ATTACHMENT B

THE V SEQUENCE: AN ENGINEERING VIEWPOINT

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ABSTRACT

The "interfacing system LOCA" (identified as the "V sequence" in WASH-1400) is regarded as a significant contributor to risk at the Surry Nuclear Power Station based on the WASH-1400 analysis, and, by implication, at other plants as well.

A new analysis, incorporating specific engineering and arrangement features not previously considered, shows that the structure in which the postulated break occurs remains intact, and at the time of fission product release, the postulated break location would actually be submerged by drainage from the refueling water storage tank. Submergence of the break provides effective scrubbing of any fission products released at the break location. The general applicability of these conclusions is addressed.

INTRODUCTION

In the Reactor Safety Study (WASH-1400) (1) published in October 1975, the Surry Nuclear Power Station was studied as a representative domestic pressurized water reactor (PWR) plant. Many hypothetical accident sequences leading to severe core damage were considered probabilistically. Two of the most severe containment release sequences studied in WASH-1400, AB and TMLB, have recently been re-analyzed as part of the Stone & Webster Engineering Corporation (SWEC) source term investigation (2).

In addition to the reanalysis of the containment release sequences, the SWEC investigation has addressed the interfacing system loss of coolant accident

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(LOCA) identified as the V sequence in WASH-1400. An interfacing system LOCA is defined as the failure of one or more of the elements of the reactor coolant pressure boundary (RCPB) which permits a loss of reactor coolant into an interfacing fluid system. In analyzing an interfacing system LOCA, it is first necessary to determine if the containment is bypassed and second, if the bypass flow path is non-isolatable. If both conditions exist, then the associated fission product release to the environment may not be bounded by that associated with containment release sequences.

The Surry V sequence identified in WASH-1400 is an example of a non-isolatable interfacing system LOCA which bypasses containment. The V sequence, as defined in WASH-1400, included the following assumptions:

- Failure of a check valve in one of the 6-inch cold leg emergency core cooling system (ECCS) injection lines (see Figure 1)
- A pre-existing condition, whereby the other check valve in the same cold leg injection line was stuck open
- Inability to close the 10-inch low pressure cold leg injection isolation valve (Surry Technical Specifications require the valve to be open with electrical power disconnected to preclude inadvertent or spurious closure resulting in complete loss of low pressure cold leg injection capability)
- A loss of coolant into the Safeguards Building (SGB) as a result of failure of low pressure piping immediately beyond the isolation valve by exposure to full RCS pressure
- Inability to achieve recirculation cooling once ECCS injection has ceased as a result of refueling water storage tank (RWST) depletion or pump failure
- Core uncovery with fission product release bypassing containment and entering the SGB
- Nearly immediate release of fission products from the SGB to the environment due to the relatively small free volume of the structure and the assumption that significant structural damage would have occurred during the initial RCS blowdown

This paper re-examines the assumptions and findings of WASH-1400 with respect to the V sequence for Surry, and generalizes the Surry findings to other PWR plants where possible.

