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PREVIOUS QUARTERLY REPORTS IN THIS SERIES

GA-A13209, for the period ending September 30, 1974

GA-A13282, for the period ending December 31, 1974

GA-A13454, for the period ending March 31, 1975

GA-A13514, for the period ending June 30, 1975

GA-A13692, for the period ending September 30, 1975

GA-A13793, for the period ending December 31, 1975

GA-A13909, for the period ending March 31, 1976

GA-A14042, for the period ending June 30, 1976

GA-A14154, for the period ending September 30, 1976

ABSTRACT

This publication continues the quarterly report series on Fort St. Vrain (FSV) Surveillance and Testing. The program will perform post-startup tests on FSV plant components and systems to increase our knowledge of operating characteristics of large HTGRs. This report contains a summary of the findings made during power runs between 25% and 28% power.

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1. INTRODUCTION

The Fort St. Vrain (FSV) Surveillance and Testing Program is directed toward acquiring FSV operating experience for application to the design of large HTGRs. Four subtasks remain funded for FY-77, none of which were scheduled for completion by the end of the quarter. The plant rise to power which was started on July 1, 1976, continues. During this quarter the plant reached 28% power, rolled the turbine, and generated a maximum electrical output of 75 MW. Numerous tests were conducted and much data were gathered and analyzed. Performance to date is as expected.

2. ACCOMPLISHMENTS

The status of the four subtasks which were active during this quarter is discussed below.

2.1. SUBTASK A: STEAM GENERATOR PERFORMANCE AND CORROSION SURVEILLANCE

2.1.1. Instrumentation

The steam generator temperature scanner collected thermocouple data on a continuous basis during the period, with an overall availability of 95%. These readings were recorded on magnetic tape at 1-minute intervals and an hourly log of the temperatures was printed and examined. In addition, the data were scanned by display on a CRT readout.

The helium temperature measurements are giving numbers close to predicted values. Thermocouples on the steam generator tubes within the PCRV lower cavity appear to be indicating tube wall readings which are affected slightly by the helium flowing over the thermocouple insulation.

As was reported in the previous quarterly report,* the exit thermocouple probes on the steam tubes (located outside the PCRV) just above the ringheaders are giving unacceptable measurements. It is now planned to replace these thermocouple probes with welded thermocouples which are fastened directly to the tube wall. Welded thermocouples have already been installed on module B-2-3. Nine of these thermocouples are being used to provide data for the steam generator scanner system. Data for the remaining nine tubes are being provided by the original probes.

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*Fort St. Vrain Surveillance and Testing Program Quarterly Progress Report for the Period Ending September 30, 1976," ERDA Report GA-A14154, General Atomic Company, October 29, 1976.

The welded thermocouples are performing satisfactorily. The temperatures measured by these thermocouples are within ±1.0% of the average module outlet temperature which is recorded in the control room. The exit thermocouple probes on the remaining steam generators will be replaced with welded thermocouples during a future plant shutdown.

2.1.2. Thermal Performance

The thermal performance of steam generator module B-2-3 is monitored by a combination of the steam generator temperature scanner system and control room data. From these data it is possible to evaluate the steam generator performance (heat duty) and to compare this performance with predicted results for the same operating conditions.

Evaluation of the steam generator performance has been performed for each startup sequence at or above 27% reactor power. Comparison of predicted performance has shown a spread of +0.2% to -2.5% in measured heat duty when it is compared to predicted heat duty. This spread is currently being evaluated.

2.1.3. Feedwater Chemistry

Water analysis samples were taken from the economizer inlet during periods of plant operation. Certain of the water chemistry limits vary depending upon whether or not the operating power level is greater or less than 25%. With the reactor above 25% power for most of the quarter, more restrictive limits are in effect. The water chemistry data demonstrated that both the startup and power operation criteria were generally met.

2.2. SUBTASK D: PCRV STRUCTURAL RESPONSE VERIFICATION

During the quarter, PCRV sensor data were collected for reactor operation between zero and 28.5% power and vessel internal pressures between 301 and 634 psia. These data were analyzed to evaluate the structural responses of the PCRV. The test data were compared with reference data corresponding to zero power and zero pressure to determine changes in prestressing forces as indicated by load cells on representative tendons and concrete strain in critical sections of the PCRV. Tendon load changes were found to be within the range recorded during initial proof pressure tests and concrete strains were well within anticipated values. Bulk concrete temperatures of less than 120°F were indicated by the embedded thermocouples during higher power operation.

The time-dependent multiaxial stress state of the PCRV concrete can be defined by a uniaxial creep function and creep Poisson's ratio. Incorporation of the variation of Poisson's ratio with temperature and age of loading in the creep function used in the finite element analysis was completed. This resulted in better correlation between the concrete cylinder creep test data and analytical prediction which included creep Poisson's ratio variation. Work was initiated to validate the creep function used in the time-dependent structural analysis with collected PCRV sensor data. The actual loading history of the FSV PCRV based on sensor logs was established for use in this validation.

