Central Files

POLICY ISSUE
(Information)

October 31, 1985

SECY-85-349

For: The Commissioners

From: William J. Dircks
Executive Director for Operations

Subject: RESOLUTION OF UNRESOLVED SAFETY ISSUE A-43, "CONTAINMENT EMERGENCY SUMP PERFORMANCE"

Purpose: To inform the Commissioners of the final technical resolution of Unresolved Safety Issue (USI) A-43, "Containment Emergency Sump Performance."

Background: USI A-43 deals with a concern for the availability of adequate recirculation cooling water following a loss-of-coolant accident (LOCA) when long-term recirculation of cooling water from the PWR containment sump, or the BWR residual heat removal system (RHR) suction intake, must be initiated and maintained to prevent core melt. This water must be sufficiently free of LOCA-generated debris and potential air ingestion so that pump performance is not impaired thereby seriously degrading long-term recirculation flow capability. The concern applies to both PWRs and BWRs. The RHR suction strainers in a BWR are analogous to the PWR sump debris screen, and adequate recirculation cooling capacity is necessary to prevent core melt following a postulated LOCA.

The technical concerns evaluated under USI A-43 are as follows:

(1) PWR sump (or BWR RHR suction intake) hydraulic performance under post-LOCA adverse conditions resulting from potential vortex formation and air ingestion and subsequent pump failure.

(2) The possible transport of large quantities of LOCA-generated insulation debris resulting from a pipe break to the sump debris screen(s), and the potential for sump screen (or suction strainer) blockage to reduce net positive suction head (NPSH) margin below that required

Contact: Karl Kniel (x27359)
A. Serkiz (x24217)

A/1

for the recirculation pumps to maintain long-term cooling.

(3) The capability of RHR and containment spray system (CSS) pumps to continue pumping when subjected to possible air, debris, or other effects such as particulate ingestion on pump seal and bearing systems.

These safety concerns have been investigated on a generic basis, beginning in 1979. The staff's proposed resolution for USI A-43 was issued for public comment on May 10, 1983. The public comment package included NUREG-0869, "USI A-43 Regulatory Analysis", the staff's technical findings report NUREG-0897, "Containment Emergency Sump Performance", proposed Regulatory Guide (R.G.) 1.82, Revision 1, "Water Sources for Long Term Recirculation Cooling Following a Loss of Coolant Accident" and proposed SRP Section 6.2.2, Revision 4 "Containment Heat Removal Systems." A summary of the public comments received and the staff's response are contained in Appendix A of NUREG-0869, Revision 1 (Enclosure 1).

Discussion: The staff's technical findings in NUREG-0897, Revision 1, (Enclosure 2) can be summarized as follows:

(1) Measurements derived from extensive, full-scale sump hydraulic tests have generally shown that low levels of air ingestion (less than 1% to 2%) will occur and have also demonstrated that vortex observations alone cannot be used to quantify levels of air ingestion (as has been done in the past). These test results have been used to develop PWR sump and BWR suction intake hydraulic design guidelines for minimizing, or eliminating, air ingestion and have eliminated the need for plant-specific sump tests or model tests.

(2) Plant insulation surveys, development of methods for estimating debris generation and transport, debris transport experiments, and information provided as public comments have shown that debris blockage effects are dependent on the types and quantities of insulation employed, the primary system layout within containment, post-LOCA recirculation patterns and velocities, and the post-LOCA recirculation flow rates. Thus, blockage effects are governed by plant-specific design features and post-LOCA recirculation flow rates that vary significantly from plant to plant.

The results also show that the screen blockage assumption endorsed in R.G. 1.82 may not be correct. The guide currently states that 50% of the sump screen area should be assumed blocked when evaluating the adequacy of pump performance (NPSH) in the recirculation mode. This assumption, if applied uniformly to all plants, may be non-conservative for many applications. Plant-specific evaluations are necessary to assess potential screen blockage by LOCA-generated debris; guidance to perform these evaluations is now given in R.G. 1.82, Revision 1 (Enclosure 3).

(3) Reviews of available data on pump air ingestion effects and discussions with the U.S. manufacturers of RHR and CSS pumps show that low levels of air ingestion (2% or less) will not significantly degrade pump performance, and that the types of pumps employed in nuclear plants will tolerate ingestion of insulation debris and other types of post-LOCA particulates that can pass through PWR sump screens or BWR suction strainers.

In summary, the A-43 findings show that the safety significance of the effects of vortex formation and air ingestion is substantially less than previously hypothesized, but that the loss of recirculation cooling capability as a result of LOCA debris generation transport and screen blockage is potentially more significant.

The staff concluded that safety reviews of future applications for new plant designs should include an evaluation of this potential and appropriate design features to minimize it. Guidance for such evaluations was developed in revisions to R.G. 1.82 (Enclosure 3) and Standard Review Plan Section 6.2.2 (Enclosure 4). Such evaluations have not generally been required for plants now in operation or under construction. The staff also considered whether the potential for a LOCA followed by debris generation and transport, sump blockage, and possible loss of long-term recirculation cooling capability is sufficient to warrant requiring all licensees and applicants to perform the evaluations. On the basis of the studies performed, the staff believes that it is likely that debris assessments would show that the designs of many (perhaps most) plants are such that debris-blockage is not a problem. The staff's regulatory analysis given in NUREG-0869, Revision 1 (Enclosure 1) indicates that, even for plants with significant potential for debris generation and transport, the conservatively estimated core melt frequency resulting from sump blockage following a LOCA would be less than about 10^{-5}/reactor-year. The regulatory analysis also considers the potential for off-site exposure of the public from such events for the five major types of plant containment. These are PWRs with large dry containments, subatmospheric containments and ice condenser containments, and BWRs with Mark I and II containments and BWRs with Mark III containments. These evaluations are based on findings that containments have inherent overpressure capabilities beyond their design basis. Containment survivability is an important factor leading to a substantial reduction in potential offsite consequences even if sump failure and core damage occur. The analyses indicate that the offsite radiation exposure resulting from loss of sump following a LOCA would be of low to medium consequence. The staff concludes that the regulatory analysis does not support a generic requirement for current licensees and applicants to perform debris assessments.

