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Nineteenth Water Reactor Safety Information Meeting

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- Structural Engineering
- Advanced Reactor Research
- Advanced Passive Reactors
- Human Factors Research
- Human Factors Issues Related to Advanced Passive LWRs
- Thermal Hydraulics
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Bethesda Marriott Hotel
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BWR STABILITY ANALYSIS AT BROOKHAVEN NATIONAL LABORATORY*

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1. Abstract

The March 9, 1988 instability at the LaSalle County-2 BWR power plant at Seneca, IL was simulated, along with related ATWS scenarios and selected operator actions. The simulations were carried out on the BNL Engineering Plant Analyzer (EPA) [Wulff et al 1991], and for the purpose of resolving ten specific, NRC-defined questions on BWR instability, which are related to core-wide power and flow oscillations. It was demonstrated that the EPA is suitable for simulating large-amplitude, limit-cycle power and flow oscillations.

The EPA simulation of the LaSalle-2 instability identified the combination of low core flow, caused by the dual recirculation pump trips, low feedwater temperature due to the inadvertent feedwater heater isolation, and power peaking as a result of fuel burn-up, to be the three causes for this instability; the absence of any one of the three causes would have prevented the instability.

By simulating the LaSalle-2 instability with postulated scram failure, it was shown that the power oscillations peak at thirteen times rated power; the peaks can reach sixteen times rated power, if all feedwater preheating is lost after a turbine trip, while feedwater flow is controlled to maintain coolant inventory in the vessel.

Ten ATWS scenarios were simulated on the EPA under conditions of existing oscillations, or conditions inducive to instability, showing that the time it takes the suppression pool to reach its temperature limit can vary between 4.3 minutes (Turbine Trip without Bypass and no feedwater pump trips) and infinity (with Boron injection). Three additional transients have been simulated to show that restarting both recirculation pumps at the occurrence of an instability leads to scram; however in the case of scram failure, the power oscillations subside after peaking shortly at approximately five times rated power.

The power vs. flow map of the LaSalle-2 plant was also reproduced at five lines of constant control rod positions. The LaSalle-2 stability boundary was established with the EPA and confirmed within ±15% accuracy by comparisons with the results of the frequency-domain code LAPUR of Oak Ridge National Laboratory. Comparisons of EPA simulation results with plant data from three Peach Bottom Stability tests show an agreement, based on mean and standard deviation, of -10±28%, -1±40% and +28±52% (low power) in the gain of the pressure to power transfer functions. This demonstrates that the time-domain code HIPA in the EPA is capable of simulating instabilities.

Modeling parameters were ranked in the order of their significance to stability. The influences of spatial increments and of time steps in the numerical solution techniques have been quantified, and the effects from using

*This work was performed under the auspices of the U.S. Nuclear Regulatory Commission

different integration routines have been assessed. It has been determined that the power amplitude is underpredicted by the factor of 3.5 if the dynamic simulation of the Balance Of Plant is omitted. It was also shown that thermo-hydraulic instabilities cannot be simulated by imposing any boundary conditions, because the instabilities are self-induced and strongly impacted by closed loop feedback and resonance mechanisms.

2. Introduction

2.1 Stability Issues

Following the unexpected, but safely terminated, power and flow oscillations in the LaSalle-2 Boiling Water Reactor (BWR) on March 9, 1988, the Nuclear Regulatory Commission (NRC) Offices of Nuclear Reactor Regulation (NRR) and of Analysis and Evaluation of Operational Data (AEOD) requested that the Office of Nuclear Regulatory Research (RES) carry out BWR stability analyses, centered around fourteen specific questions. The questions are motivated by the demand to meet the General Design Criterion 12 in Appendix A of the Code of Federal Regulation 10CFR50, which requires that reactor power oscillations violating Technical Specifications be either impossible by design, or detected and suppressed. Ten of the fourteen questions address BWR stability issues in general and are dealt with in this paper. The other four questions address local, out-of-phase oscillations and matters of instrumentation; they fall outside the scope of the work reported here.

2.2 Program Objectives

It was the purpose of the work documented in this report to answer ten of the fourteen NRC-stipulated questions. Nine questions are answered by analyzing the LaSalle-2 instability and related BWR transients with the BNL Engineering Plant Analyzer (EPA) and by performing an uncertainty assessment of the EPA predictions. The tenth question is answered on the basis of first principles. The ten answers are summarized in the next section.

Moreover, it was the objective of the work reported here to identify the effects, on BWR stability, from Balance of Plant components, primarily of systems which affect the steam and feedwater conditions, and also, to determine the consequences from operator actions taken, when the reactor is unstable or undergoing flow and power oscillations.

Finally, it was the objective of the work presented here to assist the NRC in resolving issues as they arose in the review of industry-submitted reactor analyses related to BWR stability, by utilizing the EPA's flexibility for scoping analyses, its convenient in-line interactive access capabilities and the efficiency of its high simulation speed. Table 1 lists in the first column a summary of issues and concerns resolved by the EPA for NRR, while the last column lists the industry claims which first gave rise to the NRR requests shown in the second column, and then lead to the BNL actions and EPA results listed in the third column.

