December 17, 2003

Mr. J. B. Beasley, Jr.
Vice President
Southern Nuclear Operating Company
Post Office Box 1295
Birmingham, Alabama 35201

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING SEVERE ACCIDENT MITIGATION ALTERNATIVES FOR THE JOSEPH M. FARLEY NUCLEAR PLANT UNITS 1 AND 2 (TAC NOS. MC0768, MC0769)

Dear Mr. Beasley:

The staff has reviewed the Southern Nuclear Operating Company's analysis of severe accident mitigation alternatives (SAMAs) submitted in support of its application for license renewal for the Joseph M. Farley Nuclear Plant Units 1 and 2, and has identified areas where additional information is needed to complete its review. Enclosed are the staff's RAIs. We request that you provide your responses to these RAIs by March 1, 2004, in order to support the license renewal review schedule. If you have any questions, please contact me at (301) 415-1424.

Sincerely,

/RA/

Jack Cushing, Project Manager
Environmental Section
License Renewal and Environmental Impacts Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket Nos.: 50-348, 50-364

Enclosures: As stated

cc w/enclosures: See next page

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Request for Additional Information Related to Severe Accident Mitigation Alternatives (SAMAs) for the Joseph M. Farley Nuclear Plant, Units 1 and 2 (FNP)

1. The SAMA analysis is based on the most recent version of the FNP Probabilistic Risk Assessment (PRA) for internal events, i.e., Revision 5, which is a modification to the individual plant evaluation (IPE) submittal transmitted to the NRC in June 1993. Please provide the following information regarding this PRA model:

    a. a description of the internal and external peer reviews of the level 1, 2 and 3 portions of the PRA that have been performed since the IPE;

    b. a description of the overall findings of the Westinghouse Owners Group PRA Peer Review (by element) and discussion of any findings/observations (e.g., A and B Facts and Observations) that could potentially affect the SAMA identification and evaluation process, and how SNC has addressed these findings for this application (including for example sensitivity studies of the impacts of alternative assumptions);

    c. a breakdown of the internal events core damage frequency (CDF) by major contributors, initiators or accident classes, such as loss of offsite power (LOOP), station blackout (SBO), transients, anticipated transient without scram (ATWS), loss-of-coolant accident (LOCA), interfacing systems LOCA (ISLOCA), internal floods, etc.;

    d. the approximate core damage frequency (CDF) and large early release frequency (LERF) for each revision to the PRA model, as described in Section 1.1 of Attachment F to Appendix D of the ER, and a description of the major reasons for the changes from the prior version;

    e. the changes in the level 2 methodology since the IPE submittal, including major modeling assumptions, plant response tree (PRT)/containment event tree (CET) structure;

    f. the methodology and criteria for binning endstates into the 13 accident sequences/release categories shown in Table F-6 and used in the current level 3 analysis;

    g. the specific source terms used to represent each of the 13 accident sequence/release categories, and a containment matrix describing the mapping of Level 1 results into the various accident sequences/release categories;

    h. a description of the accident sequence used to represent each of the 13 accident sequences/release categories shown in Table F-6, and how each sequence was chosen to represent a bin;

    i. a breakdown of the population dose (person-rem per year within 50 miles) by containment release mode, such as steam generator tube rupture (SGTR), ISLOCA, containment isolation failure, early containment failure, late containment failure, and no containment failure; and

j. justification for why early containment failure mechanisms are not included in the PRA quantification.

2. The CDF cited and used in the SAMA analysis is based on the risk profile for internal events at Farley Unit 1. Please provide the internal events CDF for Unit 2 if different, and a discussion of the reasons for any differences from Unit 1. Discuss the impact on the SAMA analysis and results if the analysis were based on Unit 2 rather than Unit 1.

3. The reactor coolant pump (RCP) seal LOCA previously contributed 47% to the CDF. One of the plant improvements under consideration at the time of the IPE was replacing the current RCP seal O-rings with new high temperature O-rings. SAMA 13, which addresses installation of improved seals, is labeled as already addressed by the existing plant design. Confirm that O-rings constructed from improved materials have been installed on all pumps. Discuss the RCP seal LOCA model utilized in the FNP PRA and why this is judged to provide an appropriate representation of RCP seal LOCA events. Also, indicate the current percent contribution to the CDF for RCP seal LOCA.

4. The MACCS analysis assumes all releases occur at ground level and has a thermal content the same as ambient. These assumptions could be non-conservative when estimating offsite consequences. Please provide an assessment of the sensitivity of offsite consequences (doses to the population within 50 miles) to these assumptions.