Since the changeout of primary system insulation has been performed by a number of utilities and is being planned or considered by others, a generic information letter will be issued to licensees and applicants alerting them to these safety concerns (see Appendix H, NUREG-0869, Revision 1). The generic letter will include the A-43 technical findings and Regulatory Guide 1.82, Revision 1, which provides a deterministic method for assessing debris blockage effects. This information is particularly useful for assessing the safety significance of any contemplated changes to plant insulation.

<u>Coordination</u>: The proposed resolution of USI A-43 was reviewed by both the Committee to Review Generic Requirements (CRGR) and the Advisory Committee on Reactor Safeguards (ACRS). In a memorandum from V. Stello to W. J. Dircks dated September 13, 1985 the CRGR recommended approval of the staff's proposed resolution of this issue and that the generic letter should clearly state that any NRC requirement for application of the new guidance in R.G. 1.82, Revision 1 resulting from review of the licensee's 10 CFR 50.59 analysis will be treated as a plant-specific backfit pursuant to 10 CFR 50.109.

In a letter from D. Ward to W. J. Dircks dated September 16, 1985, the ACRS also supported the proposed resolution, but recommended that certain portions of the implementation section of Regulatory Guide 1.82, Revision 1, be changed to make these positions applicable for future modifications, and not be considered a backfit. The staff has considered the ACRS recommendation and has concluded that any NRC imposition of the new guidance on existing plants should be considered a backfit pursuant to 10 CFR 50.109.

<u>Resolution</u>: The staff is implementing the resolution of USI A-43 through the following actions:

(1) The staff's technical findings (NUREG-0897, Revision 1) are being published for use as an information source by applicants, licensees, and the staff.

(2) SRP Section 6.2.2 and Regulatory Guide 1.82 are being revised to reflect the staff's technical findings reported in NUREG-0897, Revision 1. This revised licensing guidance would apply only to reviews of: (a) future construction permit applications, and preliminary design approvals (PDAs), (b) final design approvals (FDAs), for standardized designs which are intended for referencing in future construction permit applications that have not received approval, and (c) applications for licenses to manufacture. This revised guidance would become effective 6 months following issuance of Regulatory Guide 1.82, Revision 1.

(3) A generic letter for information only will be sent to all holders of an operating license or construction permit outlining the safety concerns regarding potential debris blockage and recirculation failure due to inadequate NPSH. It is recommended (but not required) that licensees utilize R.G. 1.82, Revision 1, as guidance for conduct of the 10 CFR 50.59 analysis for future plant modifications involving replacement of insulation on primary system piping and/or equipment. If, as a result of NRC staff review of licensee actions associated with replacement or modification to insulation, the staff decides that SRP 6.2.2, Rev. 4 and/or R.G. 1.82, Rev. 1, criteria should be (or should have been) applied by the licensee, and the staff seeks to impose these criteria, then the NRC will treat such an action as a plant-specific backfit pursuant to 10 CFR 50.109.

(4) The appropriate Congressional Committees will be informed that USI A-43 has been resolved.

William J. Dircks
Executive Director for Operations

Enclosures:
See next page

Enclosures: (Commissioners, SECY, OGC , OPE only)

1. NUREG-0869, Rev. 1,
   "USI A-43 Regulatory Analysis"
2. NUREG-0897, Rev. 1,
   "Containment Emergency
   Sump Performance"
3. Regulatory Guide 1.82,
   Rev. 1, "Water Sources
   for Long Term Recirculation
   Cooling Following a Loss
   of Coolant Accident"
4. Standard Review Plan
   Section 6.2.2, Rev. 4
   "Containment Heat Removal Systems"

Revision 1*
November 1985

U.S. NUCLEAR REGULATORY COMMISSION

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 1.82
(Task MS 203-4)

WATER SOURCES FOR LONG-TERM RECIRCULATION COOLING FOLLOWING A LOSS-OF-COOLANT ACCIDENT

A. INTRODUCTION

General Design Criteria 35, "Emergency Core Cooling," 36, "Inspection of Emergency Core Cooling System," 37, "Testing of Emergency Core Cooling System," 38, "Containment Heat Removal," 39, "Inspection of Containment Heat Removal System," and 40, "Testing of Containment Heat Removal System," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," require that systems be provided to perform specific functions, e.g., emergency core cooling, containment heat removal, and containment atmosphere clean up following a postulated design basis accident. These systems must be designed to permit appropriate periodic inspection and testing to ensure their integrity and operability. General Design Criterion 1, "Quality Standards and Records," of Appendix A to 10 CFR Part 50 requires that structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. This guide describes a method acceptable to the NRC staff for implementing these requirements with respect to the sumps and pools performing the functions of water source for the emergency core cooling, containment heat removal, or containment atmosphere clean up. This guide applies to light-water-cooled reactors.

The Advisory Committee on Reactor Safeguards has been consulted concerning this guide and has concurred in the regulatory position.

Any information collection activities mentioned in this regulatory guide are contained as requirements in 10 CFR Part 50, which provides the regulatory basis for this guide. The information collection requirements in 10 CFR Part 50 have been cleared under OMB Clearance No. 3150-0011.

*The substantial number of changes in this revision has made it impractical to indicate the changes with lines in the margin.

