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Radioactivity Transport Following Steam Generator Tube Rupture

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

J. Hopenfeld

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ABSTRACT

A review of the capabilities of the CITADEL computer code as well as plant experience to project radioactivity releases following a steam generator tube rupture in PWR's shows that certain experimental data is needed for reliable off-site dose predictions. This article defines five parameters which are the key for such predictions and discusses the functional dependence of these parameters on various operational variables. The above parameters can be used in conjunction with CITADEL or they can be inserted in the appropriate equations which then conveniently can be programmed as a subroutine in thermal-hydraulic system codes. A joint Westinghouse, Electric Power Research Institute and Nuclear Regulatory Commission Program aimed at obtaining the five parameters empirically is described.

INTRODUCTION

Operating experience with PWR steam generators (Reference 1) shows that the tubes in these heat exchangers are sometimes prone to corrosion and mechanically induced damage. When such damage results in a leaking tube, the potential exists for radioactive fission products to escape into the atmosphere whenever a secondary side steam dump occurs. This occurrence has long been recognized and accounted for by setting operating limits on the radioactivity levels in the primary coolant so that even if a full tube rupture occurs the activity release will not exceed allowable amounts. Indeed, although six steam generator tube rupture (SGTR) incidents have occurred in the United States none has resulted in activity doses which exceed the limits set in 10 CFR Part 100. (300 rem to the thyroid in 2 hrs). However, experience with the SGTR at R.E. Ginna (Reference 2) suggests the assumption, commonly invoked in the analysis

of SGTR events, that only dry steam is released through the relief and safety valves may not be appropriate. This is because the concentration of fission products, primarily iodine, may be higher in the wet steam. The presence of water in the released steam can result from entrained water droplets in the steam which escaped removal by the separators due to low removal efficiency of the latter during the transient.

The inclusion of water release in the design basis analysis of SGTRs could potentially lead to stricter limits on the maximum allowable iodine concentrations in the primary reactor coolant. An inadequate understanding of radioactivity transport following SGTR may dictate the use of conservative calculation of the allowable iodine limits. Should these calculated allowables turn out to be very strict they may in turn impose undue restrictions on reactor operations.

In response to this background, Westinghouse, (W), the Electric Power Research Institute, (EPRI), and the NRC have recently initiated a cooperative program aimed at the generation of the required data base which would facilitate mechanistic modelling of SGTR as well as other steam generator events. The purpose of this paper is to set forth the requirements and describe the implementation of this program.

PREVIOUS WORK

1. THEORETICAL PREDICTIONS

Previous studies (References 3, 4, 5, 6) proposed a theoretical model for the transport of radioactivity, by micron size droplets, which are carried by bubbles into the steam space. These droplets are entrained by the rising bubbles when the primary coolant flashes into the secondary side of the steam generator. Starting with first principles of heat and mass transfer, the authors have developed equations to describe the behavior of the droplets within the bubble and their deposition on solid surfaces upon impaction. These equations were then programmed into a computer code (CITADEL) for use to predict radioactivity releases.

The CITADEL code divides the steam generator into three regions: water, steam, and moisture separators. Primary coolant from a submerged leaking tube is transported to the steam space by rising bubbles while a leak above the water level results in a flashing jet which introduces primary liquid droplets into the steam space.

A number of features in CITADEL must be assessed before the code can be considered as a reliable tool for radioactivity release predictions. First, the code contains numerous input parameters, such as droplet and bubble diameters, and iodine partition coefficients, which themselves are subject to large uncertainties making the evaluation of the results very difficult. Secondly the droplet transport mechanism requires that the primary coolant flash, upon entering the secondary side, an event which sometimes may not even take place. Flashing may occur prior to reactor scram because the primary coolant temperature is higher than the secondary coolant temperature and the secondary side is in a saturated condition. Following reactor scram, reverse heat transfer may arise and for certain accident scenarios the secondary coolant temperature becomes higher than the primary coolant temperature but since the secondary coolant is still saturated flashing does not occur.

Prior to reactor scram, however, the relief/safety valves are closed so only a negligible amount of radioactivity could escape via the ejector pump and, therefore, knowledge of radioactivity transport during this period is of secondary interest. Following reactor scram the relief valve opens periodically, but since no flashing takes place within the steam generator the CITADEL code will predict no radioactivity release due to liquid entrainment. If mechanisms other than flashing exist for liquid carryover, the CITADEL results may underestimate the actual radioactivity release.

2. OPERATING EXPERIENCE

More than 300 incidents of relatively small steam generator tube leaks have been recorded in this country (Reference 1). Since plants are not required to shut down when the total primary to secondary leakage is below one gpm some plants have operated with leaking tubes for sufficiently long time to allow

measurement of radioactivity build up and distribution on secondary side components.

Table 1 presents radioactivity measurements which were obtained for various time intervals at five different steam generator units. The Table presents concentration of I-131 in the primary coolant, the steam generator water and the main line steam. The last column in the Table represents the ratio rounded off to the nearest order of magnitude (i.e., 1, 10, 100, etc.), of the concentration of iodine in main line steam to the concentration of iodine in the steam generator water. At low iodine volatility the concentration of iodine in steam would be predominantly due to droplet entrainment and therefore this ratio would be a direct measure of liquid carryover. On the other extreme when the liquid carryover in the steam is very low the last column in the Table would represent the partition coefficient which physically characterizes iodine volatility.

Table 1 leads to two main conclusions. First there is a large variation from plant to plant in the iodine retention characteristics of the secondary system. These large differences could be possibly reconciled by differences in leak location, measurement techniques and differences in system operations. In the absence of a satisfactory explanation of the plant data one must conclude that present plant experience is not sufficient to allow extrapolation of such data to the design basis SGTR. Secondly, the large plant to plant variations in the steam to water Iodine concentration ratio does not allow one to conclude whether iodine reaches the steam line predominantly by entrainment or volatility. For example, since a typical W steam generator is rated at 0.2% moisture carryover, the activity ratio in Table 1 would not be expected to exceed 10^{-4}, (dividing .2% by 20 to allow for differences in densities between water and steam), yet it does exceed this value by two order of magnitudes for both Turkey Point Units 3 and 4. If one was to draw a conclusion from this that iodine concentrations in the steam are governed by volatility and not be entrainment than such a conclusion would stand in contradiction with the Point Beach data which indicates the opposite (i.e., iodine entrainment predominates volatility).

In conclusion, neither the present analytical tools nor the accumulated steam generator experience allows reliable best estimate predictions of radioactivity releases following steam generator tube rupture.

FORMULATION OF THE PROBLEM

In formulating the present program, several parameters have been identified which are deemed to be essential for assessing the CITADEL code or for laying down a foundation for a modified computer code if required. The approach selected here was to define parameters which can be measured and at the same time are related to known physical processes. These parameters represent average quantities in time and space, and strictly speaking, are valid only for the range of conditions which they were measured. This approach risks the possibility that some SGTR events could fall beyond the bounds for which these parameters were characterized; however, looking at the problem as a whole, and particularly its complexity, it was concluded that such an integrated approach would yield the most beneficial results. The above mentioned parameters can be programmed on a subroutine in a system code such as TRAC or RELAP and they can also be used to assess CITADEL code and modify as required.

Figure 1 is a schematic representation of the various routes which fission products may take from the tube rupture site to the relief valves. 2 and 6 represent volumes containing a uniform mixture of steam and water. 3, 5 and 10 represent wall surfaces to which fission products adhere by chemical or physical sorption or by particle deposition. 9 represents the liquid in the downcomer and the liquid in the pool above the tube sheet.

A mass balance on the heated and riser sections, the separators, and the downcomer allows calculations of the amount of iodine leaving the steam generator.

