**Charles R. Pierce**
Regulatory Affairs Director

**Southern Nuclear Operating Company, Inc.**
40 Inverness Center Parkway
Post Office Box 1295
Birmingham, Alabama 35201

Tel 205.992.7872
Fax 205.992.7601

**SOUTHERN COMPANY**

July 2, 2013

Docket Nos.: 50-348
50-364

NL-13-1257

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

Joseph M. Farley Nuclear Plant – Units 1 and 2
Southern Nuclear Operating Company Response to Request for Additional
Information Regarding the Revision of the Condensate Storage Tank Level

Ladies and Gentlemen:

By letter dated August 20, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession Number ML12234A743), Southern Nuclear Operating Company (SNC) submitted a license amendment request for revision of Technical Specification (TS) 3.7.6 "Condensate Storage Tank." Subsequently, by letter dated November 26, 2012 (ADAMS Accession Number ML12320A543), the Nuclear Regulatory Commission (NRC) submitted a Request for Additional Information (RAI) to enable completion of the review. Following that set of RAIs, the NRC submitted another set of RAIs dated April 18, 2013 (ADAMS Accession Number ML13101A354). The SNC responses to the RAI dated November 26, 2012, is provided in Enclosure 1 with parts of the response included in Enclosures 2 and 3. The response to the RAI dated April 18, 2013, is included in Enclosure 4 with parts of the response included in Enclosures 5 and 6.

The included CD contains Enclosures 2, 3, 5, and 6, which are responses to both sets of RAIs. Enclosure 2 provides the requested analysis of determining condensate storage tank level for various cases. Enclosure 3 provides natural circulation cooldown procedures on CD. Enclosure 5 contains various requested drawings of the condensate storage tanks. Enclosure 6 contains part of the response to the RAIs dated April 18, 2013.

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

A001
NRR

Mr. C. R. Pierce states he is Regulatory Affairs 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

CRP/CLN/lac

*Sworn to and subscribed before me this* 2nd *day of* July, *2013.*

Laura L. Crump
*Notary Public*

*My commission expires:* 11-2-2013

Enclosures:
1. Response to Request for Additional Information Dated November 26, 2012
2. CD Containing DOEJ–FRSNC419117-M001 Version 2, "CST Volume Requirements per Plant Transient Events" (with Enclosures 3, 5, and 6)
3. CD Containing Joseph M. Farley Nuclear Plant Natural Circulation Cooldown Procedures (with Enclosures 2, 4, and 6)
4. Response to Request for Additional Information Dated April 18, 2013
5. CD Containing Condensate Storage Tank Drawings (with Enclosures 2, 3, and 6)
6. CD Containing Fauske and Associates, LLC Report No. FAI/13-0392, "Request for Additional Information by the Office of Nuclear Reactor Regulation, Joseph M. Farley Nuclear Plants, Units 1 and 2, Southern Nuclear Operating Company, Docket Nos. 50-348 and 50-364, Revision 0" (with Enclosures 2, 3, and 5)

cc: Southern Nuclear Operating Company
Mr. S. E. Kuczynski, Chairman, President & CEO
Mr. D. G. Bost, Executive Vice President & Chief Nuclear Officer
Mr. T. A. Lynch, Vice President – Farley
Mr. B. L. Ivey, Vice President – Regulatory Affairs
Mr. B. J. Adams, Vice President – Fleet Operations
RTYPE: CFA04.054

U. S. Nuclear Regulatory Commission
Mr. V. M. McCree, Regional Administrator
Ms. E. A. Brown, NRR Project Manager – Farley
Mr. P. K. Niebaum, Senior Resident – Farley
Mr. J. R. Sowa, Senior Resident – Farley

Alabama Department of Public Health
Dr. D. E. Williamson, State Health Officer

**Joseph M. Farley Nuclear Plant – Units 1 and 2**
**Southern Nuclear Operating Company Response Request for Additional**
**Information Regarding Revising the Condensate Storage Tank Level**

**Enclosure 1**

**Response to Request for Additional Information Dated November 26, 2012**

**RAI 1. Limiting Case Determination for Condensate Storage Tank Sizing Analyses**

Page E1-5 of Reference 2 identifies the following events that require cooling the Reactor Coolant System (RCS) by the Auxiliary Feedwater Water (AFW):

1. Technical Specification (TS) Bases -Design (Normal Cooldown)
2. Loss of Normal Feedwater (LNFW) without *(w/o)* Loss of Off-Site Power (LOSP)
3. TS Bases - Operability
4. Main Feedwater Line Break (MFLB) with *(w/)* LOSP
5. LOSP *w/* Seismic event
6. LOSP
7. Main Steam Line Break (MSLB) *w/o* LOSP
8. Depressurization Main Steam
9. LOSP *w/* Tornado Event
10. Small Break Loss-of-Coolant Accident (SBLOCA)
11. MSLB *w/* LOSP

The analyses of cases 1 through 4 for determination of the minimum amount of water in the Condensate Storage Tank (CST) are included in pages 6 through 22 of Reference 3. The U.S. Nuclear Regulatory Commission (NRC) staff cannot find the analyses for cases 5 through 11 in the submittal. Please provide the results of the analyses of cases 5 through 11 for the staff to review. If the cases are not analyzed, provide rationale for each case unanalyzed. All the concerns about CST sizing discussed in Request for Additional Information (RAI) 3 below are also applicable to cases 5 through 11 for the cases that are assumed to initiate coincidental with a LOSP, and should be addressed, accordingly.

