-IN	1E11D002A In Service
FL-1E11D002A-IN	1E11D002A In Service
FL-1E11D002B-IN	1E11D002B In Service
FL-1E11D002B-IN	1E11D002B In Service
FL-1P51C001A-AS	SSAC 1P51C001A in Auto Standby
FL-1P51C001B-OFF	SSAC 1P51C001B in Pull-To-Lock Off
FL-1P51C008B-NR	SSAC CCW pump 1P51C008B in standby
FL-1P52D102A-IS	IA after-filter 1P52D102A in service
FL-1P52D102A-IS	IA after-filter 1P52D102A in service
FL-1P52D102C-IS	IA prefilter 1P52D102C in service
FL-1P52D102C-IS	IA prefilter 1P52D102C in service
FL-1Z41B003B-NR	MCR AHU 1Z41B003B in standby
FL-1Z41B003BPSW1	MCR AHU B cooling aligned to PSW div 1
FL-MODESWITCHU1	Unit 1
FL-RPSBUS-S037	Selected to R25S037
FL-S004-S032A	DG 1A battery selected to charger 1R42S032A
FL-S004-S032A_U2	DG 2A battery selected to charger 2R42S032A
FL-S005-S032B	DG 1B battery selected to charger 1R42S032B
FL-S006-S032C	DG 1C battery selected to charger 1R42S032C
FL-S006-S032C_U2	DG 2C battery selected to charger 2R42S032C
FL-S016-S026/27	Selected to chargers 1A and 1B (1R42S026 1R42S027)
FL-S017-S029/30	Selected to chargers 1D and 1E (1R42S029 1R42S030)
FL-SSWP-D003A-NR	Strainer D003A in service

Station Battery 1A Out of Service
Top: CDF Scope: Internal Events.
In Service Equipment Results

Item	Base RAW	RAW	Ratio	Description
1R22S0164T	1.6	467.4	292.1	125/250VDC BUS 1A SUPPLY FROM BATTERY CHARGER
1R26M031A	1.7	473.8	278.7	125VDC THROWOVER SW 1A
1R26M031B	1.7	473.8	278.7	125VDC THROWOVER SW 1B
1R22S017	477.4	18272.1	38.3	125V/250V DC SWGR 1B
1R42S026	1	37.1	37.1	STA BATTERY CHARGER 1A
1R42S027	1	37.1	37.1	STA BATTERY CHARGER 1B
1B21N690D	8.6	309.6	36.	RPV PRESSURE LOW PIS
1B21N690B	8.6	309.6	36.	RPV PRESSURE LOW PIS
1B21N090B	8.6	307.9	35.8	RPV LOW PRESSURE PT
1B21N090D	8.6	307.9	35.8	RPV LOW PRESSURE PT
1E21AK309D	8.7	311.1	35.8	

Top: LERF Scope: Internal
In Service Equipment Results

Item	Base RAW	RAW	Ratio	Description
1R22S0164T	1.2	434.7	362.3	125/250VDC BUS 1A SUPPLY FROM BATTERY CHARGER
1R26M031A	1.5	450.6	300.4	125VDC THROWOVER SW 1A
1R26M031B	1.5	450.6	300.4	125VDC THROWOVER SW 1B
1R24S027	1.5	156.4	104.3	MCC 1C D/G BLDG 600V/208V
1R42F216	1.6	163.2	102.	
1R25S00632	1.5	152.5	101.7	
1R23S0043B	1.5	152.5	101.7	Diesel Bldg MCC 1C, 1R24-S027
1R42S002C	1.7	164	96.5	125VDC DG SYS BATTERY 1C
1R23S003	6.7	597.1	89.1	600V STA SERV SWGR/XFMR1C
1R26M032C	3.2	168.6	52.7	125VDC THROWOVER SW 1G
1R22S017	456.5	17842.2	39.1	125V/250V DC SWGR 1B

Station Battery 1B Out of Service
Top: CDF Scope: Internal
In Service Equipment Results