Heated section (above water level) and riser:

1. $$\frac{d(Mc)_H}{dt} = (mc)_p + (mc)_d - (mc)_s - S_H$$

Separators:

2. $$(mc)_o = (mc)_s - (mc)_\varepsilon - S_s$$

Downcomer:

3. $$\frac{d(Mc)_d}{dt} = (mc)_\varepsilon + (mc)_f - (mc)_d - S_d$$

In the above, M represents total mass within the specified volume, m is mass flow rate, c is the concentration of iodine per unit mass, and S represents the rate at which iodine either deposits on or redepcsits from the walls. The subscripts are defined in the nomenclature.

It should be noted that $m_i c_i = (mc)_i$ and the concentration c refers to iodine concentration in water/steam mixture or the concentration of iodine in liquid alone or steam alone when the subscripts ℓ and g are employed.

If K is specific activity (which converts iodine mass to radioactivity), then the total offsite dose is given by:

4. $$a = K \int (m_o c_o - S_o)dt$$

Equations 1-3 can be reduced into a more tractable form by introducing the following variables.

Fraction of primary liquid reaching the separators directly without prior mixing

5. $$\beta = \frac{(m_{p,\ell})_s}{m_{p,\ell}} = \frac{\text{mass of primary liquid entering separator}}{\text{mass of primary liquid entering secondary side}}$$

Overall separator efficiency for liquid removal:

6. $$\eta = \frac{m_{\ell\varepsilon}}{m_{\ell,s}} = \frac{\text{liquid entering the downcomer from the separator}}{\text{total liquid at the entrance to separator}}$$

Partition of iodine between gas and liquid:

7. (a) $$P \equiv \frac{c_\ell}{c_g}\frac{\rho_\ell}{\rho_g} = \frac{\text{mass of iodine per unit volume of liquid}}{\text{mass of iodine per unit volume of vapor}}$$

Overall mass transfer coefficient for iodine deposition on wall surfaces.

8. $$h_i \equiv \frac{S_i}{(\overline{AC})_i}$$

The iodine entering and exiting the separators can now be written as:

at the entrance to the separator

9. $$(mc)_s = \beta\,(1-x)\,(mc)_p + (mc)_H$$

$$= \beta\,(1-x)\,(m_\ell c)_p + (m_g + m_\ell)_H\, c_H$$

at the exit from the separator into the steam dome:

10. $$(mc)_o = [(mc)_g + (mc)_\ell]_o =$$

$$= \left[\frac{m_g}{P}\frac{\rho_g}{\rho_\ell} + m_\ell\right]_o c_\varepsilon$$

at the exit from the separator into the downcomer:

11. $$(mc)_\varepsilon = \eta m_{\ell,s}\, c_\varepsilon$$

The first term on the right hand side of equation 9 represents the mass of iodine in the primary liquid which did not participate in the mixing with the secondary side fluid. The second term in equation 9 represents the iodine carried by the steam/water mixture (m_H) in the riser section.

The first and second terms on the right of equation 10 represent the iodine in the steam and in the liquid respectively.

The Appendix discussed in more detail the derivation of equations 9 and 10.

Equations 1, 2, 3, 9, 10, and 11 represent a general mass balance and therefore are not restricted to a specific accident scenario. Some of the terms appearing in these equations are governed by different mechanisms depending on leak location and the time during recovery from the SGTR event. For example, the total moisture content $m_{\ell,s}$ entering the separator will be expressed differently prior and following steam generator isolation. In the first case the flow will be characterized by a bubbly flow while in the second case by a misty flow. As an example only we shall proceed and consider the second case when the liquid entering the separator is in a form of entrained droplets and the break site, under all conditions, is covered by the steam/liquid mixture. This situation is especially of interest because it is only during this period of time that the steam relief value allows the escape of the contaminated steam/water mixture into the atmosphere. Prior to S.G. isolation the escape of iodine to site boundaries occurs through the steam air ejectors.

During the time period when the relief valve is open equation 11 can be written as:

11a $(mc)_{\varepsilon} = (\beta\ (1-x)\ m_p + E_s)\eta\ c_{\varepsilon}$

The term $\beta\ (1-x)\ m_p$ represents the primary liquid entering the separator while E_s represents the liquid entrained by steam rising from the saturated liquid pool. The term m_H in equation 9 can now be expressed as

12 $m_H = m_g + E_s$.

A substitution of equations 5-11 in equations 1-3 results in three equations for the three concentrations c_{ℓ_H}, c_{ε}, and c_d.

12. $$\frac{d(Mc)}{dt}_H = (1-\beta)\ (1-x)\ (mc)_p + (mc)_d - (m_g+E)c_H - (hAc)_H$$

or

12a. $$\frac{d}{dt}\left[c_{\ell}\left(M_{\ell}+\frac{1}{P}\frac{\rho_{\ell}}{\rho_g}M_g\right)\right]_H = m_p c_p\left[(1-\beta)(1-x)+\frac{1}{P}\frac{\rho_{\ell}}{\rho_g}x\right]_p +$$

$$c_d m_d - c_{\ell H}\left[E+\frac{1}{P}\frac{\rho_{\ell}}{\rho_g}m_g\right]_H - (hAc)_H$$

13. $$\left[\left[\frac{m_g}{P_H}\frac{\rho_{\ell}}{\rho_g}\right]+1-\eta\left[(1-x)\beta m_p + E_s\right]\right]c_{\varepsilon} =$$

$$\beta(1-x)(mc)_p - \eta\left[E_s+(1-x)\beta m_p\right]c_{\varepsilon} + c_{\ell H}\left[E_s+\frac{1}{P}\frac{\rho_{\ell}}{\rho_g}m_g\right]_H -$$

$$(hA)_s\frac{\left[(1-x)\beta(mc)_p + c_{\ell H}\left[E_s+\frac{1}{P}\frac{\rho_{\ell}}{\rho_g}m_g\right]_H\right]}{\beta m_p + m_g + E_s}$$

14. $$\frac{d(Mc)d}{dt} = \eta\left[\beta(1-x)m_p + E_s\right]c_{\varepsilon} + (mc)_f - (mc)_d - (hAc)_d$$

A total of 21 parameters are required to solve these equations, 8 parameters (M_g, M_{ℓ}, m_g, m_{ρ}, $\left(\frac{\rho_{\ell}}{\rho_g}\right)$, m_d, x) are obtainable from a mass and heat balance which can be performed by RELAP or TRAC, the three surface areas (A_s, A_d, A_H) are known from geometrical considerations and the two concentrations of iodine (c_p, c_f) are given inputs. The remaining 8 parameters β, η, P_p, P_H, h_s, h_d E, are obtained as discussed below. It should be noted that β and E are also required inputs for the total mass and heat balance.

Again, equations 12 through 13 are written for a certain period of time during the transient and a certain location of the break site. For other situations these equations can be relatively easily modified.

DATA REQUIREMENTS

1. Primary fluid mixing - β

The degree to which the primary fluid mixes with the secondary fluid depends primarily on break location. A break near the tubesheet will result in extensive mixing ($\beta \rightarrow 0$) in comparison to a break at the top of the U bend ($\beta \rightarrow 1$) where little mixing exists. The difficulty in estimating β between these two bounds stems from the hydrodynamics of the jet at the break site. Prior to reactor scram when the secondary side is in the circulating mode a leak at the top of the tube bundle, will cause a flashing jet of primary fluid to mix with a flowing two phase mixture of secondary fluid. A leak at the bottom of the tube bundle, near the tube sheet, will cause a flashing jet of primary fluid to mix with a flowing subcooled secondary liquid. Following reactor scram and steam line isolation, a leak located anywhere in the tube bundle, will result in either a flashing or nonflashing jet of primary fluid, to enter stagnant pool of saturated liquid. The direction of heat flow between the primary and the secondary coolants will determine whether flashing will or will not occur. At the beginning of the transient the primary fluid is hotter than the secondary fluid, however, as the primary fluid is cooled by steam release through the intact nonleaking steam generators, the secondary side of the faulted steam generator becomes hotter than the primary fluid and therefore reverse heat transfer takes place. The enthalpy of the secondary fluid is now higher than the enthalpy of the primary fluid and therefore no flashing would occur.