**SNC Response to RAI 1**

The analysis of cases 5 through 11 of the submittal is included in the attached documentation of engineering judgment, DOEJ–FRSNC419117-M001 Version 2, "CST Volume Requirements per Plant Transient Events" (Enclosure 2 to this letter).

**RAI 2. Operator Action Times**

a. Page E1-6 of Reference 2 indicates that the operator action time assumed in the analysis of the normal cooldown case is 30 minutes for isolation of the recirculation lines from all three AFW pumps.

Provide information to justify the use of the operator action time of 30 minutes in the normal cooldown analysis.

b. The bottom portion of the same page indicates that the operator time for isolation of the AFW of 15 minutes and 30 minutes is assumed in the analyses of MSLB and MFLB, respectively. The justification for the operator action time is discussed in Attachment A to Reference 3, which contains the results of simulator exercises showing that the

AFW isolation time is 15 minutes for the MSLB and 14 minutes for the MFLB. Appendix A indicates that the simulator exercises for both MSLB and MFLB events are based on scenarios with Reactor Coolant Pumps (RCPs) in operation.

Also, the break sizes used in the simulator exercises are different from that assumed in the analysis of the MSLB and MFLB events.

As shown on page E1-5 the analysis for the MSLB and FLB assumes that both events initiate with concurrence of LOSP (i.e., no RCPs in operation). Justify the adequacy of use of the above operator action times in Appendix A to support that assumed in the analysis. Alternatively, provide the simulator exercise results for conditions compatible with the analysis in terms of break sizes and LOSP conditions (i.e., no RCPs in operation), and show the adequacy of the operator action time of 15 minutes and 30 minutes assumed in the analysis of MSLB and Main Feedwater Line Break (MFLB), respectively.

Also, provide a discussion of the plant administrative controls, procedures, and training programs to show that the operator action times assumed in the analysis of normal cooldown discussed in above Item 2.a, and the MSLB and MFLB discussed in above Item 2.b will remain valid for the duration of the plant life time.

**SNC Response to RAI 2a**

The normal cooldown case analyzed per Reference 3, "SNC Calculation BM-95-0961-001 Version 6.0" considers the failure of all AFW pumps' recirculation lines to the CST in determination of the minimum CST volume required. Considering the CST inventory loss due to the recirculation and instrumentation lines failure concurrent with normal cooldown case is conservative and included to add margin to the minimum volume required for normal cooldown.

The justification of the use of the operator action time of 30 minutes in the normal cooldown analysis for isolation of the recirculation lines from all three AFW pumps is based on the procedure FNP-0-AOP-21.0, "FNP Abnormal Operating Procedure - Severe Weather". This procedure requires isolation of the affected CST's recirculation lines within 30 minutes in the event of CST missile damage following a tornado or sustained high wind and unexpected CST inventory loss.

**SNC Response to RAI 2b**

The operator action time demonstrated in the simulator exercise is less than the operator action times assumed in the FSAR. As summarized in Attachment B to Reference 3 (the simulator exercise), the faulted steam generator (SG) was isolated in 6 minutes or less in each case. This time is based on the actual isolation of the valves per implementation of steps FNP-1-EEP-0, "Reactor Trip or Safety Injection". These times are not based on the final completion of the procedure per FNP-1-EEP-2, "Faulted Steam Generator Isolation", the final step of each exercise.

MSLB

The 15 minute isolation time of the faulted SG used in the design analysis (Reference 3) was chosen considering FSAR sections 6.5.3 and 15.4.2.1.2.2 and provide margin in the minimum volume required of the CST for the MSLB event.

The 15 minutes isolation time supports Chapter 15 Section 15.4.2.1.2.2. This section states, "[f]ollowing blowdown of the faulted steam generator, the plant can be brought to a stabilized hot standby condition through control of the AFW flow and SI flow as described by plant operating procedures. The operating procedures would call for operator action to limit RCS pressure and pressurizer level by terminating SI flow and to control steam generator level and RCS coolant temperature using the AFW system. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of 10 min following SI."

