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SOUTHERN COMPANY
Energy to Serve Your World℠

September 12, 2008

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NL-08-1365

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

Joseph M. Farley Nuclear Plant, Units 1 & 2
Response to Request for Information Regarding
Revision to Technical Specifications 3.3.1, 3.3.2, 3.3.6, 3.3.7, and 3.3.8

Ladies and Gentlemen:

In letter dated December 20, 2007, Southern Nuclear Operating Company (SNC) requested an application for amendment to Facility Operating License Nos. NPF-2 (Unit 1) and NPF-8 (Unit 2) for Joseph M. Farley Nuclear Plant (FNP), in accordance with the provisions of 10 CFR 50.90. The proposed amendment would revise Technical Specification (TS) 3.3.1, "Reactor Trip System (RTS) Instrumentation," TS 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," TS 3.3.6, "Containment Purge and Exhaust Isolation Instrumentation," TS 3.3.7, "Control Room Emergency Filtration/Pressurization System (CREFS) Actuation Instrumentation," and TS 3.3.8, "Penetration Room Filtration (PRF) System Actuation Instrumentation" to adopt Completion Time, bypass test time, and Surveillance Requirement (SR) Frequency changes approved by the NRC in WCAP-14333-P-A, Revision 1, "Probabilistic Risk Analysis of the RTS and ESFAS Test Times and Completion Times," October 1998 and WCAP-15376-P-A, Revision 1, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times," March 2003. In addition, the proposed amendments would revise SR 3.3.1.8 to adopt Surveillance Frequency changes approved by the NRC in Industry/TSTF STS Change Traveler 242, Revision 1, "Increase the time to perform a COT on Power Range and Intermediate Range Instruments." Also, the proposed amendments would revise the Completion Times of LCO 3.3.1, Condition F from 2 hours to 24 hours consistent with changes approved by the NRC in Industry/TSTF STS Change Traveler 246, Revision 0, "RTS Instrumentation, 3.3.1 Condition F Completion Time." Finally, the proposed amendments would provide for minor editorial changes.

SNC requested approval of the proposed amendment request by December 1, 2008. It is anticipated that the license amendment, as approved, will be effective upon issuance and will be implemented within 90 days from the date of issuance.

(Affirmation and signature are provided on the following page.)

On August 26, 2008, a telecon was held with the NRC Staff to discuss questions on this proposed amendment request. The SNC response to the requested information is provided in Enclosure 1.

A copy of the proposed changes has been sent to Dr. D. E. Williamson, the Alabama State Designee, in accordance with 10 CFR 50.91(b)(1).

Mr. M.J. Ajluni states he is the Manager, Nuclear Licensing 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.

If you have any questions, please advise.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

Mark J Ajluni

M.J. Ajluni
Manager, Nuclear Licensing

Sworn to and subscribed before me this 12th *day of* September*, 2008.*

Patricia H. Raymond
Notary Public

My commission expires: 7-21-2012

MJA/BDM/phr

Enclosure: 1.SNC Response to Request for Information

cc: Southern Nuclear Operating Company
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Mr. J. R. Johnson, Vice President – Farley
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Mr. L. A. Reyes, Regional Administrator
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Mr. E. L. Crowe, Senior Resident Inspector – Farley

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

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Enclosure 1

SNC Response to Request for Information

NRC Question

1. The licensee's license amendment request does not discuss external events with regard to implementing WCAP-14333 and WCAP-15376. Discuss the impact on external event risk including seismic, fire, and high wind, floods or other (HFO) events.

SNC Response:

(1) Seismic:

Although a seismic PRA has not been developed, a seismic margins assessment (SMA) for resolution of the seismic portion of NRC GL 88-20, Supplement 4 entitled "Individual Plant Examination of External Events (IPEEE) for Severe Accidents," was performed for the Farley Nuclear Plant (FNP). The SMA review level earthquake for FNP is a 0.1 g peak ground acceleration (PGA) as recommended for a reduced-scope plant in NUREG-1407. FNP structures and equipment were designed using a modified Newmark spectrum with a horizontal PGA of 0.1 g for the Safe Shutdown Earthquake (SSE) and 0.05 g for the operating basis earthquake. Based on the results of the SMA evaluations, FNP Units 1 and 2 have a high-confidence-low-probability-of-failure capacity of at least 0.1 g PGA.