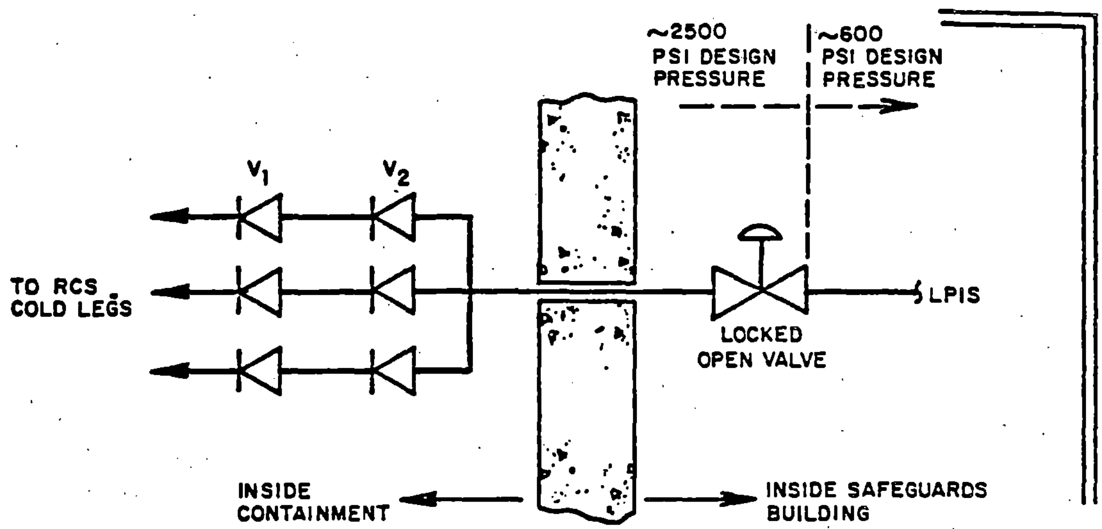

Figure 1. Surry ECCS Low Pressure Injection Configuration (Cold Leg)

SUMMARY OF CONCLUSIONS

Re-examination of the Surry V sequence has revealed the following:

- In analyzing the consequences of an interfacing system LOCA which bypasses containment, a simple diagrammatic treatment may be insufficient. A supplemental engineering evaluation may reveal plant features that will greatly mitigate the release of fission products.

- The SGB will remain largely intact during the initial RCS blow-down, although it is expected that the access door located at grade (see Figure 2) will fail.

- The most likely break location will be covered by drainage from the RWST, which will provide effective scrubbing of the fission product release.

- Due to fission product retention in the RCS and ECCS piping, and the scrubbing effect of RWST drainage covering the break in the intact SGB, the fission product release would be expected to be less than 1 percent of the iodine and cesium inventories and even less of the tellurium inventory.