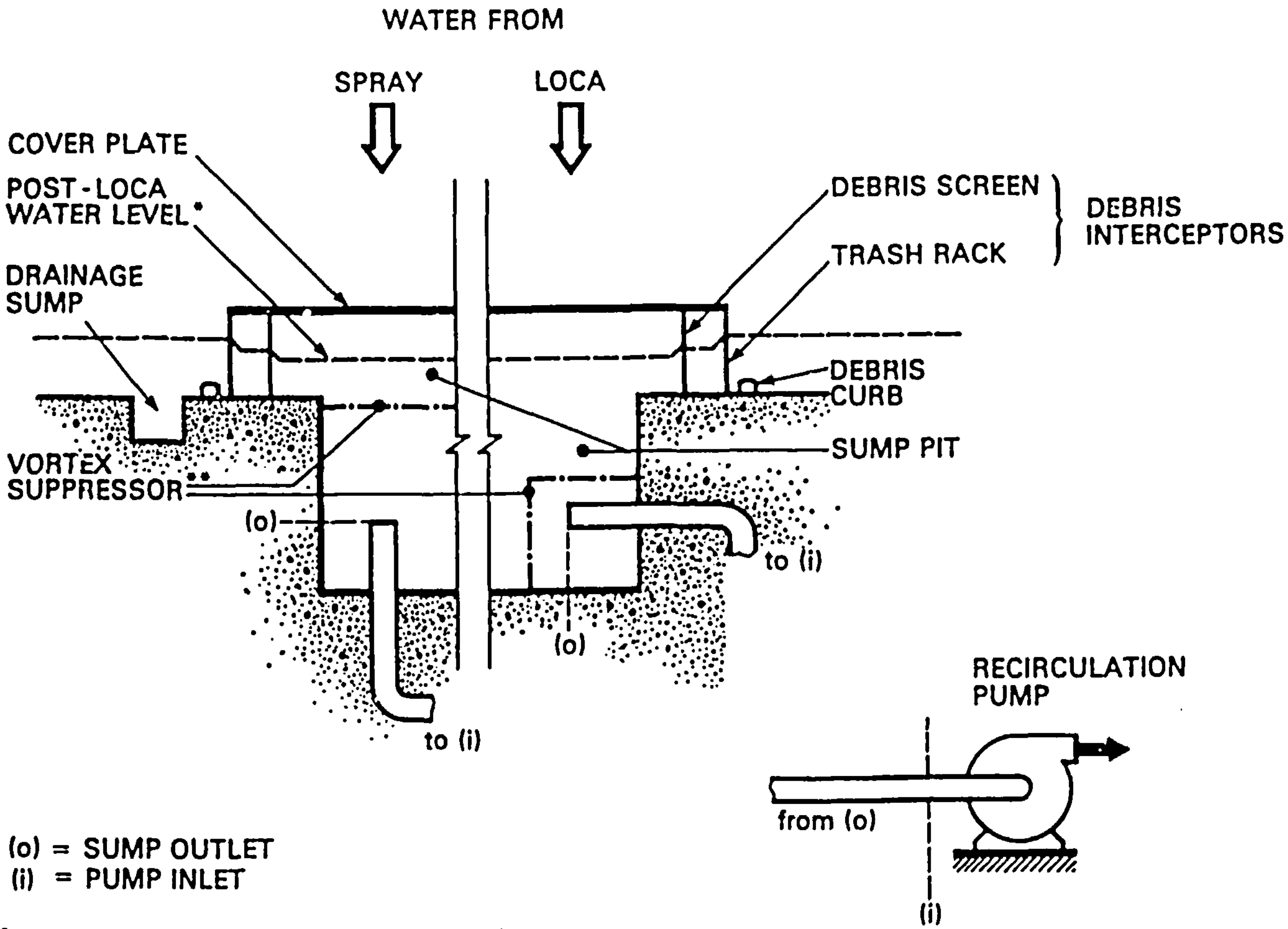
B. DISCUSSION

1. Pressurized Water Reactors

In pressurized water reactors (PWRs), the containment emergency sumps provide for the collection of reactor coolant and chemically reactive spray solutions following a loss-of-coolant accident (LOCA); thus the sumps serve as water sources to effect long-term recirculation for the functions of residual heat removal, emergency core cooling, and containment atmosphere cleanup. These water sources, the related pump inlets, and the piping between the sources and inlets are important safety components. The sumps servicing the emergency core cooling systems (ECCS) and the containment spray systems (CSS) are referred to in this guide as ECC sumps. Features and relationships of the ECC sumps pertinent to this guide are shown in Figure 1.

The primary areas of safety concern regarding ECC sumps and pump inlets are (1) post-LOCA hydraulic effects, particularly air ingestion, (2) blockage of debris interceptors resulting from LOCA destruction of insulation and its transport, and (3) the combined effects of items (1) and (2) relative to recirculation pumping operability (i.e., impact on net positive suction head (NPSH) available at the pump inlet).

Debris resulting from a LOCA has the potential to block ECC sump debris interceptors (i.e., trash racks, debris screens) and sump outlets resulting in degradation or loss of margin. Such debris can be divided into the following categories: (1) debris that is generated early in the LOCA period and is transported by blowdown forces (i.e., jet forces from the break), (2) debris that has a high density and will sink, but is still subject to fluid transport if local recirculation flow velocities are high enough, (3) debris that has an effective specific gravity near 1.0 and will float or sink slowly but will nonetheless be transported by very low velocities, and



(o) = SUMP OUTLET
(i) = PUMP INLET

* AS DETERMINED DURING SAFETY ANALYSIS

** CUBIC OR HORIZONTAL SUPPRESSOR MAY BE USED WITH EITHER SUMP OUTLET

FIGURE 1. PWR

(4) debris that will float indefinitely by virtue of low density and will be transported to and possibly through the debris screen. Thus, debris generation, early transport due to blowdown loads, long-term transport, and attendant blockage of debris interceptors must be analyzed to determine head loss effects. Appendix A provides relevant information for such evaluations; References 1 through 12 provide additional information relevant to the above concerns.

The design of sumps and their outlets includes consideration of the avoidance of air ingestion and other undesirable hydraulic effects (e.g., circulatory flow patterns, outlet designs leading to high head losses). The location and size of the sump outlets within ECC sumps is important in order to minimize air ingestion since ingestion is a function of submergence level and velocity in the outlet piping. It has been experimentally determined for PWRs that air ingestion can be minimized or eliminated if the sump hydraulic design considerations provided in Appendix A are followed. References 1, 3, 6, 7, and 8 provide additional technical information relevant to sump ECC hydraulic performance and design guidelines.

