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SOUTHERN COMPANY

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

Edwin I. Hatch Nuclear Plant
Request for Additional Information on TSTF-500, "DC Electrical Re-write"

Ladies and Gentlemen:

On August 11, 2015, Southern Nuclear Operating Company (SNC), submitted a request to revise the Plant Hatch Technical Specifications (TS) to implement TSTF-500, "DC Electrical Rewrite – Update to TSTF-360".

On September 30, 2015, a teleconference was held between Nuclear Regulatory Commission (NRC) and SNC staff members. In that conference, it was determined that the NRC needed additional information to proceed with its review. The information request dealt with the Probabilistic Risk Analysis done in support of increasing the Completion Time on the station service batteries from 2 to 12 hours. The analysis was originally provided as Enclosure 2 of the August 11, 2015 TS revision request letter.

Specifically, the NRC requested information on the Incremental Conditional Core Damage Probability (ICCDP) and the Incremental Conditional Large Early Release Probability (ICLERP); these values were not provided as part of the original evaluation in the August 11 letter.

Accordingly, the enclosure to this letter contains the Tier 1 evaluation of the Probabilistic Risk Analysis for the increased battery Completion Time, which completely supersedes the Tier 1 analysis provided in Enclosure 2 to the August 11, 2015 letter. The Tier 2 and 3 analyses, also previously supplied in Enclosure 2 of the August 11, 2015 letter remain valid.

Additionally, note that, as a result of the new ICCDP and ICLERP calculations, the Core Damage Frequency (CDF) and Large Early Release Fraction (LERF) numbers changed slightly; these changes were due to different assumptions regarding the unavailability numbers used on select components, as well as rounding differences. The numbers are provided on the Table on page E2-2.

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

Mr. C.R. Pierce states he is the Nuclear Licensing Director 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,

C. R. Pierce

C. R. Pierce
Regulatory Affairs Director

Sworn to me and subscribed this 27 day of October

Laura Crumpton
Notary Public

My commission expires: 10-8-2017

CRP/OCV/

Enclosure: Revised Tier 1 Evaluation; Description of Probabilistic Risk Analysis

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Edwin I. Hatch Nuclear Plant Unit 1 and 2

Enclosure

Revised Tier 1 Evaluation; Description of Probabilistic Risk Analysis

Evaluation of Risk Impact

The overall risk evaluation for this proposed completion time extension for each inoperable Plant Hatch Station Service Battery is based on the following three-tier approach described in Regulatory Guide 1.177:

- **Tier 1: PRA Capability and Insights**
- **Tier 2: Avoidance of Risk-Significant Plant Configurations, and**
- **Tier 3: Risk-Informed Configuration Risk Management**

1. Tier 1: PRA Capability and Insights

The quantified risk impact associated with this proposed Station Service Battery completion time extension was evaluated using the Plant Hatch Unit 1 Revision 4 PRA Average Risk Model. Results from this model are directly applied to Unit 2 because of the high degree of similarity between Hatch Plant units.

Evaluations for risk associated with this proposed amendment were performed using the following:

- Internal Events (including Internal Flooding) using the Plant Hatch Average Risk Model
- Internal Fires
- Seismic Events
- Low Power/Shutdown Risk discussed qualitatively
- Other External Events

The guidance documented in Regulatory Guide 1.177 was used to evaluate the risk impact of the requested 12 hour completion time for each Plant Hatch Station Service Battery.

Definitions

ΔCDF = CDF(TSTF-500 Case) – CDF(Base)

ΔCDF

This value shows the difference or change in average quantified Core Damage Frequency based on new values for Station Service Battery maintenance unavailability as opposed to the presently used values. This value is designed to show the average risk difference in increasing the completion time for each Station Service Battery from its present value of approximately 1 hour to 12 hours.

CDF(TSTF-500 Case)

This value is the quantified average Core Damage Frequency considering the new Station Service Battery maintenance unavailabilities. The new Station Service Battery maintenance term considers the potential for battery work being done while the units are "at Power."