Furthermore, the probability of an earthquake greater than 0.1 g PGA occurring during the additional time of the proposed Reactor Trip System (RTS) and Engineered Safety Feature Actuation System (ESFAS) extended surveillance test interval (STI) and completion times (CTs) is on the order of 2.37E-07 (based on Farley-specific hazard curve from NUREG 1488 assuming that the worse case contribution is attributed to the CT extension for the RTS). Even if it is assumed that the conditional probability of core damage is 0.1, the seismic contribution would likely be on the order of 2.37E-08. Therefore any seismic-related increase in risk due to the proposed RTS and ESFAS extended STI and CTs, although not quantified, is expected to be negligible.

Consequently, it is expected that the conclusion made for the proposed RTS and ESFAS extended STI and CTs will remain unaffected with a lack of accounting for seismic risk contribution.

(2) Fire:

A Fire Induced Vulnerability Evaluation (FIVE) analysis was performed for FNP in response to the request of Generic Letter 88-20, Supplement 4 entitled "Individual Plant Examination of External Events (IPEEE) for Severe Accidents." An evaluation was performed to determine the potential impact of a fire in the significant fire compartments identified in the IPEEE on the proposed RTS and ESFAS extended STI and CTs.

The compartments identified as having significant fire impacts can be grouped into nine general categories with respect to the significant contributors to fire risk and potential plant improvements:

Switchgear Rooms

This category includes fire compartments 18A, 19A, 21A, 41A, 56A, and 56B. The significant contributors to risk for the switchgear rooms are fires in the oil-filled transformers for 600-V load centers located in the rooms and fires originating in the Control Rod Drive Mechanism Motor-Generator sets. Loss of the on-service train switchgear leads to a loss of RCP seal cooling support systems. Consistent with NEI guidelines for assessing IPEEE vulnerabilities, Southern Nuclear Operating Company (SNC) identified procedural enhancements to improve response to a fire-induced loss of Reactor Coolant Pump (RCP) seal cooling.

Electrical Penetration Rooms

This category includes fire compartments 34B and 35A. The significant contributors to risk for these compartments are fires in the motor control centers (MCCs) resulting in spurious closure of valves in the service water supply to the component cooling water heat exchanger or in the discharge paths for the motor-driven auxiliary feedwater pumps. Since these spurious closures would require smart hot-shorts in the valve control circuits, the risk for these compartments was considered conservatively high. Consistent with Nuclear Energy Institute (NEI) guidelines for assessing IPEEE vulnerabilities, SNC identified procedural enhancements to improve response to the potential spurious valve closures.

Main Control Room

This category consists of fire compartment 44A. The major contributors to risk in this compartment are fires in the main control boards which result in loss of control of both trains of safe shutdown (SSD) equipment and require plant shutdown using controls on the hot shutdown panels. This analysis is also considered conservative, since the configuration of the control board and distance between controls for various SSD systems make it unlikely that both trains of every SSD system will be damaged before the fire is extinguished. Consistent with NEI guidelines for assessing IPEEE vulnerabilities, SNC verified that procedures were in place to address loss of control from the main control board due to fire.

Service Water Pump Room

This category consists of fire compartment 72A. The major contributor to risk in the compartment is transient fires in areas where both trains of service water are impacted. This would result in loss of RCP seal cooling. Consistent with NEI guidelines for assessing IPEEE vulnerabilities, SNC identified procedural enhancements to improve response to a fire-induced loss of RCP seal cooling.

