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High-Temperature Gas-Cooled Reactor Safety Studies for the Division of Reactor Safety Research Quarterly Progress Report, July 1–September 30, 1980

S. J. Ball
J. C. Cleveland J. C. Conklin
R. M. Harrington

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HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF REACTOR SAFETY RESEARCH QUARTERLY PROGRESS REPORT, JULY 1—SEPTEMBER 30, 1980

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S. J. Ball, *ORECA-I: A Digital Computer Code for Simulating the Dynamics of HTGR Cores for Emergency Cooling Analyses*, ORNL/TM-5159 (April 1976).

T. W. Kerlin, *HTGR Steam Generator Modeling*, ORNL/NUREG/TM-16 (July 1976).

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FOREWORD

High-temperature gas-cooled reactor (HTGR) safety studies at Oak Ridge National Laboratory (ORNL) are sponsored by the Division of Reactor Safety Research, which is part of the Office of Nuclear Regulatory Research of the Nuclear Regulatory Commission (NRC).

This report covers work performed from July 1 to September 30, 1980. Previous quarterly reports and topical reports published to date are listed on p. v. Copies of the reports are available from the Technical Information Center, U.S. Department of Energy, Oak Ridge, TN 37830.

HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF REACTOR SAFETY RESEARCH QUARTERLY PROGRESS REPORT, JULY 1—SEPTEMBER 30, 1980

S. J. Ball, Manager
J. C. Cleveland J. C. Conklin
R. M. Harrington

ABSTRACT

Development work continued on codes for simulating high-temperature gas-cooled reactor (HTGR) dynamics. Several postulated transients were run to evaluate the steam turbine plant code ORTURB. Further calculations were completed to assist in the evaluation of possible thermal stress problems in the Fort St. Vrain (FSV) reactor core support following a postulated design-basis earthquake accident.

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1. HTGR SYSTEMS AND SAFETY ANALYSIS

S. J. Ball

Work for the Division of Reactor Safety Research (RSR) under the High-Temperature Gas-Cooled Reactor (HTGR) Systems and Safety Analysis Program began in July 1974, and progress is reported quarterly. Work during the present quarter included code development and assistance to the Nuclear Regulatory Commission (NRC) on Fort St. Vrain (FSV) reactor licensing questions.

1.1 Steam Turbine Dynamic Response to Postulated Transients

J. C. Conklin

During development of the computer code ORTURB,[1] postulated transients were simulated to assess behavior of the computer algorithm. Results of these postulated transients were not compared with any actual data but were used to determine whether or not the computational model was predicting physically reasonable values (i.e., no out-of-limit steam temperatures, reverse flows, or liquid temperatures greater than saturation). The four postulated transients were: (1) condenser pressure rise to atmospheric, (2) high-pressure turbine (HPT) exhaust pressure rise to main steam pressure, (3) cessation of extraction steam flow to feedheater 6, and (4) cessation of extraction steam flow to feedheater 3. A detailed schematic of the steam turbines and feedwater heaters of the FSV power station as modeled with ORTURB is provided in Fig. 1.