5. According to Table F-10 of the Environmental Report (ER), SNC evaluated 124 SAMA candidates (SNC states there are 128 SAMAs, however four were not used). SNC indicates that the set of SAMAs was developed from review of lists for other plants, NRC documents, and advanced power reactor designs. It is not clear that the set of SAMAs evaluated in the ER addresses the major risk contributors for FNP. In this regard, please provide the following:

   a. a description of how the dominant risk contributors at FNP, including dominant sequences and cut sets from the current PRA and equipment failures and operator actions identified through importance analyses (e.g., Fussell-Vesely, Risk Reduction Worth, etc.) were used to identify potential plant-specific SAMAs for FNP;

   b. the number of sequences and cut sets reviewed/evaluated and what percentage of the total CDF they represent;

   c. a listing of equipment failures and human actions that have greatest potential for reducing risk at FNP based on importance analysis and cut set screening;

   d. for each dominant contributor identified in the current PRA (Revision 5), a cross-reference to the SAMA(s) evaluated in the ER that address that contributor. If a SAMA was not evaluated for a dominant risk contributor, justify why SAMAs to further reduce these contributors would not be cost beneficial; and

   e. a listing of the industry and NRC documents used to derive the set of SAMAs for FNP.

6. The set of SAMAs considered in the FNP ER appear to have originated from a compilation of potential plant improvements developed as part of SNC's license renewal application for Hatch Nuclear Plant. In license renewal applications for subsequent plants, several additional SAMAs have been identified that might also be applicable for FNP. These include SAMA numbers 59, 60, 149, 166, 169, 175, 177, 210, 211, and 216 in Table F.4-1 of the ER for Summer Nuclear Station. Please provide rationale for eliminating each of these SAMAs from further consideration at FNP, e.g., justification that the objective of the candidate SAMA and the associated risk reduction is addressed by one or more of the Phase 1 SAMAs identified in Table F-10 of the FNP ER (with reference to the appropriate Phase 1 SAMAs), or that the candidate SAMA is not relevant to FNP.

7. The SAMA analysis did not include an assessment of SAMAs for external events. The FNP IPE for External Events (IPEEE) has shown that the CDF due to internal fire initiated events is about 1.66×10^{-4} per reactor year for Unit 1, and 1.28×10^{-4} per reactor year for Unit 2 which is substantially greater than the internal events CDF on which the SAMA evaluation is based. The risk analyses at other commercial nuclear power plants also indicate that external events could be large contributors to CDF and the overall risk to the public. In this regard, the following additional information is needed:

    a. NUREG-1742 ("Perspectives Gained From the IPEEE Program," Final Report, 4/02), lists the significant fire area CDFs for FNP (page 3-21 of Volume 2). While these fire-related CDF estimates may be conservative, they are still large relative to the FNP internal events CDF. For each fire area, please explain what measures were taken to further reduce risk, and explain why these CDFs can not be further reduced in a cost effective manner;

    b. Table 3.5 of NUREG-1742 lists several fire-related plant improvements for FNP (pages 3-55 and 3-56 of Volume 2). Indicate whether all of the "Plant improvements" have been implemented. If not, please explain why within the context of this SAMA study;

    c. NUREG-1742 lists seismic outliers and improvements for FNP (page 2-28 of Volume 2). Indicate whether the "Plant improvements" that address the outliers have been implemented for all outliers. If not, please explain why within the context of this SAMA study;

8. SNC has opted to double the estimated benefits (for internal events) to accommodate any contributions for external events. This is acceptable when sound reasons exist to support such a numerical adjustment. However, based on the information in the ER and in the FNP IPEEE report, the fire CDF is approximately a factor five greater than the internal events CDF, which suggests that the baseline CDF should be increased by a factor of six to account for external events. In order to determine if external events have been satisfactorily accounted for, please provide the following information:

    a. the current CDF for fire-initiated events, and justification that doubling the estimated benefits for internal events will bound the risk from fire events;

b. a description of the impact on the fire CDF from the plant/procedure modifications that were made in conjunction with SNC's decision to retract the two Appendix R exemption requests as described in a letter to the NRC dated June 29, 2000;

c. an assessment of the impact on the Phase 1 screening if the internal events risk reduction estimates are increased by a factor that would bound the risk from fire and seismic events; and

d. an assessment of the impact on the Phase 2 evaluation if risk reduction estimates are increased by a factor that would bound the risk from fire and seismic events.