Component Cooling Water Heat Exchanger/Pump Room

This category consists of fire compartment 6C. The major contributor to risk in the compartment is a fire in the on-service Component Cooling Water (CCW) pump, resulting in loss of RCP seal cooling and damage to control cables for the turbine-driven auxiliary feedwater pump. Consistent with NEI guidelines for assessing IPEEE vulnerabilities, SNC identified procedural enhancements to improve response to a fire-induced loss of RCP seal cooling.

Low Voltage Switchyard

This category consists of fire compartment 84A. The major contributor to risk in this compartment is fire in the auxiliary transformers or startup transformers resulting in a total or partial loss of offsite power. Consistent with NEI guidelines for assessing IPEEE vulnerabilities, SNC verified that procedures were in place to address loss of offsite power due to fire.

Cable Spreading Room

This category consists of fire compartment 40A. The major contributors to risk in this compartment are fires in electrical cabinets and transient combustible fires resulting in a loss of control of SSD equipment from the control room. Consistent with NEI guidelines for assessing IPEEE vulnerabilities, SNC verified that procedures were in place to address loss of control from the main control board due to fire.

Turbine Building

This category consists of fire compartment 80A. The major contributors to risk in this compartment are oil-filled transformer fires resulting in loss of offsite power. Equipment required for safe shutdown is not located in the Turbine Building, and does not have cables routed through the turbine building. In addition, turbine generator fires were verified not to contribute to loss of offsite power. Consistent with NEI guidelines for assessing IPEEE vulnerabilities, SNC verified that procedures were in place to address loss of offsite power due to fire.

Other Compartments

This category encompasses fire compartments 6A, 4A10, and 4A17. The major contributors to risk in these compartments are electrical cabinet fires, indoor transformer fires, and emergency air compressor fires resulting in loss of SSD equipment. These compartments were screened by evaluation external to the FIVE methodology, but were included in the IPEEE summary table to provide a complete accounting of all compartments not screened through FIVE, Phase II, step 3. Consistent with NEI guidelines for assessing IPEEE vulnerabilities, SNC verified that procedures were in place to address the fire risks in these compartments.

As can be seen, the major fire risks were associated with fires causing loss of offsite power and/or loss of RCP seal cooling support systems. In these sequences, core damage can be mitigated by operator actions to manually start any required mitigation equipment due to the slow developing nature of the event. Therefore, it is expected that the increase in fire risk due to the proposed RTS and ESFAS extended STI and CTs would be small. Consequently, it is expected that the conclusion made for the proposed RTS and ESFAS extended STI and CTs will remain unaffected with a lack of accounting for fire risk contribution.

(3) High Winds and Other External Events

The evaluation conducted for the FNP IPEEE, concluded that high winds, tornadoes, external flooding, transportation, and nearby facility accidents were not significant contributors to risk for FNP. Also there have been no significant changes that would adversely affect the high winds design basis at FNP since the issuance of the operating license. Thus, it is expected that the risk associated with high winds and other external events is small and that the conclusion made for the proposed RTS and ESFAS extended STI and CTs will remain unaffected.

NRC Question

2. The combined risk metric results are presented on page 10 of 49 of Enclosure 1A. The discussion of the risk metrics on page 11 of 49 of Enclosure 1A states that the ΔCDF and ΔLERF estimates are from the current licensing basis (WCAP-0271 to the implementation of WCAP-15376. It is stated that from the above table, the ΔCDF acceptance criterion is slightly exceeded. This is not clear from the table.

SNC Response

When the Δ CDF (Core Damage Frequency) for the changes from WCAP-10271 to WCAP-14333 is added to the change from WCAP-14333 to WCAP-15376 the change exceeds the acceptance criterion of < 1E-06. For the 2/4 logic the Δ CDF is 1.15E-6 and for the 2/3 logic, the Δ CDF is 1.4 E-6.

When the Δ LERF (Large Early Relief Frequency) for the changes from WCAP-10271 to WCAP-14333 is added to the change from WCAP-14333 to WCAP-15376 the 2/4 logic for the Δ LERF is 5.09 E-8 and for the 2/3 logic, the Δ LERF is 7.8 E-8.

