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VICE PRESIDENT
NUCLEAR OPERATIONS

November 1, 1982

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Virgil C. Summer Nuclear Station
Docket No. 50/395
Operating License No. NPF-12
Independent Design Verification

Dear Mr. Denton:

In response to our discussions in a meeting with the staff on October 28, 1982, we wish to provide additional information concerning two findings made in the Stone and Webster (SWEC) final report. Our letter dated October 12, 1982, on this same subject stated that the damping information used in piping analysis would be included for future reference purposes in an appropriate document. It has been decided that this information is to be included in the next revision of "Piping Engineering Section - Nuclear Criteria for Piping Stress Analysis and Pipe Support Design." The next scheduled revision of this document is expected to be complete by January 1, 1983. Piping analyses for the Virgil C. Summer Nuclear Station are complete and no reanalyses are expected. The damping information being included in this design criteria, as indicated in the October 12, 1982 response, meets or is more conservative than the information in Table 3.7-1 of the FSAR.

The second finding made in the design control audit portion of the SWEC report was in regard to a lack of totally complete packages or a tendency toward informality in the analysis documentation packages. SCE&G's audit program recognized these informalities prior to the beginning of the independent design verification effort. Examples of the types of problems addressed in our audit findings are:

1. In following the documentation trail for closeout of RFI's, analysis package EF-03 contained a telecon memo with no letterhead addressed to an individual (first name only) and signed by "Fred." The telecon memo was confirming that the loads for supports EFH-170 and EFH-171 are to be doubled. Subsequent review and follow up investigation found that the information was included in the designs for these supports.

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2. Analysis package CC-03 contained an EDS internal memo concerning the global coordinate load for support CCH-765. The memo indicated that GAI questioned if the load could be a global X direction only. The item was resolved by Rev. 3 of the support drawing. According to DS-8, this item should have shown up as a comment in section 1.6 of the package verification.

3. Analysis package MS-15 indicates that support MSH-180 has been voided. The support had, in fact, been removed but the computer listing of supports indicated the support as still valid.

4. An analysis review comment sheet was noted to be in the wrong section of the verification package for CC-03.

5. Terminal end movements of the anchor at node 142 (interface of analysis codes CC-08 and CC-09) were not shown on the isometric drawing. Subsequent review of the analysis indicated that the appropriate information had been used.

6. Review of analysis package CC-09 indicted that the weight of the flanges for flow element FE-7132 was 98 pounds. This was obviously not correct because the flanges are on a one (1) inch line. However, review of the analysis inputs showed that 30 pounds was used. Subsequent review verified that the 30 pounds was an acceptable and conservative design input. The analysis input had been changed based on later information. However, there was no documented auditable link to explain the weight change.

The ultimate conclusion of SCE&G's audit program has been that, although administrative inconsistencies/informalities have been found to exist in documentation, these situations have consistently been found to be non-safety significant. As indicated in our October 12, 1982 letter, SCE&G has been working to improve the quality and completeness of these packages. This is a continuing activity whose satisfactory completion is scheduled by the end of the year. If additional problems are discovered in future audits, it is expected they will be of similar character (non-safety significant). Any finding which is determined to be safety significant will be reported to the NRC within seven (7) days.

If you require additional information, please advise.

Very truly yours,

O. W. Dixon, Jr.

GM:OWD/fjc

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