Table 1. MAJOR ISSUES RESOLVED FOR NRR BY THE ENGINEERING PLANT ANALYZER

ISSUE	NRR REQUEST	BNL-EPA ACTION & RESULT	Number of EPA Simul.	GE CLAIM
1. Scram Failure				
1.1. Rise of Time-Mean Fission Power with Growth of Power Amplitude During Limit Cycle Oscillations	Nov. 1988: EPA Simulation of LaSalle with Scram Failure	Dec. 6, 1988: 40% Rise in Mean Fission Power, or 2% rise per 100% growth in ampl. (confirmed by March-Leuba Jan. 1989)	12	Dec. 1988: no rise in mean power June 1989: yes, "slight rise".
1.2. Maximum Amplitude of Fission Power During Limit Cycle Oscillations	Nov. 8, 1989: Resolve Difference between BNL & GE Prediction 2,000 vs 200%	Jan. 9, 1990: Comprehensive Error Analysis Imposed GE B.C., Identified Important Effects of Resonance Feedback from Control System in generating large power amplitudes, up to 2000%	2	Oct. 1989: Max. Amplit. is 200% March 6, 1990: Acknowledge Importance of BOP Feedback.
2. Impact from Oscillations on ATWS EOP	April 15, 1990 & later: Simulate ATWS Scenarios as per NRR Requests	April 25, 1990: Simulated Nine ATWS Scenarios, Developed Event TREE, for Pool Temperature Heat-Up Time.* Identified two New ATWS Turb. Trip Scenarios with rapid pool heating.	9	Oct. 1989: No New ATWS Issues March 6, 1990: Revise ATWS EOP/EPG
3. Stability Boundary & Decay Ratio	Jan. 1990: Develop Power vs. Flow Map with 80%, 100% & higher Control Rod Positions	March 6, 1990: Mapped power vs. flow, confirmed with GE data, compared decay ratios along stability bdy. with Lapur: ±15% agreement.	34	Not Applicable
4. Code Assessment	EPA vs. LASALLE Comparisons, Simulation of Peach Bottom Stability, Error Assessments	Dec. 1988: LaSalle Simulation Parameter Ranking Effects from B.C. April 1990: Systematic bias estimates August 1990: Spectral Anal. for Peach Bottom Stability Tests.	21 10 4	Not Applicable
Total Number of Calculations for LaSalle Analysis *			92	-----

*see Figure 5 below

3. Results and Conclusions

The above objectives have been met. Over ninety transients of the complete BWR system have been simulated on the Engineering Plant Analyzer (EPA) and summarized in this paper (by including all the undocumented trials, code and input data revisions and corrections, one would arrive at over four hundred simulations). This number of simulations was performed (i) to resolve ten NRC-stipulated questions, (ii) to determine the effect of power oscillations on fuel and suppression pool temperatures for ten NRC-selected ATWS scenarios, (iii) to rank modeling parameters according to their impact on stability, (iv) to check out results of BWR stability-related analyses submitted to the NRC by the industry, and (v) to assess the EPA's capability of analyzing BWR stability. We summarize our results in the above order.

3.1 Resolution of Ten NRC Questions

The ten NRC-Stipulated Questions and their resolutions are summarized as follows:

Question No. 1: What are the causes of large amplitude oscillations and under what conditions can they occur in a BWR?

Answer: The instability at LaSalle-2 was a thermohydraulic instability and caused by the combination of three phenomena, namely by:

(1) core flow reduction due to the tripping of both Recirculation Pumps,

(2) radial power peaking and an axial power shape with strong bottom peaking as a result of fuel burn-up, and

(3) feedwater temperature reduction due to inadvertent closure of some of the valves admitting extraction steam to the feedwater heaters.

All three phenomena were necessary to cause the instability; according to the EPA predictions, the instability would not have occurred in the absence of any one of the above phenomena.

Question No. 2: What are the inherent limits, if any, on the amplitude of power oscillations in the case of scram failure? If limit-cycle oscillations occur, what then are the amplitude-limiting mechanisms?

Answer:

(1) The EPA predicts for LaSalle-2 conditions, but with postulated scram failure, power peaks as high as 13 times the rated power (see Fig. 1). Under circumstances with lower feedwater temperature (turbine trip, and no extraction steam for feedwater preheating) and no feedwater flow reduction (100 % Bypass flow, no operator intervention), the power peaks could be higher (up to 16 times rated power), as shown in Fig. 2.

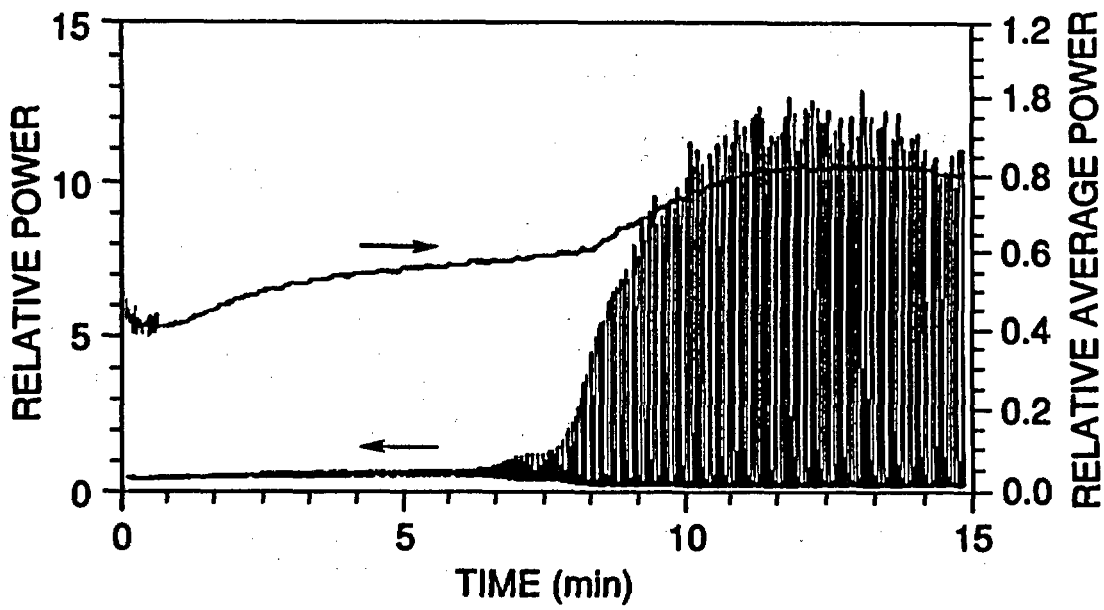

Figure 1. **Fission Power and Time-Averaged Fission Power (smooth top curve) for LaSalle-2 with Postulated Scram Failure. Feedwater controller maintaining inventory.**