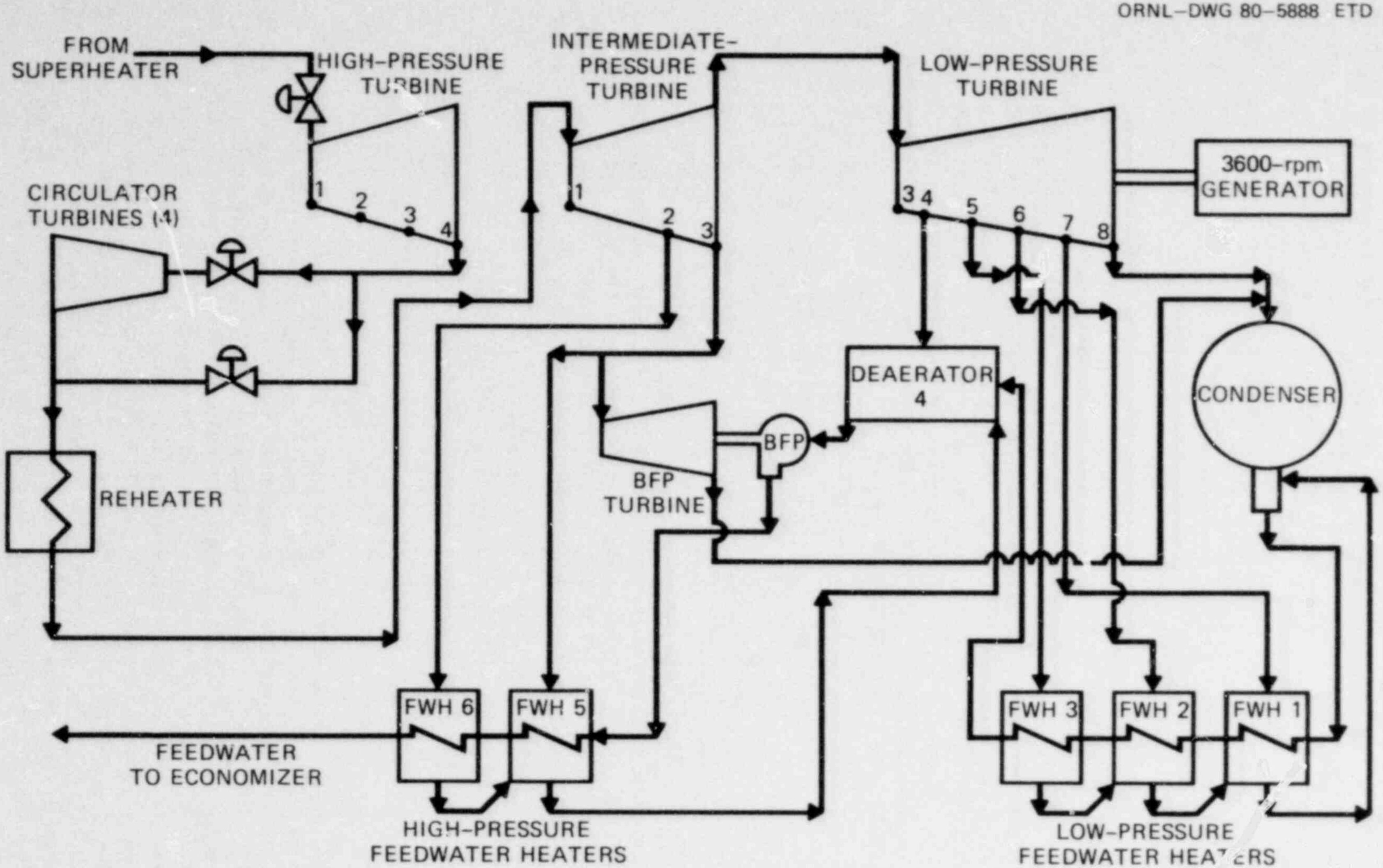

Fig. 1. Fort St. Vrain reactor steam turbine plant schematic.

1.1.1 Condenser pressure rise to atmospheric

The condenser was postulated to pressurize to the atmosphere linearly over 60 s. The pressure response of turbine extraction points 6 and 7 and the pressure response of the shell side of feedheaters 1 and 2 are plotted in Fig. 2.

The pressure at extraction point 7 does not begin to rise until the pressure ratio (P_{cond}/P_7) is greater than the critical pressure ratio. The shell-side pressure of feedheater 1 does not begin to increase until after the corresponding extraction point 7 pressure rise. The increase in extraction pressure at point 7 also increases the extraction enthalpy carried to feedheater 1, which will increase the temperature of the feedwater exiting feedheater 1. The increase in the temperature of the feedwater entering feedheater 2 accounts for the feedheater 2 shell-pressure rise occurring slightly before the extraction point 6 pressure rise. The rise in extraction point 6 pressure is due to both the effects of decreasing extraction flow to the shell of feedheater 2 and the increase in the stage pressure ratio ($point_7/point_6$) to greater than critical.