Placement of the ECC sumps at the lowest level practical ensures maximum use of available recirculation coolant. However, since there may be places within the containment where coolant could accumulate during the containment spray period, these areas can be provided with drains or flow paths to the sumps to prevent coolant holdup. This guide does not address the design of such drains or paths. However, since debris can migrate to the sump via these drains or paths, they are best terminated in a manner that will prevent debris from being transported to and accumulating on or within the ECC sumps.

Containment drainage sumps are used to collect and monitor normal leakage flow for leakage detection systems within containments. They are separated from the ECC sumps and are located at an elevation lower than the ECC sumps to minimize inadvertent spillover into the ECC sumps due to minor leaks or spills within containment. The floor adjacent to the ECC sumps would normally slope downward, away from the ECC sumps, toward the drainage collection sumps. This downward slope away from the ECC sumps will minimize the transport and collection of debris against the debris interceptors. High-density debris may be swept along the floor by the flow toward the trash rack. A debris curb upstream of and in close proximity to the rack will decrease the amount of such debris reaching the rack.

It is necessary to protect sump outlets by debris interceptors of sufficient strength to withstand the vibratory motion of seismic events, to resist jet loads and impact loads that could be imposed by missiles that may be generated by the initial LOCA, and to withstand the differential pressure loads imposed by the accumulation of debris. Considerations in selecting materials for the debris interceptors include long periods of inactivity, i.e., no submergence, and periods of operation involving partial or full submergence in a fluid that may contain chemically reactive materials. Isolation of the ECC sumps from high-energy pipe lines is an important consideration in protection against missiles, and it is necessary to shield the screens and racks adequately from impacts of ruptured high-energy piping and associated jet loads from the break. When the screen and rack structures are oriented vertically, the adverse effects from debris collecting on them will be reduced. Redundant ECC sumps and sump outlets are separated to the extent practical to reduce the possibility that an event causing the interceptors or outlets of one sump to either be damaged by missiles or partially clogged could adversely affect other pump circuits.

It is expected that the water surface will be above the top of the debris interceptor structure after completion of the safety injection. However, the uncertainties about the extent of water coverage on the structure, the amount of floating debris that may accumulate, and the potential for early clogging do not favor the use of a horizontal top interceptor. Therefore, in computation of available interceptor surface area, no credit may be taken for any horizontal interceptor surface; the top of the interceptor structure is preferably a solid cover plate that will provide additional protection from LOCA-generated loads and that is designed to provide for the venting of any trapped air.

Debris that is small enough to pass through the trash rack and thus could clog or block the debris screens or outlets needs to be analyzed for head loss effects. Screen and sump outlet blockage will be a function of the types and quantities of insulation debris that can be transported to these components. A vertical inner debris screen would impede the deposition or settling of debris on screen surfaces and thus help to ensure the greatest possible free flow through the fine inner debris screen. Slowly settling debris that is small enough to pass through the trash rack openings could block the debris screens if the coolant flow velocity is too great to permit the bulk of the debris to sink to the floor level during transport. If the coolant flow velocity ahead of the screen is at or below approximately 5 cm/sec (0.2 ft/sec), debris with a specific gravity of 1.05 or more is likely to settle before reaching the screen surface and thus will help to prevent undue clogging of the screen.

The size of openings in the screens is dependent on the physical restrictions that may exist in the systems that are supplied with coolant from the ECC sump. The size of the mesh of the fine debris screen is determined based on consideration of a number of factors, including the size of openings in the containment spray nozzles, coolant channel openings in the core fuel assemblies, and such pump design characteristics as seals, bearings, and impeller running clearances.

As noted above, degraded pumping can be caused by a number of factors, including plant design and layout. In particular, debris blockage effects or debris interceptor and sump outlet configurations and post-LOCA hydraulic conditions (e.g., air ingestion) must be considered in a combined manner. Small amounts of air ingestion, i.e., 2% or less, will not lead to severe pumping degradation if the "required" NPSH from the pump manufacturer's curves is increased based on the calculated air ingestion. Thus the combined results of all post-LOCA effects need to be used to estimate NPSH margin as calculated for the pump inlet. Appendix A provides information for estimating NPSH margins in PWR sump designs where estimated levels of air ingestion are low (2% or less). References 1 and 8 provide additional technical findings relevant to NPSH effects on pumps performing the functions of residual heat removal, emergency core cooling, and containment atmosphere cleanup. When air ingestion is 2% or less, compensation for its effects may be achieved without redesign if the "available" NPSH is greater than the "required" NPSH plus a margin based on the percentage of air ingestion. If air ingestion is not small, redesign of one or more of the recirculation loop components may be required to achieve satisfactory design.

To ensure the operability and structural integrity of the racks and screens, access openings are necessary to permit inspection of the ECC sump structures and outlets. Inservice inspection of racks, screens, vortex suppressors, and sump outlets, including visual examination for evidence of structural degradation or corrosion, can be performed on a regular basis at every refueling period downtime. Inspection of the ECC sump components late in the refueling period will ensure the absence of construction trash in the ECC sump area.

2. Boiling Water Reactors

In boiling water reactors (BWRs), the suppression pool in conjunction with the drywell, downcomers, and vents, serves as the water source for effecting long-term recirculation cooling and for fission product removal. This source, the related pump inlets, and the piping between them are important safety components. These components are referred to in this guide as the suppression pool. Features and relationships of the suppression pool pertinent to this guide are shown in Figure 2. There are concerns with the performance of the suppression pool and pump inlets that are similar to those associated with the ECC sumps in PWRs, i.e., post-LOCA hydraulic effects (particularly air ingestion), blockage of debris interceptors resulting from LOCA destruction of insulation and its transport (including suppression pool bulk velocity effects), and the combined effects of these items relative to the operability of the recirculation pump (e.g., the impact on NPSH available at the pump inlet). References 1 and 7 provide data on the performance and air ingestion characteristics of BWR configurations.