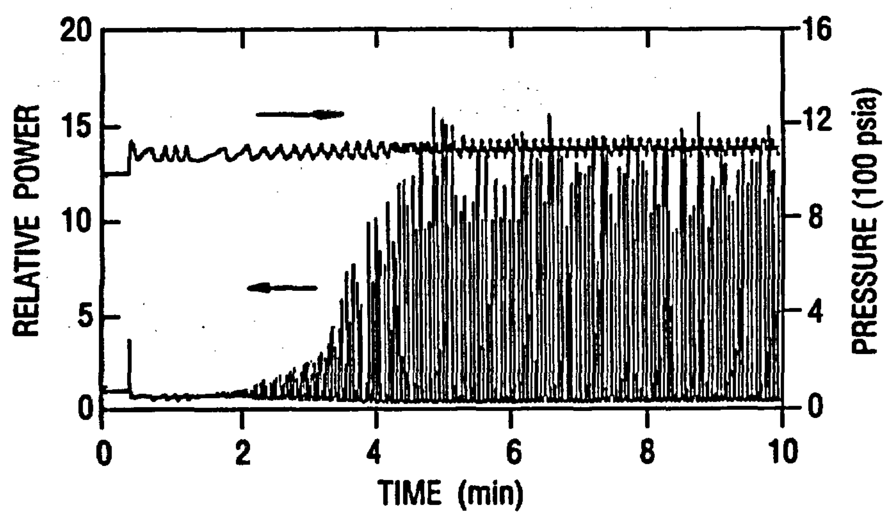

Figure 2. **Fission Power and Pressure for ATWS With Simultaneous Turbine and Recirculation Pump Trips, starting from steady-state, no operator interventions (Event Tree Code No. 1, Fig. 5.)**

Even higher power peaks are possible in the unlikely event that the feedwater control system failed in the maximum demand position after a turbine trip with 100% Bypass flow.

(2) Doppler and void reactivities limit the growth of the fission power amplitude (see Fig. 3). During large-amplitude, limit-cycle power and flow oscillations, the reactor remains subcritical on the average over an oscillation period, with the mean total reactivity of approximately -4.0 \$, while the instantaneous total reactivity swings between -9.3 \$ and +1.04 \$.

For very large power oscillations, both void and Doppler reactivity curb the fission power rise, but the Doppler reactivity feedback determines both peaking time and magnitude of the peak, because the Doppler reactivity drops off very sharply before the void reactivity peaks, as seen in Fig. 4.

Question No. 3: Can core-wide power and flow oscillations occur during any type of Anticipated Transient Without Scram (ATWS)? What effects can power and flow oscillations have during ATWS events, especially on Suppression Pool Temperature?

Answer:

Ten ATWS scenarios were defined by NRR and simulated on the Engineering Plant Analyzer. Figure 5 presents an overview of the scenarios and the EPA simulation results.

(1) An ATWS, caused by scram failure and the simultaneous tripping of turbines and both recirculation pumps would lead, without any further operator intervention, to:

(a) large core-wide power and flow oscillations with power peaks of 1,600 % of rated power, larger than are predicted for the conditions of the LaSalle-2 instability in 1988 (Fig. 2).

(b) the rise in the pool temperature to its limit of 353 K (or 80°C, 175°F) in only 7.2 minutes (see Fig. 5, Event Tree No. 1).

(2) Large limit-cycle power and flow oscillations in a BWR give rise to an increase in time-mean fission power above that which is attained during stable natural circulation after a dual recirculation pump trip. The rise is 2.2% for a 100% increase in peak power.

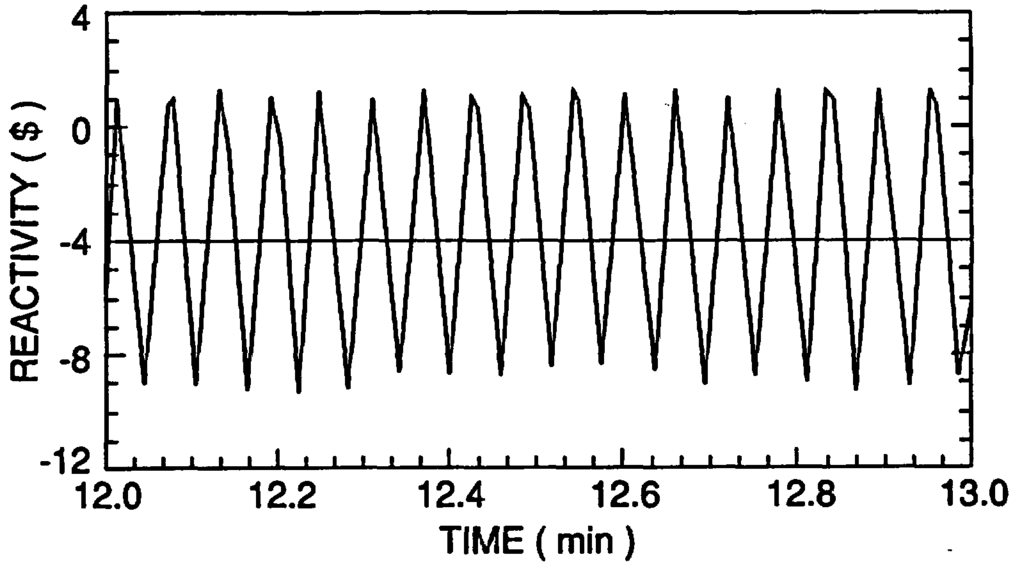

Figure 3. **Expanded View of Instantaneous and Time-Averaged Total Reactivity for LaSalle-2 Transient With Postulated Scram Failure.**