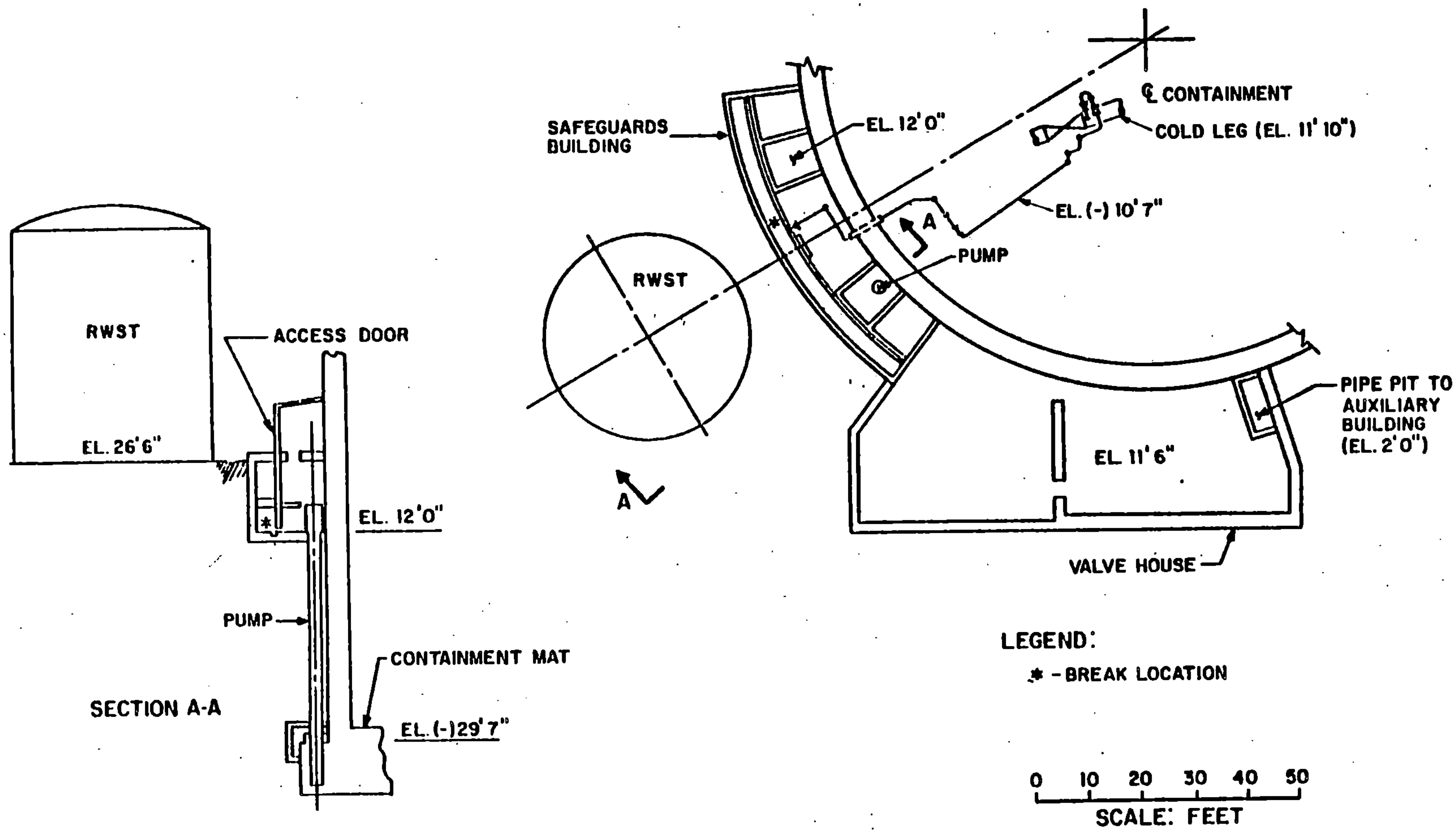

Figure 2. SGB and Cold Leg Injection Piping Arrangement

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Generalization of these findings to other PWR plants would include the following observations:

- Detailed engineering analyses, beyond the level of an "FSAR review," should be applied in the analysis of interfacing system LOCAs which could potentially bypass containment.
- All PWR low pressure ECCS configurations are sufficiently similar to suggest that flooding of the break for a V sequence is likely.
- Interfacing system LOCAs which could potentially bypass containment may actually result in breaks inside containment if the locations of highest stress (relative to capability) clearly exist there.
- For plant configurations where the break is outside containment and is not covered by RWST drainage, ventilation system characteristics will be important.

It is characteristic of the V sequence, as well as other interfacing system LOCAs which could potentially bypass containment, that any necessary preventive or corrective measures are relatively straightforward and easily implemented.

SURRY V SEQUENCE

Surry ECCS Configuration

The Surry cold leg injection path configuration, shown diagrammatically in Figure 1, resulted from a change made late in construction and is not typical of domestic PWRs. Surry was originally designed with the ECCS normally aligned for hot leg injection. The hot leg injection flow path has three check valves in series and two normally closed, parallel isolation valves which were originally intended to open automatically on a safety injection signal. These two isolation valves can be closed remotely.

In the fall of 1971, the Surry ECCS was redesigned to normally align the system for cold leg injection, thus improving ECCS performance so that the already designed plant would comply with the ECCS Interim Acceptance Criteria. To preclude inadvertent or spurious closure of the single, normally open, cold leg injection path isolation valve, the Surry Technical Specifications required that the valve be opened and electrical power to the valve be disconnected prior to power operation. It should be noted that this configuration is for the most part unique to Surry, and most other plant configurations would be less susceptible to a V sequence.

Break Size and Location

A review of the Surry low pressure ECCS piping arrangement outboard of the 10-inch cold leg injection path isolation valve indicates that the most likely break would be an elbow split just beyond the valve, with an estimated flow area of 0.2 to 0.5 ft^2. Since most of the hydraulic losses occur within the ~150 feet of primarily 6-inch ECCS injection line leading from the cold leg to the break, the mass and energy release rates would not be sensitive to the exact break area. (Note that K = 18.1 for the injection line, with D = 6 inches and no additional restriction at the break itself). The break is located approximately 2 feet off the floor of the lowest level in the SGB, in the piping and valve area outside the pump cubicles (see Figure 2).