As in the case of PWRs, it is desirable to include consideration of the use of debris interceptors in BWR designs to protect the pump inlets. However, the location of the debris interceptors need not be restricted to the pool itself. Debris interceptors or equivalent plant structures in the drywell in the vicinity of the downcomers or vents could serve effectively in reducing debris transport to the pump inlets.

Similarly, the smallest opening in the debris interceptors is dependent on the physical restrictions that may exist in the systems served by the suppression pool. For example, spray nozzle clearances, coolant channel openings in the core fuel assemblies, and such pump design characteristics as seals, bearings, and impeller running clearances will need to be considered in the design.

C. REGULATORY POSITION

1. Pressurized Water Reactors

Reactor building sumps that are designed to be a source of water for the functions of emergency core cooling, containment heat removal, or containment atmosphere cleanup following a LOCA should meet the following criteria:

1. A minimum of two sumps should be provided, each with sufficient capacity to service one of the redundant halves of the ECCS and CSS.

2. To the extent practical, the redundant sumps should be physically separated by structural barriers from each other and from high-energy piping systems to preclude damage to the sump components (e.g., racks, screens, and sump outlets) by whipping pipes or high-velocity jets of water or steam.

3. The sumps should be located on the lowest floor elevation in the containment exclusive of the reactor vessel cavity. The sump outlets should be protected by at least two vertical debris interceptors: (a) a fine inner debris screen and (b) a coarse outer trash rack to prevent large debris from reaching the debris screen. A curb should be provided upstream of the trash racks to prevent high-density debris from being swept along the floor into the sump.

4. The floor in the vicinity of the ECC sump should slope gradually downward away from the sump.

5. All drains from the upper regions of the reactor building should terminate in such a manner that direct streams of water, which may contain entrained debris, will not impinge on the debris interceptors.

6. The strength of the trash racks should be adequate to protect the debris screens from missiles and other large debris. Each interceptor should be capable of withstanding the loads imposed by missiles, by the accumulation of debris, and by head differentials due to blockage.

7. The available interceptor surface area used in determining the design coolant velocity should be calculated to conservatively account for blockage that may result. Only the vertical interceptor area that is

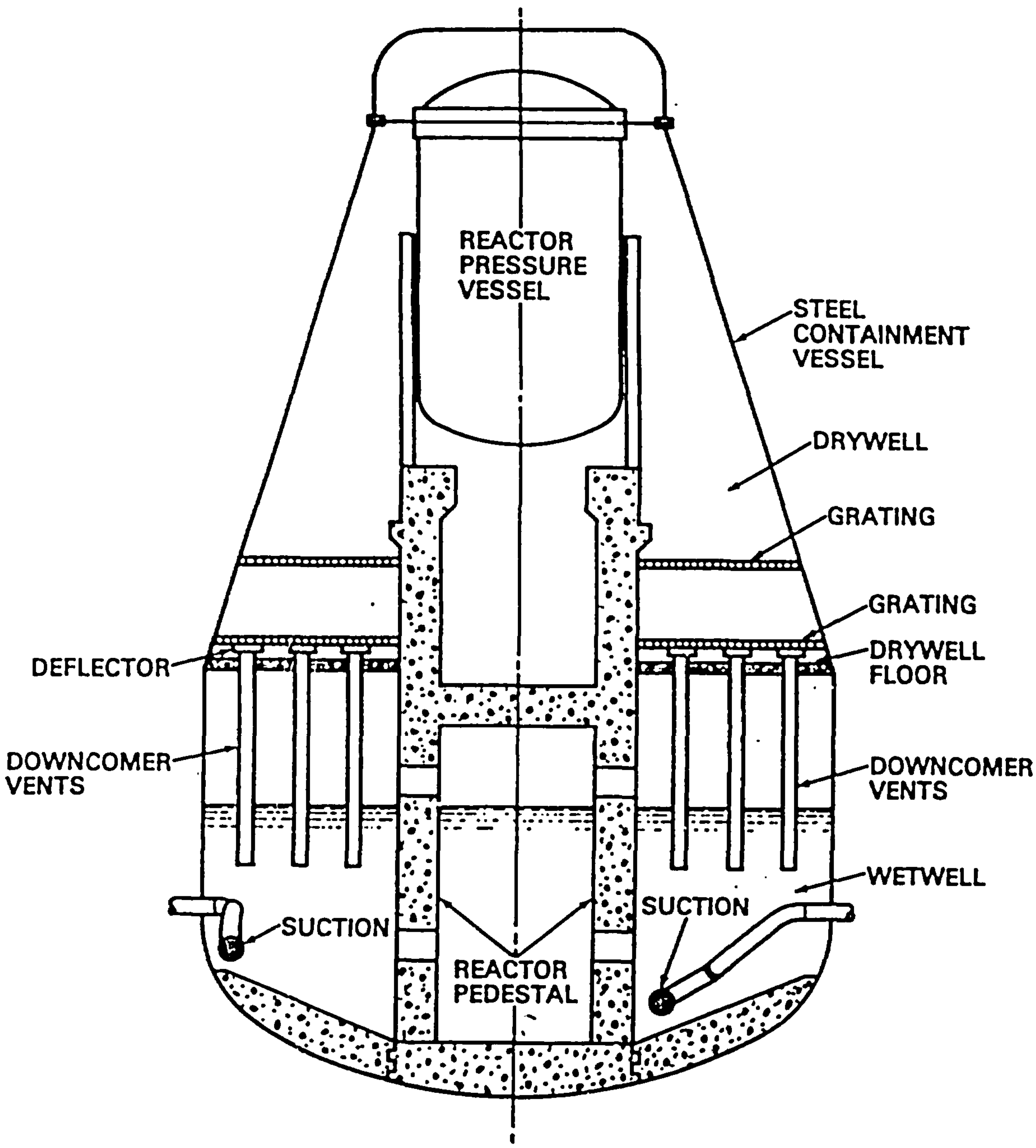

REACTOR
PRESSURE
VESSEL
STEEL
CONTAINMENT
VESSEL
DRYWELL
GRATING
GRATING
DEFLECTOR
DRYWELL
FLOOR
DOWNCOMER
VENTS
DOWNCOMER
VENTS
WETWELL
SUCTION
SUCTION
REACTOR
PEDESTAL

FIGURE 2. BWR

below the design basis water level should be considered in determining available surface area. Fibrous insulation debris should be considered as uniformly distributed over the available debris screen area. Blockage should be calculated based on levels of destruction estimated (Refs. 1 and 12).

