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Georgia Power
the southern electric system

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Docket Nos. 50-424
50-425

Tac Nos. M74485
M74486

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

Gentlemen:

VOGTLE ELECTRIC GENERATING PLANT
REVISED RESPONSE TO IPE QUESTION

Enclosed is a revision to the response to IPE Front End Question number 2(b) which was previously transmitted to the NRC with our letter LCV-0636-B, dated September 13, 1995.

Sincerely,

C.K. M'Coy

C. K. McCoy

CKM/HWM

Enclosure

cc: Georgia Power Company
Mr. J. B. Beasley, Jr.
Mr. M. Sheibani
NORMS

U. S. Nuclear Regulatory Commission
Mr. S. D. Ebneter, Regional Administrator
Mr. L. L. Wheeler, Licensing Project Manager, NRR
Mr. C. R. Ogle, Senior Resident Inspector, Vogtle

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Response 1b (continued)

The manual control of the turbine driven AFW pump is applicable to the loss of all AC power event (Station Blackout). Station Blackout, as an event, is a dominant contributor to CDF, therefore this procedural enhancement significantly reduces core damage frequency. The establishment of one NSCW pump operation is applicable to the loss of NSCW initiating event, however, this event has a very low frequency of occurrence and the procedural enhancement has a marginal benefit in reducing core damage frequency. Credit for high temperature Reactor Coolant Pump seals provides additional benefit for this event. The opening of the inverter room doors on a loss of CB ESF HVAC is applicable to all events where one or both trains of Control Building ESF Electrical Room HVAC fail due to component or support system failures. With the loss of Control Building ESF Electrical Room HVAC system, several rooms with important ESF electrical equipment, such as DC buses and panels and 480 V motor control centers reach temperatures that cause electrical equipment failures. This procedural enhancement significantly reduces core damage because the impacted electrical equipment is critical for actuating, controlling, and powering other equipment necessary to mitigate the consequences of all initiating events.

Question 2

According to the IPE submittal, the freeze date of the analysis was January 1, 1991, "with some exceptions." Subsection 2.1 of the submittal states that these exceptions are explicitly cited throughout the report; however, no explicit discussion of these exceptions was found in the submittal. It appears that one of these exceptions is related to the pending installation of new reactor coolant pump (RCP) O-rings in Unit 1 as of the IPE date, as credit for new RCP O-rings was taken in the analysis for both units. The only other possible exception to the analysis freeze date appears to be the procedure enhancements described above in question 1.

a. Please identify and describe all exceptions to the analysis freeze date.

b. If available, describe the impact of the "exceptions" on the CDF, both individually and collectively.

Response 2

a. The freeze date, January 1, 1991, referred to in Subsection 2.1 was the date established for the initial modeling and quantification of the PSA. It was established to provide a baseline date for design and equipment reliability data. After the initial quantification, the recovery process commenced, from which model changes were expected and subsequently implemented. As noted above credit was taken for the RCP high temperature O-rings installed on Unit 2 and scheduled for installation on Unit 1 (they have since been installed). Procedure enhancements were also identified and implemented (see response to 1a above) as a result of the recovery process. Two additional post freeze date items were: 1) the diesel generator reliability data (see IPE report Subsection 1.4.1 and 3.3.2) and 2) essential chilled water reliability data (see Subsection 3.2.2). Both of these systems had benefited from reliability program enhancements. The Independent Review Group recommended that the results of these program enhancements be included in the PSA in order to more accurately reflect the plant as built, operated, and maintained status.

b. The response to question 2a identifies the following exceptions to the freeze date credited in the analysis;

- RCP high temperature O-rings,

- procedural enhancements (see response to 1a),

- diesel generator, and

- essential chilled water reliability data.

Analysis that determines the impact on CDF of old reactor coolant pump (RCP) O-rings is not available. No analysis using old RCP O-rings is available because the decision to credit high temperature RCP O-rings was made early in the IPE development process. This decision was based upon the fact that, at the time of the IPE analysis, high temperature O-rings were already installed on Unit 2 and scheduled for installation on Unit 1. The high temperature O-rings have since been installed on Unit 1.

Procedural enhancements credited in the analysis and the impact of each enhancement on CDF is detailed in the responses to questions 1a and 1b.

To assess the impact of using post freeze date diesel generator and essential chilled water reliability data, sensitivity analyses were performed. Cases were run to assess the impact on CDF of using failure data up to the freeze date for components within the diesel generator and essential chilled water systems. Cases to assure each system individually and a case to collectively assure the combined impact of the systems were run. Table 1 contains the results (impact on CDF) of the sensitivity case for the diesel generator and essential chilled water systems.

The percent decrease in CDF shown in Table 1, is calculated using the following formula:

Percent Decrease in CDF = $(CDF_{Freeze} - CDF_{Reported}) / CDF_{Freeze}$

where, CDF_{Freeze} = CDF calculated using failure data up to the freeze date for components within the diesel generator and essential chilled water systems

$CDF_{Reported}$ = CDF calculated using failure data beyond the freeze date for components within the diesel generator and essential chilled water systems

TABLE 1 CHANGE IN CDF DUE TO APPLICATION OF POST "FREEZE DATE" DG and ECW RELIABILITY DATA		
System	Dominant Contributor	Percent Decrease in CDF
Diesel Generator System	Failure to start DG	19%
Essential Chilled Water System	Failure to start ECW chiller	62%
DG and ECW System	Failure to start ECW system	67%

Question 3

The IPE includes loss of 120 volt AC instrument panels and dc buses as initiating events. However, no mention is made in the submittal of possible consideration of failures of 4,160-Vac and 480-Vac buses as initiating events. Please provide the basis for omitting, as initiating events, equipment failures related to 4,160-Vac and 480-Vac buses.

Response 3

Special initiating events (plant-specific initiating events) are those systems or component failures which result in a reactor trip or LOCA and simultaneously disable or degrade the performance of accident mitigation systems required to respond to the event. These events generally involve the loss of support systems, such as loss of nuclear service cooling water, loss of two 120V vital AC buses or one 125V DC bus, or loss of instrument air. For special initiating events, the loss of a system or component directly results in a reactor trip and the need for decay heat removal.

Because these events are plant specific in nature, a review of plant information such as the plant's design, abnormal operating procedures and operating history was conducted to identify these initiators.

In order to determine whether the loss of a plant system or component should be treated as a special initiating event, several factors were considered:

1. If the event frequency was below the frequency of approximately 1E-08/year, and the expected level of degradation to other plant systems was not significant, then the event was eliminated from further consideration.