The condenser pressure rise to atmospheric is abruptly stopped at 60 s with the discontinuous slope as shown, which is physically unrealistic. The extraction pressure at point 7 also has a discontinuous slope at that time because of the ORTURB modeling assumption that the dynamics of mass and energy storage in the turbine stages are negligible compared with the dynamics of the remainder of the system. The shell pressure of feedheater 1 is slightly less than that of the condenser from 55 to 65 s, which would cause reverse flow from the condenser to the drain cooler section of feedheater 1. However, reverse flow is not accounted for by ORTURB; thus, this situation indicates that the transient was modeled in an inappropriate manner for ORTURB. However, because the maximum pressure discrepancy is small (0.3 psid) and there are no existing data for comparison, this particular transient was not reevaluated with a better estimate for the condenser pressure rise. This transient was also used to improve the pressure-flow convergence schemes for ORTURB.

Feedheater 2 shell pressure is still rising at the termination of the condenser pressure ramp at 60 s, lagging slightly behind the shell pressure plot of feedheater 1. This lag is due to the effect of the rising feedwater temperature from feedheater 1, which increases the saturation pressure of the vapor in the shell of feedheater 2. The extraction point 6 pressure is influenced by the feedheater 2 shell pressure at 60 to 65 s in an appropriate manner because the shell pressure affects the extraction flow, which in turn affects the turbine extraction point pressure by modifying the turbine mass flow distribution. This pressure at point 5 is also changing throughout the transient, but very little (<0.1 psid).

The electrical output of the low-pressure turbine (LPT) was calculated throughout the transient and decreased as the condenser pressure rose. However, the exhaust enthalpy loss exceeded the values published by Spencer et al.[2] used in ORTURB; the electrical output therefore is not reported.