8. Evaluation or confirmation of (a) sump hydraulic performance (e.g., geometric effects and air ingestion), (b) debris effects (e.g., debris transport, interceptor blockage, and head loss), and (c) the combined impact on NPSH available at the pump inlet should be performed to ensure that long-term recirculation cooling can be accomplished. Such evaluation should arrive at a determination of NPSH margin calculated at the pump inlet. An assessment of the susceptibility of the recirculation pump seal and bearing assembly design to failure due to particulate ingestion and abrasive effects should be made to protect against degradation of long-term recirculation pumping capacity.

9. The top of the debris interceptor structures should be a solid cover plate that is designed to be fully submerged after a LOCA and completion of the ECC injection. It should be designed to ensure the venting of air otherwise trapped underneath.

10. The debris interceptors should be designed to withstand the vibratory motion of seismic events without loss of structural integrity.

11. The size of openings in the debris screens should be based on the minimum restriction found in systems served by the pumps performing the recirculation function. The minimum restriction should take into account the requirements of the systems served.

12. Sump outlets should be designed to prevent degradation of pump performance by air ingestion and other adverse hydraulic effects (e.g., circulatory flow patterns, high intake-head losses).

13. Materials for debris interceptors should be selected to avoid degradation during periods of inactivity and operation and should have a low sensitivity to such adverse effects as stress-assisted corrosion that may be induced by the chemically reactive spray during LOCA conditions.

14. The debris interceptor structures should include access openings to facilitate inspection of these structures, any vortex suppressors, and the sump outlets.

15. Inservice inspection requirements for ECC sump components (i.e., debris interceptors, any vortex suppressors, and sump outlets) should include:

a. Inspection during every refueling period downtime, and

b. A visual examination for evidence of structural distress or corrosion.

2. Boiling Water Reactors

The suppression pool, which is the source of water for such functions as emergency core cooling, containment heat removal, and containment atmosphere cleanup following a LOCA in conjunction with the vents and downcomers between the drywell and the wetwell, should contain the following features:

1. The inlet of pumps performing the above functions should be protected by two debris interceptors:

a. A fine downstream debris screen and

b. A coarse upstream trash rack to prevent large debris from reaching the debris screen.

It should be noted that certain design features of BWRs may perform a function equivalent to that of trash racks and debris screens. Design features such as deflectors and suction strainers may be considered equivalent to trash racks and debris screens. The terms "trash rack" and "debris screen" include equivalent plant features.

2. If it is demonstrated that significant amounts of debris will not be generated within the wetwell, the trash rack may be located in the drywell or the downcomer system between the drywell and wetwell.

3. All drains from the upper regions of the reactor building should terminate in such a manner that direct streams of water, which may contain entrained debris, will not impinge on the debris interceptors.

4. The strength of the trash rack should be adequate to protect the debris screen from missiles and other large debris. Each interceptor should be capable of withstanding the loads imposed by missiles, by debris, and by head differentials due to blockage.

5. Bulk suppression pool velocity due to recirculation operation should be considered for both debris transport and coolant velocity computations.

6. The available interceptor area used in determining the design coolant velocity should conservatively account for blockage that may result. Fibrous debris should be assumed to be uniformly distributed over the available debris screen surface. Blockage should be calculated based on levels of destruction estimated. (See Refs. 1 and 12.)

7. Evaluation or confirmation of (a) suppression pool hydraulic performance (e.g., geometric effects and air ingestion), (b) debris effects (e.g., debris transport, interceptor blockage and head loss, and clogging of pump seals by particulates), and (c) the combined impact on NPSH available at the pump inlet should be performed to ensure that long-term recirculation cooling can be accomplished. An assessment of the susceptibility of the recirculation pump seal and bearing assembly

design to failure due to particulate ingestion and abrasive effects should be made to protect against degradation of long-term recirculation pumping capacity.

8. The debris interceptors should be designed to withstand the vibratory motion of seismic events without loss of structural integrity.

9. The size of openings in the screens should be based on the minimum restriction found in systems served by the suppression pool. The minimum restriction should take into account the operability of the systems served.

10. The pool outlets to the recirculation pumps should be designed to prevent degradation of pump performance through air ingestion and other adverse hydraulic effects (e.g., circulatory flow patterns, high intake-head losses).

11. Material for debris interceptors should be selected to avoid degradation during periods of inactivity and normal operations and should be compatible with the characteristics of the spray during LOCA events.

12. Inservice inspection requirements should include:

(a) inspection during every refueling period downtime,

(b) a visual examination for evidence of structural distress or corrosion, and

(c) an inspection, for evidence of debris or trash, of the wetwell air spaces and the drywell floor region, including the vents, downcomers, and deflectors.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants regarding the NRC staff's plans for using this regulatory guide. This regulatory guide has been developed from an extensive experimental and analytical data base. The applicant is free to select alternative calculation methods that are founded on substantiating experiments or limiting analytical considerations. Except in those cases in which the applicant proposes an alternative method for complying with the specified portions of the Commission's regulations, the methods described in this guide will be used by the NRC staff in its evaluation of all:

1. Construction permit applications and applications for preliminary design approval that are docketed after May 1986.

2. Applications for final design approval of standardized designs that are intended for referencing in future construction permit applications and have not received approval by May 1986.