NRC Question

3. The PRA quality discussion on page 15 of 49 of Enclosure 1A and the table starting on page 16 provide disposition level B facts and observations (F&Os). Confirm that all F&Os have been dispositioned and the impacts on the proposed LAR have been assessed and documented in the included table.

SNC Response

The disposition status for level B facts and observations (F&Os) in the table starting on page 16 is based on a review of the Farley Revision 7 PRA model which was also used as the source of information contained in Tables 1A, 1B, 2, and 3 (pages 28-34) of Enclosure 1A. The disposition table addresses all level B F&Os from the Farley peer review. No level A F&Os were identified for Farley.

NRC Question

4. Confirm that the equipment out of service (EOOS) risk monitor utilizes the same fault tree and data based used for the PRA model evaluated for this amendment. Does this model include SSCs incorporating the proposed WCAP-14333 completion times (CTs) and bypass test times? Discuss the modifications to the FNP PRA and EOOS risk monitor to accommodate the increased CTs, bypass times implemented under WCAP-14333 and WCAP-15376. Also, will the FNP PRA and EOOS incorporate the proposed increase surveillance intervals?

SNC Response

The equipment out of service (EOOS) risk monitor utilizes the same Probabilistic Risk Assessment (PRA) model and database used for the evaluation of this amendment with the exception that all maintenance and test unavailability events are set to 0.0 in the EOOS database. Therefore, no modifications to the EOOS model are required to accommodate the proposed completion times and bypass test times of this amendment. EOOS does have the capability to evaluate the out of service condition of a train of RTS/ESFAS equipment to allow assessment of the risk associated with testing or unplanned unavailability as it occurs. The unreliability data for RTS/ESFAS will be updated to reflect the new surveillance intervals as part of the implementation process. SNC stated in the December 20, 2007 letter that the approved amendment would be implemented within 90 days from the date of issuance.

NRC Question

5. Page 24 of 49 of Enclosure 1A references implementation guideline tables 1A, 1B, 2, 3, and 4. Table 4 is not identified in Enclosure 1A.

SNC Response

See Enclosure 1A page 36 of 49 for Table 4.

NRC Question

6. The following regulatory commitment is identified in Enclosure 4.

For channels with built-in bypass capability, or for inoperable channels bypassed for surveillance testing of other channels, implement administrative controls to ensure that analog channels are not routinely removed from service at-power for channel calibration, if such calibration would take more than 4 hours.

A discussion of this regulatory commitment is not provided in Enclosure 1A.

SNC Response

See pages 12 of 49 and 13 of 49 for a discussion of the commitments identified in Enclosure 4, List of Regulatory Commitments.

NRC Question

7. Confirm if procedural enhancements concerning fires are a result of this amendment request or the IPEEE review.

SNC Response

The procedure enhancements referred to in the response to Question 1 are those identified in the IPEEE, not a result of this amendment request. The enhancements were aimed at improving response to a loss of RCP seal cooling and prevention of consequential RCP seal Loss of Coolant Accident (LOCA) and are noted in NUREG-1742 Volume 2 Table 3.5. All fire-related plant improvements listed in NUREG-1742 Table 3.5 have been verified as implemented in past Request for Additional Information (RAI) responses (SNC letter NL-04-0287 dated February 26, 2004 page 59 of 75 item b). Therefore, no commitments are being made to implement procedure enhancements with respect to fire risk as part of this amendment request.

NRC Question

8. Indicate if the CTs and STIs impact previously identified risk significant fire zones or result in additional risk sensitive fire zones specific to the proposed CTs and STIs.