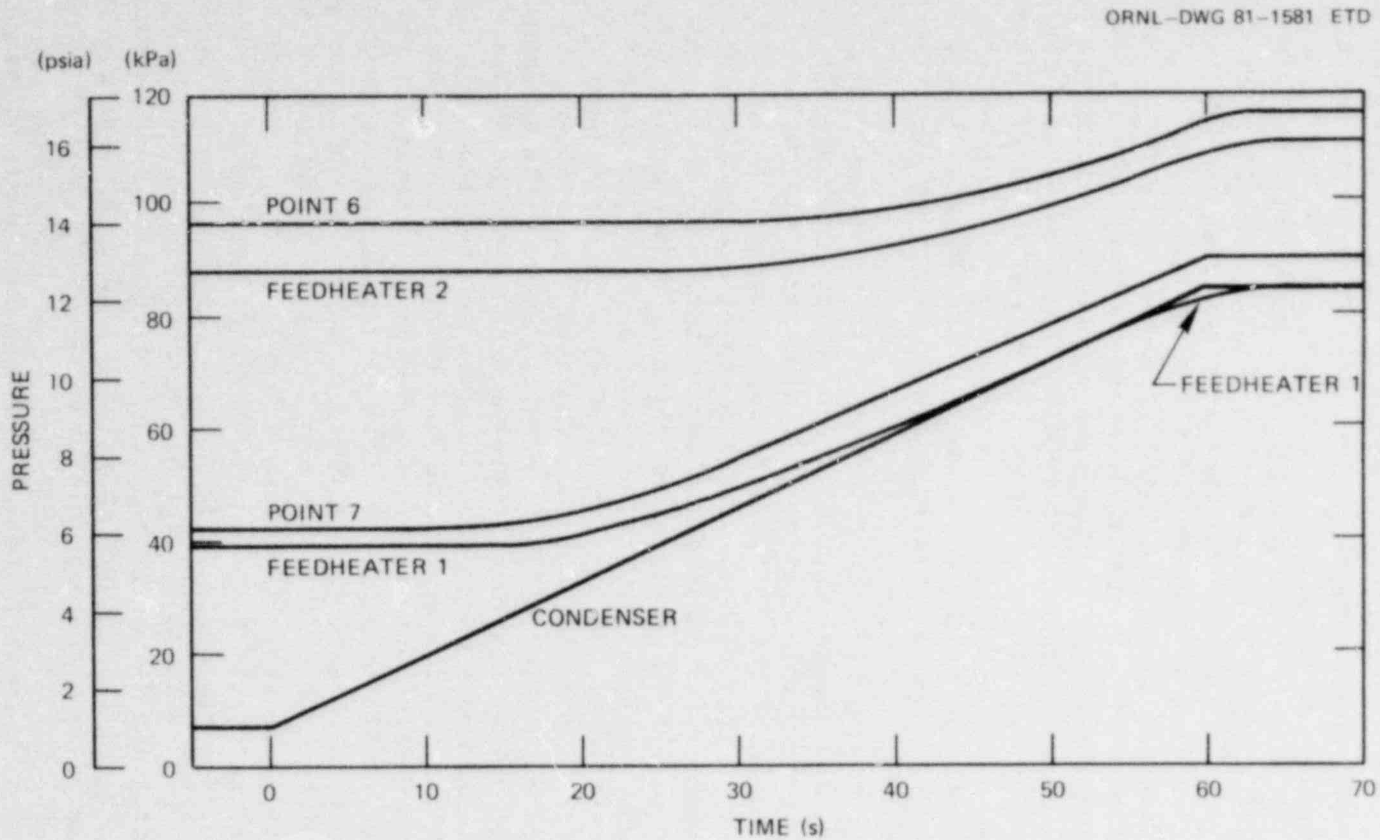

Fig. 2. Condenser pressure rise to atmospheric.

1.1.2 High-pressure turbine exhaust pressure rise

The exhaust pressure of the HPT was linearly ramped from the 100% power exhaust pressure of 893 psia to the constant, 100% power main steam pressure (2412 psia) while maintaining the 100% power value of mass flow through the turbine. The transient pressure response to both the governing-stage shell and the following reaction-stage shell (point 3) are plotted in Fig. 3. The pressure-flow computational algorithm had convergence difficulties at 35 s into the transient which was expected, as this particular transient is severe in nature and probably unrealistic. A prudent plant operator would probably trip the HPT off-line when this situation was evident. The intent of modeling this transient was to (1) ensure that variations in exhaust pressure would be appropriately