3. Applications for licenses to manufacture that are docketed after May 1986.

APPENDIX A

GUIDELINES FOR REVIEW OF SUMP DESIGN AND WATER SOURCES FOR EMERGENCY CORE COOLING

The ECC sump performance should be evaluated under possible post-LOCA conditions to determine design adequacy for providing long-term recirculation. Technical evaluations can be subdivided into (1) sump hydraulic performance, (2) LOCA-induced debris effects, and (3) pump performance under adverse conditions. Specific considerations within these categories, and the combining thereof, are shown in Figure A-1. Determination that adequate NPSH margin exists at the pump inlet under all postulated post-LOCA conditions is the final requirement.

Sump Hydraulic Performance

Sump hydraulic performance (with respect to air ingestion potential) can be evaluated on the basis of submergence level (or water depth above the sump outlets) and required pumping capacity (or pump inlet velocity). The water depth above the pipe centerline (s) and the inlet pipe velocity (U) can be expressed non-dimensionally as the Froude number:

$$\text{Froude number} = U/\sqrt{gs}$$

where g is the acceleration due to gravity. Extensive experimental results have shown that the hydraulic performance of ECC sumps (particularly the potential for air ingestion) is a strong function of the Froude number. Other nondimensional parameters (e.g., Reynolds number and Weber number) are of secondary importance.

Sump hydraulic performance can be divided into three performance categories:

1. Zero air ingestion, which requires no vortex suppressors or increase of the "required" NPSH above that from the pump manufacturer's curves.

2. Air ingestion 2% or less, a conservative level at which degradation of pumping capability is not expected based on an increase of the "required" NPSH (see Figure A-2),

3. Use of vortex suppressors to reduce air ingestion effects to zero.

For PWRs, zero air ingestion can be ensured by use of the design guidance set forth in Table A-1. Determination of those designs having air ingestion levels of 2% or less can be obtained using correlations given in Table A-2 and the attendant sump geometric envelope. Geometric and screen guidelines for PWRs are contained in Tables A-3.1, A-3.2, A-4, and A-5. Table A-6 presents design guidelines for vortex suppressors that have shown the capability to reduce air ingestion to zero. These guidelines (Tables A-1 through A-6) were developed from extensive hydraulic tests on full-scale sumps and provide a rapid means of assessing sump hydraulic performance. If the PWR sump design deviates significantly from the design boundaries noted, similar performance data should be obtained for verification of adequate sump hydraulic performance.

For BWRs, full-scale tests of pool outlet designs for recirculation pumps have shown that air ingestion is zero for Froude numbers less than 0.8 with a minimum submergence of 6 feet, and operation up to a Froude number 1.0 with the same minimum submergence may be possible before air ingestion levels of 2% may occur (Refs. 1 and 7).

LOCA-Induced Debris Effects

Assessment of LOCA debris generation and determination of possible debris interceptor blockage is complex. The evaluation of this safety question is dependent on the types and quantities of insulation employed, the location of such insulation materials within containment and with respect to the sump location, the estimation of quantities of debris generated by a pipe break, and the migration of such debris to the interceptors. Thus blockage estimates are specific to the insulation material and the plant design and require consideration of such effects as are outlined in Table A-7.

Since break jet forces are the dominant debris generator, the predicted jet envelope will determine the quantities and types of insulation debris. Figure A-3 provides a three-region model that has been developed from analytical and experimental considerations as identified in Reference 1. The destructive results of the break jet forces will be considerably different for different types of insulation and must be individually addressed. The insulation type, how and whether it is encapsulated, and how it is fastened to the insulated surfaces all have significant influence on the maximum volume of insulation debris generated. Region I represents a total destruction zone; Region II a region where high levels of damage are possible depending on insulation type, whether encapsulation is employed, methods of attachment, etc.; and Region III, a region where dislodgement of insulation in whole, or as-fabricated, segments is likely occur. A more detailed discussion of these considerations is provided in Reference 1. Use of the outer boundary of Region II for estimating maximum volumes of total insulation destruction is considered a conservative bounding condition.

References 1, 9, 10, 11, and 12 provide more detailed information relevant to assessment of debris generation and transport.

Pump Performance Under Adverse Conditions

The pump industry historically has determined net positive suction head requirements for pumps on the basis of a percentage degradation in pumping capacity. The percentage has been at times arbitrary, but generally in the range of 1% to 3%. A 2% limit on allowed air ingestion is recommended since higher levels have been shown to initiate degradation of pumping capacity.

The 2% by volume limit on sump air ingestion and the NPSH requirements act independently. However, air ingestion levels less than 2% can also affect NPSH requirements. If air ingestion is indicated, correct the NPSH requirement from the pump curves by the relationship:

$$NPSH_{required(\alpha_p < 2\%)} = NPSH_{required(liquid)} \times \beta$$

where $\beta = 1 + 0.50\alpha_p$ and α_p is the air ingestion rate (in percent by volume) at the pump inlet flange.

Combined Effects

As shown in Figure A-1, three interdependent effects (i.e., sump hydraulic performance, debris effects, and pump operation under adverse conditions) require evaluation for determining long-term recirculation capability. Figure A-2 provides a logic diagram for combining these considerations to evaluate the ECC sump design and expected performance. The same logic applies to BWR design evaluations of suppression pools and the outlets to recirculation pumps.

TABLE A-1

HYDRAULIC DESIGN GUIDELINES* FOR ZERO AIR INGESTION

Item	Horizontal Outlets	Vertical Outlets
Minimum Submergence, s (ft)	9	9
(m)	2.7	2.7
Maximum Froude Number, Fr	0.25	0.25
Maximum Pipe Velocity, U (ft/s)	4	4
(m/s)	1.2	1.2

*These guidelines were established using experimental results from References 3, 4, and 5 and are based on sumps having a right rectangular shape.

