Docket File

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

July 11, 1985

Docket No. 50-395

Mr. O. W. Dixon, Jr., Vice President
Nuclear Operations
South Carolina Electric and Gas Company
Post Office Box 764
Columbia, South Carolina 29218

Dear Mr. Dixon:

Subject: Virgil C. Summer Nuclear Station Facility Operating License NPF-12, NUREG-0737, Item II.D.1

References: (1) Letter from T. Nichols, Jr. (SCE&G) to H. Denton (NRC) dated 4/1/82
(2) Letter from O. Dixon, Jr. (SCE&G) to H. Denton (NRC) dated 7/30/82
(3) Letter from O. Dixon, Jr. (SCE&G) to H. Denton (NRC) dated 7/30/82

Along with our consultants (EG&G, Idaho), we have reviewed your submittals pertaining to TMI Item II.D.1 of NUREG-0737, "Performance Testing of Relief and Safety Valves." From this, we find that we need additional information in order to complete our review (see enclosure).

We request that you contact the project manager assigned to your plant in order to negotiate an acceptable schedule for submittal of this information.

This request for additional information was approved by the Office of Management and Budget under clearance number 3150-0011 which expires September 30, 1985. Comments on burden and duplication may be directed to the Office of Management and Budget, Reports Management Room 3208, New Executive Office Building, Washington, D.C. 20503.

Sincerely,

DESIGNATED ORIGINAL
Certified By Angela Henry

Darl Hood
for Elinor G. Adensam, Chief
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Enclosure:
As stated

cc: See next page

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SAFETY EVALUATION QUESTIONS TMI ACTION NUREG-0737 II.D.1 FOR V. C. SUMMER UNIT 1

Questions related to selection of transients and inlet fluid conditions:

1. The Westinghouse valve inlet fluid conditions report stated that liquid discharge through both the safety and Power Operated Relief Valves (PORVs) is predicted for a FSAR feedline break event. The Westinghouse report gave expected peak pressure, pressurization rate, and fluid temperature range for an FSAR feedline break at the V. C. Summer Plant. The V. C. Summer Plant specific submittal, however, does not address this event. NUREG-0737 requires analysis of accidents and occurrences referenced in Regulatory Guide 1.70, Revision 2, and one of the accidents so required is the feedline break. Therefore, assure that the fluid conditions for this were enveloped in the EPRI tests and that the time period of water relief in the EPRI tests was as long as expected at the plant. Demonstrate operability of the safety valves and PORVs for this event and assure that the feedline break event was considered in analyses of the piping system.

2. Results from the EPRI tests on the Crosby safety valves indicate that the test blowdowns exceeded the design value of 5% for both "as installed" and "lowered" ring settings. If the blowdowns expected for the plant (see Question 4) also exceed 5%, the higher blowdowns could cause a rise in pressurizer water level such that water may reach the safety valve inlet line and result in a steam-water flow situation. Also the pressure might be sufficiently decreased such that flashing occurs in the primary loop or the reactor vessel, natural circulation is interrupted, and adequate cooling for decay heat removal is not achieved. Discuss these consequences of higher blowdowns if increased blowdowns are expected.

Questions related to valve operability:

3. The submittal does not identify the ring settings to be used on the Crosby 6M6 safety valves or what effect these settings have on valve performance in the V. C. Summer installation. Provide the final ring settings selected for the V. C. Summer safety valves. Identify the expected blowdowns corresponding to these plant ring settings and explain how the blowdowns were extrapolated or calculated from test data. Verify that at these ring settings the valves can perform their pressure relief function and the plant can be safely shutdown with the blowdown and fluid conditions occurring at the plant.

4. Results from EPRI tests on the Crosby 6M6 safety valve with loop seal internals show that during some tests the valve attained rated lift and rated flow at 3% accumulation while during other tests it did not. Provide a demonstration that the plant safety valves will pass their rated flow with the ring settings used.

5. During two EPRI hot loop seal-steam tests and one subcooled water test on the 6M6 safety valve, the valve fluttered and chattered upon closure. These tests were terminated by manually opening the valve to stop the chatter. The hot loop seal tests appear to be representative of conditions at the V. C. Summer plant and the liquid flow tests may be representative of a feedline break event (see Question 1). Justify that the valve behavior exhibited in these tests is not indicative of the performance expected for the V. C. Summer valves.

6. NUREG-0737, Item II.D.1 requires that the plant-specific PORV control circuitry be qualified for design-basis transients and accidents. Please provide information which demonstrates that this requirement has been fulfilled.

7. Bending moments are induced on the safety valves and PORVs during the time they are required to operate because of discharge loads

and thermal expansion of the pressurizer tank and inlet and outlet piping. Make a comparison between the predicted plant moments with the moments applied to the tested valves to demonstrate that the operability of the valves will not be impaired.