Because of the complex flow geometry it is difficult to estimate how much liquid will actually reach the separator and, furthermore, it is even more difficult to determine what fraction of this liquid was originated during flashing and what fraction originated from the secondary side.

Data reported in the literature on aerosol scrubbing by suppression pools, jet mixing in simple geometries, droplet distribution from flashing jets, were all obtained under condition which are significantly different from those existing in a steam generator during shutdown following a tube rupture.

2. Separator Efficiency - η

PWR steam separators consist of two devices commonly known as the primary and the secondary separators. The primary separator employs a swirl vane which utilizes centrifugal force to separate water droplets from steam. Such separation depends on flow velocities, droplet size, droplet size distribution, and droplet concentration. The secondary separator, also commonly referred to as the steam dryer, is constructed from corrugated plates and separation is achieved by impaction of droplets on these plates. As steam velocity increases, a point is reached where the steam reentrains water droplets from the wall liquid film and consequently the dryer loses its effectiveness. Separators are very effective in drying steam near or at their design conditions. The SGTR events, however, may give rise to flow velocities and particle loadings which are outside the design envelope and, therefore, tests are required to establish separator efficiencies for these off-design conditions. Of particular concern here are (a) loadings with droplets which are less than 5 microns in diameter, (b) large steam velocities immediately following steam line break, (c) very high liquid loadings, and (d) flow reversal in downcomer. Because of the uncertainties in scaling laws, separator efficiencies under these off-design conditions can be obtained only from prototype testing.

3. Pool Entrainment - E

When the condenser is not available for steam dump following SGTR, after a brief period of time the secondary side pressure reaches a certain level and is maintained at that level by periodic steam releases through the relief valve. Each time the valves open the water pool level swells due to rapid void expansion on depressurization. The mechanism for liquid entrainment under these conditions may be different than the mechanism for liquid entrainment from a boiling pool at constant pressure where entrainment results from the breakup of liquid ligaments and protrusions following bubble bursting at the surface. If the entrainment mechanism during depressurization indeed is different than the entrainment mechanism during boiling, the available data on entrainment rates (Reference 5) maybe not be applicable. Tests are required to check this point.

4. Mass Transfer Coefficient - h

Molecular iodine, I^-, and possible IO_3^- can be either physically or chemically adsorbed on surfaces which are immersed in water or steam. Metal composition, coolant chemistry, temperatures and Reynold numbers are the main parameters which could control the deposition of these species on steam generator surfaces.

5. Feed Water Concentration - c_f

The concentration of iodine in feed water depends on the history of the SGTR event under consideration. Prior to reactor scram, iodine contaminated steam from the faulted unit is the source of iodine in feed water. A mass balance on the entire secondary system is required to determine the feed water concentration c_f. Therefore, correlations for h should also include the appropriate conditions existing in the secondary system outside the faulted unit.

6. Partition Coefficient - P

When the steam at the outlet from the steam generator is very dry (X=1), steam will be the sole carrier of iodine into the environment, and its amount is governed solely by the partition coefficient. When the steam is wet, the magnitude of the partition coefficient will determine whether for a given steam quality water or steam will be the predominant carrier of iodine. Figure 2 compares the relative contributions of iodine in steam and iodine in water to the total radioactivity release. Unless the moisture in the steam exceeds 1% most of the iodine is carried by steam for $P<1000$. If the steam, on the other hand, contains more than 15% moisture ($x<.85$) entrained liquid will control iodine release. For $P>10^5$ even minute amounts ($x\sim.99$) of entrained liquid contribute significantly to iodine release. Presently P is not sufficiently known to determine which mode of iodine predominates.

Following a steam generator tube rupture accident, the aqueous/gaseous iodine partitioning will be determined by iodine hydrolysis and the effect of radiation. Both hydrolysis and radiolysis are, in turn, influenced by system pH, temperature, concentration, and impurities. The overall reaction for iodine hydrolysis may be written as

$$3I_2 + 3H_2O = 5I^- + IO_3^- + 6H^+.$$

Rapid iodine hydrolysis is favored by high pH. Styrikovich et al., Reference 7, have published the results of studies of the partitioning of iodine between steam and water. The measurements were made on saturated steam and pressures of 0.19, 0.4, and 1 MP_a. Their results show that I_2 volatility decreases with increase in pH. The partition coefficients measured in these studies with an initial form of I_2 varied from 17.0 to 25000 depending on the pH (6-10) temperature (118°C, 143°C, 179°C), and time. Reference 8 indicates that several hours may be required before the partition coefficient reaches a steady state value. Since the SGTR shut down event takes much less time, the use of equilibrium partition coefficient as published in the literature may not be justified.

In addition, radiation effects have the potential to override hydrolysis in controlling iodine volatility (Reference 9). Radiation can change the chemistry of the system by producing a host of products such as OH, H_2O_2, H, and e^-_{aq} that can react with iodine species. At low pH (<N7), the hydroxyl radical, OH, or hydrogen peroxide, H_2O_2, can react with I^- to form I_2. Conversely, at high pH the hydrogen peroxide can convert I_2 to I^-

It should be noted that iodine concentration in the primary coolant can vary by two orders of magnitude following power changes (Iodine spike) and, therefore, one can not be certain that equilibrium iodine concentrations have been reached during the SGTR event.

Coolant impurities, pH, and radioactivity levels are the main variables to be investigated.

7. THERMAL-HYDRAULIC

The thermal-hydraulic history of the SGTR event provides the boundary conditions and flowrates required to solve Equations 1-3. Figure 3 shows typical mass flows, temperatures and pressures during an SGTR event calculated by TRAC. These boundary conditions strongly depend on the accident scenario as can be seen from Figures 3 and 4 which make comparison of two different accidents. In the base case the temperature of the secondary coolant is higher than the

temperature of the primary coolant and consequently no flashing occurs when the relief valve is opened. In the second case (safety valve stuck open), the primary temperature exceeds the secondary liquid temperature and flashing does occur. These two different accident scenarios could result in different radiological consequences if flashing is the main source of radioactivity release as currently calculated by the CITADEL code. On the other hand, if flashing is not the predominant mechanism for iodine release, less pronounced dependence on accident scenarios is expected.

PROGRAM DESCRIPTION

The variables governing h, β, η, E, P together with the expected occurrence of a given SGTR accident provide the required input for the implementation of the overall program objectives: (1) obtain the required data base to assess the ability of CITADEL to estimate offsite radioactivity releases following various postulated SGTR events; and (2) optimize the mix between full size prototypic tests and small scale laboratory tests to minimize cost.

The scope of the present program is limited to the W type steam generator. The partition coefficient P is equally applicable to once through steam generators however, the mixing parameter β and the surface deposition coefficient h most likely are different for the two designs. A separate study will be required to determine β and h for the once-through design.

Table 2 outlines how the experimental work is divided between the three program participants. Westinghouse's main responsibility is to conduct tests using their prototypic steam generator facility MB-2. The data will be used to check correlations which were obtained in laboratory tests and to provide data which cannot be obtained in small scale tests. Oak Ridge National Laboratory (ORNL) responsibility is to elucidate the chemistry of radioactive iodine under simulated SGTR environments. Northwestern University (NW) responsibility is to generate correlations for liquid carryover for a wide range of conditions. Los Alamos National Laboratory (LANL) provides analytical support to the above organizations, with current emphasis on droplet formations and breakup. Battelle Columbus Laboratory (BCL) is responsible for updating the CITADEL code

as the data becomes available. A brief description of the three facilities is presented below.