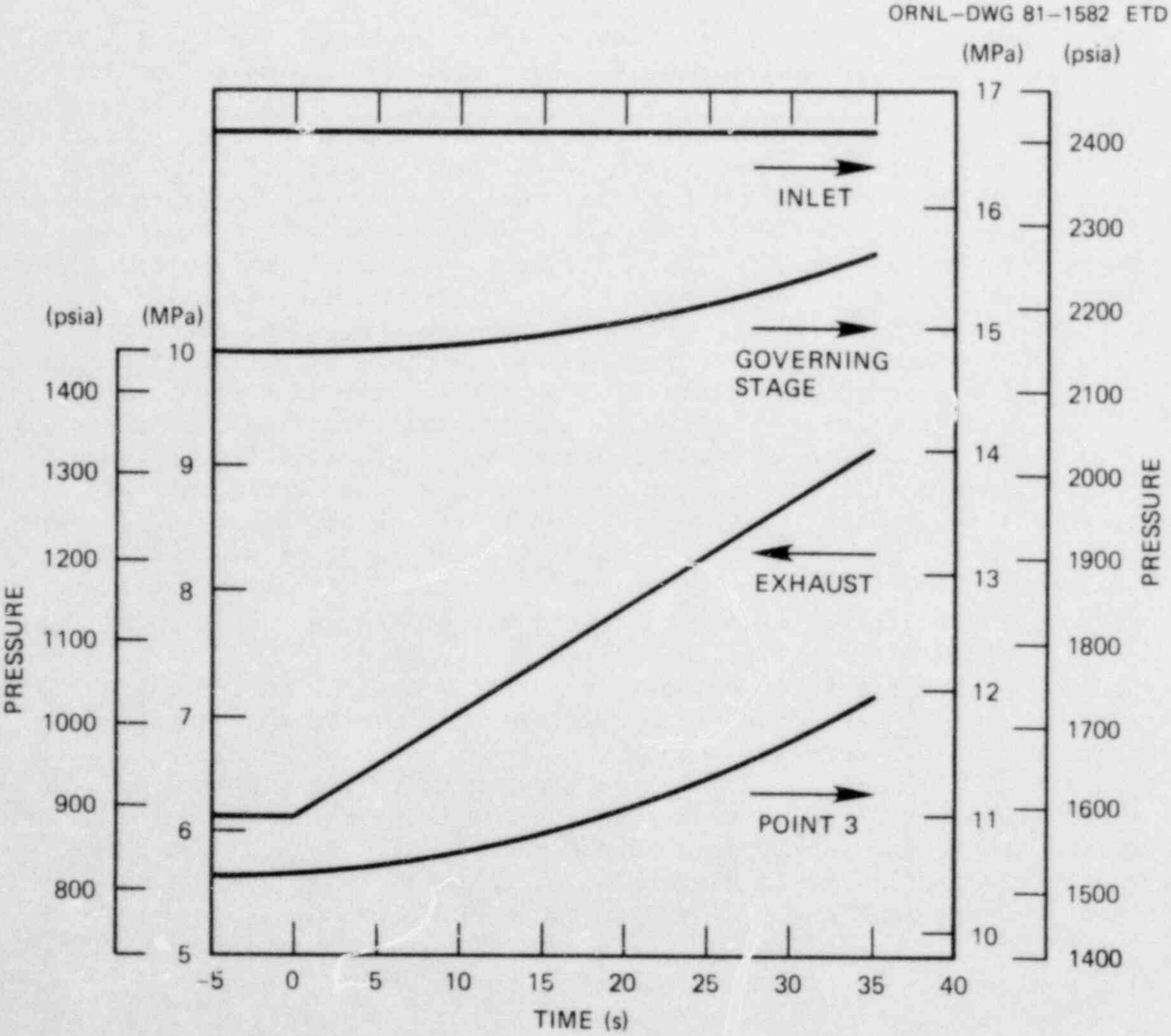

Fig. 3. High-pressure turbine exhaust pressure rise.

reflected upstream and (2) optimize the convergence scheme of the computational algorithm.

As shown in Fig. 3, variation in the exhaust pressure is indeed reflected upstream to the governing-stage shell. Although pressure-flow convergence problems stopped the calculation at 35 s, it appears that in the limit, as the exhaust pressure approaches the inlet pressure, the governing-stage and following reaction-stage shell pressure also approach the inlet pressure appropriately.

1.1.3 Cessation of extraction flow to feedheater 6

Utilities have been known to turn off extraction flow to the highest-pressure feedheater during periods of heavy electrical demand. This extraction flow cessation sends more steam through the turbine to generate more power, at the expense of turbine thermal efficiency. This situation was modeled with ORTURB and the most interesting results are shown in Fig. 4.

Point 2 (the extraction point) pressure increases immediately upon stoppage of the extraction flow to feedheater 6 and remains constant. The perturbation in mass flow then causes extraction point 3 pressure to increase. Greater pressure at extraction point 3 increases the extraction flow to feedheater 5, thus causing an increase in feedheater 5 shell pressure. This higher pressure then causes a decrease in extraction flow, thus increasing the extraction point pressure. The end result of these feedback effects is a slight increase ($\sim$1 psi) in extraction point 3 pressure. This pressure increase at point 3 should, ideally, also increase extraction point 2 pressure, because downstream pressure variations should be reflected upstream for reaction stages, except for the final stage, which should have critical flow. This variation (probably within the experimental error of a pressure tranducer) does indicate proper response of feedheater 5 shell pressure and extraction flow to an upstream turbine flow perturbation. The other downstream extraction pressures and feedheater shell pressures also showed appropriate responses.

The shell pressure of feedheater 6 decays appropriately to the decreased saturation temperature of the incoming feedwater (note the pressure scale change). The mass flow out of the drain cooler of feedheater 6 to feedheater 5 also decreased appropriately.

The feedwater inlet temperature to the steam generator decreased as expected, and because the steam conditions at the steam generator and reheater outlets were set constant to the 100% power conditions, an additional 29.4 MWt (3.5%) was transferred to the working fluid. The total gross electrical output of the plant increased as expected, with ORTURB predicting an increase of 3.6 MWe (1.1%). This output increase represents a decrease in overall plant thermal efficiency, as expected when the regeneration of a Rankine cycle is decreased.