V Sequence Systems Transients

Blowdown Phase. The blowdown phase was calculated using RELAP4-MOD5(3) (0.0 to 1400 sec). In the RELAP calculation, the RCS was represented by a 21-node model, with the ECCS injection piping modeled as a 6-inch orifice with effective resistance factor K = 18.1. WASH-1400 concluded that ECCS injection could continue for as little as 1 hour, or as long as 10 hours, depending on the pumping capacity assumed to be available. In the SWEC reanalysis, it was determined that drainage from the RWST to the break location would limit the duration of ECCS injection to approximately 2 hours independent of the number of ECCS pumps operating. Because of the potential for ECCS damage due to the break itself, or associated adverse environmental effects, no active ECCS was assumed to function in the SWEC analysis. The blowdown phase ends with the RCS at low pressure (approximately 100 psia), the accumulators completely discharged, and the core reflooded.

Core Damage Phase. The core damage phase (1400 to 9000 sec) was calculated with a hand-calculation model. In this model, the primary coolant system was represented as a single control volume with two regions: vapor and liquid. All decay heat generated in the covered portion of the active fuel zone (liquid region) was allowed to generate steam, and a simple heat transfer model was applied in the uncovered portion (vapor region) to calculate superheating of the steam generated in the covered portion. A portion of the steam generated was assumed to oxidize zircalloy cladding at the rates given in BMI-2104 for the V sequence(4). The fission product release rates and the timing of the start of core melt, core slump and collapse, and the vessel lower head failure were also taken from BMI-2104. Since the BMI-2104 analysis assumed a 6-inch (0.2 ft^2)

frictionless break, this information is not consistent with the SWEC system transient analysis which included friction. The result of this inconsistency is that the entire sequence would actually progress at a somewhat slower pace than presented here, with the fission product release occurring as much as an hour later. The RCS pressure transient (with friction) is shown in Figure 3.

Building Integrity

Analyses performed by SWEC show that buildings, such as the SGB, can withstand the pressure transient imposed by the V sequence event. The pressure transient consists of a pressure "pulse" immediately after the rupture, followed by a rapid building decompression to nearly atmospheric pressure, as shown in Figure 4. The magnitude of the pressure pulse depends on the effective break area and the amount of vent opening to the atmosphere in the building. With an effective break area of 0.2 ft^2 and two vent openings of 23 ft^2 and 5.25 ft^2, the magnitude of the pressure pulse is less than 20 psia, as shown in the figure. Most of the venting is calculated to be through the SGB access door which does not have sufficient latch strength to remain shut under pressurization.

Fission Product Retention

SGB Flooding. The initial RWST water level (with the tank at the minimum technical specification inventory of 380,000 gallons) is approximately 35 feet above the elevation of the break (see Figure 2). With the exception of several check valves, all valves between the RWST and the RCS injection points are normally open. Therefore, as soon as the pressure on the downstream side of the pump discharge falls below approximately 40 psia, RWST drainage will begin.

For large breaks with little communication between the two sides, drainage will begin almost immediately. For small breaks, it is possible that the start of drainage could be delayed as much as 30 to 40 minutes. Once drainage begins, the break will be flooded in 2 to 5 minutes, with the entire SGB flooded to its maximum depth in 10 to 15 minutes. The maximum water depth in the SGB is fixed by an opening in the SGB wall 4.75 feet above the SGB floor.

Water leaving the SGB will flow into the auxiliary building complex. Water will continue to drain from the RWST for as long as several hours, or until the operator isolates the tank. The mass of water in the vicinity of the break becomes saturated approximately 20 to 30 minutes after the water first covers the