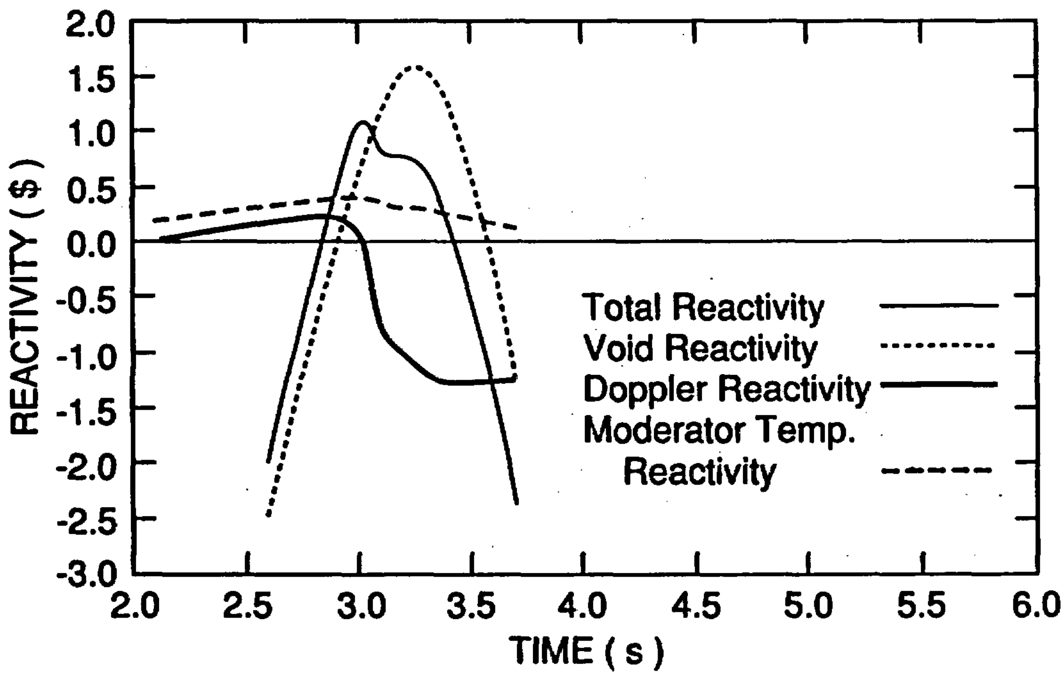

Figure 4. **Enlarged View of Reactivity Components and Total Reactivity, for LaSalle-2 Transient With Postulated Scram Failure.**

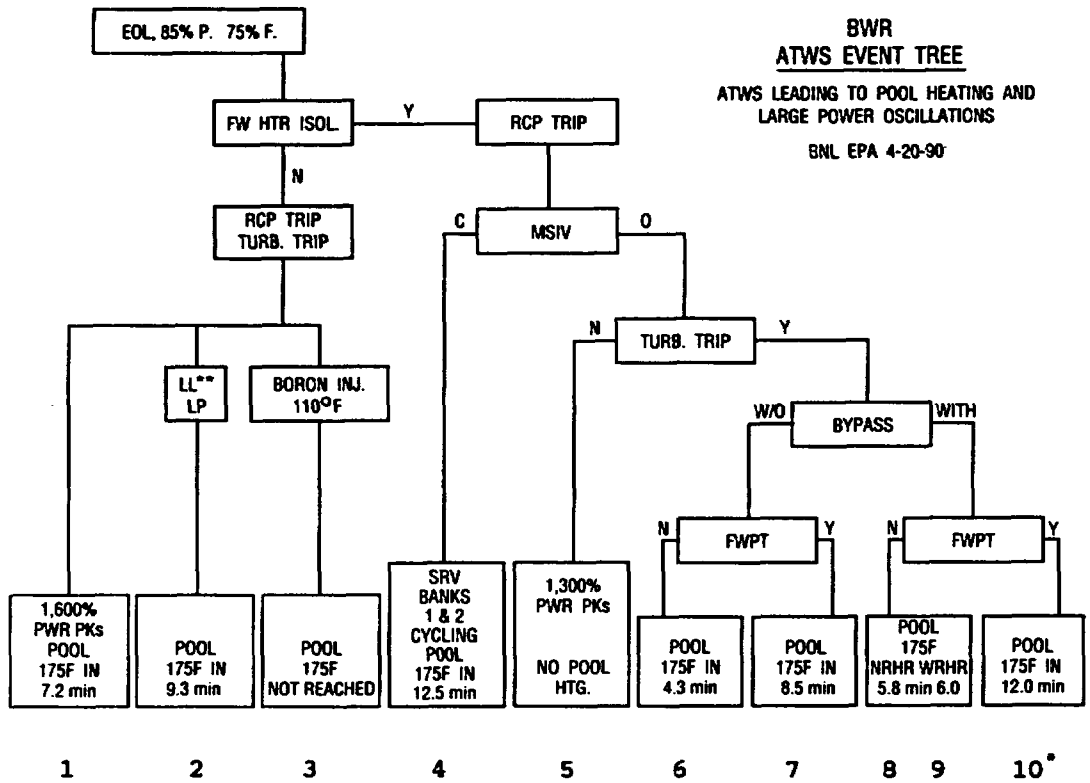

Figure 5. ATWS Event Tree (Scenario selection per USNRC-NRR Request). All EPA simulations started from LaSalle-2 conditions on March 9, 1988 prior to instability.

*) The numbers in the last row are the ATWS Event Tree Code Numbers shown for cross reference in the last column of Table 1.

**) Manual lowering of pressure and coolant level.

(3) If steam were to be discharged into the suppression pool during an ATWS with large power and flow oscillations in the core, then the elevated mean fission power would cause the suppression pool temperature to reach its limit of 353 K (80°C, 175°F) faster than it would during normal ATWS conditions with over-pressurization of the vessel. See Fig. 5 for the time spans.

Question No. 4: What are the amplitudes of fuel pellet and cladding temperature oscillations associated with limit-cycle power oscillations?

Answer:

(1) For the March 9, 1988 LaSalle-2 conditions, but with postulated scram failure, the EPA predicts the fission power to oscillate between 30 and 1,300% of rated power, and the fuel centerline temperature between 1,200 and 1,755 K (1,700 and 2,700 °F), while the fuel cladding temperature is oscillating between 563 and 569 K (554 and 565 °F).

(2) Under conditions with 100% Bypass flow and automatically controlled feedwater flow, the fission power is predicted to oscillate between 40 and 1,700% of rated power, the fuel centerline temperature between 1,033 and 2,088 K (1,400 and 3,300 °F), the fuel mean temperature between 726 and 1,089 K (850 and 1,500 °F), and the cladding temperature between 563 and 569 K (553 and 565 °F).

(3) Should the feedwater regulator fail in the full demand position, at 100% Bypass flow, then one would have to expect even larger temperature oscillations. The EPA predictions are based on a rewet model which could not be confirmed for periodic flow conditions, because no experimental data were available.

Question No. 5: Can the safety limit of minimum critical power ratio (MCPR = 1.05) be violated during limit-cycle oscillations?

Answer:

(1) The Minimum Critical Power Ratio does not fall below MCPR = 1.05 during power and flow oscillations, if the scram system shuts down the reactor before the power peaks exceed 118 % of full power. Then there is no fuel damage expected.

(2) If the scram system fails to shut off the reactor, then the safety limit will be temporarily violated. However, the widely accepted MCPR correlation in the EPA could not be confirmed for oscillatory flow conditions, because there were no data available.

Question No. 6: For Isolation events, how do the time rates of Suppression Pool temperature and of containment atmosphere temperature rise depend on the amplitude of limit-cycle power oscillations?