The 15 minutes isolation time also supports Chapter 6 Section 6.5.3, which states, "[i]n the event of a main steam or feedwater line break, the auxiliary feedwater pumps will start, and within 60 seconds will pump auxiliary feedwater through the flow restriction orifices to the three steam generators. The flow restriction orifices limit flow to the faulted steam generator and establish flow to the two intact steam generators. After 10 minutes, valves MOV 3764A, B, C, D, E, or F (drawing D-175007) will be manually activated from the control room, as required, to isolate flow from the motor-driven pumps to the faulted steam generator."

The 15 minutes isolation time utilized in the analysis to determine the minimum required volume for the CST during a MSLB event is a conservative design value which provides margin to the minimum required volume. The greater isolation time used in this design analysis as compared to that specified in the FSAR (15 min. vs. 10 min.) results in a larger inventory loss from the CST for a MSLB which increases the minimum required CST volume for this event.

MFLB

The 30 min isolation time of the faulted SG used in the design analysis (Reference 3), was chosen to meet the operator action time specified in FSAR section 15.4.2.2.2.1 in determining the minimum CST volume required for the MFLB events. The isolation time is input in defining the inventory loss from the CST prior to isolation which is accounted for in determining the minimum required volume for this event.

FSAR section 15.4.2.2.2.1 states, "[a]n analysis has also been performed to demonstrate that the operator has at least 30 min to increase AFW flow to the intact steam generators without hot leg boiling prior to transient turnaround." This requirement applies to the Case B where AFW flow is initiated one minute after the reactor trip and a minimum flow of 150 gpm is supplied to the intact steam generators.

Based on these above requirements, the 30 minute isolation time represents the worst case in determining the minimum required volume for the MFLB event and therefore, was used in the design analysis (Reference 3).

In both initial and continuing training, licensed operators are trained in the proper use of Westinghouse Owner's Group Emergency Response Guidelines (ERGs) and the FNP Emergency Response Procedures (ERPs). These symptom based procedures are designed to prioritize operator actions and are to be performed in the order written. However, so long as required actions are not displaced, allowance is made for early or additional actions to be taken which improve the plant response. Current training also focuses on operator fundamentals. The applicable fundamental, emphasized for this event, is minimizing the effects of excessive cool down caused by continued feeding of a faulted SG. It should be noted that, for this particular case, the time delta between early performance and procedure directed performance was negligible and well below the times assumed in the various analyses. Therefore, in the absence of ERG/ERP changes, there is no reason to anticipate any response degradation as a result of training.

**RAI 3. Main Feedwater Line Break with Loss of Off-Site Power**

Pages 12 through 16 of Reference 3 discuss the CST sizing based on the analysis of the main feedwater line break (MFLB) with loss of off-site power (LOSP). With the assumed LOSP, the RCPs will not be in operation throughout the duration of the MFLB event, thus, heat removal of RCS will be from natural circulation. The analysis assumes that the reactor is maintained at hot standby (at temperature of 550 °F) for 2 hours and cool down to the Residual Heat Removal (RHR) entry temperature of 350 °F. It further assumes that the Natural Circulation Cooldown (NCC) from 550 °F to 350 °F is completed in 4 hours based on a cooldown rate of 50 °F per hour.

The NRC staff has concerns about the adequacy of the use of 4-hour for the NCC completion time for the following reasons:

Generic Letter (GL) 81-21, "Natural Circulation Cooldown", discusses the NCC phenomena for an event occurred in a Pressurized-Water Reactor (PWR) and indicates that steam bubble (voiding) will occur in the reactor vessel during NCC when the operator reduces the RCS pressure to conditions where the corresponding saturation temperature drops to the temperature of the relatively stagnant fluid in the Reactor Vessel Upper Head (RVUH). The GL states that "... any significant vessel voiding produced during controlled cooldown conditions increases the susceptibility of the plant to more serious accidents. For these reasons reactor vessel voiding during controlled natural circulation cooldown should be avoided."

In response to the GL 81-21 concerns, the PWR owner groups develop the NCC procedures and incorporate them into their respective Emergency Response Guidelines. For the ERGs applicable to Westinghouse plants, the NCC procedures are included in ES-0.2, "Natural Circulation Cooldown", ES-0.3, "Natural Circulation Cooldown with Steam Void in Vessel (With Reactor Vessel

Level Instrumentation System (RVLIS))", and ES-0.4, "Natural Circulation Cooldown with Steam Void in Vessel (Without RVLIS)".

The NCC procedures for Westinghouse plants provide guidelines to the operator for steam void identification and elimination, and specify acceptable criteria for "controlled cooldown" applying to the key plant parameters including: (1) the RCS subcooling margin; (2) presurizer water level range; (3) cooldown limits; (4) RCS temperature and pressure limits; (5) RCS hot-leg temperature limits; and (6) the limit of the steam bubble size in the RVUH.