CDF(Base)

This value is the average Core Damage Frequency quantified using the present Station Service Battery maintenance unavailabilites.

ΔLERF = LERF(TSTF-500 Case) – LERF(Base)

ΔLERF

This value shows the difference or change in average quantified Large Early Release Frequency based on new values for Station Service Battery maintenance unavailability as opposed to the presently used values. This value is designed to show the average risk difference in increasing the completion time for each Station Service Battery from its present value of approximately 1 hour to 12 hours.

LERF(TSTF-500 Case)

This value is the quantified average Large Early Release Frequency considering the new Station Service Battery maintenance unavailabilities. The new Station Service Battery maintenance term considers the potential for battery work being done while the units are "at Power."

LERF(Base)

This value is the average Large Early Release Frequency quantified using the present Station Service Battery maintenance unavailabilites.

ICCDP and ICLERP are calculated using the following equations:

ICCDP = [(conditional CDF with the subject equipment out of service and nominal expected equipment unavailabilities for other equipment permitted to be out of service by the TS) – (baseline CDF with nominal expected equipment unavailabilities)] x (total duration of single CT under consideration)

ICLERP = [(conditional LERF with the subject equipment out of service and nominal expected equipment unavailabilities for other equipment permitted to be out of service by the TS) – (baseline LERF with nominal expected equipment unavailabilities)] x (total duration of single CT under consideration)

<u>Delta CDF and Delta LERF Calculation</u>

Internal Events and Internal Flooding Considerations

To determine the risk associated with the extended CT for the Station Service Battery, the Hatch PRA "At Power" model was used. In order to perform a risk evaluation, the unavailability hours for each station service battery were increased to 12 hours. The following results were obtained.

	U1			U2		
	Base Case	TSTF-500 Case	**Delta**	Base Case	TSTF-500 Case	**Delta**
CDF	7.88E-06	7.95E-06	7.05E-08	7.34E-06	7.40E-06	6.71E-08
LERF	1.14E-06	1.14E-06	6.97E-09	1.03E-06	1.04E-06	6.92E-09

Fire Risk Contribution

Plant Hatch does not have a Fire PRA model that can be used for risk-informed applications. A draft Unit 1 model is available but the large number of open peer review comments prevents direct use of model results.

To determine order of magnitude bounding conditions, the Unit 1 pre-generated cutsets available from the Fire PRA peer review were used to evaluate relative risk increases due to changes in the battery failure rates. The results showed that the change in fire risk is about half of the internal events risk. It is believed that resolution of the open items will result in a fire risk approximately two to three times the internal events risk.

Based on the above insights, and because the delta CDF from internal events PRA is very small, it is conservatively assumed that fire risk contribution is three times as much as the internal events (including internal flooding) risk. Therefore, the bounding delta CDF due to internal fires would be 2.12E-07 for Unit 1 and 2.01E-07 for Unit 2.

The delta CDF for fire is 2.12E-7 for Unit 1 and 2.01E-07 for Unit 2.

The delta LERF for fire is 2.10E-8 for Unit 1 and Unit 2.

Seismic Risk Contribution

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

NUREG-1488 gives the seismic initiating event frequencies for each nuclear plant in the US. NUREG/CR-4840 provides fragility curves for various components and a methodology to combine the seismic hazard probabilities with the seismic fragility probabilities to get conditional probabilities of component failures given different seismic accelerations.

Frequencies of Seismically-Induced LOOP Events for SPAR Models, a paper which estimates values for seismic LOOPs (accession number ML062540239 in ADAMS), indicates a value of 4.2E-5/yr is appropriate for Hatch. This value is mentioned in Table 1 Frequencies of Seismically-Induced LOOP Events of RASP Handbook (Volume 2, Version 1.01, ML080300179). The value 4.2E-05 is derived by multiplying 6.13E-04 (seismic initiating event frequency) with 6.83E-02 (Conditional probability of LOSP due to seismic event). In a simplistic approach, a CCDP (due to LOSP) is multiplied to obtain delta CDF or delta LERF. However, a detailed conservative evaluation was performed in which ground acceleration is divided into three bins with each bin having different seismic initiating event frequency and conditional probability of LOSP due to seismic event.