Answer:

(1) The Suppression Pool temperature rises whenever steam is discharged from the vessel into the pool, after the pressure in the reactor vessel exceeds the lowest relief valve pressure setpoint. The vessel is over-pressurized after Main Steam Isolation Valve (MSIV) closure or Turbine Trip with or without Bypass.

(2) The time rates of pool temperature and of containment pressure rise are, in the case of MSIV closure or Turbine Trip without Bypass, directly proportional to the rise in time-mean fission power generation. The time-mean rises 2.2% for every 100% increase in peak power, as can be seen in Fig. 6. The rise of time-mean fission power and consequent rises in suppression pool temperature and containment pressure (if there is no operator intervention to prevent pool saturation) are strongly affected by the systems of pressure and feed water regulations.

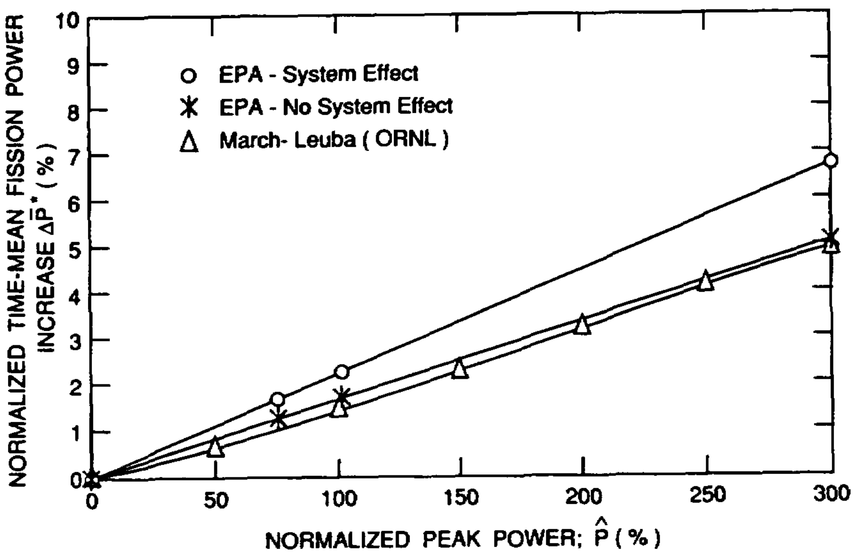

Figure 6. Dependence of Time-Mean Fission Power on Amplitude of Fission Power. Circles: EPA result with dynamic simulation of feedwater flow and temperature and of steam flow. Stars: EPA results, without system effects. Deltas: ORNL result, without system effects.

(3) The time the Suppression Pool takes to reach its temperature limit is given above under Question No. 3 (Answer Part 4), for ten different scenarios.

Question No. 7: Can suppression pool temperature and pressure exceed technical specification limits?

Answer:

Yes, but only in the unlikely event that several failures occur, and that there is no operator intervention. As shown in Fig. 5, the Technical Specification of Suppression Pool temperature limit (353 K, 80°C, 175°F) is exceeded according to the EPA predictions, in eight of the ten scenarios selected by NRR.

A dual Recirculation Pump trip with scram failure (ATWS), followed by a delayed Turbine Trip without Bypass, and without a feedwater pump trip (Event Tree Code No. 6 in Fig. 5) could cause the Suppression Pool temperature to rise at the rate of 11 K/min (21 °F/min), and after saturation is reached in the Suppression Pool water, the Suppression Pool pressure should rise at the rate of 3 bar/s (44 psi/ sec) up to the wetwell vent set point.

Question No. 8: Are available computer codes reliable for predicting BWR instability?

Answer:

(1) The Engineering Plant Analyzer (EPA) has been demonstrated to be reliable for predicting and analyzing BWR instability. This was demonstrated by assessing the EPA through comparisons with plant data, with results from the frequency-domain code LAPUR, and with results from analytical solutions and independent numerical analyses. The results are summarized in Section 3.5 below.

(2) Concerning NRC's Question No. 8 about other available computer codes being reliable for predicting BWR instability, we state that codes with

(a) implicit first-order Euler integration with respect to time, and upwind space differencing with zeroth-order interpolation, or

(b) stabilizer steps in the numerical integration algorithm [Rouhani et al 1988, pp. 2-1-14 to 15; Liles et al 1988, pp. 2-29 to 30], or

(c) artificial numerical viscosity in the finite-difference term for the momentum flux [Ransom et al 1985, p. 43; Dimenna et al 1988, p. 2-24], or

(d) time averaging over two successive time steps [Dimenna et al 1988, p. 7-58]

have inherent numerical damping which renders the predictions more stable than the power plant or the test facility.

Question No. 9: Are stability analyses useful if they are performed with imposed neutron flux oscillations?

Answer:

No. Thermohydraulic instabilities are self-exited and self-sustained, by internal forces, through internal, closed loop feedback mechanisms, which are in resonance with each other. Analyses performed with externally imposed fission power or any boundary condition, either at core or vessel boundaries (such as core inlet flow, system pressure, steam or feedwater conditions) are in general misleading. They are misleading because their results are dominated by imposed conditions. They lack the capability to account for resonance feedback effects; small deviations between natural and imposed frequencies and phase shifts have a strong influence on the prediction.

Question No. 10: When should frequency- or time-domain computer codes be used? When should point kinetics, and when should space-time kinetics codes be used?

Answer:

We answer this four part question without any calculations.

(1) Frequency-domain computer codes are based on linearized equations. They involve no time integration, as they are designed to obtain the stability boundary via the so-called Decay Ratio, or ratio of two successive amplitudes (which is a growth ratio, if greater than unity) from the leading eigenvalue of the characteristic systems

equation.

Having no time discretization errors, frequency-domain codes should predict more precisely the Decay Ratio than time-domain codes. With no need for numerical time integration of partial differential equations, frequency-domain codes are less expensive to use than time-domain codes.

Being based on linearized equations, frequency-domain codes are restricted to the determination of decay ratios and related parameters of linear perturbation analysis.

Thus, frequency-domain codes are superior to time domain codes in the restricted realm of their capability to predict decay ratios and stability boundaries.

(2) Time-domain codes are indispensable for the analysis of all nonlinear effects, i.e. for the determination of amplitudes of power, flow or fuel temperatures during oscillations, for the determination of plant responses to operator actions, malfunctions and functions of control systems.