A cooldown following the NCC procedures by controlling the size of the steam bubble and maintaining plant conditions within the acceptable criteria can increase the time required to achieve the RHR entry conditions, and thus, increase the time auxiliary feedwater is dependent upon to remove decay heat (specifically, for the LOSP cases). The existing NCC analyses using an acceptable thermal-hydraulic code and following the NCC procedures for United States PWRs show that the time required to achieve cooldown from hot standby to RHR entry conditions ranges from 8 to 24 hours, which are significantly greater than 4-hour assumed in the MFLB analysis.

Based on (1) the NCC phenomena in GL 81-21, (2) guidance for NCC procedures in Westinghouse ERGs, and (3) extended cooldown time from the existing NCC analyses discussed above, the NRC staff requests the licensee to provide information to support the adequacy of the assumed NCC completion time of 4 hours. In the material to be submitted, the following information should be provided:

1. The Required NCC time

    a. Determine whether vessel voiding will occur or not during the period of NCC from hot standby at temperature of 550 °F to the RHR entry temperature of 350 °F.

    b. Assess the effects of the vessel voiding on the time required to cool down the plant from hot standby to the RHR entry conditions, if the existence of void is determined.

    c. Provide the new calculated CST water volume, if the required NCC time is determined to be greater than 4-hour assumed in the current analysis.

    d. Describe the methods used to determine the void formation (Item 3.1.a) and required NCC time to achieve the RHR entry conditions (Item 3.1.b), and address the acceptability of the methods used (including thermal hydraulic codes simulating the RCS response during NCC).

    e. List the nominal values with measurement uncertainties and the corresponding values used in the NCC analysis to address Items 3.1.a and 3.1.b for the following parameters: (1) initial power level, and decay heat model and initial value in percentage of the rated thermal power, (2) initial RCS and Steam Generator (SG) pressure, (3) initial pressurizer and SG water volume, (4) AFW temperature and flow rate per SG, (5) SG Power-Operated Relief Valve (PORV) steam flow rate from intact and affected SGs, (6) pressurizer PORV flow rate and (7) auxiliary spray flow rate.

The discussion should include rationale to show that the value of each of the above parameters used in the subject NCC analysis is conservative, resulting in a longest cooldown time.

f. Provide the sequence of the event for the NCC analysis used to address Items 3.1.a and 3.1.b above, and time response for key RCS parameters including RCS flow, pressurizer pressure, pressurizer level, RCS hot and cold leg temperatures, SG pressure, RCS hot and RVUH sub cooling, RVUH steam volume, integral AFW flow, AFW flowrate to each of the SGs, charging and safety injection pump flowrates, SG and pressurizer PORV flow, and auxiliary spray flow for the analysis of the Nee from hot standby to RHR entry conditions.

g. Justify that, if the plant NCC procedures and/or a thermal hydraulic code are not used in addressing above Items 3.1.a and 3.1.b, the licensee's approach is conservative, resulting in a longest time to achieve the RHR entry conditions.

2. Plant NCC Procedures

a. Provide a copy of current plant NCC procedures, and verify that the procedures are consistent with the corresponding Nee procedures in Westinghouse ERGs.

b. Identify operator actions and associated action times credited in the NCC analysis used to addressing Items 3.1.a and 3.1.b. Where an operator action is credited, confirm that such action is consistent with the plant NCC procedures, and action times are conservative, resulting a longest time to achieve the RHR entry conditions.

c. List the assumptions used in the NCC analysis in addressing above items criteria applicable to key plant parameters for "controlled cooldown". Identify the assumptions and acceptance criteria that are different from that of the NCC procedures, and justify the differences. (The key plant parameters that the acceptance criteria are applied include the RCS subcooling margin, pressurizer water level range, cooldown limits, RCS temperature and pressure limits, RCS hot-leg temperature limits, and steam bubble size limit).

d. Under the assumed LOSP conditions, address the functionality of each SG and pressurizer PORV, or auxiliary spray. Discuss what, if any, function of the PORV, or auxiliary spray provides, and its capability to perform that function assumed in the NCC analysis used to address Items 3.1.a and 3.1.b. If the valve's actuation must be manual, provide information to show that the operator is capable of actuating the valve within the analytical assumed time. Provide justification for the case when the PORVs or auxiliary spray are not used in the subject NCC analysis.

e. List the single failure events considered in the NCC analysis for addressing Items 3.1.a and 3.1.b above, and identify the worst single failure used in the subject NCC analysis that results in a longest cooldown time. Provide justification if single failure event is not considered in the NCC analysis for the MFLB event with LOSP, a design basis accident included in the UFSAR Chapter 15.

f. Provide a list of systems and components which are used in the NCC analysis to address above Items 3.1.a and 3.1.b. Specify whether each system and component specified is safety grade. For pressurizer and SG PORVs, auxiliary spray and control valves, specify the valve motive power and confirm whether the motive power, valve controls, and valve motive air system are safety grade. For non-safety grade systems and components, state whether safety grade backups are available which can be expected to function or provide the desired information within a time frame compatible with the cooldown shown by the subject NCC analysis, or justify that non-safety grade component can be used for the MFLB event, a design basis accident. Specify the plant parameters that are monitored during the subject NCC analysis, and confirm that all instrumentation used by the operator to measure these parameters is safety grade. If any of the above instrumentation is non-safety grade, justify its use in the subject NCC analysis.

