December 22, 2000

Mr. Steve Byrne
Vice President, Nuclear Operations
South Carolina Electric & Gas Company
Virgil C. Summer Nuclear Station
Post Office Box 88
Jenkinsville, South Carolina 29065

SUBJECT: VIRGIL C. SUMMER NUCLEAR POWER STATION RE: CRACK IN WELD AREA REACTOR COOLANT SYSTEM (TAC NO. MB0251)

Dear Mr. Byrne:

During the meeting on December 20, 2000, you presented a number of conclusions to us based on your assessment of the root cause and extent of the cracking in the weld in the “A” reactor coolant pipe and the non-destructive testing of the nozzle welds in all 6 coolant pipes. We have determined that we need additional, more detailed, information regarding the data and analyses upon which you based these conclusions. One of these conclusions was “reasonable assurance exists that the other nozzle welds in the reactor coolant system are acceptable, with margin for 2 years, 9 months until Refuel 14.”

In order to understand the bases for your conclusions and to complete our independent assessment prior to plant restart, we request that you provide the information requested in the enclosed list of questions. We have recently received WCAP-15615, “Integrity Evaluation for Future Operation Virgil C. Summer Nuclear Plant: Reactor Vessel Nozzle to Pipe Weld Regions,” and “Crack Growth of Alloy 182 Weld Metal in PWR Environments (PWR MRP-21)”, and we have begun our review of these documents. If you believe these documents provide the information requested in the enclosed questions, you may reference these documents in your response without repeating the information. We request that you submit your response to these questions by December 29, 2000. If you have any questions regarding the enclosure, contact us as soon as possible to obtain clarification. If you can not submit some part of this information by December 29, 2000, contact us as soon as possible to inform us when you can provide the information.

Sincerely,

***/RA/***

Karen R. Cotton, Project Manager, Section 1
Project Directorate II
Division of Project Licensing Management
Office of Nuclear Reactor Regulation

Docket No. 50-395

Enclosure: Request for Additional Information

cc w/encls: See next page

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Vice President, Nuclear Operations
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South Carolina Electric & Gas Company

cc:

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**Questions for V. C. Summer**

1. Discuss your plans for re-inspecting the nozzle welds, including the method(s) and scope of the inspections and the schedule for such inspections.

2. Discuss the basis for your statement that the frequency and extent of cracking at Ringhals is more severe than that at V. C. Summer because of the double-V weld configuration.

3. Provide data and appropriate figures that characterizes all indications found in all nozzle welds, including lengths (and depths, as available).

4. For all indications found in all nozzle welds, discuss how the ASME Section XI proximity rules combine these indications.

5. Provide the bounding circumferential and axial cracks for each nozzle weld, and the technical justification supporting these determinations, including the ASME Code methodology used to evaluate these flaws.

6. Provide your technical basis for the crack growth rate assumed in your determination that the nozzle welds are safe to operate for an additional cycle. Include your basis for the stress intensity (K) dependency and applications to various weld geometries. In particular, include a discussion of the finite element analysis model utilized.

   a. If a limit load analysis was performed, provide the Z factors utilized.
   b. If a LEFM/EPFM analysis was performed, provide the material properties used.

7. Provide the technical justification for your statement that the residual stresses in the hoop direction are greater than those in the axial direction, and quantify these differences.

   a. Provide all operating stresses in both the circumferential and axial directions in all nozzle welds.

   b. Provide a comparison of the expected axial and circumferential residual stress distributions in the "A" loop nozzle weld which underwent extensive repairs to the axial and circumferential residual stress distributions in the other nozzle welds without extensive repairs. Make this comparison as quantitative as possible based on existing information and provide the bases for the expected stress distributions and a description of the models or testing used to develop them.

   c. Provide a discussion of any available data regarding the threshold temperatures and stresses for PWSCC in Alloy 182/82. Using these threshold values discuss the susceptibilities of the "A" loop nozzle weld which cracked to the other nozzle welds.

   d. Using your assumed stress distributions and crack growth assumptions, provide your calculated estimate of (1) the time it took for the axial through-wall crack in the "A" hot leg to grow through-wall and (2) the time it would take for an assumed axial flaw in one of the other hot legs to propagate through wall after initiation.

8. Discuss your plans for enhancing your leakage detection capabilities, including trending and revised action levels. Describe administrative procedures you have in place or plan to put in place to provide operators guidance on how to monitor for and respond to potential unidentified reactor coolant system leakage. Include the types and frequencies of monitoring methods as well as action levels and responses.

9. Identify all ASME Class I components at V. C. Summer with Alloy 82/182/Inconel welds. Include a discussion of the safety consequences if these welds develop cracks similar to that seen on the "A" hot leg nozzle weld. Address your plans for inspection of these wells.

10. Since leak before break (LBB) is approved for your RCS main loop piping, and PWSCC has been identified as a cracking mechanism for the Alloy 82/182 welds, discuss your plans to mitigate this cracking mechanism such that your LBB analysis remains valid.