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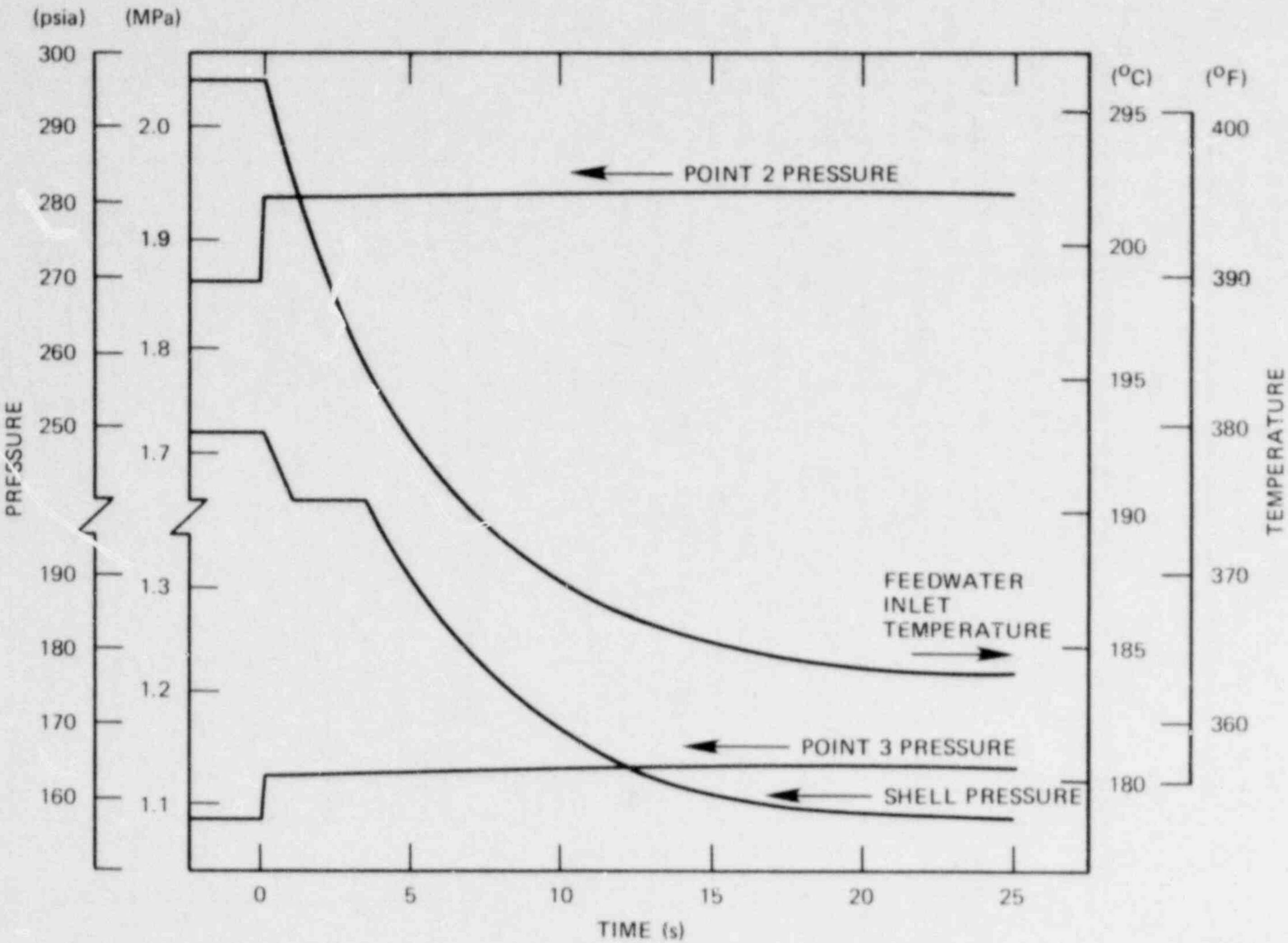

Fig. 4. Feedheater 6 extraction flow stoppage.

1.1.4 Cessation of extraction flow to feedheater 3

An ORTURB simulation of an extraction flow stoppage to feedheater 3 was also performed, with all other 100% power operating conditions remaining the same. The most interesting results are shown in Fig. 5. The shell pressure of feedheater 3 decays exponentially to the saturation pressure of the inlet feedwater. There is a step increase in pressure at extraction point 5 because of this perturbation of the turbine mass-flow distribution, which also steps the extraction point 6 (downstream) pressure, but not the upstream point 4. The pressure at point 4 should rise immediately, but does not, as this phenomenon is not modeled. There is, however, a feedback effect on the extraction point 4 pressure because of deaerator pressure changes.

