UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 26, 2012

Mr. M. J. Ajluni
Nuclear Licensing Director
Southern Nuclear Operating Company, Inc.
40 Inverness Center Parkway
Post Office Box 1295
Birmingham, AL 35201-1295

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 – REQUEST FOR ADDITIONAL INFORMATION (TAC NOS. ME9293 AND ME9294)

Dear Mr. Ajluni:

By letter dated August 20, 2012, Southern Nuclear Operating Company (SNC) submitted a license amendment request (LAR) for the Joseph M. Farley Nuclear Plant, Units 1 and 2. The LAR would revise the condensate storage tank (CST) level requirement specified in Technical Specification Surveillance Requirement 3.7.6.1. The change is related to the calculational basis for the level with respect to potential vortexing and assumptions regarding heat loads on CST volume. The U.S. Nuclear Regulatory Commission staff finds that additional information is needed as set forth in the Enclosure.

Please provide the additional information within thirty (30) days of the date of this letter.

Sincerely,

Robert Martin

Robert E. Martin, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-348, 50-364

Enclosure:
Request for Additional Information

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REQUEST FOR ADDITIONAL INFORMATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2

SOUTHERN NUCLEAR OPERATING COMPANY

DOCKET NOS. 50-348 AND 50-364

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.

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.

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 (ERGs). 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 NCC 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 NCC 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.

References:

1. Letter from M. J. Ajluni (SNC) to NRC, "Joseph M. Farley Nuclear Plant – Units 1 and 2, Response to Supplemental Information Request Regarding Technical Specifications Condensate Storage Tank Minimum Level, License Amendment Request," Received on October 15, 2012.

2. Enclosure 1 to Reference 1, "Response to Request for Supplemental Information."

3. Enclosure 2 to Reference 1, "SNC Calculation BM-95-0961-001, 'Verification of CST Sizing Basis', Version 6.0.

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