**SNC Response to RAI 3.1a. – f.**

Responding to RAI 3 is complicated because it contains discussions of a 1981 generic issue, emergency response procedures (ERP), and deterministic safety analysis assumptions. The RAI also challenges the four-hour cooldown time assumed in the safety analysis and requests the detailed NCC analysis. The cooldown time of four hours is not directly related to the proposed Technical Specification change request on vortexing, but cooling down at a rate less than 50 degrees per hour increases the time to reach 350 degrees and RHR entry conditions. This will increase inventory requirements for secondary side heat removal.

The following paragraphs discuss these three issues and how they relate to the proposed Technical Specification change on CST volume and "sizing".

In 1981, the NRC issued a generic letter (GL) following a voiding event at a PWR domestic plant during NCC. Alabama Power Company (APCo) responded to the GL with a series of submittals and RAI responses. The resolution included a Farley Unit 2 test and incorporation of NCC into PWR and Farley ERPs. As the RAI states, it is preferable to cool down slowly to RHR entry conditions under NCC conditions. Since this issue was resolved many years ago and the resolution incorporated into ERPs, it does not relate directly to CST volume requirements under NCC conditions or the requested Technical Specification change on CST volume due to non-conservative sizing assumptions.

The Technical Specifications for CST volume support the safety analysis assumptions. Technical Specification Bases 3.7.6 and Final Safety Analysis Report Section 9.2.6.3 list the two hour in hot standby followed by a 50 degree per hour cooldown for four hours to RHR entry conditions for the MFLB case concurrent with loss of offsite power. The Farley ERPs recommend cooldown rates below 50 degrees per hour for NCC conditions following an accident such as a line break. This apparent inconsistency between deterministic safety analyses and the ERGs/ERPs occurs because the safety analyses and ERPs were developed for different purposes and have different assumptions.

Risk-informed regulation and severe accident management procedures are other examples of programs that have improved safety at nuclear facilities, but in some cases are inconsistent with the original licensing basis based on deterministic safety analyses, single failure, etc. The Emergency Response Guidelines provide a network of predefined and prioritized symptom-based response strategies that guide the operator in management of emergency transients. Event-related recovery and function-related restoration strategies are combined to guide diagnosis and plant recovery to the optimal end state while ensuring explicit diagnosis and restoration of the plant safety state independent of event sequence, thus enabling the operator to address multiple or simultaneous events.

The following discussion provides insights on the deterministic safety analysis assumptions and the ERPs relative to CST volume and level. The Technical Specification volume requirement assumes an instantaneous pipe rupture, loss of offsite power, and a design basis tornado missile damaging the CST at a critical location. No quantitative assessment has been made of the probability of such an event for this submittal, but it would be very low. The tornado missile protected portion of the CST used as the basis for the proposed Technical Specification change is approximately the first 13 feet and corresponds to the 164,000 gallons of inventory. The CST and all piping and components required to supply the auxiliary feedwater pumps are Safety Class 2B and Seismic Category I. However, only the lower portion of the tank (below 16.4) feet is considered missile protected. Not only does the deterministic analysis not credit the additional water volume in the tank, it also does not credit alternate sources of make-up if the CST was unavailable. Therefore, the proposed Technical Specification change increasing CST inventory volume requirements from 150,000 gallons to 164,000 gallons for vortexing assumed no additional water was in the CST at the onset of the postulated accident.

The ERPs are more symptom-based. The CST has a 500,000 gallon capacity and the level is administratively required to be held at 26 feet (approximately 323,000 gallons) under normal operation. The ERPs allow the operator to use this inventory, and if necessary, other sources of inventory, for auxiliary feedwater pumps.

A confirmatory calculation was performed to assess the impact of a longer cooldown time to RHR entry conditions. If the unit remained in hot standby for two hours and then took an additional eight hours to reach RHR entry conditions, it would require approximately 180,000 gallons of inventory for secondary side heat removal. This is well below the administrative limits, but the water level for this volume is slightly above the missile protected volume of the tank credited in the development of the required Technical Specification volume.

An NRC accepted computer program (TORMIS) has been used to calculate the probability of a tornado missile damaging the CST. For FNP, the specific acceptance criterion for tornado damage for unprotected systems and components required for a tornado event is that the cumulative sum of the mean failure probabilities for these systems and components be less than 10E-06 per year per unit. For the CST, results show a probability of damage for all sections, including the unprotected sections, to be no greater than 5.63E-07 (identified as the Upper Section, which is defined as the section of the CST at a height from 24.6 feet to 41.0 feet). This is within the acceptance criterion of 10E-06.