When the extraction flow to feedheater 3 is stopped, exiting feedwater that enters the deaerator cools. This cooling decreases the deaerator pressure, which then increases the extraction flow to the deaerator, depressing the extraction point 4 pressure. The pressure at the downstream extraction points decreases as shown. The magnitude of this extraction point 4 perturbation is <2 psi, and is not sufficient to cause a pressure perturbation upstream to extraction point 3.

The turbine electrical output calculated by ORTURB increased by 1.9 MW immediately upon stoppage of the extraction flow to feedheater 3. This power output increase is caused by more mass going through the turbine. However, as the deaerator cools, it receives more turbine extraction flow, decreasing the mass flow through the remainder of the turbine. The electrical output calculated by ORTURB then decreases to

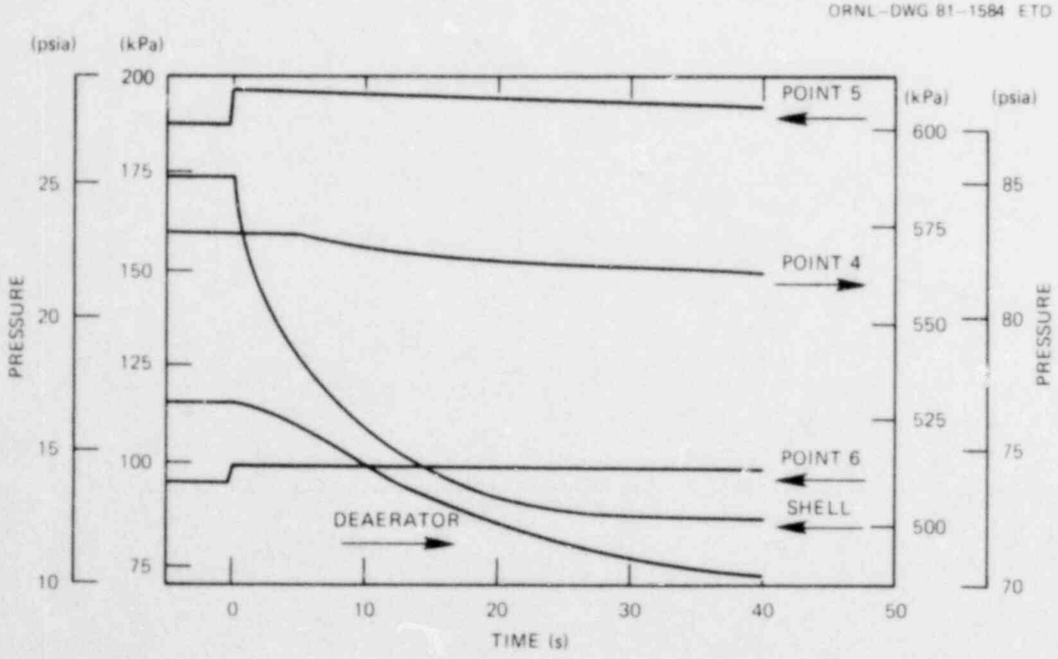

Fig. 5. Feedheater 3 extraction flow stoppage.

~0.5 MW above the 100% power load. The inlet feedwater to the steam generator was calculated to be slightly cooler than the 100% power conditions by an amount equivalent to 0.14 MW. This calculation, if correct, would indicate that an increase in both thermal efficiency and plant electrical output could be obtained by turning off feedheater 3 at 100% power.

Comparison of actual plant data with the calculated results of the previous transients would be highly desirable. Intentional failure of the plant for the first two postulated transients would not be feasible, however, because these transients are rather severe and would probably result in permanent turbine damage. The two transients of extraction flow perturbations do not appear to have severe consequences if the turbine extraction flow is turned off. Comparison of actual extraction flow transients of the FSV steam turbines would be extremely valuable for improving or correcting the computational model of ORTURB, and perhaps also for improving plant performance.