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

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

Delta CDF = seismic initiating event frequency * conditional probability of LOSP due to seismic event * delta CCDP from the internal events *

Estimation of Seismic Risk Contribution for Unit 1 delta CDF is 1.68E-12

Estimation of Seismic Risk Contribution for Unit 1 delta LERF is 1.61E-13

Estimation of Seismic Risk Contribution for Unit 2 delta CDF is 1.68E-12

Estimation of Seismic Risk Contribution for Unit 2 delta LERF is 2.57E-14

Other External Events Risk Contribution

The risk contribution from a tornado is treated as a tornado induced LOSP. The switchyard structures and incoming transmission lines are constructed to NESC wind and ice loading requirements as described in FSAR chapter 8. The wind design for Hatch is 130 MPH, which is equivalent to an F3 or greater tornado. NUREG/CR-4461 provides an updated method of calculating the probability of a tornado with winds exceeding this amount striking the switchyard (approximately 1 sq km in size). Using this method results in a tornado induced LOSP probability of 3.35E-06. The delta CDF was obtained by using the following formula.

Delta CDF = (Delta LOSP/Normal LOSP) * F3 or greater Tornado Frequency

Estimation of Risk Contribution for Unit 1 delta CDF is 1.41E-13

Estimation of Risk Contribution for Unit 1 delta LERF is 1.35E-14

Estimation of Risk Contribution for Unit 2 delta CDF is 1.41E-13

Estimation of Risk Contribution for Unit 2 delta LERF is 2.15E-15

Low Power/Shutdown Risk

The AOT increase request for the Station Service Batteries is not applicable to operation Mode 4 (cold shutdown) and Mode 5 (refuel). Therefore, these operational conditions will not be evaluated.

The Internal Events review, although it considers Mode 1 or the "at Power" case, bounds Mode 2 (Startup/Hot Standby) and Mode 3 (Hot Shutdown). In these cases, the reactor can be cold (just above 212° F) or in excess of 500 psig; each case, however, considers the shutdown reactor. Shutdown reactor water systems such as condensate are abundant. Their redundancy, required to keep an operating reactor at 100% , makes this so. Consideration of the low pressure cases shows that there are several motor driven pumps capable of supplying the vessel with water. For the high pressure cases, there is an extra reactor feed pump, HPCI, RCIC, or the condensate booster pumps—the service of which depends on the particular reactor pressure. The transition from the high pressure to low pressure sources is by normal means and is the same that is modeled in the PRA for Mode 1. The overall difference is that there is a longer time frame allowed for the depressurization because power or decay heat is not as demanding as in the "at Power" model. Level control is an important consideration for shutdown as well as for the operating reactor. The shutdown cases tend to be less severe, however, because decay heat (or even the potential for approximately 5% reactor power in Mode 2) does not demand the full function of the systems under consideration as in the "at Power" case.

LOCAs, which tend to pose the most restrictive level control problems, are normally evaluated for a pressurized system which means that most of the time the consideration is for the "at Power" condition. The time a shutdown reactor is pressurized is short compared to the time at power. LOCA is possible during a depressurized condition, but it would tend to be caused by valve misalignment or operator error more so than actual pipe rupture. This type of event typically has more evaluation time and a longer time frame for recovery than at-power LOCAs, and the problem is corrected prior to catastrophic core damage. The overall LOCA initiating event frequencies are reasonably small (E-04 to E-05) for the range of LOCAs considered and are not a significant contribution during the shutdown or full power case.

The LERF condition is not as significant in Modes 2 and 3 because of the low reactor power. In order to have LERF, there needs to be core damage as well as a release of the damaged core to primary containment and ultimately to the environment. The availability of sources to cover the core in the low power condition has previously been discussed. The next phase of the LERF condition should water sources fail, however, is release of this damaged product to primary containment or out via a failed isolation pathway. If the material does not get into primary containment, the capability to penetrate the containment via some failure mode such as overpressure is such that the time frame involved would no longer make it an Early Release. This does not take into account the availability of sources for containment cooling or pressure control.