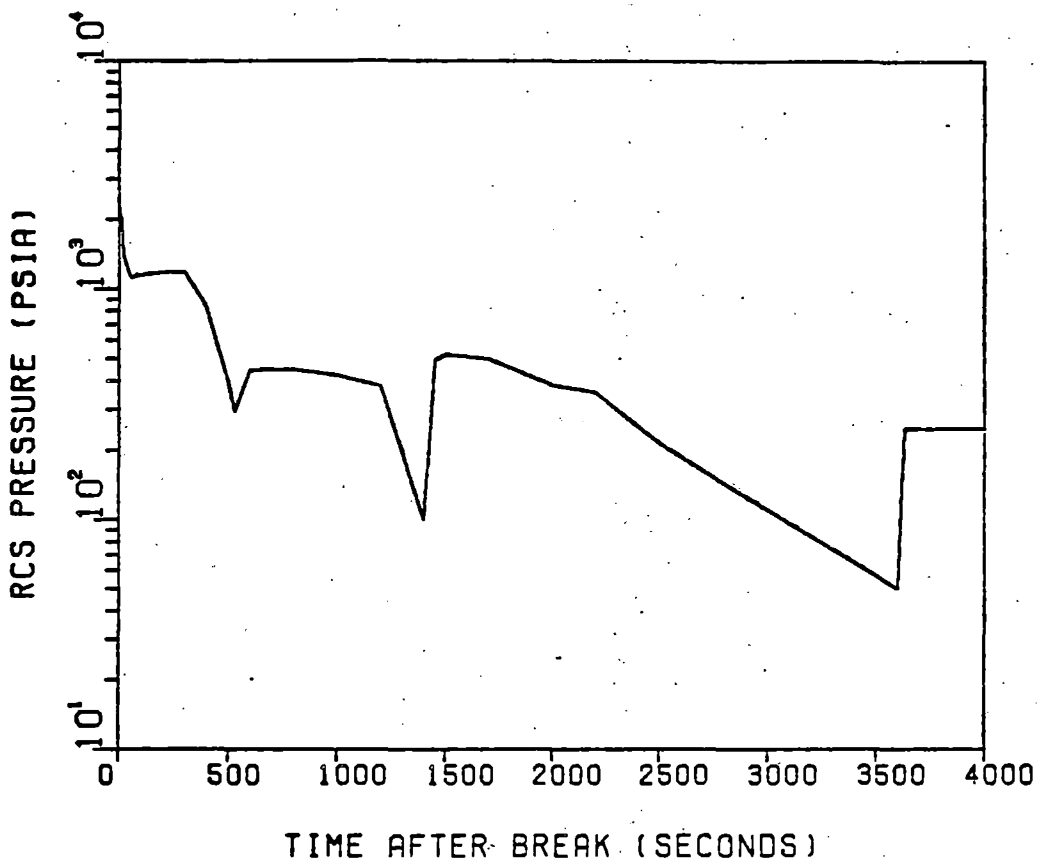

Figure 3. RCS Pressure Transient

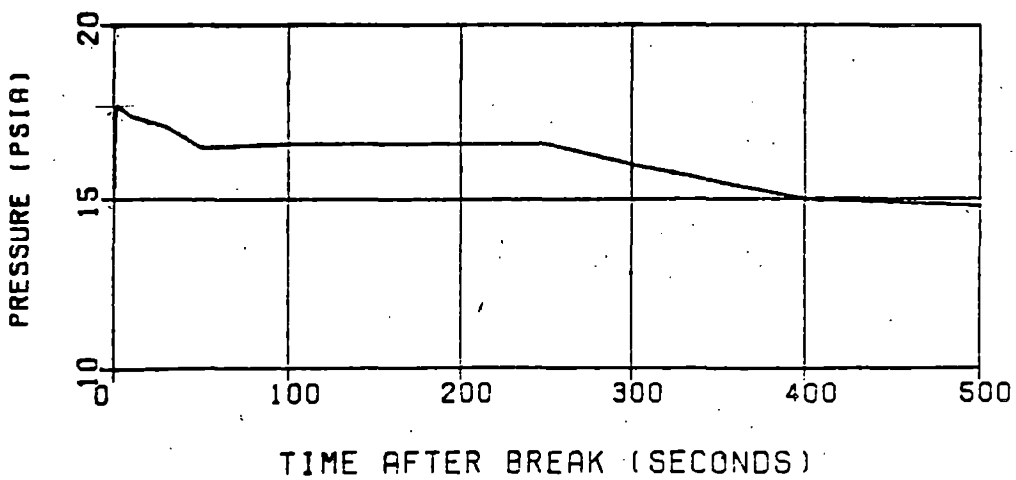

Figure 4. SGB Pressure Transient

break. Therefore, it is expected that the fission product release will be into a saturated pool.

Consideration of water loss from the SGB by effects other than overflow has revealed the following:

- Evaporation by the sensible heat of hydrogen passing through the pool will result in a level loss of 1 to 2 inches.
- Continuous operation of the two safety-related SGB sump pumps will decrease the level by approximately 6 inches per hour once the RWST is depleted.
- Decay heat generation from the scrubbed fission products (~ 50 percent of the cesium and iodine, and ~ 30 percent of the tellurium released from the RCS and ECCS per BMI-2104) will decrease the level approximately 2 inches per hour (after RWST depletion), but only after the fission product scrubbing has already been effective.