The TORMIS results are not part of the current licensing basis assumptions for the Chapter 15 safety analysis or the sizing of the CST protected by the Technical Specifications. Regardless, it should be noted that the water level administratively maintained in the CST is within the Upper Section (26 feet) and that the Middle Section of the CST, defined as 16.4 feet to 24.6 feet, has an even lower probability of damage according to the TORMIS evaluation. While the Middle and Upper Sections cannot be credited with tornado missile protection, the damage probability to these portions of the CST containing the administratively held level is within the acceptance criterion.

SNC currently does not have a detailed thermal-hydraulic model for NCC supporting the MFLB safety analysis. As discussed in the paragraphs below, the Chapter 15 accident analysis covers only the short term portions of the postulated accident when stored energy and decay and sensible heat are high. Acceptance criteria for the thermal-hydraulic codes are analyzed until primary side and secondary cooling capacity exceed decay/sensible heat. The required CST inventory is calculated using different models and methods and does not require a detailed T-H model. The required CST inventory has been recalculated and submitted several times over the life of the plant, but has been based on the increased volume necessary for the proposed change itself. Relatively recent examples included the FNP power uprate and steam generator replacement projects. More detailed NCC evaluations are being performed as part of the severe accident management efforts. However, there are no current plans to revise the deterministic safety analyses based on these efforts.

The Chapter 15 transient analysis only demonstrates the acceptability of changes to AFW flow requirements in terms of short-term consequences (i.e., prior to event turnaround, which is when the cooling capacity equals the decay heat production, which is on the order of 15 minutes to 1 hour. After this time period, the operators gain control and transition the plant to hot standby, which is where RAI 3 begins). Therefore, RAI 3 questions the methodology/assumptions for portions of the analysis that happen well after the end of the non-LOCA analysis scope.

FSAR Section 15.4.2.2.2.1 states "[a] detailed analysis using the RETRAN-02([52, 53, 54]) code is performed in order to determine the plant transient following a feedline rupture. The code describes the plant thermal kinetics and the RCS, including natural circulation, pressurizer, steam generators, and feedwater system; and computes pertinent variables, including the pressurizer pressure, pressurizer water level, and reactor coolant average temperature." The discussion of natural circulation above refers to the modeling of the loss of offsite power MFLB case. From the time of RCP trip/coastdown, to the end of the time period analyzed (up until event turnaround) there is no forced primary coolant flow and natural circulation is the driver for any flow.

The same FSAR section goes on to state that the "RETRAN code is used to calculate the course of the system transient through the time that the auxiliary feedwater system heat removal capacity exceeds decay heat generation". This defines the time to which the transient is analyzed. The Chapter 15 MFLB analysis is concerned with demonstrating the ability of the AFW system to provide adequate cooling to prevent core damage. The time period of interest for demonstrating this is until the secondary side heat removal exceeds decay heat production, since after this point in time, there is a net removal of heat and the most challenging time has passed. At this point, the Chapter 15 portion of the analysis is over, and the EOPs are utilized to safely take the plant to a Cold Shutdown condition.

**SNC Response to RAI 3.1.a**

The Wesinghouse Owner's Group ERGs do not explicitly provide a transition to ES-0.2, after SI termination following a secondary fault, should a natural circulation cool down be desired or required. The ERG transition is to "plant specific procedure". FNP does not have any plant specific natural circulation cool down procedures other than the ESP-0.2, 0.3, 0.4 series. Therefore, FNP has elected to utilize these instead of developing and training on an additional procedure to accomplish the same result.

As stated, a cool down without head voiding is preferred and would be accomplished per ESP-0.2 and the cool down rate would be limited to 25 °F/hr in any active loop. A check of available CST inventory is made to determine if this cool down rate must be exceeded. If there is insufficient CST inventory, a transition is made to either ESP-0.3 or ESP-0.4. These procedures allow a more rapid cool down rate, not to exceed 100 °F/hr in any active loop, so long as RCS subcooling is maintained and the magnitude of the head void is managed so that natural circulation is not impeded.

As symptom based procedures, these ERPs attempt to prevent any head voiding during natural circulation cooldown if plant conditions permit. If this is not possible, alternative guidance is provided which allows a more rapid cool down and accommodates the resulting head voiding while maintaining core subcooling and preventing obstruction of natural circulation.

The Background Document for ESP-0.2 cites certain conservatisms used in the natural circulation cooldown analysis that help to ensure that the upper head fluid will actually be subcooled when the RCS is depressurized to 400 psig, even though the analysis shows that the fluid will be cooled only to the saturation temperature for 400 psig. These conservatisms assure that the recovery strategy is technically sound, and provides the necessary guidance for operators to perform a natural circulation cooldown and depressurization to cold shutdown of the RCS without forming a void in the upper head region for $T_{hot}$ upper head plants, even if the CRDM fans are not in operation.

**SNC Response to RAI 3.1.b**

Refer to the response to RAI 3.1.a.

