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SIMULATIONS OF THE RECENT LASALLE-2 INCIDENT WITH THE BNL PLANT ANALYZER

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ABSTRACT

This paper presents the results of simulations of the recent power oscillation incident at the LaSalle-2 Nuclear Power Plant using the BNL Plant Analyzer. The causes of the oscillation were investigated and the sensitivity of the oscillation to key parameters was studied. It is concluded that the observed power oscillation was caused by boiling instability (i.e., density wave oscillation) reinforced by the reactivity feedback in neutron kinetics, and that the density wave oscillation resulted from flow reduction due to recirculation pump trip and feedwater temperature reduction due to partial loss of feedwater heating capabilty as well as power peaking.

INTRODUCTION

On March 9, 1988, an Instrument Maintenance (IM) technician at LaSalle Unit 2, while performing a functional test on a differential pressure switch, caused both recirculation pumps to trip off due to a valving error (Diederich 1988). Because of the large and rapid power reduction, feedwater heater high level alarms caused a partial isolation of feedwater heaters, resulting in a reduction of 51 °F in feedwater temperature. Approximately 5 minutes into the event, the local power range monitor (LPRM) up- and downscale alarms began annunciating and the average power range monitors (APRM) were observed to be oscillating with an ~2.3 s period. Realizing the unit's unfavorable location on the power/flow map, the operating staff was preparing to scram the reactor manually, when an automatic scram occurred on high-flux trip (118% trip on APRM). Prior to the scram, the operators attempted to remedy the situation by trying to restart the recirculation pumps, but failed.

The growing power oscillation observed in this incident raises concerns about the stability of BWRs. The important questions are: Why did the oscillation occur? Was there a possibility for divergent oscillations? What if the operator did restart the recirculation pumps? What if the Main Steam Isolation Valves (MSIV) were inadvertently closed right after the pump restart? Brookhaven National Laboratory (BNL) was asked by the USNRC to simulate the LaSalle-2 event with the BNL Plant Analyzer (BPA) (Wulff 1984 and Cheng 1986). This paper reports the results of the BPA simulations as well as the important findings of the present analysis.

*Work performed under the auspices of the U.S. Nuclear Regulatory Commission.

EVENT DESCRIPTION

After the control systems had been ruled out as the cause of the instability by using the BPA, it was postulated that the observed growing power oscillation was caused by a nuclear thermal hydraulic instability brought about by the recirculation pump trip and the partial feedwater heater loss plus power peaking.

The transient was initiated from the 85% power and 75% flow condition by a recirculation pump trip followed by a partial feedwater heater loss. Table 1 summarizes the sequence of events for the transient.

Table 1

Sequence of Events

Event/Action	Time (m)
1. Steady state at 85% power and 75% flow	-5.0
2. Recirculation pumps tripped	0.0
3. Reactor power dropped to 37%	0.4
4. Core flow reached natural circulation (29%)	0.5
5. Feedwater heaters partially isolated	1.0
6. Reactor power reached 45% & "beat" phenomena began	2.0
7. Modulated limit cycle oscillations continued	5.2
8. Enhanced limit cycle oscillations began	5.9
9. Growing oscillations started	7.3
10. Reactor power reached 118%, and reactor tripped	8.1
11. End of transient	9.0

THE BPA SIMULATION

The proper initial conditions are essential for the analysis of this event. A steady-state run was first made to obtain the desired initial conditions. The initial conditions obtained by the BNL Plant Analyzer are summarized as follows:

MASTER

1.	Reactor Power	2808 MWt
2.	Core Inlet Flow Rate	10,210 kg/s (81 Mlb/h)
3.	System Pressure	69.5 MPa (1007 psia)
4.	Steam Flow Rate	1424 kg/s (11.3 Mlb/h)
5.	Feedwater Flow Rate	1424 kg/s (11.3 Mlb/h)
6.	Recirculation Drive Flow Rate	3641 kg/s (28.9 Mlb/h)
7.	Recirculation Pump Speed	1590 rpm
8.	Core Average Void Fraction	41.7 %
9.	Core Average Fuel Temp.	601°C (1113°F)
10.	Core Average Coolant Temp.	285°C (545°F)
11.	Core Inlet Subcooling	11°C (20°F)
12.	Feedwater Temperature	206°C (402°F)

Reactivity feedback plays an important role in a BWR for both the steady-state and transient analyses. The feedback coefficients used in the BPA simulation were obtained from the earlier work on the void feedback (Cheng 1977) and the Doppler feedback (Cheng 1978). Axial power distribution is known to affect the core instability of a BWR (Yokomizo 1987). The axial power profile used in this work is a typical bottom-peaked power shape with an axial peak of 1.38.

The results of the BPA simulations are presented in Figures 1 through 9. Figure 1 presents the simulated power oscillation which shows a remarkable resemblance to the actual APRM traces (Kaufman 1988). The zoomed display of the power oscillation just prior to the automatic scram is shown in Figure 2, which shows a period of oscillation of ~2.8 s as compared to ~2.3 s as observed in the actual event. That the power oscillation is the result of the density wave oscillation is clearly demonstrated by the oscillatory void behavior as shown in Figure 3.