In consideration of failed containment isolation, it is possible that the main steam isolation valves may be closed already due to the operational variations involved in startup and hot shutdown; therefore, in these states their probability of failure to close would be less. HPCI and RCIC steam line isolations could be treated in a similar fashion as the MSIV's; however as the steam line low pressure alarms cleared, they would be opened. Their failure to close would provide a high energy pathway. If, however, all sources of core coverage failed and a HPCI or RCIC steam line failed to isolate, the actual release rate would decrease rapidly because the motive force (i.e. the steam pressure attributed to low power or decay heat) would not last. This plus the holdup time involved with the reactor building would severely retard the LERF capabilities of such scenarios.

In the shutdown or startup conditions, not only are more physical attributes available to prevent core damage, the number of initiating event contributions are less. One such example is the case with the Anticipated Transient Without Scram (ATWS). Losses of condenser vacuum and feedwater or MSIV closure are not as severe or they would be at power. These accidents have their most significant contributions when these Balance of Plant (BOP) systems are required to keep the unit operating. Failure of these systems limits the use of the condenser as a heat sink and the use of high pressure feedwater injection. During the shutdown or startup condition, failure of these systems or functions would tend to be more of an inconvenience to operation than a threat to core damage. Reactor scram is not considered for the Mode 3 case but is for Mode 2, but even this would be a very low power event. The main events to consider would be LOSP or Loss of Electrical Bus cases. These events tend to take away the redundancy associated with extra systems during the non "at Power" case.

In general, Modes 2 and 3 are not normally sustained. Mode 2 is the startup case. Transition through this mode can certainly be more than a few hours, but it is not designed as a convenient holding point to perform various activities without going to cold shutdown. It is an allowance for the physical restrictions of control rod manipulation during startup (and certain Refuel Mode cases) and maintenance on Station Service Batteries would be an administrative hindrance. Use of Mode 2 is controlled by Technical Specifications and procedures.

Mode 3 is a unique end state that accounts for any requirements to end full power operation. It is convenient to perform certain required maintenance in this condition in order to save time restoring the unit to full power operation from cold shutdown (Mode 4). It is possible to enter this condition by necessity during the time that a Station Service Battery is undergoing maintenance on an extended completion time. The transition into Mode 3 for those unique times when a Battery is already in maintenance while in Mode 1 are still low risk as discussed previously.

Summary of results from each contribution

Risk	Unit 1		Unit 2	
	Delta CDF	Delta LERF	Delta CDF	Delta LERF
Internal Events PRA	7.05E-08	6.97E-09	6.71E-08	6.92E-09
Fire	2.12E-07	2.10E-08	2.01E-07	2.10E-08
Seismic	1.68E-12	1.61E-13	1.68E-12	2.57E-14
Other External Events	1.41E-13	1.35E-14	1.41E-13	2.15E-15
Shutdown	Bounded by Internal Events PRA			
Total	**2.83E-07**	**2.80E-08**	**2.68E-07**	**2.79 E-08**

ICCDP and ICLERP Calculation

Regulatory Guide 1.177 (Revision 1.0, Section 2.4) acceptance guidelines specify that a Technical Specification change may be classified as having a small quantitative impact on plant risk if it has an ICCDP of less than 1E-06 and an ICLERP of less than 1E-07 .

As stated in the RG 1.177 (Revision 1), the TS conditions addressed by CTs are entered infrequently and are temporary by their very nature. However, TS do not typically restrict the frequency of entry into conditions addressed by CTs. In order to demonstrate that the TS CT change has only a small quantitative impact on plant risk, the RG 1.177 considers ICCDP of less than 1E-06 and an ICLERP of less than 1E-07 as small for a single TS condition entry. The following calculation demonstrates that the ICCDP and ICLERP are significantly less than the threshold values mentioned in the RG 1.177 for a single TS condition entry.