Westinghouse - MB-2 Facility

The W Model Boiler No. 2 (MB-2) facility is approximately 1% power scaled model of the Model F steam generator. The MB-2 is geometrically and thermal-hydraulically similar to the Model F; however, since its total volume is scaled to correspond to its power a full size tube rupture cannot be simulated by one to one scaling because this will overfill the unit in a nonprototypic manner. Table 3 presents a comparison between MB-2 and SG-F with respect to the number of tubes, tube specifications, volumes and unit heights and separator specification. Figure 5 presents a schematic of secondary and primary loop instrumentation. The MB-2 presently is equipped to simulate SGTRs at two elevations only; one is near the tube sheet elevation, and the other is at the U-bend elevation. The primary liquid injection is controlled by a removable tube which penetrates the shell. A calibrated ventury tube, differential pressure transmitter, system pressure transmitter and fluid thermocouples are used to measure injection rates. 106 thermocouples are positioned at various axial and radial locations in the bundle to measure the primary and secondary liquid temperatures and the outer wall tube temperature. The secondary side includes 36 pressure taps of which 9 are located in the tube bundle region, 12 in the downcomer pipes, and 15 in the upper shell region. Steam pressure measurements are made at three different locations in the outlet piping. Primary, feedwater and steam flow rates are all measured with a calibrated orifice and narrow, middle and wide range differential pressure transducers. A collection tank is used to weigh and analyze any carryover liquid. Different nonvolatile traces for the primary and secondary fluids are used to characterize primary/secondary fluid mixing as discussed above.

ORNL - Facility

The apparatus shown schematically in Figure 6 is located in a hot cell in the ORNL radiochemical complex. This apparatus consists of (a) 3-½" diameter vessel, 5' long; (b) tube, bundle and separators; (c) relief valve; and (d) a condenser. Radioactive iodine will be injected into the vessel which

contains saturated water at the 1000 to 1200 psi range. The residence time of iodine in the liquid will be varied by means of the relief valves, thus allowing the measurements of the partition coefficient as a function of time. I-131 and Ba-140 are used to measure iodine deposition and liquid carryover. Procedure for measurements and valve activation will be developed.

Northwestern University

Figure 7 is a schematic of the NWU facility which in main is very similar to the ORNL facility but unlike ORNL it employs nonradioactive iodine (sodium-Iodine) and operates at relatively low pressures. This facilitates the attainment of a large number of data points required for entrainment (E) and mixing (β) correlations. Briefly, the NWU facility consists of a 10 in., 6-ft long vessel containing approximately 40 3/4-in. tubes. The primary liquid pressures, up to 20 bars, can be injected into this vessel after the desired steam flow conditions have been established. Traces in the primary and secondary liquids are used to determine the respective amounts of liquid in the carryover. The vessel is also equipped with viewports to allow photographs of the flashing jet. Resistivity probes at various elevations allow measurements of local void fractions. The validity of the correlations will be tested against the prototypic MB-2 data.

Battelle's Columbus Laboratories (BCL)

The CITADEL code was developed by BCL for the NRC to analyze the transport and deposition behavior of iodine in the secondary loop of a PWR during SGTR accidents.[5,6] The previous version of the code did not include the iodine partitioning with steam from the secondary water, and had some numerical problems. BCL is currently modifying the code to include the iodine partitioning and to correct the numerical problems.

Following the code modifications BCL will use the code to assess the radioactivity releases to the environment from the three vendors steam generators; Westinghouse, Combustion Engineering and Babcock and Wilcox, for four accident sequences, namely, (1) SGTR with loss of offsite power, (2) SGTR with a PORV stuck open for 20 minutes after the PORV opens initially, (3) SGTR with an ARV

stuck open for 20 minutes after the ARV opens initially, and (4) SGTR with ARV stuck open for accident duration. BCL will also keep abreast of the experimental data as it comes in to compare the correlations in the code with the data and to see to what extent the code should be modified. The resulting code could then be used by itself or as a subroutine for dose calculations in a thermal-hydraulic systems code.

SUMMARY

The relatively high probability of SGTR accidents dictates that a better understanding of the mechanism of radioactivity release is required. Determination of Technical Specification limits for radioactivity levels in the primary and secondary requires reliable calculations on radioactivity transport for steam generator tube rupture events. A program aimed at the generation of the required data to test and modify if required the CITADEL code has been defined and is now being implemented. The experimental results will also be used to program special iodine transport subroutines for thermal-hydraulic system codes such as TRAC or RELAP.

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Nomenclature

A - surface area

c - iodine concentration per unit mass

E - entrainment

h - mass transfer coefficient defined by equation 8

m - mass flow rate

M - total mass within a given volume

P - partition coefficient, EQ 7

S - iodine removed from fluid due to deposition

t - time

1-x - liquid mass fraction upon flashing

Subscripts

d - downcomer

f - feed water

g - gas phase

H - volume bounded by separator and liquid pool

o - exit from steam generator

ℓ - liquid phase

p - primary

s - entering separator

Greek

β - mixing parameter, EQ 5

ε - entering downcomer

η - separator efficiency, EQ 6

ρ - density

Table 1 OPERATING EXPERIENCE DURING SGTL FOR SELECTED PLANTS

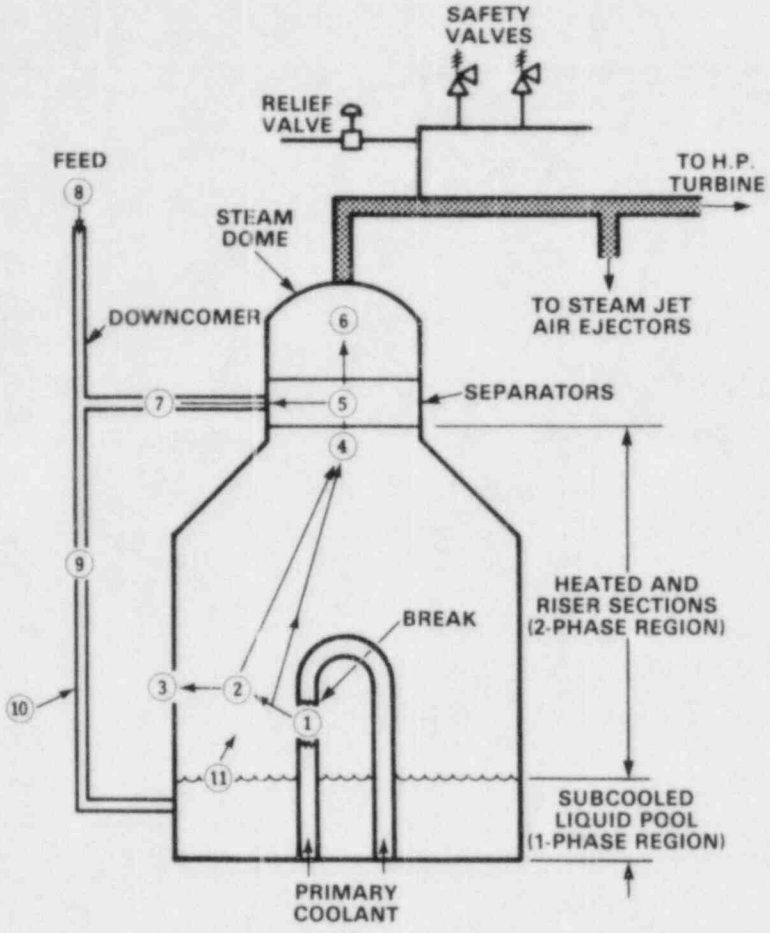
Plant	Date	Leak Rate GPM	Steam Flow GMS/ CC(5)	Iodine 131 Concentrations, µCi/cc			M.S./ S.G. Liq.	Source
				Reactor Coolant	S.G. Liquid	Main Steam		
Point Bch.	2-20-76	0.76	3.75	.007	1.39×10^{-4}	3.78×10^{-9}	10^{-5}	Reference 10
Turkey Point Unit 3C	11-09-77	.05-	4.00	.06	1.8×10^{-4}	2.9×10^{-6}	10^{-2}	
	11-17-77	.13	4.00	.014	4.3×10^{-5}	3.6×10^{-7}	10^{-2}	Reference 11
Unit 4A	1-18-78		4.00	.007	2.7×10^{-6}	5.6×10^{-8}	10^{-2}	
	1-20-78	.01-	4.00	.007	3.9×10^{-6}	1.0×10^{-7}	10^{-2}	
	1-24-78	.33	4.00	.007	1.3×10^{-5}	2.1×10^{-7}	10^{-2}	
	1-26-78		4.00	.047	7.4×10^{-4}	1.1×10^{-5}	10^{-2}	
Ft. Calhoun	2-18-84	.001	--	--	7.0×10^{-7}			Pvt. Communic tion
San Onofre	6-14-84	.1	--	.014	2.5×10^{-4}	6.3×10^{-8}	10^{-4}	Pvt. Communication
	6-19-84	.2	--	N.A.	3.5×10^{-5}	N.A.	--	

