Lewis Sumner
Vice President
Hatch Project Support

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

Tel 205.992.7279
Fax 205.992.0341

SOUTHERN COMPANY
Energy to Serve Your World™

HL-5982

August 31, 2000

Docket Nos. 50-321
50-366

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

Edwin I. Hatch Nuclear Plant
Additional Information Related to the Staff's Review of Severe
Accident Mitigation Alternatives (TAC Nos. MA8096 and MA8098)

Gentlemen:

By letter dated May 30, 2000, the NRC requested additional information (RAI) related to the review of severe accident mitigation alternatives for the Edwin I. Hatch Nuclear Plant, Units 1 and 2. Southern Nuclear Operating Company (SNC) responded by letter dated July 26, 2000. Subsequently, the NRC asked for clarification of SNC's response to RAIs 5 and 11 in that response. This letter provides that clarification.

If you have any questions regarding this submittal, please contact J. T. Davis at 205-992-7692.

Respectfully submitted,

Lewis Sumner

H. L. Sumner, Jr.

HLS/JTD

Attachment: Clarification of SNC's Response to SAMA RAIs 5 and 11

cc: Southern Nuclear Operating Company
Mr. P. H. Wells, Nuclear Plant General Manager
Mr. C. R. Pierce, License Renewal Services Manager
SNC Document Management (R-Type A02.001)

U. S. Nuclear Regulatory Commission, Washington, D.C.
Mr. C. I. Grimes, Branch Chief, License Renewal and Standardization Branch
Mr. L. N. Olshan, Project Manager - Hatch
Mr. W. F. Burton, Project Manager - Hatch License Renewal
Mr. J. H. Wilson, Environmental Project Manager – Hatch License Renewal
Ms. Brenda J. Shelton, Chief, Information and Records Management Branch

U. S. Nuclear Regulatory Commission, Region II
Mr. L. A. Reyes, Regional Administrator
Mr. J. T. Munday, Senior Resident Inspector – Hatch

A001

ATTACHMENT

Clarification of SNC's Response to SAMA RAIs 5 and 11

5. **Original RAI - "Please provide a breakdown of leading contributors to dose consequences (e.g., containment bypass, early containment failure, late containment failure, intact containment). Results may be presented in either a table or figure that provides general risk insights-percent contributions to the population dose."**

Verbal Request: The results in the table provided in your July 26, 2000 do not match the application and it appears to have an error in it. Please respond to the differences and correct the table if necessary or provide a justification for the differences

Supplemental Response:
The table in the original RAI response contained inaccuracies in the population dose column. The table below contains the corrected information. The 3.48 person-rem is slightly different than the ER because of the addition of sequences related to late containment failure (sequence 12 & 14), as requested in the RAI.

Release Mode	Sequence	Population Dose (Person-rem)	Contribution (%)
Containment bypass	5 (Loss-of-coolant accident (LOCA) Outside Containment)	0.189	5.44
Early containment failure	2 (SBO), 4 (Loss of containment heat removal (CHR)/Drywell Failure), 11 (Anticipated transient without scram (ATWS) Drywell Failure)	3.18	91.21
Late containment failure	12 (High pressure transient w/loss of CHR), 14 (SBO w/containment isolation failure)	0.113	3.32
Intact containment (venting)	15 (High pressure transient w/Venting)	1.05E-03	0.03
TOTAL		**3.48**	**100**

11. **Original RAI - "The SAMA submittal does not provide sufficient detail about the release sequences to readily determine if the times specified for declaring a general emergency are appropriate. Please provide this information."**

Verbal Request: The table provided in the RAI response did not provide the details about what triggers the declaration for a general emergency. Please provide information on what governs declaration of the general emergency for the sequences described in the table.

Supplemental Information:

This information discusses the declarations of the emergency classifications that can be applied to the five large early release fraction (LERF) accident sequences used in the severe accident mitigation alternative (SAMA) report. Normally, the probabilistic risk assessment (PRA) does not directly address timing of Emergency Classifications. However, for the case of this analysis, the timing of our LERF sequences envelop any declaration of SITE AREA or GENERAL Emergency classification. Probabilistic Safety Analysis (PSA) conditions are so severe that classification and notifications will have been made in advance of vessel failure or containment failure. This is an Emergency Planning function addressed in our Emergency Implementing Procedures (EIPs). The EIPs provide guidance to ensure conservative calls for recommendations to state and local authorities for protection of the general population within our Emergency Planning Zone.

The following information is based on an interpretation of plant procedure for Emergency Classification and Initial Actions. In many cases, the declaration of emergency class is not direct. It is an interpretation of many events and operator decision. The procedure, as was stated, is designed to be conservative in this respect by providing the user with broad-based conditions. These conditions are interpreted directly for the following scenarios because the situations are such that follow up activities are not considered. In brief, the scenario has occurred and there are no recovery actions to consider.

Sequence 2
This is a station blackout sequence with no possibility of obtaining offsite or onsite power within the mission time frame. Vessel failure is at high pressure and there is no containment heat removal available.

A SITE AREA EMERGENCY (SAE) is declared when it is known that all AC power has been lost for longer than 15 minutes and there is no possibility of energizing any 4160VAC Emergency Bus with a diesel generator.

A GENERAL EMERGENCY (GEN) is declared because a shutdown has occurred but no decay heat removal systems are available. In addition, core degradation or melt could occur in about 10 hours with subsequent containment failure.

Sequence 4
A loss of all Plant Service Water has occurred. High Pressure Coolant Injection (HPCI) is inoperable and there is no Residual Heat Removal (RHR) Service Water available. In addition there is no containment vent or heat removal capability. The vessel is depressurized and low pressure injection is available; but, without containment heat removal, Safety Relief Valve (SRV) activity is precluded and the vessel repressurizes to

the mechanical lift setpoint. Low pressure injection is lost due to the repressurization and eventually the containment fails due to lack of heat removal.

A SAE is declared based on containment radiation levels and low reactor water level.

A GEN is declared because a shutdown has occurred but no decay heat removal systems are available. In addition, core degradation or melt could occur in about 10 hours with subsequent containment failure.

Sequence 5
An unisolable leak outside primary containment has occurred. Core damage occurs when low pressure injection sources are finally depleted.

A SAE is declared on an unisolable primary containment failure with subsequent required entry conditions for the Secondary Containment EOP.

A GEN is declared on a Loss of Coolant Accident (LOCA) with containment performance being unsuccessful and affecting long term Emergency Core Cooling System (ECCS) success. This could lead to core degradation or melt in hours.

Sequence 11
An ATWS with Main Steam Isolation Valve (MSIV) closure has occurred. There has also been a failure to inject with Standby Liquid Control. In addition, there is a failure to remove containment heat load. The containment fails on overpressure and injection is subsequently lost.

A SAE is declared on a valid scram signal (automatic and manual) with failure to bring reactor power less than 3% and injection of Standby Liquid Control required.

A GEN is declared based on a transient occurring with failure of a core shutdown system and containment failure likely.

Sequence 15
This sequence is one where no water injection capabilities are available. Core damage and vessel failure occur. The containment is vented, however, for heat removal via the drywell path.

A SAE is declared based on containment radiation alarms and reactor water level being low.

A GEN is declared based on drywell wide range radiation and pressure indicating failure of two fuel boundaries with a third boundary showing potential failure. A high gas content in primary containment requiring venting demonstrates the third boundary potential failure.