Figure 4 presents the simulated core flow response and Figure 5 shows the zoomed display of the core flow oscillation with the same period of ~2.8 s as the power oscillation. The core flow quickly reaches the natural circulation condition within a minute due to the recirculation pump trip as shown in Figure 6 for the recirculation pump speed and in Figure 7 for the recirculation drive flow.

Figure 8 shows the feedwater flow response and Figure 9 presents the feedwater temperature response along with the actual plant data. The temperature response shows good agreement with the plant data except for the value at 1 min.

SUMMARY AND CONCLUSIONS

The BPA has been used to simulate the recent LaSalle-2 power oscillation incident. Extensive sensitivity studies were performed. The simulation results support the following conclusions:

1. The best-estimate conditions for LaSalle-2 led to limit cycle oscillations only.

2. The LaSalle-2 conditions within the uncertainty envelope produced growing oscillations leading to automatic scram as actually observed in the event.

3. The cause of the LaSalle-2 event was the coupled nuclear thermal-hydraulic instability originated by the density wave oscillations and reinforced by the reactivity feedback. The instability was brought about by the combination of:

   - Power peaking (especially radial peaking),
   - Flow reduction due to the recirculation pump trip,
   - Feedwater temperature reduction due to the partial feedwater heater loss.

4. The amplitude of the power oscillation remains bounded even after a postulated scram failure.

5. Reactor should be scrammed when LaSalle-type power oscillations occur.

6. These studies reinforce the importance of continued monitoring and controlling of peaking factors.

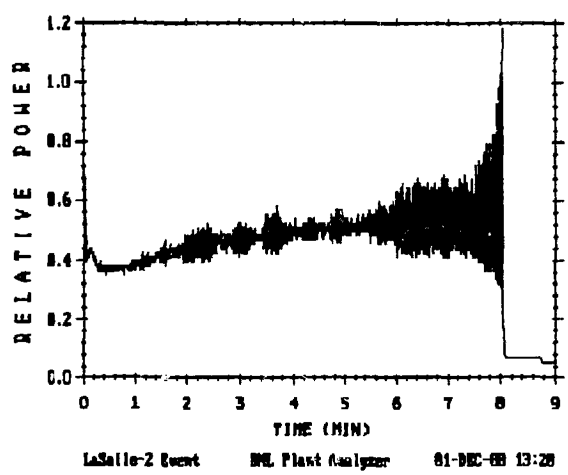

Figure 1 Simulated Reactor Power Oscillation for the LaSalle-2 Event.

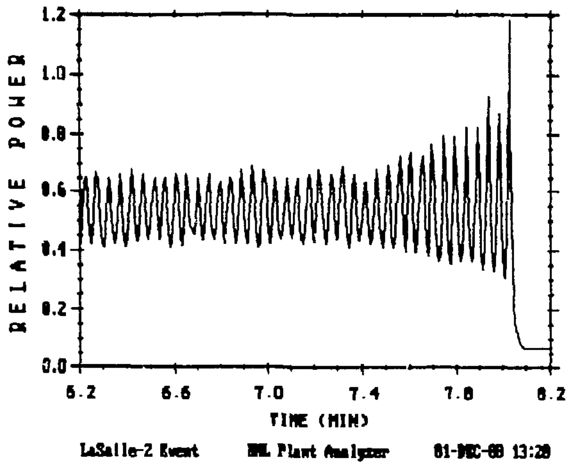

Figure 2 Zoomed Display of the Power Oscillation.

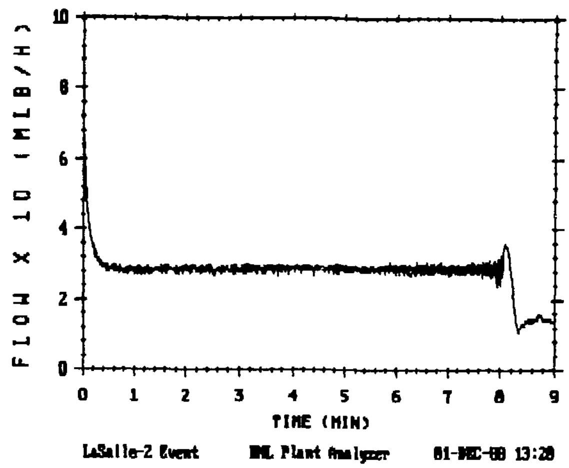

Figure 4 Simulated Core Flow Response for LaSalle-2 Event.

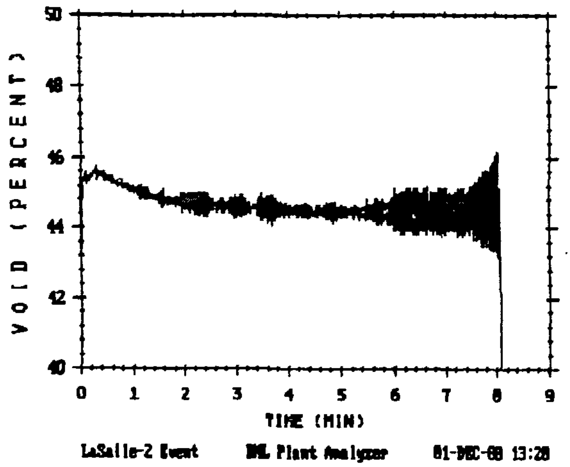

Figure 3 Oscillating Behavior of Core Average Void Fraction.