SNC Response

The proposed CT and STI changes are not expected to impact risk significant fire zones or result in additional risk significant fire zones for the following reasons:

- Fires in the switchgear rooms (1-21A, 1-41A, 2-21A, 2-41A, 56A, 56B), turbine building (1-84A, 2-84A), and transformer yards (1-80A, 2-80A) result in Loss of Offsite Power which will result in a reactor trip without action of the RTS due to loss of power to the rod control system.
- Response to fire in the control room (44A) or cable spreading room (1-40A, 2-40A) directs the operators to manually trip the reactor and locally verify that the reactor trip breakers are open prior to transferring control to the hot shutdown panel (HSP). Once control is transferred to the HSP, ESFAS is not credited due to circuit design which defeats automatic operation of equipment following transfer to HSP control.
- Other areas likewise assume that the reactor will be tripped manually in response to the fire and that all equipment important to the mitigation strategy will be manually started if available.

NRC Question

9. The IPEEE results show Farley to have a high fire CDF estimate of about 1.66E-4 compared to internal events CDF of 2.3E-5/2.03E-5. Based on this, indicate whether the CTs and STIs impact on previously identified risk significant fire zones or result in additional risk sensitive fire zones specific to the proposed CTs and STIs. Discuss why total baseline risk is not expected to be greater than 1E-4/year after implementation of the proposed CTS and STIs.

SNC Response

The 1.66E-04/year CDF estimate for fire is from the IPEEE submittal. At the time of this submittal, the internal events CDF was 1.30E-04/year. The relationship between fire CDF and internal events CDF from the IPE and IPEEE was therefore:

Ratio of Fire to Internal CDF = 1.66E-04 per year / 1.30E-04 per year
= 1.28

The most recent update of the fire contributions for significant compartments was performed as part of a previous Service Water Intake Structure (SWIS) fire protection exemption request (NRC approved exemption in letter dated August 16, 2005 MC 0627 and MC 0628). At that time, the contribution to CDF from fire in the significant compartments identified in the IPEEE was calculated to be:

Unit 1

Fire Compartment	Description	Average CDF	Average LERF
1-41A	Aux Bldg SWGR Room Train A	1.57E-05	3.33E-09
44A	Control Room	1.16E-05	3.10E-09
1-21A	Aux. Bldg SWGR Room Train B	1.04E-05	2.20E-09
72A	SW Intake Structure	3.77E-06	8.01E-10
1-35A	Train A Elec. Pen. Room	2.18E-06	4.63E-10
1-34B	Train B Elec. Pen. Room	1.54E-06	3.26E-10
1-4A10	Aux Bldg, Elev 121' Elev.	9.95E-07	2.68E-10
1-6C	Aux Bldg (CCW Pumps)	7.36E-07	1.56E-10
56B	DG Bldg SWGR Room B	6.05E-07	1.28E-10
1-19A	Aux Bldg. SWGR Room 1B (DC)	6.00E-07	1.27E-10
1-18A	Aux Bldg. SWGR Room 1A (DC)	4.67E-07	9.90E-11
1-80A	XFMR Yard	3.08E-07	6.53E-11
56A	DG Bldg SWGR Room A	2.63E-07	5.53E-11
1-84A	Turbine Bldg	2.18E-07	4.62E-11
1-4A17	Aux Bldg 155' Elev.	2.12E-07	4.49E-11
1-40A	Unit 1 Cable Spread Rm	1.74E-07	4.27E-11
1-6A	Aux Bldg (TDAFWP)	2.18E-08	4.62E-12
	Total	4.98E-05	1.13E-08