1.2 Development of the ORECA Code for Simulating FSV Reactor Core Emergency Cooling Transients

S. J. Ball

A most helpful review of the ORECA code[3] was done by Brookhaven National Laboratory (BNL),[4] which noted several problems and made many good suggestions for improvement. A number of changes to the code were made as a result.

The major improvement made was to the internode heat-conduction algorithm. Previously, the effective thermal conductance between a given node and its neighbors was assumed to be a function of only that node's temperature. In the corrected version, each conductance term is based on the average temperature of the given node and its neighbor. Corrections were also made to the acceleration pressure-drop term and the orifice coefficient temperature multiplier for cases in which flow is reversed. Improvements were made in the algorithms that account for the ratios of conduction areas and directional conductivity relationships for axial vs radial conduction between refueling region blocks.

1.3 ORECA Code Calculations of Postulated FSV Reactor LOFC/FWCD Accidents for Core Thermal Stress Evaluations

S. J. Ball

Results of previous analyses of the postulated design-basis earthquake loss-of-forced-convection (LOFC) accident followed by a firewater cooldown (FWCD) were used by Los Alamos Scientific Laboratory (LASL) to calculate thermal stresses in parts of the core support structure.[5] These stresses result from large temperature differences between adjacent refueling regions caused by preferential heating and cooling of the regions during the LOFC and FWCD phases of the accident. Recent LASL calculations of maximum stresses in the core support blocks indicated that the stresses were large enough for some concern about possible crack formation and propagation in the support blocks. Several significant uncertainties, however, in both the thermal analyses (ORECA code) and the stress analyses (LASL calculations) required refinements in both analyses.

Sensitivity studies were completed using the ORECA code to determine the effect of changes in the reference case assumptions. These studies indicated that a reduction of initial core power and an increase in the firewater booster pump output could significantly reduce the maximum region-to-region temperature difference. The reference case analyses had been done assuming a 105% operating power level; thus, further analyses at the current FSV operating limit ($\sim$70%) were recommended as an interim means of alleviating concern about the safety of present operation, at least until more detailed analyses of the full-power case were available. Booster pump tests had shown that the FWCD flow estimates used in the analysis were conservative. The sensitivity studies also showed that the problem was less severe for high-flow-resistance cores because the redistribution of the coolant flow in the FWCD phase is less sensitive to the hot coolant-channel flow resistance.

Further refinements to the ORECA code in the core support region were also found to be needed to provide more detailed information for LASL's stress analysis code inputs. Output was sent to LASL from a revised ORECA model that had ten axial nodes per refueling region — one for the upper reflector, one for each of the six fueled regions, two equal-sized nodes for the bottom reflector, and one for the core support block. The revised ORECA version also provided outputs of heat flows into selected nodes via conduction and convection.

1.4 Long-Range Program Planning

R. M. Harrington   S. J. Ball

A "white paper" review of HTGR safety code availability and needs was written, primarily to stimulate internal discussions of long-range plans. The review focused on licensing needs for the next generation of HTGRs and identified probable major safety issues.

Plans were also outlined for a proposed program expansion that would include several FSV experiments. Possibilities considered included tests that could be used to infer core (i.e., refueling region) bypass flow fraction, "cool core" tests of the refueling region reverse-flow phenomenon, noise analysis and other tests to investigate the mitigation of the oscillation phenomenon with the Luci locks (region constraint devices), and tests of interactions between the control and safety systems.

A proposal was also outlined for additional program effort on severe accident sequence analyses (SASA), which would be similar to, though at a much lower level than, the recently-organized RSR SASA program for light-water reactors.

2. MEETING ATTENDED UNDER PROGRAM SPONSORSHIP: VISIT TO LASL TO DISCUSS THE FSV CORE SUPPORT THERMAL STRESS PROBLEM — AUGUST 25, 1980

S. J. Ball

The FSV core thermal stress problem is discussed in Sect. 1.3. This LASL meeting was held to bring NRC and Public Service Company of Colorado (PSC) personnel up to date on the status of the analysis and to decide what further work needed to be done. The conclusion was reached that ORNL should refine the modeling of the core support region and provide LASL with input data from a 70% power cycle-two core accident simulation, as well as data from the reference full-power equilibrium core runs.

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