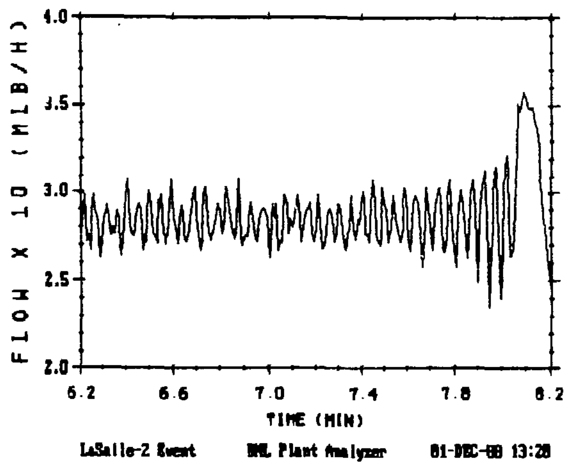

Figure 5 Zoomed Display of the Core Flow Oscillation.

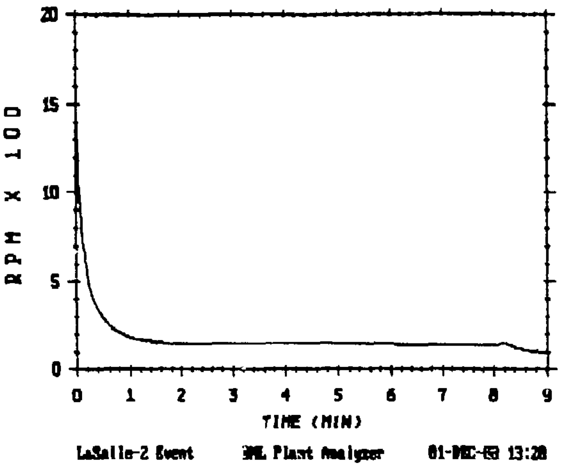

Figure 6 Recirculation Pump Speed Response.

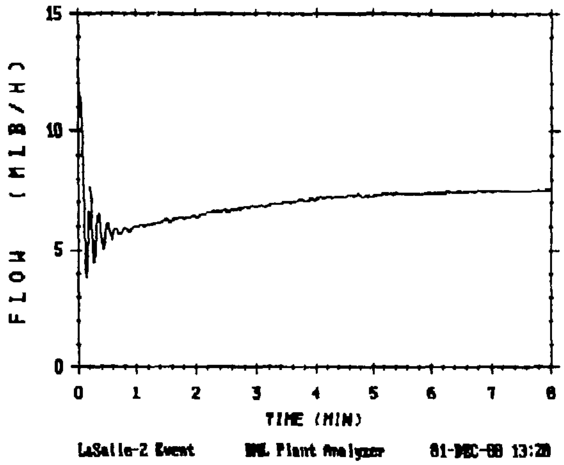

Figure 8 Simulated Feedwater Flow Response for the La-Salle-2 Event.

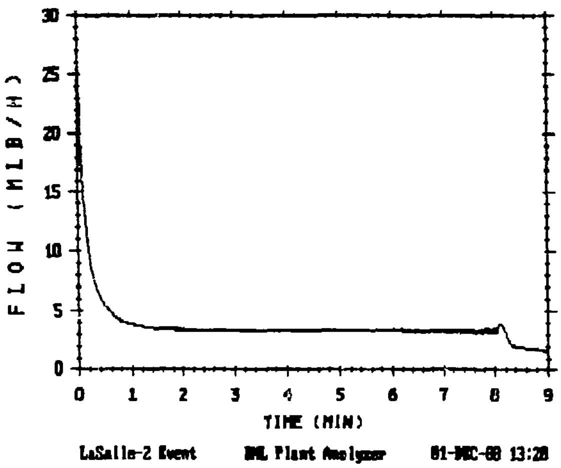

Figure 7 Recirculation Drive Flow Response.

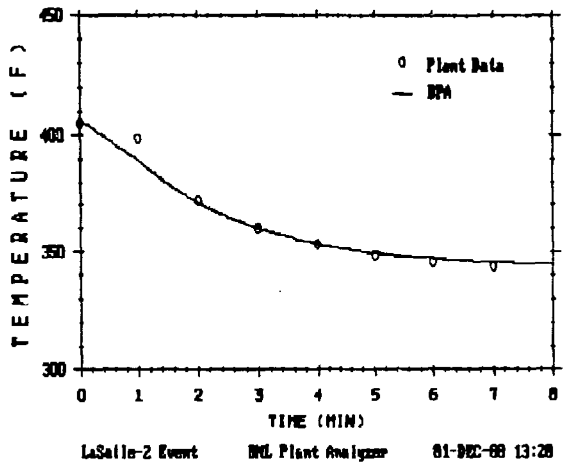

Figure 9 Feedwater Temperature Response for the LaSalle-2 Event

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