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SOUTH CAROLINA ELECTRIC & GAS COMPANY
POST OFFICE BOX 764
COLUMBIA, SOUTH CAROLINA 29218

E. H. CREWS, JR.
VICE-PRESIDENT AND GROUP EXECUTIVE
ENGINEERING AND CONSTRUCTION

September 20, 1979

Mr. James P. O'Reilly, Director
U. S. Nuclear Regulatory Commission
Region II, Suite 1217
230 Peachtree Street, N. W.
Atlanta, Georgia 30303

Subject: Virgil C. Summer Nuclear Station
Inspection & Enforcement Bulletin 79-13
Revision 1 - Docket No. 50-395
Nuclear Engineering File 2.8950

Dear Mr. O'Reilly:

NRC I&E Bulletin 79-13, Revision 1 regarding cracking in feedwater system piping required action by South Carolina Electric & Gas Company within twenty (20) days. This letter provides information in response to this bulletin as it applies to the Virgil C. Summer Nuclear Station.

As a designated applicant for an operating license item 3 of the bulletin applies to the Virgil C. Summer Nuclear Station and requires submittal of reports described in items 4, 5 and 6 of the bulletin. At this time only 5 requires action.

(1) Item 5(a) - Our projected schedule for hot functional testing is February 1, 1980 and the projected schedule for fuel loading is July 1, 1980. Inspection required by item 1 will be performed between these two dates.

In lieu of the inspections prescribed in item 1 of the bulletin, SCE&G proposes the alternate inspection program as follows:

(a) Following hot functional testing and prior to fuel load, an ultrasonic examination of all feedwater system welds and adjacent pipe and nozzle areas inside containment, shall be performed on all feedwater nozzle-to-pipe welds inside containment.

(b) Visual inspection of all feedwater system pipe supports and snubbers inside containment shall be performed to verify operability and conformance to design.

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(c) During the first refueling outage, ultrasonic examinations of all feedwater system piping welds inside containment, radiographic examination of all feedwater system nozzle-to-pipe welds inside containment and visual inspection of all feedwater system pipe supports and snubbers inside containment shall be performed. The results of these examinations shall be compared to the results of the examinations discussed in B.1 for determinations of any degradation of feedwater system piping welds and to verify operability of feedwater pipe supports and snubbers.

(d) Any indications of cracking or other unacceptable Code discontinuities shall be reported in accordance with the requirements of Section 4 of IE Bulletin 79-13, Rev. 1.

(e) A written report shall be submitted in accordance with Section 6, of IE Bulletin 79-13, Revision 1.

(f) Future examinations of feedwater system piping shall be based on the results of the examination performed during the first refueling outage. If no indication of cracking is discovered, it is assumed that the examinations required by the ASME B&PV Code Section XI Inservice Inspection program will suffice.

Our reasons for proposing this alternate inspection program are:

(a) The physical location, size and installation of the feedwater piping will not allow for 100% radiographic examination of the feedwater system piping welds.

(b) Radiographic examination will require a minimum of 45 days with the system void of water. This will increase corrosion, both in the feedwater system and the steam generators and result in increased down time of the plant.

(c) Decreased radiation exposure to workers.

(d) With an ultrasonic examination as a baseline for comparison, ultrasonic examination of the feedwater system would provide sufficient information to determine whether or not cracking had occurred in the feedwater water system.

(2) Item 5(b) and (c) - Operating procedures are presently being finalized and are not complete at this time. Emergency Operating Procedure (EOP-2) for a main steam line break and a main feedwater line break will have adequate coverage for symptoms of a feedwater line break inside the reactor building so that the operator can readily recognize the accident and take immediate corrective action to limit the accident by isolating the affected feedwater line and steam generator.

For feedwater leaks inside the reactor building the detection system for the reactor coolant system, as listed in FSAR Section 5.2.7 is also used for detection of feedwater and steam leaks. An increased condensate drain flow in the reactor building cooling unit or a leak detection sump level increase, without an increase in radioactivity in the reactor building environment, would indicate the possible presence of a feedwater or steam leak. Followup reactor building entry and investigation should locate the source.

Presently, SCE&G is evaluating an acoustical monitoring system by Westinghouse which may be utilized to detect and locate feedwater leaks inside the reactor building. If evaluated favorably, this system may be placed in service at a later date.

If further information regarding this bulletin is required, please let us know.

Very truly yours,

E. H. Crews, Jr.

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cc: Office of Inspection & Enforcement
Washington, D. C.

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