Stephen A. Byrne
Vice President, Nuclear Operations
803.345.4622

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A SCANA COMPANY

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RC-01-0060

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Gentlemen:

Subject: VIRGIL C. SUMMER NUCLEAR STATION
DOCKET NO. 50/395
OPERATING LICENSE NO. NPF-12
FINAL REPORT PURSUANT TO 10 CFR PART 21
(SSH 2000-001)

South Carolina Electric & Gas (SCE&G) submits this letter in accordance with the requirements of 10 CFR 21.21(a)(2) as a final report of an identified defect which was potentially associated with a substantial safety hazard.

On 10/12/00, at 0630 hours, visual inspection revealed small amounts of boron buildup on the weld between the vessel nozzle and the hot leg pipe. Within hours, the suspect area was cleaned and a dye penetrant (PT) examination of the pipe identified a 4 inch indication at the weld between the hot leg piping and the reactor vessel nozzle. The weld is located approximately 3 feet from the vessel in pipe near the nozzle. The indication was located about 17" from the top of the pipe. This pipe has a nominal inside diameter of 29 inches and is approximately 2.5 inches thick.

Subsequent ultrasonic examination from the inside diameter identified an axial flaw less than 3 inches long. The same examination determined that the original indication was not the source of the leak.

The interim 10 CFR Part 21 report (RC-00-0368) was submitted to allow additional evaluation and cause determination.

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SCE&G Engineering review of the Root Cause Evaluation Report for potential 10 CFR Part 21 concerns, has determined that a Significant Safety Hazard did not exist. Attached is the engineering justification for this position.

Should you have any questions, please call Mr. Jeffrey Pease at (803) 345-4124.

Very truly yours,

Stephen A. Byrne

JWP/SAB/dr
Attachment

c: N. O. Lorick
N. S. Carns
T. G. Eppink
R. J. White
L. A. Reyes
K. R. Cotton
J. B. Knotts, Jr.
NRC Resident Inspector
G. G. Williams
NSRC
RTS (SSH 2000-001)
File (818.18)
DMS (RC-00-0060)

ENGINEERS	Serial:	CR14992	
	Engineer:	Rice, C	
TECHNICAL WORK RECORD	Date:	3-12-01	
Project Title NCN 00-1396 10CFR50.21 Evaluation	Tab 3.28	Page 1	of 2

DESCRIPTION OF CONDITION:

One circumferential flaw and several axial flaws were indicated in the "A" loop Hot Leg to reactor vessel nozzle weld. The circumferential flaw was contained within the cladding under the carbon steel nozzle, and its propagation in depth was arrested when it reached the ferritic material. One end of this flaw turned to the axial direction. One axial crack was through-wall. This crack was contained within the weld, and was approximately 2.5 inches long on the ID of the pipe, and extended to a small weep-hole on the OD of the pipe. Leakage was estimated to by 0.3 gpm.

For more complete descriptions of these flaws refer to Root Cause Report C-00-1392 "V. C. Summer Nuclear Station Root Cause Investigation "A" Hot Leg Nozzle Weld Cracks" and WCAP-15615 "Integrity Evaluation of Future Operation Virgil C. Summer Nuclear Plant: Reactor Vessel Nozzle to Pipe Weld Regions.

The primary root cause is attributed to extensive weld repairs on the nozzle to pipe weld which created high residual stresses in a material and in an environment known to cause Primary Water Stress Corrosion cracking (PWSCC). A secondary root cause is attributed to the fact that neither codes, standards, or the welding process recognized or required consideration of the cumulative effect of multiple repair welds and weld grinding in the creation of high residual stresses.

FACTS:

The cracking within the subject weld is axially oriented, and limited in length to the Alloy 182 weld, which has a length of about 2.5 inches. The likelihood of a longer crack occurring is vanishingly small, because the cracking cannot extend into either the carbon steel or the stainless steel by a stress corrosion mechanism.

A fracture assessment summarized in WCAP-15615 has shown that a very large, through-wall flaw would be required to cause failure in the hot leg piping and welds. The critical axial flaw sizes calculated based on plastic limit load methodology are 26 inches for the stainless steel pipe and 35 inches for Alloy 182/82 weld material.

WCAP-15665 also presents the results of leak rate calculations as a function of crack length in both Alloy 182 and Type 304N stainless steel. Under normal operating loads an axial through-wall crack 2.2 inches long in Alloy 182 would be expected to leak at a rate of 1.0 gpm, while an axial through-wall crack 2.5 inches long in Alloy 182 would be expected to leak at a rate of approximately 1.5 gpm.

EVALUATION:

1. The axial through wall crack in the "A" hot leg to reactor vessel nozzle weld is a hardware deficiency in a basic component.
    a. This deficiency is a defect in the in the manufacturing/installation of this weld which is part of the Reactor Coolant Pressure Boundary because it has been primarily attributed to the extensive weld repairs and grinding which occurred during the installation of the Hot Leg.

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b. This deficiency is not considered to be a design deficiency because the high residual stresses were due to the repair effort, which was outside the original design considerations.
c. This deficiency is considered to be a result of improper guidance provided by maintenance instructions because the second root cause has been attributed to the fact that neither codes, standards nor the welding process recognized or required consideration of multiple weld repairs in creating high residual stresses.

2. Could the deficiency cause a loss of safety function to the extent that there is a major reduction in the degree of protection provided to public health and safety?

   This condition does not create a Substantial Safety Hazard for the following reasons:

   a. The safety function of this weld is to maintain the Reactor Coolant System Pressure Boundary so that the reactor core can be maintained in safe (cool) condition, and so that dose rates remain within normal levels.

      Axial cracks in the weld are limited to the 2.5 inch length of the weld. This is considerably shorter than the crack length that would be required for structural failure of the weld or associated piping. Also, the expected leak rate through such small cracks is approximately 1.5 gpm which is well within the capacity of normal make-up systems. In fact, the estimated leakage from the axial, through-wall crack experienced at VCS was only 0.3 gpm. Therefore, neither the crack that was found nor any other cracks that could be reasonably postulated to occur in these welds, could cause a condition whereby the structural integrity of the Reactor Coolant System, or its core cooling function would have been challenged.

   b. The subject welds are all located inside the reactor building. Any increased radiological dosage attributable to leakage through cracks in reactor vessel nozzle welds would be contained inside the reactor building. There would be no radioactive material released to an unrestricted area, nor would any personnel in an restricted area receive an increased dose due to this condition.

   c. The VC Summer Technical Specifications contain safety limits regarding thermal power, RCS temperature and pressure, pressurizer pressure and Reactor Trip System instrumentation setpoints. This condition involving cracked reactor nozzle welds could not result in a situation were these parameters would be outside the safety limits specified in the Technical Specifications.

CONCLUSION:

The described condition, involving a PWSCC induced, through-wall crack in a Reactor Vessel Hot Leg Nozzle Weld does not constitute a substantial safety hazard. Therefore this condition is not reportable under 10CFR50.21