(3) Both time and frequency-domain codes can be used effectively as scoping analysis tools, if they are designed for efficiency. This is known in the case of frequency-domain codes and demonstrated in this report for the time-domain code HIPA of the EPA.

(4) Computer simulations with point kinetics are suitable for analyzing core-wide, in-phase power and flow oscillations, provided the time-dependent radial and axial distortions of the fission power distributions can be modeled as in HIPA and confirmed through the use of plant data, such as the data from the LaSalle-2 instability of March 9, 1988.

(5) Computer simulations with three-dimensional neutron kinetics are indispensable for the analysis of region-wise, out-of-phase power and flow oscillations, as well as for the study of all transients with asymmetric power and flow distributions.

This completes the summary of our answers to the ten NRC-stipulated questions.

3.2 Effects of Oscillations on ATWS

Figure 5 above summarizes the results obtained from simulating the ten NRC-selected ATWS transients. The last row of blocks in Fig. 5 shows that, <u>counting from the times of turbine trips</u>, the suppression pool reaches the temperature limit

(1) in the **shortest** time of 4.3 minutes, if the feedwater pumps are not tripped but maintain the coolant inventory in the vessel, and if there is no extraction steam available for preheating the feedwater and thereby for preventing positive reactivity insertion through cold feedwater (see ATWS with Event Tree Code No. 6 in Fig. 5).

(2) in the **longest** time of 12 minutes for all **Turbine Trips**, if the feedwater pumps are tripped, after a Turbine Trip with Bypass (ATWS with Event Tree Code No. 10 in Fig. 5).

(3) in 12.5 minutes for the ATWS with MSIV closure immediately after the power oscillations reached their maximum amplitudes. This time is arbitrary to the extent that the time of MSIV closure is arbitrary (cf. Event Tree Code No. 4 in Fig. 5).

(4) never, if either the vessel pressurization is prevented as all the steam is passed through the still running turbines or through a large capacity bypass, or if Boron is injected after a turbine trip (cf. Event Tree Code Nos. 3 and 5 in Fig. 5).

For additional ATWS-related results from EPA simulations see also the answer to NRC Question No.3 in Section 3.1 above.

3.3 Parameter Ranking

Eight parameters representing models and operator actions have been varied and ranked. The parameters were varied by their range of estimated uncertainty, or by selecting alternative correlations. The parameter variations were carried out for the simulation of the LaSalle-2 plant conditions on March 9, 1988, and the results are summarized as follows:

(1) Of the parameters analyzed, the void reactivity coefficients, the axial and radial peaking factors, then the form losses and direct heating, have the strongest impact on stability.

(2) The oscillation period is affected strongly by the void coefficient, the fuel response time (fuel rod diameter), both peaking factors and direct (gamma) heating. The mean power is sensitive to fuel response time, axial power shape, void coefficient and to fluid friction (wall shear). The mean flow is sensitive only to wall shear and axial power peaking.

EPA simulations of ATWS scenarios demonstrated that:

(3) Reactor stability is strongly impacted by the predictions of mixture level elevation and of steam condensation in the space of the downcomer, between the feedwater spargers and the mixture level. The more steam is condensed onto the subcooled feedwater, the smaller is the feedwater subcooling temperature at the core entrance, and the more stable is the reactor.

(4) Predictions of fission power peak are strongly affected by system effects, and therefore by the models for control systems, valve dynamics, rotating machinery (feedwater pumps and their turbines) and thermal responses of heat exchangers. For more details, see Section 3.6 below.

3.4 Vendor Submittal

Table 1 shows how the EPA assisted the NRC to confirm or reject claims made by the industry in conjunction with their efforts to resolve the stability issues which were raised after the LaSalle-2 instability.

The last column of Table 1 shows the claims advanced by the industry at first, and the changes of their positions afterward. The most notable position change is the recognition of new ATWS issues, based chiefly on the EPA simulation of the ATWS with Event Tree Code No. 1 in Fig. 5 above.

3.5 Assessment of Engineering Plant Analyzer

(1) The BNL Engineering Plant Analyzer (EPA) has been assessed for stability-related analyses with

(a) plant data from three Peach Bottom Stability Tests, for **small-amplitude** oscillatory transients,

(b) plant data obtained from the STARTREC system during the LaSAlle-2 instability of March 9, 1988, for **large-amplitude** oscillatory transients,

(c) the vendor-established power vs. flow map at the Control Rod Lines of 80 and 100% of full-power rod positions.

(d) results from the frequency-domain code LAPUR,

(e) results from analytical solutions and separate effects analyses by numerical methods, and

(f) results from RAMONA-3B simulations of six FRIGG stability tests, by indirect comparisons based on the identities between the modeling and numerical methods in RAMONA-3B and in HIPA of the EPA.

(2) The comparisons of EPA results with STARTREC data, show that the EPA with its High-Speed Interactive Plant Analyzer (HIPA) code can be used for analyzing reliably core-wide, in-phase thermohydraulic instabilities as had occurred at LaSalle-2 on March 9, 1988. The EPA simulates such BWR instabilities, using only documented best-estimate modeling parameters, without any destabilizing boundary conditions, modeling changes or simulated operator actions which did not happen. There was no "code tuning" used to match EPA predictions with plant data. This comparison of EPA results with LaSalle-2 plant data is unprecedented in detail and scope among comparisons published to date, between available LaSalle-2 plant data and a computer simulation. Inasmuch as the LaSalle-2 instability event provides the only available, large-amplitude test data, this comparison is crucial for assessing the code's fidelity.

(3) Seven parameters are available from the STARTREC plant recording system and were compared with EPA predictions: fission power, reactor vessel pressure, core inlet flow rate, coolant level in the downcomer, feedwater temperature and the flow rates of steam and feedwater. The following results were obtained:

(a) The timing of the power oscillations was correctly reproduced, i.e. the onset of oscillations and the occurrence of scram trip are predicted correctly.

(b) The EPA-predicted growth of the power amplitude prior to reactor scram could not be compared with plant data, because the continuous strip chart recordings did not cover a sufficiently long time span of the transient.

(c) The EPA-predicted frequency of the power and flow oscillation is 11% smaller than the recorded frequency, and the predicted time-mean fission power prior to scram is 4.7% larger than the (presumably time-averaged,) recorded fission power, with the standard deviation of 4.2%.