Unit 2

Fire Compartment	Description	Average CDF	Average LERF
2-41A	Aux Bldg SWGR Room Train A	1.98E-05	4.19E-09
44A	Control Room Unit 2	1.19E-05	3.16E-09
2-21A	Aux. Bldg. SWGR Room Train B	1.01E-05	2.13E-09
2-4A16	Aux Bldg (155 Elev.)	4.07E-06	8.63E-10
2-6C	Aux Bldg (CCW Pumps)	3.69E-06	7.83E-10
72A	SW Intake Structure Unit 2	3.63E-06	7.71E-10
2-40A	Unit 2 Cable Spreading Room	3.62E-06	9.71E-10
56B	DG Bldg SWGR Room B Unit 2	9.69E-07	2.05E-10
2-80A	XFMR Yard	4.59E-07	9.72E-11
56A	DG Bldg SWGR Room A Unit 2	2.63E-07	5.58E-11
2-84A	Turbine Bldg	1.60E-07	3.39E-11
2-6A	Aux Bldg (TDAFWP)	9.63E-08	2.04E-11
2-4F	Aux Bldg (Chem. Drain Tank Rm)	1.94E-08	4.11E-12
2-4C	Aux Bldg (Boric Acid Area)	9.08E-09	1.92E-12
	Total	5.87E-05	1.33E-08

These values compare to internal events CDF values of 3.86E-05/year for Unit 1 and 5.81E-05/year for Unit 2. Therefore, the relationship between internal events CDF and fire CDF for the Revision 5 Farley PRA model was:

Unit 1 Ratio of Fire to Internal CDF = 4.98E-05 per year / 3.86E-05 per year = 1.29

Unit 2 Ratio of Fire to Internal CDF = 5.87E-05 per year / 5.81E-05 per year = 1.01

Since the most recent update of the fire PRA values noted above, modifications have been completed on the Unit 2 service water pumps which has reduced the internal events CDF to a value more comparable to the Unit 1 values. The modifications involved replacement of the Unit 2 service water pump assemblies such that support from an external booster pump for the lube and cooling system was no longer required. The new pump assemblies are self-lubricated. Therefore, the fire results for Unit 2 would be expected to be more similar to Unit 1 with the current design.

Both the IPEEE and Revision 5 results are considered conservative in that many components are not credited in the fire results due to uncertain cable routing (see response 10 below). In addition, screening fire modeling was performed only in a few important compartments since the purpose of the IPEEE fire analysis was vulnerability screening.

Therefore, based on past evaluations, it is estimated that the baseline internal events plus fire CDF can be approximated by multiplying the internal events CDF by a factor of 2.3. This would produce baseline internal events plus fire CDF values of:

Unit 1 Internal plus Fire CDF = 2.35E-05 per year * 2.3
= 5.41E-05 per year

Unit 2 Internal plus Fire CDF = 2.03E-05 per year * 2.3
= 4.67E-05 per year

The proposed changes to the CT and STI changes are not expected to impact risk significant fire zones or result in additional risk significant fire zones for the reasons outlined in response 8. Therefore, the total baseline risk is expected to be less than 1.0E-4 after implementation of CT and STI changes when fire CDF is considered consistent with the acceptance guidelines of Regulatory Guide 1.74.

NRC Question

10. Discuss the Five methodology (including conservatisms)

SNC Response

The FIVE analysis performed for the Farley IPEEE used the standard FIVE analysis methods to calculate fire ignition frequencies for the plant fire compartments combined with conditional core damage probabilities (CCDP) given the expected damage due to the fire. The determination of fire impacts for each fire compartment was based on information contained in the Raceway Database for Appendix R electrical raceways. The Appendix R database documents the routing of cables associated with the equipment designated as safe-shutdown equipment in the Appendix R compliance review. However, the PRA model credits some equipment for which the cable routing is not documented in the Raceway Database. Therefore, a global assumption was made that the equipment in the PRA model which is not included in the Raceway Database would be failed for any fire. The specific equipment not included in the Appendix R analysis is:

- Instrument Air Compressors A and B
- Containment Fan Coolers
- Containment Spray System
- Steam Dumps
- Main Feedwater Pumps
- Condensate Pumps
- SW from the opposite unit
- Instrument Air from the opposite unit

Therefore, the FIVE analysis was similar to many IPEEE fire PRAs in that the analysis calculated actual CCDPs for each fire compartment and combined those values with appropriate fire ignition frequencies. However, the results are also believed to be conservative because many of the components which were globally failed could have been credited in many compartments with more detailed circuit analysis and fire modeling.