Table 2 PARTICIPANTS/RESPONSIBILITIES

ORGANIZATION	Main feature of work	Parameters to be characterized	Variables	Begin/ Complete
W	Prototypic (very close), Geometry of W steam generator	η, E	Leak Location	Aug 84 May 85
ORNL	Irradiated Iodine, Prototypic, pressures, temperatures, coolant chemistry and pH.	P, h, E	pH; (4 to 10) 1-131; (10^{-5} - 10^{-10} molar) Ba-140	Sept 84 Sept 86
NWU	Prototypic, steam and break velocities	E,β	Leak Location and size, steam velocities, depressurization rates.	Sept 84 Sept 86
LANL	Analytical characterization of Jets and sprays			Sept 83 Sept 84
BCL	CITADEL code modification, licensing calculations			April 84 Oct 84

Table 3 PRIMARY AND SECONDARY BUNDLE SCALING

	SG-F	MB-2	MB-2/SG-F
Number of U-Tubes	5646	52	0.00921
Tube Specifications			
O.D. cm (in.)	1.75 (0.688)	1.75 (0.688)	1.0
I.D. cm (in.)	1.54 (0.608)	1.54 (0.608)	1.0
Pitch cm (in.)	2.49 (0.98)	2.49 (0.98)	1.0
Volume m^3 (ft^3)	18.41 (650.)	0.139 (4.90)	0.00755
Secondary Bundle Volume m^3 (ft^3)	44.88 (1585)	0.283 (10.0)	0.0063



1	Iodine entering secondary side from broken tube	$(mc)_p$
2	Iodine mass in volume bounded by separators and liquid pool	$(Mc)_H$
3	Iodine deposited on walls in heated and riser sections	$(S)_H$
4	Iodine entering separators	$(mc)_s$
5	Iodine deposited on separator walls	$(S)_s$
6	Iodine entering steam dome	$(mc)_o$
7	Iodine from separators entering downcomer	$(mc)_\epsilon$
8	Iodine from feedwater entering downcomer	$(mc)_f$
9	Iodine mass in downcomer and in liquid pool	$(Mc)_d$
10	Iodine deposited on downcomer and liquid pool surfaces	$(S)_d$
11	Iodine from downcomer entering 2-phase region	$(mc)_d$

Figure 1 Overall iodine mass balance.

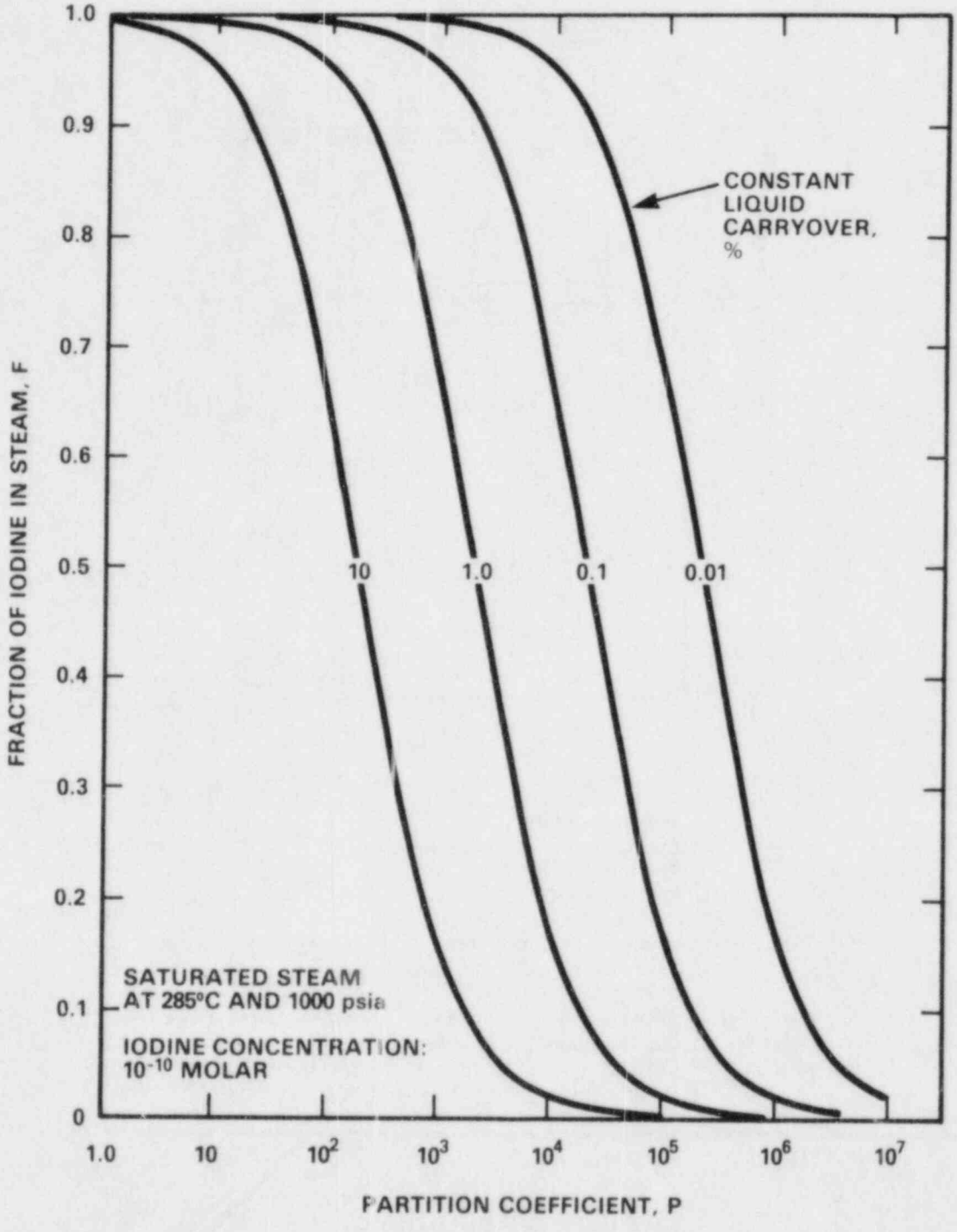


Figure 2 Fraction of iodine in steam, F, as a function of the partition coefficient, P.

