PSE&G

Public Service Electric and Gas Company P.O. Box 236 Hancocks Bridge, New Jersey 08038

Nuclear Department

October 13, 1983

Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, MD 20014

Attention: Mr. Steven A. Varga, Chief
Operating Reactors Branch 1
Division of Licensing

Dear Mr. Varga:

CONTAINMENT INTEGRATED LEAK RATE TEST
NO. 2 UNIT
SALEM GENERATING STATION
DOCKET NO. 50-311

Attached as Enclosure 1 is a report on the first inservice Containment Integrated Leak Rate (Type A) Test for Salem Unit No. 2. The test was performed during the 24 hour period ending May 23, 1983, during the first refueling outage in compliance with Appendix J of 10 CFR 50 and Plant Technical Specification 4.6.1.2. The conclusion of the report is that the containment leakage rate, measured as .048%/day of the containment volume, with a 95% upper confidence level of .054%/day, meets the acceptance criteria of Appendix J and the Technical Specification.

Included in the report is a summary of local leak rate (Type B & C) tests conducted immediately prior to the first inservice Type A test. This summary includes "as found" data for informational purposes.

One condition of the Unit No. 2 inservice testing was not exactly the same as the original preoperational test condition, in that some liner plate monitor channels were plugged shut during the inservice test. During a preliminary review of the results of the Type A test, USNRC Region I Inspector, Mr. Sada Pulani, suggested that PSE&G provide a justification for the



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difference in test conditions. We, therefore, delayed submittal of the report to incorporate our justification.

General Description of Reactor Containment Building

The reactor containment structure is described in Section 3.8 of the Salem Generating Station Updated Final Safety Analysis Report (UFSAR). Included in Section 3.8.1.6.8.6 of the UFSAR is a description of the inspection and testing (which included ultrasonic, vacuum box, strength and leak tests) that took place during erection of the containment liner. All documentation of these tests and results, as well as mill test reports on the materials is on file as microfiche in the Salem Technical Document Room.

Background

The liner plate leak detection system for Salem Generating Station originally consisted of a channel over each containment liner plate weld, connected via 1/4 inch pipe to various manifolds located in the containment. Utilizing station air, each manifold could pressurize a zone of monitor channels to test for liner plate weld leakage. However, this concept was abandoned when 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Watercooled Power Reactors" was published. Since there was no longer a need to test liner plate welds separately, the monitor channel system was abandoned in place. At the time it was decided to abandon this system, all the monitor channels throughout the entire containment had been installed and tested and all the embedded portions of the 1/4 inch connecting piping had been installed. However, only 30% of the cylinder wall piping and none (0%) of the dome piping had been installed.

At roughly the same time that the system was abandoned, Salem Unit 2's completion date was set back approximately two years. After Unit 2's completion, there was a delay in the issuance of the operating license due to the events at Three Mile Island. Concern about the possibility of condensation and corrosion taking place within the monitor channels while the plant was inoperative prompted the implementation of a procedure to purge the oxygen laden air from the space enclosed by the monitor channel and liner plate with nitrogen. After the residual oxygen in this volume was reduced to 5% or less, as determined

by an oxygen analyzer, the threaded sleeve couplings were plugged, thus leaving the monitor channels inerted with nitrogen. These plugs were removed for the structural integrity test and initial type A test and after completion of the tests, the channels were purged again with nitrogen and the plugs reinstalled. After the seal plugs were installed, access scaffolding was removed from within the containment in preparation for power operation.

The scaffolding required to reach all the zones of channels in the dome and cylinder portion of the liner is quite extensive, and for this reason an evaluation was made to determine whether the channels could remain plugged.