Item	Base RAW	RAW	Ratio	Description
1R22S0174T	5.4	473.4	87.7	125/250 VDC BUS 1B SUPPLY FROM BATTERY CHARGER
1R26M031C	5.7	488.1	85.6	125VDC THROWOVER SW 1C
1R26M031D	5.7	488.1	85.6	125VDC THROWOVER SW 1D
1R25S001	1.3	51	39.2	125VDC CABINET 1A
1R22S016	474.1	18347.4	38.7	125V/250V DC SWGR 1A
1P70F020	875	32078.8	36.7	PNEUM HDR ISOL CHECK VLV
1P70F026	875	32078.8	36.7	NITROGEN SUPPLY CHK VLV
1T48F010	901.6	32466.6	36.	DW PNEUM N2 SUPPLY GLOBE
1R22S0163M	1.7	60.7	35.7	FEEDS 1R25-S001
1R42S029	1.3	42	32.3	STA BATTERY CHARGER 1D
1R42S030	1.3	42	32.3	STA BATTERY CHARGER 1E

Top: LERF Scope: Internal
In Service Equipment Results

Item	Base RAW	RAW	Ratio	Description
1R22S0174T	3.6	437.9	121.6	125/250 VDC BUS 1B SUPPLY FROM BATTERY CHARGER
1R26M031C	4.2	456.7	108.7	125VDC THROWOVER SW 1C
1R26M031D	4.2	456.7	108.7	125VDC THROWOVER SW 1D
1P70F020	642.7	27261.2	42.4	PNEUM HDR ISOL CHECK VLV
1P70F026	642.7	27261.2	42.4	NITROGEN SUPPLY CHK VLV
1T48F010	711.2	28796.4	40.5	DW PNEUM N2 SUPPLY GLOBE
1S11S005	2	80.4	40.2	S/U AUX XFMR 1D 24KV/4160
1R22S016	452	17870.3	39.5	125V/250V DC SWGR 1A
1R42S029	1.2	37.5	31.3	STA BATTERY CHARGER 1D
1R42S030	1.2	37.5	31.3	STA BATTERY CHARGER 1E
1R23S0048T	1.1	31.8	28.9	FEEDS 1R42-S029

Importance Calculator Results – Unit 2
Plant Alignment

Item	Description
DWPN_U2	N2 aligned to Unit 2 tank
FL-1Z41B003B-NR	MCR AHU 1Z41B008B in standby
FL-1Z41B003BPSW1	MCR AHU B cooling aligned to PSW div 1
FL-2P51C001A-AS	SSAC 2P51C001A in Auto Standby
FL-2P51C001B-OFF	SSAC 2P51C001B in Pull-To-Lock Off
FL-2P51C005B-NR	SSAC CCW pump 2P51C005B in standby
FL-DG1B-SSWP	DG 1B aligned to Standby Service Water Pump
FL-RPSBUS-S037	RPS Throwover Switch selected to Essential Bus 2R25S037
FL-RPSBUS-S037	RPS Throwover Switch selected to Essential Bus 2R25S037
FL-S004-S032A_U2	DG 2A battery selected to charger 2R42S032A
FL-S005-S032B	DG 1B battery selected to charger 1R42S032B
FL-S006-S032C	DG 2C battery selected to charger 2R42S032C
FL-S016-S026/27	Selected to chargers 2A and 2B (2R42S026 2R42S027)
FL-S016-S027/28	Selected to chargers 2B and 2C (2R42S027 2R42S028)
FL-S017-S029/30	Selected to chargers 2D and 2E (2R42S029 2R42S030)
RHRSWASTR2A	Strainer 2E11D002A aligned for RHRSW
RHRSWBSTR2B	Strainer 2E11D002B aligned for RHRSW
SSWPSTRN3A	Strainer 2P41D003A aligned for SSWP

Station Battery 2A out of service.
Top: CDF Scope: Internal Events
In Service Equipment Results

Item	Base RAW	RAW	Ratio	Description
2R22S0162T	1.5	459.5	306.3	BATT CHGRS
2R26M031A	1.7	466.8	274.6	THROW-OVER SWITCH 2A
2R26M031B	1.7	466.8	274.6	THROW-OVER SWITCH 2B
2R22S017	469.5	17950.9	38.2	250V DC SWGR 2B
2R42S002C	38.9	1477.4	38.	125V DIESEL SYS BATT 2C
2R42S028	1	37	37.	125 VOLT BATTERY CHARGER
2R42S027	1	36.9	36.9	125 VOLT BATTERY CHARGER
2R42S026	1	36.9	36.9	125 VOLT BATTERY CHARGER
2R23S0047B	13.4	487	36.3	2R42-S031
2E21K307D	8.6	310	36.	
2E21K309D	8.6	310	36.	

