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August 3, 1979

Mr. C. Burger
Structural Engineering Branch
Div. of Reactor Safety Research
U.S. Nuclear Regulatory Commission
Willste Building
7915 Eastern Avenue
Silver Spring, Maryland 20910

Dear Chuck:

Enclosed for your reference is a short paper published by Vreeland in the ANS transactions. On reexamining it, I see that his expression contains a condensation rate σ and is therefore not inertially limited. If an inertially limited expression is desired, you will have to reference our existing one-dimensional model. Similar models are reported in the cavitation literature for spherical geometries (e.g., Rayleigh bubble collapse).

It occurred to me that you may not have a copy of our technical paper on steam generator waterhammer. A copy is enclosed. It provides a convenient summary of our analytical efforts and may be helpful to you in discussing inertially limited processes.

Sincerely yours,

CREARE INCORPORATED

Paul

Paul H. Rothe
Senior Engineer

PHR/sr

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Enclosures: Vreeland, John A.; ANALYSIS OF WATERHAMMER OBSERVED IN FEEDWATER SYSTEMS OF CERTAIN PWR SYSTEMS.
Rothe, P. H., et. al.; WATERHAMMER IN THE FEEDWATER SYSTEMS OF PWR STEAM GENERATORS.

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which water flows vertically upward is surrounded by sodium flowing countercurrent in a concentric annulus. The instrumentation shown in Fig. 1 is located at fluid inlets and exits to the test section as well as along the inner tube wall. The test section and SGTF are ~90% complete at the time of writing.

1. H. C. STEVENS, "System Design Description for Steam Generator Test Facility—AI Test Section," ANL-G0033-0004-SA-00 (Apr. 1975).

5. Analysis of Water Hammer Observed in Feedwater Systems of Certain PWR Systems, *John A. Vreeland (Cal State Univ)*

This paper summarizes a quantitative analysis of the kind of water hammer experienced by several operating plants including Indian Point 2. Although the severity of the Indian Point experience has attracted a great deal of attention with NRC and ACRS recently referring to the "Indian Point 2 Type of Water Hammer," similar, though less destructive, events have been documented in more than a dozen plants with comparable designs.

To facilitate a comprehensive review of related phenomena, the Atomic Energy Commission requested of all licensed operators information concerning unanticipated piping movement and bulges in containment lines which may not have been previously reportable.[1] Responses to this request—generously made available by the Atomic Energy Commission—have been reviewed along with documentation in the individual dockets and contacts with representatives of the plant operators to fully understand the nature of the phenomenon.

It has been concluded that the water hammers are the result of the collapse of a steam bubble, as previously proposed,[2,3] although the initiating mechanism is quite different and requires the water level in the steam generator to be at the bottom of the feedwater sparger. An analytical description of this model has been developed and used to study various plant geometries. The phenomenon is found to be heat transfer limited and largely independent of the presence or magnitude of dissipative or nonconservative effects. For low levels of heat transfer, the volume of the steam bubble decreases to zero with a pressure differential remaining approximately equal to the dissipative effects. As the rate of heat transfer increases, the moving liquid acquires sufficient momentum that deceleration by the higher pressures is inadequate. Analysis of systems with feedwater line diameters between 12 and 18 in. indicates that condensation rates in excess of 200 lbm/sec are necessary before such inertial effects are observed at operating temperatures and pressures. If the system has a 20-ft^2 condensing surface and a feedwater temperature 450°F less than the saturated steam, a condensation coefficient of 78 lbm/(h ft^2 °F) is required. This exceeds empirical and theoretical data by a factor of 10 (Ref. 4). Since this coefficient for water has been shown to decrease rapidly with the time the condensing interface is exposed,[5] this factor could probably be understood in terms of the turbulence associated with high feedwater flow rates without postulating increases in heat transfer surface area. The depressurization produced by this condensation is predicted to last for approximately $\pi/24\ \sqrt{\alpha_1 \alpha_2}$, where α_1 and α_2 are the times required for sound to traverse the liquid replacing the steam and the steam bubble, respectively.

Following bubble collapse with high condensation rates, a pressure surge with a magnitude

$$(P - P_s)|_{max} = \frac{a \rho v_g}{9 \pi g} \frac{\beta}{D} \sigma \text{ psia}$$

occurs at about $\pi/2\,(4 + 7\beta)^{-1/2}\alpha_1$ sec after the bubble collapse and lasts for about $\pi \beta^{-1/2} \alpha_1$ sec. Here, a is the velocity of sound in ft/sec; ρ, the density of water in lbm/ft^3; v_g, specific volume of steam in ft^3/lbm; g, the acceleration of gravity in ft/sec^2; D, diameter of the pipe in ft; σ, rate of condensation in lbm/sec; and β, ratio of volume of bubble to volume of entire FW pipe. These expressions describe quantitatively all of the known experimental data related to these events and can be used as appropriate forcing functions in lieu of solving the far more complex problem on the computer.

1. Information Request No. 74-1, AEC Director of Regulatory Operation.

2. R. M. ROIDT, "Steam Water Slugging in Steam Generator Feedwater Lines," 74-7E9-FLINE-M1 (Jan. 1975); unpublished.

3. W. E. BENNETT, "Water Hammer in Steam Generator Feedwater Lines," NSD-TB-75-7 (June 1975); unpublished.

4. J. G. COLLIER, *Convective Boiling and Condensation*, p. 307, McGraw Hill (1972).

5. JOHNSTONE and SMITH, "Rate of Condensation During Short Exposures," *Intern. J. Heat Mass Transfer*, 9, 581 (1966).

6. The Influence of Secondary Failures on LMFBR Steam Generator System Design, *Joseph C. Mills (AI)*

The current philosophy in LMFBR steam generator design is to base the faulted design loads on a large leak sodium/water reaction (SWR) accident consisting of a single, resistance-free, instantaneous guillotine failure of a water tube followed by the simultaneous guillotine failure of the six surrounding tubes. This assumption, in many cases, has placed undesirable hardships on the design of the steam generator systems. The intent of this paper is to provide analytical results, which, combined with existing test data, imply that the simultaneous seven-tube guillotine failure assumption is overly conservative.

The definitive characteristics of secondary failures are: (a) the temporal spacing between each secondary failure; (b) the resistive nature of the failure; and (c) noninstantaneous failure openings. These effects were parametrically investigated using a recently developed[1,2] version of the TRANSWRAP[3] computer code. A simulated LMFBR secondary sodium system, the recently designed Large Leak Test Rig (LLTR) (Ref. 3) with a full-size 1700-psia evaporator, was modeled for the analysis. Several cases, each assuming a primary tube failure followed by nine secondary failures representing various characteristics, were analyzed. The first set of parametrics assumed that each of the secondary failures was resistance-free guillotines temporally spaced from 0 to 20 msec apart. The second set of parametrics consisted of full guillotine secondary failures spaced 5 msec apart with various resistive coefficients. The third set of parametrics provided for resistance-free secondaries spaced 5 msec apart, but assuming various noninstantaneous rupture opening profiles.

The results indicated that 40, 15, and 5% peak pressure reductions can be expected by individually allowing for the effect of each phenomenon (temporal spacing,