(d) The EPA-predicted reactor pressure agrees with the STARTREC recording within 2.4% of the recorded value.

(e) The predicted core inlet flow agrees with the plant data within plotting accuracy.

(f) The EPA-predicted collapsed-liquid level differs from the recorded level by 7 cm (2.8 in), with the standard deviation of 9 cm (3.5 in).

(g) The EPA-predicted feedwater temperature agrees within plotting accuracy with the recorded feedwater temperature.

(h) The EPA over-predicted the steam and feedwater flow rates by 36±8% and 22%, respectively, because the feedwater regulator failure, which had actually occurred, could not be simulated due to a lack of information.

The differences between plant data and EPA-predicted steam and feedwater flow conditions reflect the uncertainty of available information about the plant conditions just prior to component failures, and during the instability.

(4) The assessment of the EPA with plant data from the Peach Bottom Stability Tests demonstrates that the EPA is reliable also for analyzing small-amplitude oscillations. The assessment is for small amplitudes, but it encompasses neutron kinetics, thermal fuel response, coolant thermohydraulics, and control systems.

(a) The EPA predicted for Peach Bottom Tests PT1, PT2 and PT4 the gain of the power to pressure transfer function with the biases and standard deviations of -10 ±28%, -1 ±40% and +28 ±52%, respectively.

(b) The respective frequencies at peak gain were predicted with errors of +6%, +3% and -28%.

The differences between experiments and EPA predictions are comparable to the associated experimental uncertainties, they are larger than the discrepancies between RAMONA-3B results and data from non-nuclear FRIGG experiments [Rohatgi et al 1990,1991] and they are all smaller than the differences between non-nuclear test data and results from the frequency-domain code NUFREQ, as reported by Yadigaroglu [Delhaye et al 1981, p. 376], and from the time-domain code TRAC-BF1 [Rouhani 1990]. No assessment appears to have been published that is similar to the comparison of a time-domain code, such as HIPA in the EPA, with spectral analysis data from a nuclear reactor power plant.

(5) EPA predictions are in agreement with the vendor-supplied power vs. core flow relationship along the constant Control Rod Lines (see Fig. 7).

(6) The comparison between the EPA results and the results from the frequency-domain code LAPUR have shown, that the two codes agree in locating the stability boundary within ±15 % of the decay ratio. To the best of our knowledge, there has been no comparison published for any other time-domain systems code with a better agreement.

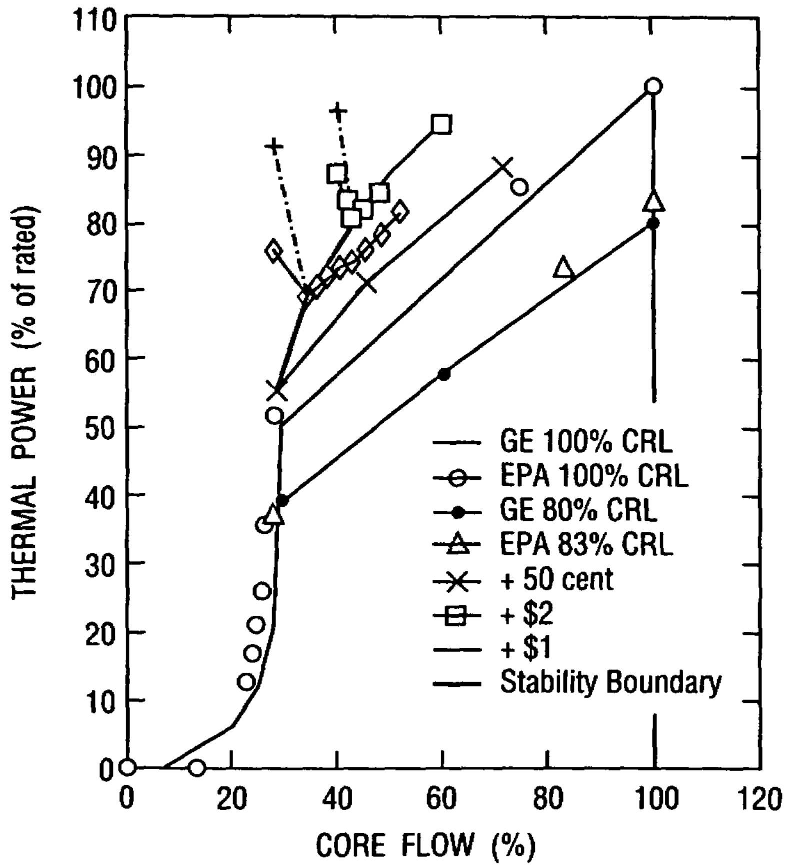

Figure 7. Comparison of EPA-Predicted With Official POWER vs. FLOW MAP, and EPA-Predicted Stability Boundary.
(solid square and diamond are unstable conditions obtained with fixed feedwater conditions, crosses with dynamically simulated feedwater conditions; total of 33 simulations)

(7) The analysis of uncertainties of modeling, computing and controlling boundary conditions in the EPA and showed that:

(a) The leading contributor to **modeling** uncertainty in simulating limit-cycle power oscillations in the EPA is the lack of experimental information on rewetting. The uncertainty spans the difference between limit-cycle oscillations with temporary, local dryout on the one hand, and on the other, an escalating fuel temperature with clad melting.

(b) Significant uncertainties in power peak predictions arise from the **modeling** of Doppler (±31%) and void (±25%) reactivities, of radial peaking (±17%), of gap conductance (+10%, -100%), and fuel heat transfer (+8%).

(c) The only significant uncertainty from numerical methods comes from numerical diffusion. It is estimated indirectly to be less than 15%. The EPA simulation results were demonstrated to be independent of the algorithm selected for time integration.

(d) Setting aside the uncertainties from simulating the consequences of operator actions, or of control system failures, the prescription of boundary conditions introduces an error as large as the factor of 3.5. Steam and feedwater conditions must be dynamically simulated for analyzing nonlinear effects on stability in a BWR.

(8) Figure 8 shows the comparison of the results from two different computer codes. The amplitude difference shown in that figure may be considered as an indication of the uncertainty encountered with current computer codes in predicting fission power amplitudes.