Top: LERF Scope: Internal Events
In Service Equipment Results

Item	Base RAW	RAW	Ratio	Description
2R22S0162T	1.2	478.4	398.7	BATT CHGRS
2R26M031A	1.5	498	332.	THROW-OVER SWITCH 2A
2R26M031B	1.5	498	332.	THROW-OVER SWITCH 2B
2R42S002C	3.2	547.6	171.1	125V DIESEL SYS BATT 2C
2R23S003	5.7	715.9	125.6	600V STA SERV SWGR 2C
2R22S007	1.5	174	116.	4160V STA SERV SWGR 2G
2R23S0047B	1.6	167.9	104.9	2R42-S031
2R24S027	1.7	172.2	101.3	D/G BLDG 600/208V MCC 2C
2R25S00630	1.7	167.9	98.8	
2R26M032C	3.3	187.8	56.9	125 VOLT DC THROW-OVER SW
2R22S017	503.2	19635.5	39.	250V DC SWGR 2B

Station Battery 2B out of service.
Top: CDFScope: Internal
In Service Equipment Results

Item	Base RAW	RAW	Ratio	Description
2R22S0172T	5.4	463.8	85.9	
2R26M031C	5.7	469.4	82.4	THROW-OVER SWITCH 2C
2R26M031D	5.7	469.4	82.4	THROW-OVER SWITCH 2D
1S22S019	1.2	63.6	53.	500KV/230KV TRANSFORMER
179537	1.2	63.6	53.	500KV/230KV TRANSFORMER
179706	1.2	63.6	53.	500KV/230KV TRANSFORMER
2S14	1.2	63.6	53.	500KV/230KV TRANSFORMER
2S22S011	1.2	63.6	53.	500KV/230KV TRANSFORMER
2P70F020	805.8	31676.8	39.3	PNEUM HEADER ISO CHECK VL
2P70F026	805.8	31676.8	39.3	NITROGEN SUPPLY CHECK VAL
2T48F010	833.6	32182.8	38.6	NITROGEN SUPPLY TO DRYWELL PNEUMATIC SYSTEM

Top: LERF Scope: Internal
In Service Equipment Results

Item	Base RAW	RAW	Ratio	Description
2R22S0172T	3.9	481.9	123.6	
2R26M031C	4.6	501.5	109.	THROW-OVER SWITCH 2C
2R26M031D	4.6	501.5	109.	THROW-OVER SWITCH 2D
2R23S004	8.4	486	57.9	600V STA SERV SWGR 2D
2P70F020	655.3	29859.1	45.6	PNEUM HEADER ISO CHECK VL
2P70F026	655.3	29859.1	45.6	NITROGEN SUPPLY CHECK VAL
179706	1.1	49.8	45.3	500KV/230KV TRANSFORMER
2S14	1.1	49.8	45.3	500KV/230KV TRANSFORMER
2S22S011	1.1	49.8	45.3	500KV/230KV TRANSFORMER
179537	1.1	49.8	45.3	500KV/230KV TRANSFORMER
1S22S019	1.1	49.8	45.3	500KV/230KV TRANSFORMER

9. *Regulatory Position 2.3.5 of RG 1.177 states that sensitivity analyses may be necessary to address the important assumptions in the submittal made with respect to TS change analyses. Regulatory Position 4 of RG 1.177 states that documentation to support risk-informed TS change requests should include, among other items, sensitivity and uncertainty analyses performed.*

   No sensitivity analyses were reported in the submittal. Briefly describe the sensitivity and uncertainty analyses performed to support the proposed TS change analyses or justify that no sensitivity analysis is warranted for this application.