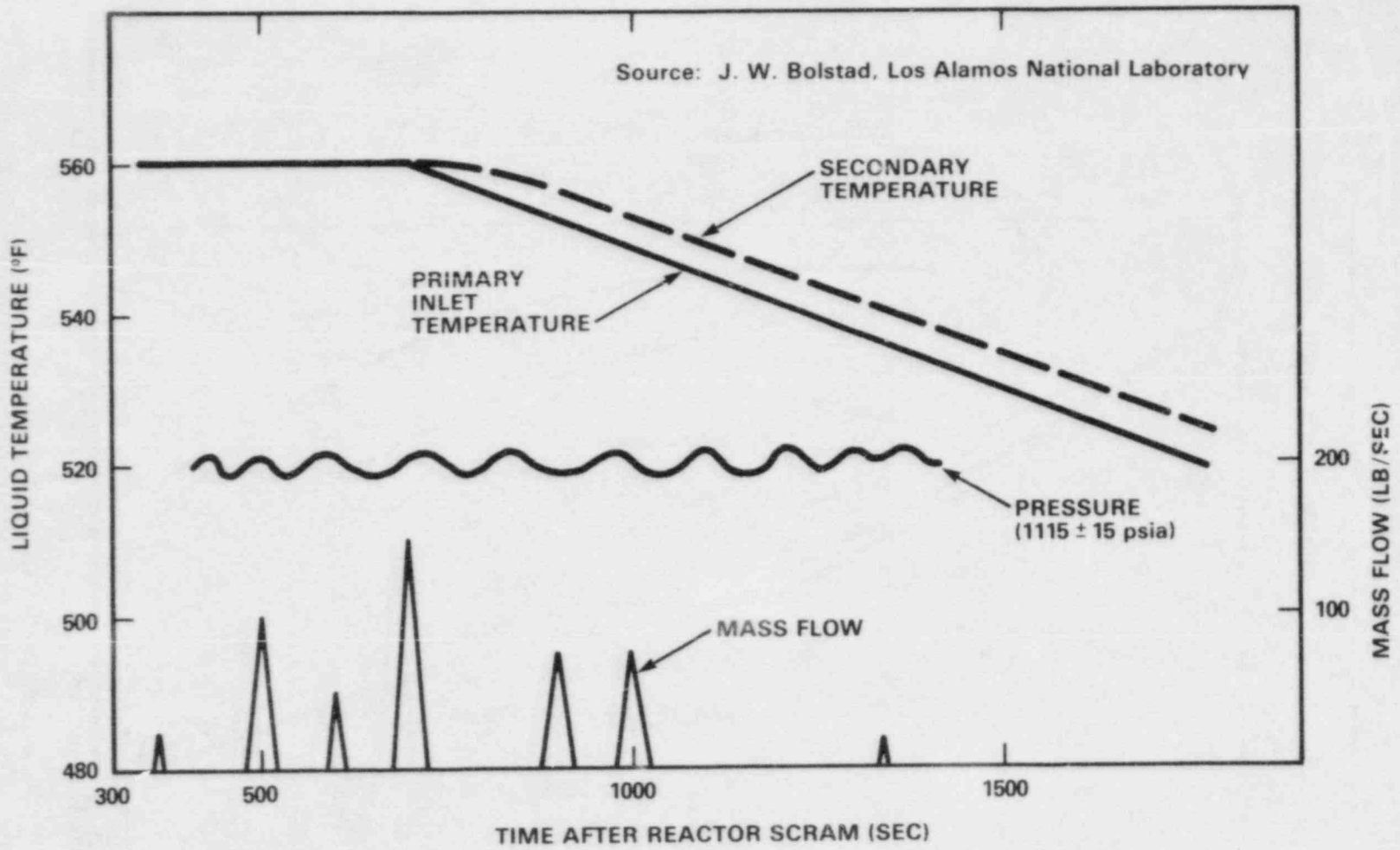


Figure 3 Pressure, temperatures, and mass flow rates following SGTR.

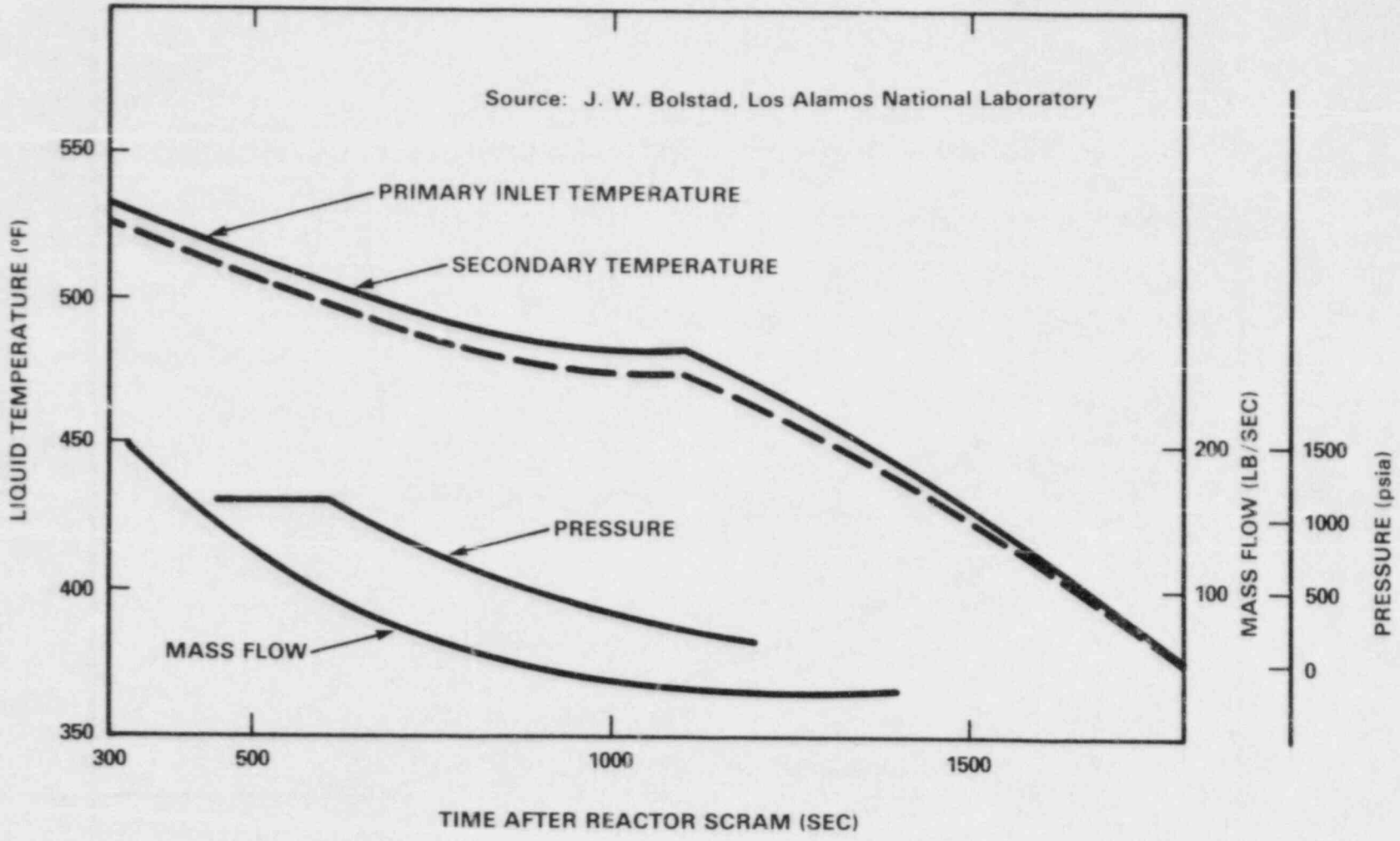


Figure 4 Pressure, temperatures and mass flow rates following SGTR with a stuck open safety valve.