Determination

Although the containment liner test channels (monitor channels) were originally designed for testing the leak tightness of the liner seam welds, they were designed as safety related, Seismic Category 1. The monitor channels, as such, are capable of withstanding all upset loading conditions, as well as all test loads, without any loss of function or impairment of the performance of the containment liner. Although the monitor channels were not originally designed as part of the pressure boundary, our evaluation shows that they are capable of performing this function, and in so doing, provide additional containment leak protection. This evaluation leads to our determination that a valid Type A leak rate test can be performed whether the monitor channels are open or plugged shut.

The maximum stress in the liner plate that would cause the maximum stress in the monitor channels is a tensile stress which is a result of the 47 psig integrated leak rate test of the containment (see the following table). This case induces a higher tensile stress in the liner plate than the design basis accident. The reason for this is that there is no temperature rise associated with the test condition. Compressive stresses are created by the high temperatures associated with an accident condition, which overcome the tension in the liner. The stresses listed in the following table have been divided by the appropriate capacity reduction factor, which is 0.95 for tension. Our original computations for the liner plate indicate that there would be no inelastic buckling of the plate.

This table gives the stresses for the reinforcing bars and liner for various loading conditions and was previously provided to the NRC as additional information to the SAR.

| Max. Stress K/□ / Loading | REBARS | | | | | | LINER | | | |
|---|---|---|---|---|---|---|---|---|---|---|
| | DOME | | WALL | | DISC. | AT | DOME | | WALL | |
| | Merid. | Hoop | Merid. | Hoop | @ FDN | OPNG | Merid. | Hoop | Merid. | Hoop |
| P | +9.7 | +19.3 | +9.5 | +22.5 | +19.0 | +31.0 | +11.0 | +11.8 | +11.2 | +21.1 |
| 1.15P | +12.6 | +22.2 | +12.7 | +26.1 | +23.0 | +35.5 | +13.0 | +15.3 | +14.4 | +25.0 |
| P+T | +27.4 | +37.0 | +28.3 | +31.2 | +36.5 | +36.0 | -7.3 | -6.5 | -13.3 | -5.1 |
| 1.5P+T | +42.5 | +52.2 | +45.0 | +49.0 | +57.0 | +51.2 | -3.0 | -2.2 | -9.3 | +1.0 |
| P+T+E | +28.6 | +40.3 | +30.5 | +31.5 | +39.0 | +42.8 | -9.0 | -14.6 | -23.9 | -6.4 |
| 1.25P+T+1.25E | +37.0 | +49.5 | +40.5 | +41.3 | +51.0 | +53.5 | -8.5 | -9.3 | -25.7 | -4.6 |
| P+T+1.0E' | +29.2 | +41.4 | +31.2 | +32.3 | +40.0 | +43.8 | -9.7 | -10.4 | -27.7 | -6.9 |

**Note:** E stands for design earthquake, E' stands for maximum credible earthquake and P is the condition for 47 psig. Plus (+) is tension, minus (-) is compression.

The above table is based on ground accelerations of 0.10g for the design earthquake and 0.20g for the maximum credible earthquake. Where creep of concrete reduces the stresses in the rebars or the liner it was not considered for conservatism. Where creep increased the stress it was included to obtain the maximum stress.

The channels were checked for stresses due to test conditions by subjecting them to the combination of an exterior pressure of 47 psig and the movement of the channel legs due to elongation of the liner plate, in both meridional and hoop directions. The results show that the channels are not overstressed.

Since the channels are not overstressed for the test condition, and this is the most severe loading case, we conclude that the Type A tests can be conducted with the monitor channels either open or plugged shut and meet the requirements of Appendix J, 10 CFR 50.

Included as Enclosure 2 for your information are Chicago Bridge & Iron Company Drawing Nos. 19, 20, 55, 56, 57, 58, 58A and 59 showing the arrangement and details of the monitor channels in Unit 2.

Very truly yours,

E. A. Liden
Manager - Nuclear
Licensing and Regulation

Enclosures

CC: Mr. Donald C. Fischer (with Enclosures)
Licensing Project Manager

Mr. James Linville (with Enclosures)
Senior Resident Inspector