At the time of vessel lower head failure, the break will still be covered by approximately 2 feet of water.

Containment Response. Prior to vessel lower head failure, containment response to the V sequence would be minimal. At the time the vessel lower head fails, the water level in the SGB will be approximately 2 feet above the break and 4 feet above the RCS loop centerline. Combined with the initially subatmospheric pressure of the Surry containment, the water will begin to return to the RCS through the injection line at an initial rate of approximately 400 gpm. As the containment pressurizes, the return rate will slow, finally reaching equilibrium about 1 hour after the vessel lower head failed. At this point, a "loop seal" will have formed in the ECCS injection line (see Figure 2) which will not be even partially displaced until approximately 24 hours later. Although a fission product deposition analysis has not been performed for the Surry containment under these conditions, it is expected that the post-melt through contribution to the V sequence source term will be very small.

Source Term Estimate. Fission product retention in the RCS and ECCS was not calculated in this analysis. However, an estimate of the releases associated with this accident sequence was developed based on the NRC-sponsored analysis reported in BMI-2104. The BMI-2104 analysis resulted in the release of ~ 50 percent of the core inventory of cesium iodide and cesium hydroxide and ~ 30 percent of the tellurium to the SGB.

The volatile fission product release occurs in the SGB with the break submerged to a depth of approximately 2 feet. The decontamination factor for such a submergence would be high, and is estimated to be approximately 50. In view of the above, the amount of aerosol released to the environment for a V sequence at Surry is expected to be small; most likely, it will be less than 1 percent of the cesium and iodine inventories, and even less of the tellurium inventory.

APPLICABILITY TO OTHER PLANTS

The applicability of the conclusions to other plants, with respect to the Surry V sequence, depends on the degree to which the Surry analysis can be extended in terms of the potential for flooding of the break. In cases where the break is not flooded, building ventilation characteristics need to be considered carefully.

Potential for Flooding of the Break

All PWR low pressure ECCS pumps have net positive suction head (NPSH) requirements which the plant design must meet. These requirements dictate that the pumps be placed low in the overall plant arrangement with respect to the RWST. Since valves are normally open between the RWST and the pump suction, drainage from the RWST to the V sequence break location is considered likely. Typically, the pumps and associated piping are located in confined areas where flooding of the break is possible. However, in analyzing the potential for flooding of the break, attention must be paid to the presence of drains, sump pumps, wall openings for piping, ductwork and electrical cables, etc., which could serve to conduct water away from the area.

For an interfacing system LOCA bypassing containment other than the V sequence, flooding of the break may still result from the break effluent itself. The building geometry, the degree of effluent subcooling, the characteristics of droplet formation and settling, and the condensation rate would be important parameters in such an analysis.

Break Location

Some interfacing system LOCAs would not bypass containment. If points of high stress exist in portions of the interfacing fluid system inside containment and it can be clearly demonstrated that such points are the likely break locations, then the interfacing system LOCA will simply be another case involving a fission product release to containment. On the other hand, if the break location is outboard of an operable isolation valve, then the event can be terminated by simple closure of the valve.

Non-Flooded Breaks

For non-isolatable interfacing system LOCAs bypassing containment, which do not result in flooding of the break, ventilation systems may play an important role. With respect to the V sequence, ventilation systems serving areas where low pressure ECCS equipment and piping are located generally include radiation monitoring and provision for filtration of the exhaust. Such systems may be capable of processing five to ten times the volumetric break flow existing at the time of fission product release. However, effectiveness of these systems may be reduced by damage sustained during the intial blowdown, excessive moisture or aerosol loading, high temperature due to the hydrogen carrier gas and the fission product decay heating, and actuation of fire protection systems which may isolate the filter train.

An engineering study which considers the specific plant and system arrangement details is required in addressing interfacing system LOCAs. Reactor accident analyses, including probabilistic risk assessments, should not simply assume the existence of direct release paths to the environment.

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