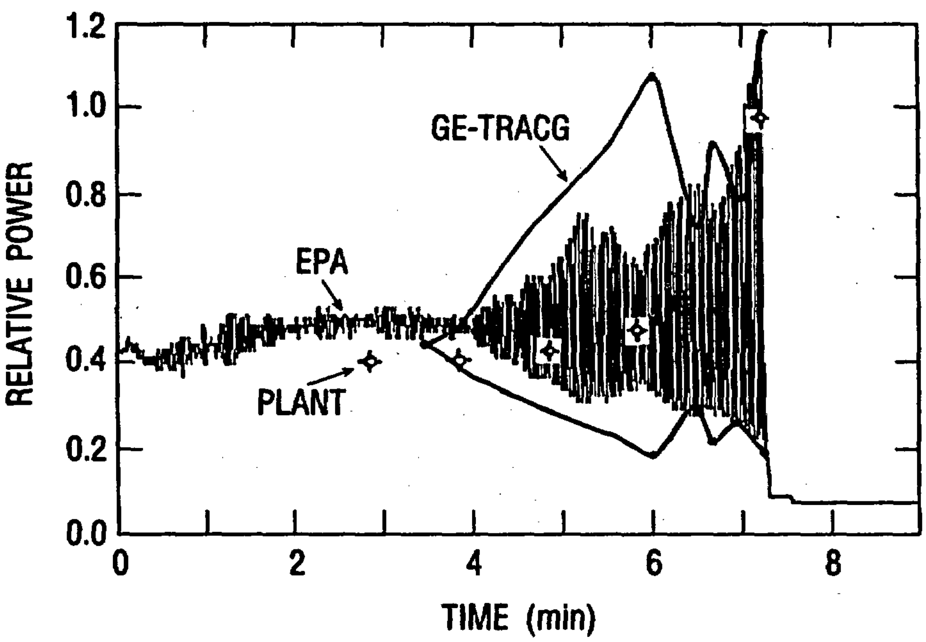

Figure 8. Comparison of EPA-predicted Fission Power (thin line) with TRACG (GE) Prediction, carried out with imposed steam and feedwater flows (bold envelope) and Discrete Plant Data (circles), for LaSalle-2 Instability Event.

3.6 Importance of System Effects

(1) Figure 7 demonstrates the strong influence of system effects on the relation between fission power amplitude and time-mean fission power. This relation is very sensitive to minute changes in the wave form of the power history, which in turn depends on system effects. System effects simulation cannot be replaced by imposition of boundary conditions.

(2) As in Item (1) above, Fig. 7 demonstrates the same strong influence that boundary conditions have on the time-mean of fission power: dynamically simulated steam and feedwater conditions produce much higher mean fission powers than statically imposed ones.

(3) The imposition of static boundary conditions, instead of their dynamic simulation, has been shown to result in an under-prediction of the fission power peaks by the factor of 3.5. Two important phenomena are excluded from, and one new phenomena is introduced in, the simulation by the imposition of static boundary conditions:

(a) The feedwater control system maintains the reactor coolant inventory for the increasing steam generation. The resulting **upward trend** of cold feedwater flow provides increasingly positive reactivity and increases further the power amplitude and the mean fission power which, in turn calls on the feedwater regulator to increase the feedwater flow. This trend continues until negative void reactivity feedback balances the positive reactivity insertion through cold feedwater. By imposing a fixed feedwater mass flow rate in the calculations, one misses the **upward trend** and the resulting growth in fission power amplitude.

(b) The dynamic simulation of the pressure regulating and feedwater control systems produces pressure and feedwater flow oscillations, which resonate with the power and flow oscillations in the core and, through closed-loop feedback, exite and enhance the core power and flow oscillations.

(c) The coolant inventory is not maintained with an imposed fixed feedwater flow rate and the reactor restabilizes, because the coolant level falls below the elevation of the feedwater spargers in the vessel, thereby exposing the incoming subcooled feedwater to the saturated steam in the vessel dome. The feedwater is consequently heated up while it falls from the sparger to the coolant level in the downcomer; and the core inlet subcooling approaches zero. That means a reduction in reactivity and in power, which restabilizes the reactor. This all would not occur if the inventory were maintained.

Items a, b and c above constitute the three reasons for underpredicting the fission power amplitude in calculations with imposed fixed feedwater conditions. It must be recognized that even if information were available on the <u>trend</u> of the feedwater temperature and flow, and imposed as boundary condition, one would still miss the resonance effects explained in Item b above.

4. Recommendations

Time-domain analyses of limit-cycle oscillations in a BWR reactor require the simulation of at least ten minutes of actual time, except when overriding phenomena shut off the oscillations sooner. The amplitude of limit-cycle oscillations may not grow monotonically; it may grow with amplitude modulation for a long time because of the effects from the slowly responding Balance of Plant. By terminating the simulation prematurely, one misses the true maxima of the power amplitude.

The analysis of BWR stability, primarily of its nonlinear phenomena, must include the dynamic simulation of system effects from the Balance of Plant.

Computer codes with first-order implicit Euler integration in time, and with donor-cell differencing in space should not be used for stability analyses, because of their inherent excessive numerical diffusion (see answer to NRC Question No. 8, in Section 3.1 above).

It is strongly recommended that experiments be performed on rewetting under oscillatory flow and heating conditions. These experiments are needed for the analyses of limit-cycle oscillations with large power amplitudes and local, temporary dry-out in the core.

5. Lessons Learned

(1) By far the greatest amount of resources was spent on the collection of plant data in support of code assessment. It was difficult and in part impossible to obtain reliable and consistent data from the Peach Bottom stability experiments. In spite of official and inofficial appeals, it was impossible to obtain consistent and complete information on the responses of the LaSalle-2 plant. Access to information for reactor analyses must be improved.

(2) Analytical solutions derived, and independent numerical analyses performed, to support the computer simulations and to assess the uncertainty of simulation results are indispensable for confirming efficiently any results of computer calculations or simulations.

(3) Given the resources that were available to the project, the work documented here would not have been possible, and the insight into the BWR plant behavior could not have been gained by means of traditional computer calculations, performed on general-purpose computers, with standard FORTRAN computer codes. The achievement was possible only with the simulation speed, the versatility and the in-line interactive access capabilities afforded only by a simulation environment, such as that of the Engineering Plant Analyzer.

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