8. As part of comparing inlet piping configurations of the plant safety valves and the test valves, a comparison between the two inlet piping pressure drops should be made. Provide a numerical comparison between a calculated plant pressure drop and the test pressure drop. Explain how the plant pressure drop was calculated.

9. The submittal states that backpressures at the safety valves were analyzed for steady state steam discharge from all three safety valves and were shown to be less than 500 psig. It does not, however, identify the expected backpressure for loop seal discharge from the safety valves. Provide this value for expected backpressure and assure that it was enveloped in the EPRI hot loop seal discharge tests.

<u>Questions related to the thermal hydraulic analysis</u>:

10. The submittal does not present details of the thermal hydraulic analysis. Provide a report or other documentation that contains at least the following information: For the analysis involving discharge of saturated steam with a 380°F loop seal through the safety valves, identify parameters used such as peak pressure, pressurization rate, valve opening pop time, and time step. Provide rationale for the values used. Explain how many volumes were used in pipe segments of the thermal hydraulic model. Provide a copy of the computer printout from the RELAP5 analysis of the loop seal/steam discharge through the safety valves.

11. The submittal presents the loop seal temperature distribution that was used as input to the RELAP5 analysis, but does not explain how the simmering of the loop seal water through the

safety valve was simulated in the RELAP5 calculations. Explain how the valve flow area was varied in the analysis as water passed through the valve and how long the simmering process lasted before the valve popped open. Specify the resulting water flow rate and explain why this was deemed to be appropriate.

12. The submittal states that the thermal hydraulic analysis was performed using RELAP5/MOD1 and that the RELAP5 control system was used to calculate the fluid forces. Identify the methodology used to calculate forces from RELAP5/MOD1 and provide additional verification that the methodology produces accurate force histories for similar problems.

13. Report the flow rates through the safety valves and PORVs that were assumed in the thermal hydraulic analysis. Because the ASME Code requires derating of the safety valves to 90% of actual flow capacity, the safety valve analysis should be based on a flow rate of at least 111% of the flow rating of the valve, unless another flow rate can be justified. Provide information explaining how derating of the safety valves was handled.

<u>Questions related to structural analysis</u>:

14. The submittal does not present details of the structural analysis. Provide a report or other documentation that contains at least the following information: For the analysis involving discharge of saturated steam with a 380°F loop seal through the safety valves, identify the time step used in the forcing function time histories, the time step used in the integration solution, damping valves used, the cutoff frequency if modal superposition was used, and the spacing between lumped masses in the structural model. Provide rationale for the values used. Explain how the connections to the pressurizer and relief tanks were treated in the structural model. Identify the manufacturer and model numbers of snubbers used to support the safety valve piping (down to the relief tank) and specify the stiffnesses used

in the model to represent the snubbers. Provide a copy of the computer printout from the TPIPE and TMRPIPE analyses of the loop seal/steam discharge through the safety valves. Also, provide clear readable as-built drawings of the piping configuration from the pressurizer to the relief valve showing dimensions, pipe sizes and locations of pipe supports and snubbers.

15. The submittal states that the structural analysis on the piping system was performed using the TPIPE and TMRPIPE computer codes. It further states that these programs have had application on numerous projects in the nuclear industry. Provide verification that these programs have produced accurate results for problems similar to a valve actuation in the safety valve/PORV piping system. Explain whether the dynamic piping response was obtained using the direction integration, modal superposition, or other solution technique.

16. The submittal states that pressure oscillations in the safety valve inlet piping were reported by EPRI for some fluid conditions and upstream piping configurations. According to EPRI results these oscillations commonly occurred during passage of loop seal water and were in the 170-260 Hz frequency range. The submittal states that the oscillations have been evaluated and that stresses are within code allowable for the V. C. Summer plant. It is not clear though whether this evaluation reflects the fact that the pressure oscillations could excite high frequency vibration modes in the piping causing significant bending moments in the inlet piping. Show that the bending moments caused by this dynamic response do not exceed the allowable bending moment. Provide the referenced report "Pressure Oscillations in Safety Valve Inlet Piping", EPRI, March 17, 1982.

17. The submittal presents the load combinations that were considered in the piping analysis. The combinations listed consider all those that are recommended in the report <u>EPRI PWR Safety and</u>

Relief Valve Test Program Guide for Application of Valve Test Program Results to Plant-Specific Evaluations except for an upset condition in the Class 1 piping in which PORV discharge transient, OBE, and normal loads are combined. Provide justification for not considering this load combination in the analysis.

18. The submittal states that piping stresses and support loads in both the upstream and downstream portions of the safety valve piping system are acceptable. It also states that piping stresses and support loads in the PORV system are acceptable. Provide a numerical comparison between the calculated and allowable stresses for the piping and supports for these systems to verify this conclusion. Also, identify the codes and standards from which the allowable piping stresses and support loads were obtained.