**SNC Response to RAI 3.1.c**

A confirmatory calculation was performed to assess the impact of a longer cooldown time to RHR entry conditions during natural circulation. If the unit remained in hot standby for two hours and then took an additional eight hours to reach RHR entry conditions at a cooldown rate of 25 °F/hr, it would require approximately 180,000 gallons of CST inventory for secondary side heat removal during MSLB and MFWLB events. The minimum volume required for both events would exceed the missile protected volume of the tank (164,000 gallons) by as much as 16,000 gallons, but well below the maintained CST administrative volume limits.

This assumption of an additional eight hours of cooldown to reach RHR entry conditions at a cooldown rate of 25 °F/hr is not applicable to other events that require natural circulation during LOSP. These other events have been identified as TS Basis - Operability and the LOSP events. Per the TS and FSAR accident analysis, CST volume is based on having sufficient water available to maintain the RCS in Mode 3 for 9 hours with steam discharging to the atmosphere concurrent w/LOSP. These plant events are not applicable to the lower cooldown rate assumption since the reactor is being held at Mode 3 and not in a cooldown mode, i.e., the unit is not being cooled to 350°F but held at $T_{no\text{-}load,}$ even though the RCS will be in natural circulation due to loss of power to the RCPs. The TS Bases – Operability event is described in TS Bases Section B 3.7.6 - LCO and LOSP event is described in FSAR Section 15.2.9.

**SNC Response to RAI 3.1.d**

Though there is no analysis to support the technique employed in ES-0.3, specific symptoms are relied upon to indicate that void growth is restricted to the top of the hot legs, so as not to disrupt the natural circulation flow.

By maintaining RCS subcooling, controlling pressurizer level, and monitoring RVLIS full range indications, a maximum cooldown rate of 100°F/HR and a continued RCS depressurization are permitted under natural circulation conditions. If, during the course of the cooldown/depressurization, the RVLIS indicates that void growth is approaching the top of the hot legs, RCS depressurization is stopped and the operator is instructed to repressurize the RCS to collapse the void.

Additionally, though there is no analysis to support the technique employed in ES-0.4, specific symptoms are relied upon to indicate that void growth is restricted to the upper head/upper plenum region, above the top of the hot legs, so as not to disrupt the natural circulation flow.

The growth of the vessel void is inferred from PRZR level indications. Initially a low PRZR level is established to accommodate void growth. PRZR level changes from inventory shrinkage and void growth are accounted for separately with a step-by-step cooldown/depressurization technique. This technique is based on the concept of meeting Technical Specification cooldown requirements and not creating a saturated primary system (as measured at the hot legs).

**SNC Response to RAI 3.1.e**

As stated in the introduction to the response to RAI 3, SNC currently does not have a detailed thermal-hydraulic model for NCC.

**SNC Response to RAI 3.1.f**

As stated in the introduction to the response to RAI 3, SNC currently does not have a detailed thermal-hydraulic model for NCC.

**SNC Response to RAI 3.1.g**

SNC has not used the longest time to reach RHR entry conditions for the deterministic safety analysis. However, the conservatisms and differences between the safety analysis and ERPs are discussed in the introduction to the response to RAI 3.

**SNC Response to RAI 3.2.a**

The plant's current NCC procedures are provided in Enclosure 3. These plant procedures were developed from the Westinghouse ERGs. Any deviations from the ERGs are detailed in procedures FNP-0-ESB-0.2, FNP-0-ESB-0.3, and FNP-0-ESB-0.4 (items 7, 8, and 9 below), which are also included in Enclosure 3. The included procedures are:

1. FNP-1-ESP-0.2, "Natural Circulation Cooldown to Prevent Reactor Vessel Head Steam Voiding"
2. FNP-2-ESP-0.2, "Natural Circulation Cooldown to Prevent Reactor Vessel Head Steam Voiding"
3. FNP-1-ESP-0.3, "Natural Circulation Cooldown with Allowance for Reactor Vessel Head Steam Voiding (with RVLIS)"
4. FNP-2-ESP-0.3, "Natural Circulation Cooldown with Allowance for Reactor Vessel Head Steam Voiding (with RVLIS)"
5. FNP-1-ESP-0.4, "Natural Circulation Cooldown with Allowance for Reactor Vessel Head Steam Voiding (without RVLIS)"

6. FNP-2-ESP-0.4, "Natural Circulation Cooldown with Allowance for Reactor Vessel Head Steam Voiding (without RVLIS)"
7. FNP-0-ESB-0.2, "Natural Circulation Cooldown to Prevent Reactor Vessel Head Steam Voiding"
8. FNP-0-ESB-0.3, "Natural Circulation Cooldown with Allowance for Reactor Vessel Head Steam Voiding (with RVLIS)"
9. FNP-0-ESB-0.4, "Natural Circulation Cooldown with Allowance for Reactor Vessel Head Steam Voiding (without RVLIS)"

**SNC Response to RAI 3.2.b**

Refer to the response to RAI 3.1.d.