Internal Events (including Internal Flooding) PRA Model

For this calculation, the Plant Hatch Internal Events (including Internal Events) PRA model was used to quantify the following two scenarios:

- Station Service Battery A (1R42S001A) fails
- Station Service Battery B (1R42S001B) fails

Station Service Battery A (1R42S001A) fails

The Station Service Battery A (1R42S001A) was failed in the PRA logic model by placing a flag and setting it to 1.0. Note that this scenario is postulated such that it fails station service battery A, maintenance unavailability, loss of charger input, and loss of DC bus due to premature fuse failure. The placement of flag in the logic model also increases the DC power initiating event frequency. The results summarized are in the table below:

	U1			U2		
	Modified Base Case	TSTF-500 Case	**ICCDP / ICLERP**	Modified Base Case	TSTF-500 Case	**ICCDP / ICLERP**
CDF	7.88E-06	2.92E-05	2.92E-08	7.34E-06	2.66E-05	2.64E-08
LERF	1.14E-06	2.58E-06	1.99E-09	1.03E-06	2.36E-06	1.82E-09

Station Service Battery B (1R42S001B) fails

The Station Service Battery B (1R42S001B) was failed in the PRA logic model by placing a flag and setting it to 1.0. Note that this scenario is postulated such that it fails station service battery B, maintenance unavailability, loss of charger input, and loss of DC bus due to premature fuse failure. The placement of flag in the logic model also increases the DC power initiating event frequency. The results are summarized in the table below:

	U1			U2		
	Modified Base Case	TSTF-500 Case	**ICCDP / ICLERP**	Modified Base Case	TSTF-500 Case	**ICCDP / ICLERP**
CDF	7.88E-06	4.13E-05	4.58E-08	7.34E-06	4.06E-05	4.55E-08
LERF	1.14E-06	5.39E-06	5.82E-09	1.03E-06	5.39E-06	5.97E-09

Contribution from other hazards

Conservatively assume that ICCDP and ICLERP are three times more for fire hazard compared to the Internal Events (including Internal Flooding).

For seismic and other external events, conservatively assume that ICCDP and ICLERP are the same as the Internal Events (including Internal Flooding).

The following table summarizes ICCDP and ICLERP resulting from fire, seismic, and other external events.

Risk	**Unit 1**		**Unit 2**	
	ICCDP (Fire + Seismic +Other External Events)	**ILERP** (Fire + Seismic +Other External Events)	**ICCDP** (Fire + Seismic +Other External Events)	**ILERP** (Fire + Seismic +Other External Events)
Station Service Battery A	1.17E-07	7.96E-09	1.06E-07	7.28E-09
Station Service Battery B	1.83E-07	2.33E-08	1.82E-07	2.39E-08

Total ICCDP and ICLERP

The ICCDP and ICLERP generated as a result of this evaluation for Unit 1 and Unit 2 for Plant Hatch are summarized below. Note that ICCDP and ICLERP obtained from the Internal Events (including Internal Flooding) evaluation have been conservatively multiplied by a factor of three to account for risk contributions from the Fire, Seismic, and Other External Events hazards. Despite such conservatism, the ICCDP and ICLERP are less than 1E-06 and 1E-07, respectively.

Risk	**Unit 1**		**Unit 2**	
	ICCDP	**ILERP**	**ICCDP**	**ILERP**
Station Service Battery A	1.46E-07	9.95E-09	1.32E-07	9.10E-09
Station Service Battery B	2.29E-07	2.91E-08	2.28E-07	2.99E-08

Regulatory Guide 1.177 acceptance guidelines specify that a permanent TS CT change may be classified as having a small quantitative impact on plant risk if it has a delta CDF of less than 1.0E-06, a delta LERF of less than 1.0E-07, ICCDP less than 1E-06, and ICLERP less than 1E-07. The delta CDF, delta LERF, ICCDP, and ICLERP for all cases are within the acceptable criteria. Therefore, the acceptance guidelines for Regulatory Guide 1.177 are satisfied.