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U.S. Nuclear Regulatory Commission
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Subject: Virgil C. Summer Nuclear Station
Docket No. 50/395
Operating License No. NPF-12
Response to Generic Letter 88-05

Gentlemen:

Attached is the South Carolina Electric & Gas Company response to Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," which is being provided pursuant to 10CFR50.54(f).

The statements and matters set forth in this submittal are true and correct to the best of my knowledge, information, and belief.

Very truly yours,

D. A. Nauman

MGC/DAN:lcd
Attachment

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The following are the South Carolina Electric & Gas Company (SCE&G) responses to the four programmatic suggestions contained in Generic Letter 88-05:

SUGGESTION 1

A determination of the principal locations where leaks that are smaller than the allowable technical specification limit can cause degradation of the primary pressure boundary by boric acid corrosion. Particular consideration should be given to identifying those locations where conditions exist that could cause high concentrations of boric acid on pressure boundary surfaces.

RESPONSE:

A detailed evaluation of the location of carbon steel reactor coolant pressure boundary components and their vulnerability to boric acid corrosion is being performed. This evaluation is expected to be complete by September 30, 1988.

SUGGESTION 2

Procedures for locating small coolant leaks (i.e., leakage rates at less than technical specification limits). It is important to establish the potential path of the leaking coolant and the reactor pressure boundary components it is likely to contact. This information is important in determining the interaction between the leaking coolant and reactor coolant pressure boundary materials.

RESPONSE:

Manual and motor operated valves of the reactor coolant system (RCS) which are three inches and larger are provided with double-packed stuffing boxes and intermediate lantern ring leakoff connections. Further, control valves, regardless of size, are provided with double-packed stuffing boxes and with stem leakoff connections. The leakoff connections on valves that normally operate in a radioactive fluid, are piped to a closed liquid waste system. Leakage to the containment atmosphere is essentially zero for these valves.

To complement this design, SCE&G has procedures for the inspection of various primary systems. The following surveillance test procedures (STPs) are performed either after maintenance which require the opening and closeout of the system or after each refueling outage prior to criticality:

STP-150.001 Reactor Coolant System Leak Test
STP-150.002 Chemical Volume Control System Leak Test
STP-150.003 Safety Injection System Leak Test
STP-150.004 Residual Heat Removal System Leak Test

These procedures outline a walkdown of the respective systems, each of which has two specific sections for boric acid residue inspection. The first has the inspector note any boric acid residue or discoloration on any valves, piping or other components. The second requires the inspection of joints, floors, underneath systems, and surrounding areas for evidence of leakage -- if access to these components is unavailable due to insulation.

At the beginning of refueling outages, inspection teams made up of operations and maintenance personnel make entries prior to decontamination to survey the reactor building. During these inspections, Maintenance Work Requests (MWR) are written to identify and correct leakage problems. After the fourth refueling outage, the evaluation of carbon steel reactor pressure boundary components will be used to supplement or generate the procedures for the visual inspection of the RCS pressure boundary components.

Additionally, General Maintenance Procedure (GMP) 100.017, "Controlling the Breach of System Integrity," specifically references IEB 82-02, "Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants," and requires a special quality control inspection of threaded fasteners when disassembling components or flanges in ASME Class 1 piping. STP-109.001, "Reactor Building Closeout Inspection," requires a thorough visual inspection of the reactor building for general cleanliness prior to establishing containment integrity.

SUGGESTION 3

Methods for conducting examinations and performing engineering evaluations to establish the impact on the reactor coolant pressure boundary when leakage is located. This should include procedures to promptly gather the necessary information for an engineering evaluation before the removal of evidence of leakage, such as boric acid crystal buildup.

RESPONSE:

Currently SCE&G uses the Non-Conformance Notice (NCN) program to perform evaluations to establish the impact of corrosion on safety-related components once corrosion has been identified. Normally, any components with boric acid leakage would be identified at the beginning of the outage by operations and maintenance personnel. During the course of repair, if corrosion is present, an NCN would be written for engineering evaluation.

As an aid to engineering personnel involved in the dispositioning of NCNs written due to boric acid corrosion, Design Engineering will develop guidelines for the examination and evaluation of boric acid build-up on safety-related Class 1 primary pressure boundary components.

Training for quality control, operations, mechanical maintenance, and engineering personnel involved in identifying and evaluating boric acid corrosion problems will be scheduled, and a discussion of this issue will be included in the Program Content and Criteria Training program for appropriate station personnel.

SUGGESTION 4

Corrective actions to prevent recurrences of this type of corrosion. This should include any modifications to be introduced in the present design or operating procedures of the plant that (a) reduce the probability of primary coolant leaks at the locations where they may cause corrosion damage and (b) entail the use of suitable corrosion resistant materials or the application of protective coatings/claddings.

RESPONSE:

Programs currently in place have been responsible for the remedial corrective actions which are in various stages of implementation.

These include:

A. Conoseal Modification - Industry operating experience with conoseals has shown recurring problems with leakage. Most of the problems have occurred during post refueling outage start-up.

The current design has an upper and lower seal arrangement with the bottom seal requiring a carbon steel clamp to be torqued in place. Not only is the reconnection procedure lengthy, resulting in ALARA concerns, but if a leak occurs, the procedure requires system depressurization and a lengthy rework to stop the leak.

As a result, a modification of the conoseals is being planned. This modification should simplify removal and re-connection and reduce the possibility of leakage.

B. Fitting Seal Welding - Various plants have reported problems with the fittings used to seal the incore thermocouple extension wiring at the top of the conoseal assembly on the reactor vessel head. SCE&G has experienced leakage problems after reconnecting the fittings following the second and third refueling outages. The leakage was detected during the maintenance and operations walkdown inspections and was corrected prior to startup. As a result, modification alternatives are being explored to affect a more reliable seal.

C. Resistance Temperature Devices (RTD) Loop By-Pass Removal - The RTD loop by-pass have been a source of leakage problems in the industry. Currently an engineering review is in progress to evaluate removal of the by-pass loops.

D. Valve Packing Program - A maintenance valve packing program is in the process of implementation. This program is to "live-load" the packing and reduce packing leakage by maintaining uniform packing compression for extended time periods.

During the third refueling outage, 45% of the RCS Class 1 valves which require packing were re-packed and 26% were live-loaded.

Scheduled for the fourth refueling outage, valves which were live-loaded will be examined and data taken based on the amount of live-load spring decompression which has resulted during the fuel cycle. This data will be trended and evaluated to develop a regimented packing replacement preventive maintenance schedule based on the packing life expectancy of each valve.