**SNC Response to RAI 3.2.c**

Refer to the response to RAI 3.1.d.

**SNC Response to RAI 3.2.d**

As stated in the introduction to the response to RAI 3, SNC currently does not have a detailed thermal-hydraulic model for NCC. However, for an LOSP not complicated by other failures, the Service Air (SA) system will be operating and all of these components will be functional from the main control board. If the SA system were not available, the SG PORVs (SG atmospheric reliefs) can be supplied from the Emergency Air system which can be manually placed in service, the PRZR PORVs can be supplied from their backup source which is normally maintained aligned for service requiring no additional operator action and the auxiliary spray path will be unavailable. Following the initial manual alignments, both the SG PORVs and the PRZR PORVs can be controlled from the main control board.

**SNC Response to RAI 3.2.e**

As stated in the introduction to the response to RAI 3, SNC currently does not have a detailed thermal-hydraulic model for NCC.

**SNC Response to RAI 3.2.f**

As stated in the introduction to the response to RAI 3, SNC currently does not have a detailed thermal-hydraulic model for NCC. In response to other portions of the RAI, the Service Air system provides the primary motive force for all of the specified valves. The Service Air system is not seismic Class 1, but it is provided with safety grade emergency AC power. The Emergency Air system for the SG PORVs and the backup source for the PRZR PORVs are both safety grade. The AFW FCVs would have to be controlled locally with the actuator manual hand wheel. Capability to establish this local control within thirty minutes was demonstrated during the most recent TFPI. All monitoring instrumentation used in the NCC procedures is safety grade.

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**Enclosure 2**

**CD Containing DOEJ–FRSNC419117-M001 Version 2, "CST Volume Requirements per Plant Transient Events" (with Enclosures 3, 5, and 6)**

**Joseph M. Farley Nuclear Plant – Units 1 and 2**
**Southern Nuclear Operating Company Response Request for Additional**
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**Enclosure 3**

**CD Containing Joseph M. Farley Nuclear Plant Natural Circulation Cooldown Procedures (with Enclosures 2, 5, and 6)**

**Joseph M. Farley Nuclear Plant – Units 1 and 2**
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**Enclosure 4**

**Response to Request for Additional Information Dated April 18, 2013**

**RAI**

Farley uses a vortex correlation developed by Harleman (1959) for a vertically downward configuration that differs from the 45 degree upward flow pipe used at Farley. This correlation is shown to over-predict the submergence by a factor of three when compared to Catawba and McGuire test data at the Froude number that corresponds to the maximum flow rate from the CST. This would be a sufficient margin to justify using the Harleman correlation to support the LAR if the test data were established to be applicable. Achieving this requires an acceptable description of the plant configurations, the test configurations, and the test data.

Provide a complete description of the tests and test results for each test used to substantiate determination of critical submergence that includes the following:

1. Drawing of the plant tank and the plant suction pipe within the tank that provides all relevant dimensions,
2. Drawing of the corresponding test configuration that provides all relevant dimensions,
3. Description of how quantitative air entrainment is determined during testing,
4. Description of conduct of the test that includes any observations, and
5. Summary of the test data such as a plot of critical submergence as a function of Froude number.

**SNC Response to RAI Part 1**

Refer to Enclosure 5 of this letter for requested drawings. These drawings are:

1. U-161693 Version 2.0; Unit 1 CST General Plan
2. U-213481 Version 3.0; Unit 2 CST General Plan
3. U161703 Version B; Unit 1 CST 8 Inch Auxiliary Feed Pump Suction Nozzle
4. U213493 Version A; Unit 2 CST 8 Inch Auxiliary Feed Pump Suction Nozzle

**SNC Response to RAI Part 2**

Refer to pages 11 through 13 of Enclosure 6 of this letter for the response.

**SNC Response to RAI Part 3**

Refer to pages 13 through 17 of Enclosure 6 of this letter for the response.

**SNC Response to RAI Part 4**

Refer to page 17 of Enclosure 6 of this letter for the response.

**SNC Response to RAI Part 5**

Refer to pages 17 and 18 of Enclosure 6 of this letter for the response.

**Joseph M. Farley Nuclear Plant – Units 1 and 2**
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**Enclosure 5**

**CD Containing Condensate Storage Tank Drawings (with Enclosures 2, 3, and 6)**

**Joseph M. Farley Nuclear Plant – Units 1 and 2**
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**Enclosure 6**

**CD Containing Fauske and Associates, LLC Report No. FAI/13-0392, "Request for Additional Information by the Office of Nuclear Reactor Regulation, Joseph M. Farley Nuclear Plants, Units 1 and 2, Southern Nuclear Operating Company, Docket Nos. 50-348 and 50-364, Revision 0" (with Enclosures 2, 3, and 5)**