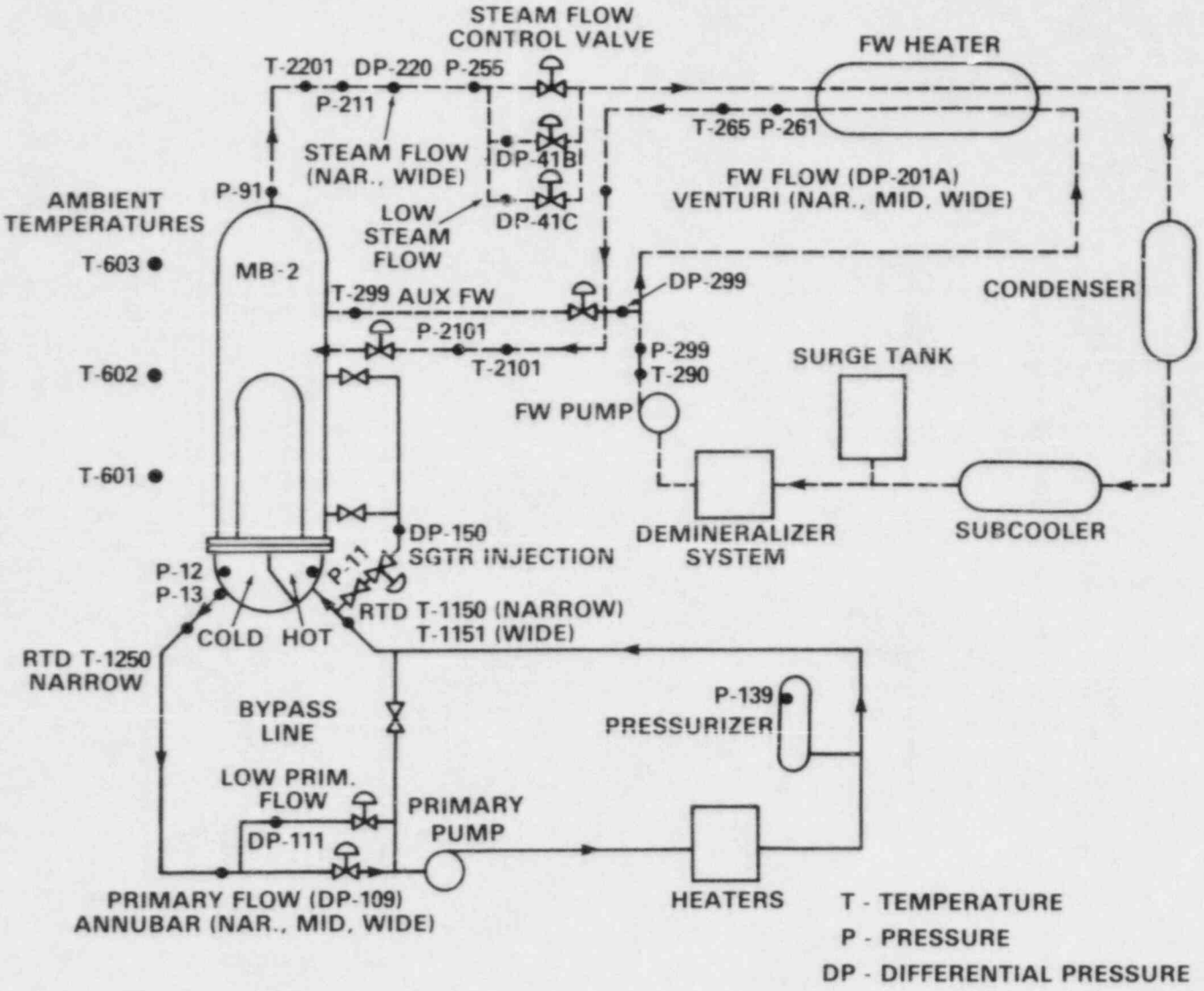


Figure 5 MB-2 primary and secondary loop instrumentation.

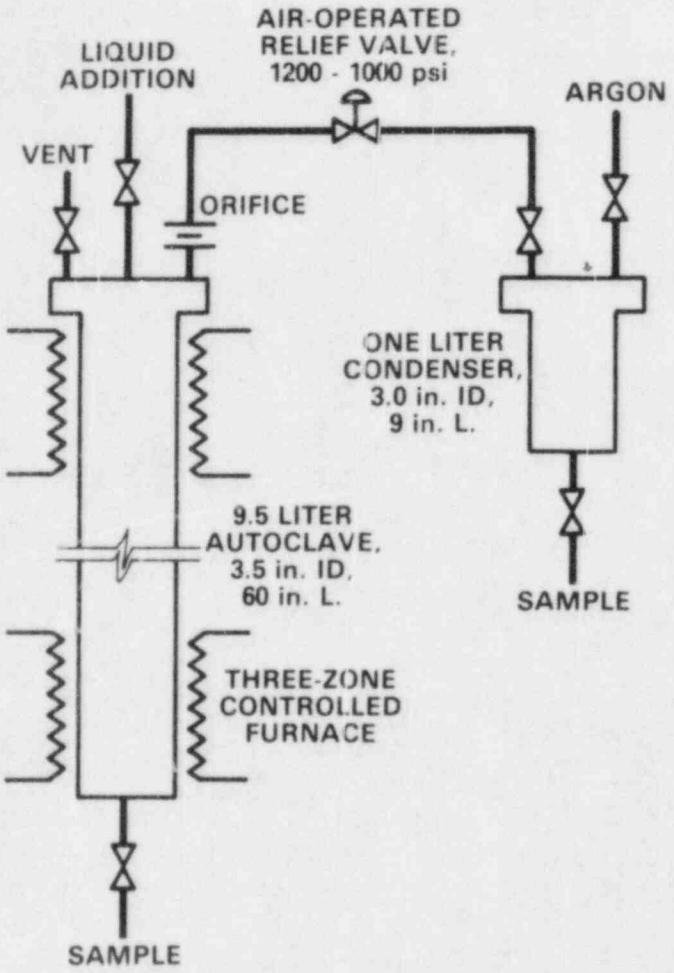


Figure 6 Schematic of ORNL experiments.