9. The SAMA analysis did not include an assessment of the impact of PRA uncertainties. On that basis, please provide the following information to address these concerns:

    a. an estimate of the uncertainties associated with the calculated core damage frequency (e.g., the mean and median internal events CDF estimates and the 5th and 95th percentile values of the uncertainty distribution);

    b. an assessment of the impact on the Phase 1 screening if risk reduction estimates are increased to account for uncertainties in the risk assessment; and

    c. an assessment of the impact on the Phase 2 evaluation if risk reduction estimates are increased to account for uncertainties in the risk assessment. Please consider the uncertainties due to both the averted cost-risk and the cost of implementation to determine changes in the net value for these SAMAs.

10. Provide the requested information on the following SAMAs:

    a. why SAMA 122 was screened in Phase 1 when it has an estimated cost of $1.4M;

    b. why SAMAs 14 and 36 were not screened in when using a 3-percent real discount rate;

    c. SAMA 19 involves "procedural guidance for use of cross-tied component cooling or service water pumps" with an estimated cost of $1.75M. Explain/provide more details on the enhancement and the associated cost;

    d. SAMA 54 proceduralizes alignment of the spare diesel to the shutdown board after LOOP and failure of the diesel normally supplying it. The screening criterion refers to SAMA 56 which involves the installation of an additional diesel generator estimated to cost $74.5M. Indicate whether procedures already exist to do this with the spare diesel, and if not, what the estimated cost is;

    e. SAMA 61 involves cross-tying the AC buses, or installing a portable diesel-driven battery charger. The criterion associated with this SAMA suggests that this has been implemented at FNP. Indicate which ability FNP has–the cross-tie or the portable charger;

f. SAMA 66 involves developing procedures to repair or replace failed 4 kV breakers with an estimated cost of $7.15M. Explain/provide more details on the enhancement and the associated cost;

g. SAMA 81 is assigned screening criterion C which means that this SAMA is addressed by another SAMA(s); however, the other SAMA that addresses SGTR coping abilities is not provided. Indicate which SAMA addresses SAMA 81; and

h. SAMA 90, which addresses increased frequency for valve leak testing, is assigned screening criterion C and refers to SAMA 93. SAMA 93 discusses providing leak testing, not increased frequency. Indicate whether increased valve frequency testing would be cost beneficial at FNP.

11. For certain SAMAs considered in the ER, there may be lower cost alternatives that could achieve much of the risk reduction. As one example, SAMA 58 evaluated the use of fuel cells instead of lead-acid batteries. The disposition of this SAMA was N/A, but no explanation was provided. It is noted that a lower cost alternative is available, but was not explored. In this regard please consider and provide estimated costs and benefits for:

    a. adding a small diesel-driven battery charger, unless this capability already exists (see RAI 10e);

    b. installing a direct-drive diesel to power an auxiliary feedwater (AFW) pump as an alternative to a motor-driven pump (SAMA 107);

12. For the remaining 11 (Phase 2) SAMAs, the following information is needed to better understand the modification and/or the modeling assumptions:

    a. the estimated benefit in terms of percent reduction in CDF and person-rem for each of the 11 SAMAs;

    b. the major cost factors that are included and not included in the estimated implementation costs, e.g., the cost of replacement power during extended outages required to implement the modifications, recurring maintenance and surveillance costs, contingency costs associated with unforeseen implementation obstacles, costs associated with procedures, engineering analysis, training, and documentation;

    c. SAMA 11 involves the use of the hydro test pump for reactor coolant pump seal injection. In Section 5.2 of Attachment F, SNC states that the hydro test pump suction isolation valve would need to be replaced with a safety-related motor-operated valve, and that the power supply to the hydro test pump would have to be changed to a class 1E supply. Discuss how much of the estimated implementation cost is attributed to the replacement of the existing valve with a safety-related MOV, and discuss why such a change-out is necessary. Discuss how much of the cost is attributed to the replacement of the existing power supply with a safety-related supply, and why a non class 1E power supply would not be acceptable; and

d. SAMA 24 involves development of procedures for actions on loss of HVAC. In Section 5.3 of Attachment F, the implementation cost is estimated to be $830,000 per unit. The cost appears to be dominated by the installation of temperature sensors in several pump rooms, and control circuits to generate an alarm in the main control room. Describe the existing capabilities to monitor temperatures in these rooms, and why such capabilities would not be sufficient to initiate operator action. Also in that section, SNC discusses a lower cost alternative involving re-labeling the fan trouble alarm annunciator window and revising procedures to instruct operators to take actions to mitigate the loss of HVAC. This low cost alternative does not appear to have evaluated. Please provide the costs and benefits associated with this low cost alternative.

Joseph M. Farley Nuclear Plant

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