2.3. SUBTASK G: TRANSIENT ANALYSIS PROGRAM (TAP) CODE VERIFICATION

Data continued to be collected and evaluated during the reporting period. Plant performance calculations based on the data received were compared to prediction made by TAP for the same operating conditions. Some minor performance differences were noted and several areas of TAP were identified as potential problem areas. Further work is under way and this effort will be reported in detail next quarter.

2.4. SUBTASK I: VALVE PERFORMANCE INSPECTION

Fort St. Vrain employs once-through steam generators which use a bypass startup system. PV-22129-1 and PV-22130-1 are Control Component, Inc. (CCI) 6 in. by 10 in. angle valves with a 4-in.-diameter balanced plug and 6-in.-long disc stack used as pressure control valves to provide steam generator backpressure control during startup and shutdown of the secondary cooling system. These valves are located on a bypass from the main steam line downstream of the steam generators and are physically mounted immediately on top of the preflash tanks. The purpose of these valves is to regulate steam generator backpressure during (1) preboiling recirculation of feedwater, (2) steam generator boilout including two-phase flow, and (3) the transition to superheated steam up to a temperature of 800°F and a flow rate of 288,000 lb/hr corresponding to approximately 28% of reactor power.

2.4.1. Valve Problems

When air actuators were used with PV-22129-1 and PV-22130-1, the valves were unable to maintain steam generator backpressure control within the desired ±50 psi pressure span due to erratic valve stem motion. This erratic motion was prevalent with the valve operating with the plug near the valve seat. Backpressure variation experienced was as much as ±400 psi from the desired values, which are very slowly ramped from 1000 to 2200 psi. As steam generator boiling was initiated, the pressure variations subsided to ±100 psi but were still outside of specified values.

Inspection of the valve while in operation showed the valve stem to jump from 1/16 to 1/4 in. instead of smoothly moving in response to the valve positioner's demand. It appeared the stem was experiencing a great deal of static friction and when the air actuator built up sufficient differential pressure across the double-acting actuator piston to initiate movement, overshoot would occur causing correspondingly large pressure variations in the steam generators.

2.4.2. Discussion

The vendor (CCI) was contacted to assist in troubleshooting and correcting the problem experienced. The problem of precise valve stem positioning apparently was not new to CCI and was attributed to packing friction, an undersized actuator, and/or insufficient actuator stiffness.

The valves, as built and delivered, contained Grafoil stem packing and Grafoil in the plug balance seal of the valve. CCI indicated that static or breakaway friction varies greatly with the technique used to install these seals.

The valves were disassembled, inspected, modified, reassembled, and tested numerous times in an effort to minimize metal-to-metal interference between the plug and disc stack, as well as to minimize packing and seal friction. The air actuators were modified in an attempt to increase their stiffness. The pressure control was still unacceptable.

General Atomic initiated procurement of self-contained electrohydraulic actuators in parallel to the continuing efforts of CCI to further reduce stem and plug friction.

Modifications to the pressure control valves are summarized as follows:

1. The valve stem packing was changed from all Grafoil to alternating rings of VG4 and Grafoil.
2. The balance seal was reduced from two rings of Grafoil to one ring installed at a reduced compression.
3. New plug and stem assemblies were designed, manufactured, and installed to accomplish better plug-to-stem alignment.
4. Clearances were increased between the valve plug and disc stack from a nominal value of 0.010 in. diametral clearance to a nominal value of 0.020 in.
5. Efforts directed at stiffening the control characteristics of the pneumatic actuator by changing the type of positioners and spring rates of return springs were unsuccessful, and a hydraulic

actuator was finally installed along with associated valve yoke stiffening supports.

6. During the evaluation of the performance of the pressure control valves, it became apparent that the minimum controllable flow exceeded desired values and the maximum desired flow of the valve was somewhat less than expected. New disc stacks are currently being designed and procurred.

2.4.3. Hydraulic Actuator

The hydraulic actuator selected for use on the CCI valve is a self-contained unit manufactured by Western Hydraulics, Anaheim, California, with a force output capability adjustable from 10,000 to 20,000 lb. This actuator is similar in circuit design to other existing actuators built by Western Hydraulics but has a custom servovalve control package fitted to the particular needs of the CCI valve.

The hydraulic actuators were installed and, after experiencing some component design problems in the electronic control package, are performing satisfactorily.

2.4.4. Summary

The pressure control valves, with modifications to the seals and to the plug geometry and with hydraulic actuators installed, are performing sufficiently well to achieve their intended design purpose using a modified startup and shutdown procedure. New disc stacks have been ordered which will allow operation at lower flow rates. Controller circuit leakage current resulted in system oscillation similar to the original problem. The controllers have been replaced with modular units eliminating the current problem and making future corrections easier.

It is anticipated that installation of the newly designed disc stacks will allow the valves, when modified, to meet their intended design goals.