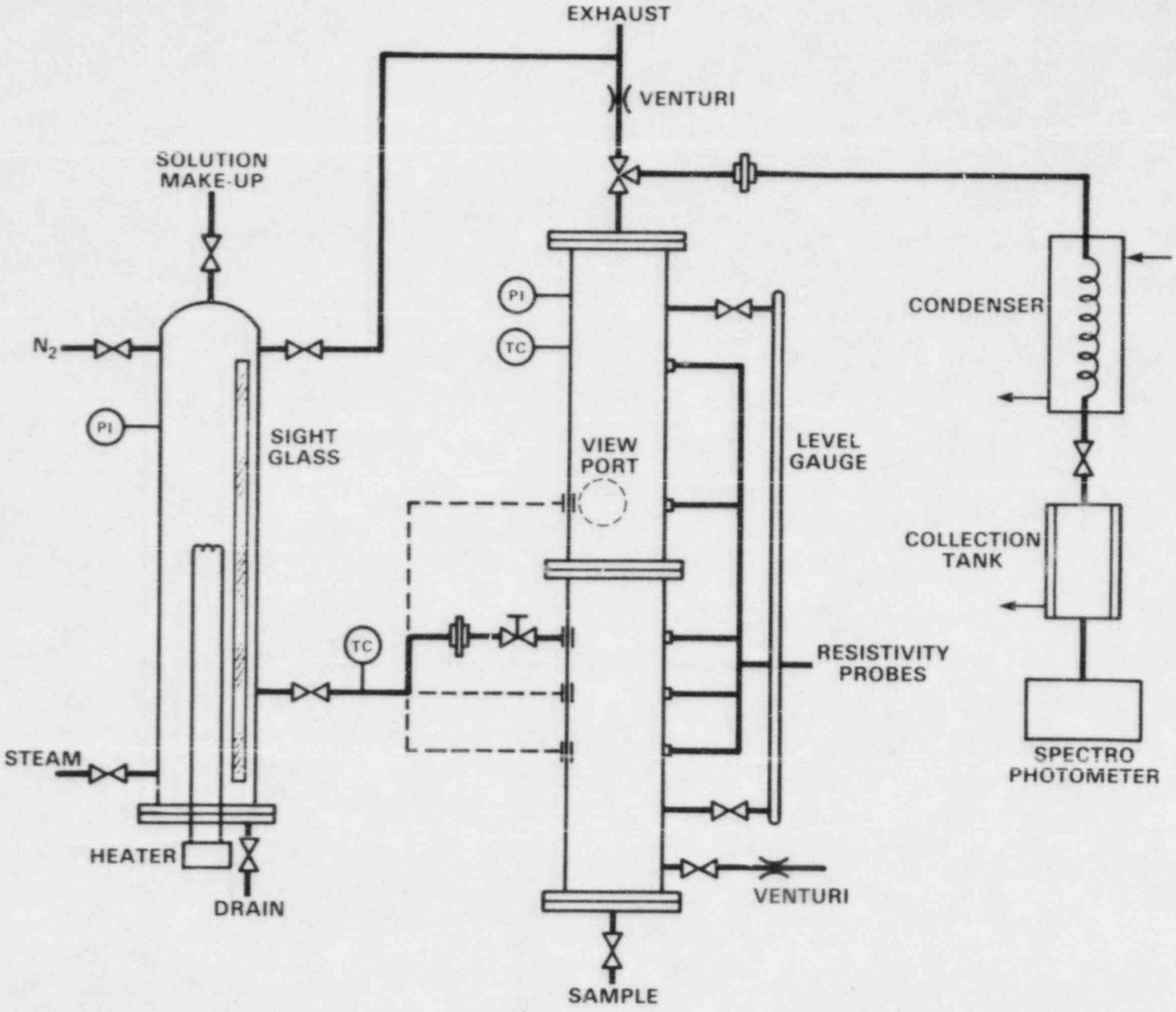
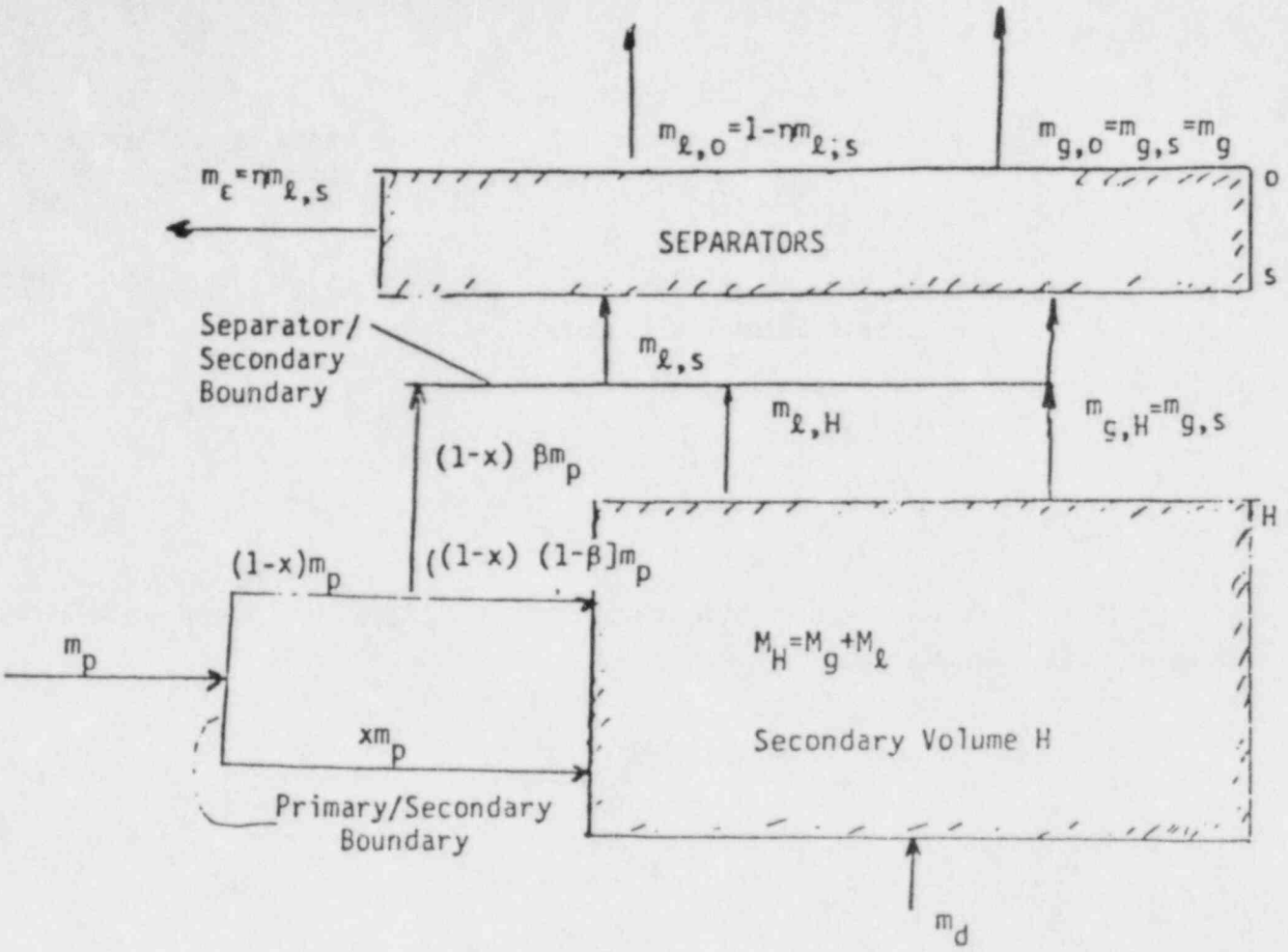


Figure 7 Schematic of Northwestern University experiments.

APPENDIX



Schematic of Mass Balance
on Control Volume H and Separator

$m_{\ell,p} = (1-x)m_p$ = primary liquid from the break entering secondary side which contains a mixture M_H of steam M_g and water M_L.

$(xm)_p$ = mass of primary liquid which flushes and mixes with the steam/water mixture.

$(1-x)\beta m_p$ = mass of primary liquid which reaches the separator by short circuiting mixing.

$[(1-x)(1-\beta]m_p$ = mass of primary liquid which undergoes mixing.

m_{gH} = total steam generation in control volume due to evaporation and flushing from primary fluid.

$m_{\ell,H}$ = total water flow in the secondary side as dictated by mass and heat balance considerations.

2. Comments on Equations 9 and 10

Equation 9

If the iodine concentrations in the primary steam and liquid immediately upon flushing β denoted as $c'_{g,p}$ and $c'_{\ell,p}$ then

1. $(mc)_p = (xmc'_g)_p + [(1-x)mc'_\ell]_p$

2. $c = x_p c'_g + (1-x_p)c'_\ell = [x_p\ c'_g + (1-x_p)]c'_\ell$

$$= [x_p \frac{1}{\beta} \frac{\rho_\ell}{\rho_g} + (1-x_p)]c'_\ell$$

In reference 3 Postma pointed out that since the time required to flash vapor from superheated liquid is very short liquid phase diffusion may be neglected. Under these conditions the interface concentration during flashing builds up to a value which corresponds to $\frac{1}{\beta} \frac{\rho_\ell}{\rho_g}$

thus from equation 2 above

$$c_p = c_{\ell,p} = c'_{\ell,p}$$

The use of c_p in equation 9 implies that the concentration of iodine in the liquid was not altered due to flashing.

Equation 10

1. $(mc)_o = [(mc)_g + (mc)_\ell]_o$

utilizing the definition of

$$P \equiv \frac{c_\ell}{c_g}\frac{\rho_g}{\rho_\ell}$$

we get:

2. $(mc)_o = [m_g \frac{1}{P}\frac{\rho_g}{\rho_\ell} + m_\ell]_o c_{\ell,o}$

The liquid exiting the separator and carried by the steam is assumed to have the same iodine concentration as the liquid which flows into the downcomer. Thus,

3. $c_{\ell,o} = c_{\ell,\varepsilon} = c_\varepsilon$

4. $(mc)_o = [m_g \frac{1}{P}\frac{\rho_g}{\rho_\ell} + m_\ell]_o c_\varepsilon$

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A review of the capabilities of the CITADEL computer code as well as plant experience to project radioactivity releases following steam generator tube rupture in PWR's shows that certain experimental data is needed for reliable offsite dose predictions. This article defines five parameters which are the key for such predictions and discusses the functional dependence of these parameters on various operational variables. A joint Westinghouse, Electric Power Research Institute, and the Nuclear Regulatory Commission program aimed at obtaining the five parameters empirically is described. Present status of the CITADEL code is also reviewed.

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