   SNC Response:

   Delta CDF and Delta LERF Evaluation:

   A sensitivity analysis was performed by increasing UA hours of station service batteries to 24 hours, and applying common cause failure probability of the station service battery.

Sensitivity Run Common cause and UA of 24 hours for each battery				
Risk	**Unit 1**		**Unit 2**	
	Delta CDF	**Delta LERF**	**Delta CDF**	**Delta LERF**
Internal Events PRA	1.48E-07	1.47E-08	1.41E-07	1.46E-08
Fire	4.44E-07	4.41E-08	4.23E-07	4.38E-08
Seismic	8.21E-12	7.90E-13	8.21E-12	1.26E-13
Other External Events	Qualitative Assessment			
Shutdown	Bounded by Internal Events PRA			
Total	**5.92E-07**	**5.88E-08**	**5.64E-07**	**5.84E-08**

   The results show that the delta CDF and delta LERF are below the threshold values.

   ICCDP and ICLERP Evaluation:

   A sensitivity analysis was performed by increasing UA hours of station service batteries to 24 hours, and the common cause failure probability of station service battery basic event was increased to have the same value as the singleton basic event. The sensitivity scenario was quantified using a flag event with a value of 1.0. The flag was placed as an input to several gates such that it failed station service battery A, failed maintenance unavailability, loss of charger input, and loss of DC bus due to premature fuse failure. The placement of the flag in the logic model also increased the DC power initiating event frequency.

The ICCDP and ICLERP are summarized below.

		IE	Fire	Seismic	Total
ICCDP (U1)	SSB – A	6.52E-08	1.96E-07	5.89E-08	3.20E-07
	SSB – B	9.96E-08	2.99E-07	5.89E-08	4.57E-07
ICLERP (U1)	SSB – A	4.42E-09	1.33E-08	5.89E-09	2.36E-08
	SSB – B	1.27E-08	3.81E-08	5.89E-09	5.67E-08
ICCDP (U2)	SSB – A	5.79E-08	1.74E-07	5.89E-08	2.91E-07
	SSB – B	9.80E-08	2.94E-07	5.89E-08	4.51E-07
ICLERP (U2)	SSB – A	4.09E-09	1.23E-08	5.89E-09	2.23E-08
	SSB – B	1.28E-08	3.84E-08	5.89E-09	5.71E-08

The results show that the ICCDP and ICLERP are below the threshold values.

10. *RG 1.177 states that the common cause failure (CCF) contributions should be modeled so that they can be modified to reflect the condition in which one or more of the components is unavailable. RG 1.177 further states that for appropriate configuration risk management and control, preventive and corrective maintenance activities need to be considered, and licensees should, therefore, have the ability to address the subtle difference that exists between maintenance activities.*

    The licensee has not provided any discussion on treatment of CCF in the submittal. Briefly describe treatment of CCF performed to support the TS change analysis and demonstrate that the treatment of CCF is consistent with the guidance provided in RG 1.177.

    SNC Response:

    Delta CDF and Delta LERF Evaluation:

    The original evaluation (i.e., non-sensitivity case) did not consider common cause failure of station service batteries because of differences in these two batteries. However, a sensitivity run was performed after adding common cause failure basic event in the PRA model. The results are summarized in response to RAI 9 above.

    The results show that the delta CDF and delta LERF are below the threshold values.

    ICCDP and ICLERP Evaluation:

    The original evaluation (i.e., non-sensitivity case) did not consider common cause failure of station service batteries because of differences in these two batteries. However, a sensitivity analysis was performed by increasing UA hours of station service batteries to 24 hours, and the common cause failure probability of station service battery basic event was increased to have the same value as the singleton basic event. The sensitivity scenario was quantified using a flag event with a value of 1.0. The flag was placed as an input to several gates such that it failed station service battery A, failed maintenance unavailability, loss of charger input, and loss of DC bus due to premature fuse failure. The placement of the flag in the logic model also increased the DC power initiating event frequency. The results are summarized in response to RAI 9 above.

    The results show that the ICCDP and ICLERP are below the threshold values.