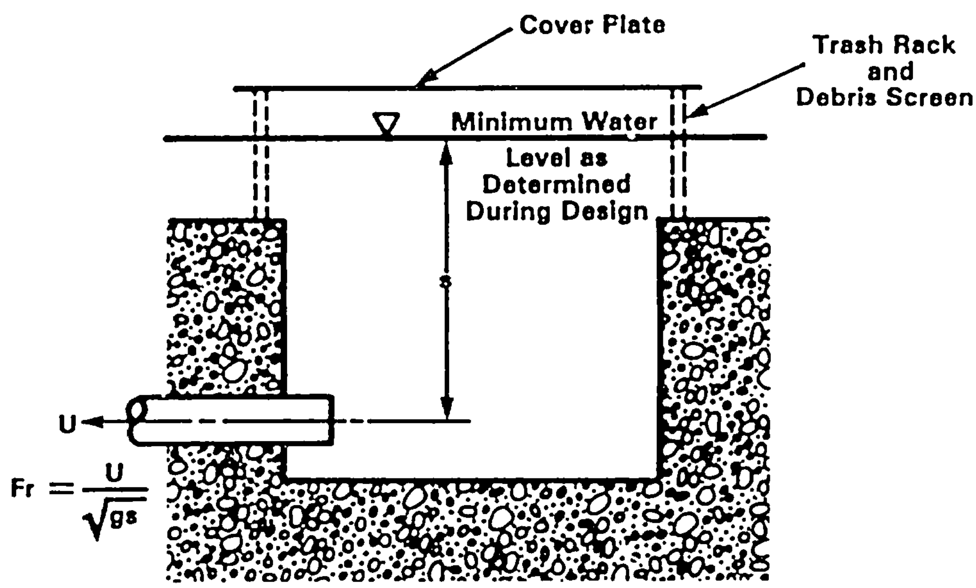
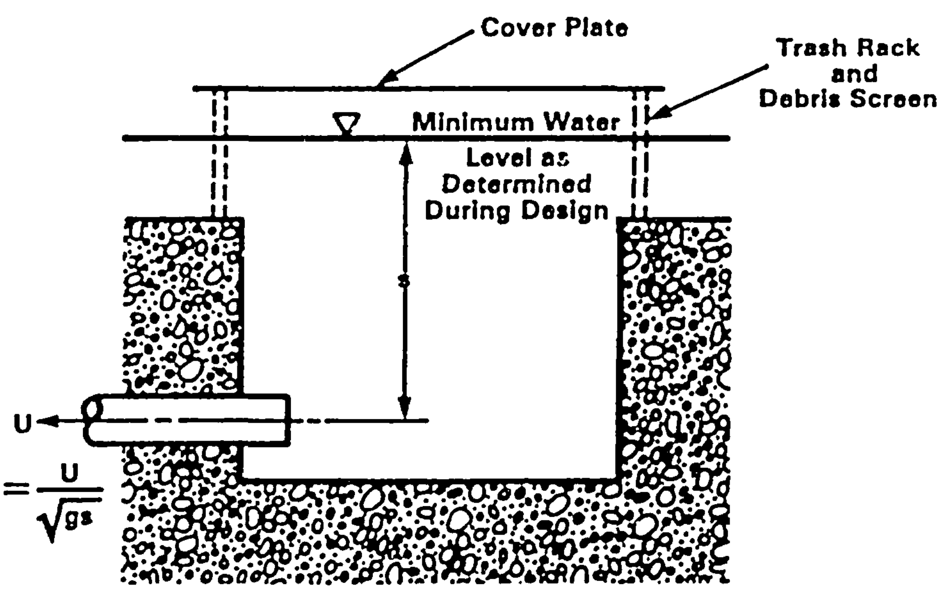


TABLE A-2

HYDRAULIC DESIGN GUIDELINES FOR AIR INGESTION <2%

Air ingestion (α) is empirically calculated as
$\alpha = \alpha_o + (\alpha_1 \times Fr)$
where α_o and α_1 are coefficients derived from test results as given in the table below.

Item	Horizontal Outlets		Vertical Outlets	
	Dual	Single	Dual	Single
Coefficient α_o	-2.47	-4.75	-4.75	-9.14
Coefficient α_1	9.38	18.04	18.69	35.95
Minimum Submergence, s (ft)	7.5	8.0	7.5	10.0
(m)	2.3	2.4	2.3	3.1
Maximum Froude Number, Fr	0.5	0.4	0.4	0.3
Maximum Pipe Velocity, U (ft/s)	7.0	6.5	6.0	5.5
(m/s)	2.1	2.0	1.8	1.7
Maximum Screen Face Velocity (blocked and minimum submergence) (ft/s)	3.0	3.0	3.0	3.0
(m/s)	0.9	0.9	0.9	0.9
Maximum Approach Flow Velocity (ft/s)	0.36	0.36	0.36	0.36
(m/s)	0.11	0.11	0.11	0.11
Maximum Sump Outlet Coefficient, C_L	1.2	1.2	1.2	1.2



$$Fr = \frac{U}{\sqrt{gs}}$$

TABLE A-3.1

GEOMETRIC DESIGN ENVELOPE GUIDELINES FOR HORIZONTAL SUCTION OUTLETS*

<table>
<tr><th rowspan="3">Sump Outlet</th><th colspan="3">Size</th><th colspan="6">Sump Outlet Position**</th><th colspan="2">Screen</th></tr>
<tr><th rowspan="2">Aspect Ratio</th><th colspan="2">Min. Perimeter</th><th rowspan="2">e_y/d</th><th rowspan="2">$(B - e_y)/d$</th><th rowspan="2">c/d</th><th rowspan="2">b/d</th><th rowspan="2">f/d</th><th rowspan="2">e_x/d</th><th colspan="2">Min. Area</th></tr>
<tr><th>(ft)</th><th>(m)</th><th>(ft^2)</th><th>(m^2)</th></tr>
<tr><td>Dual</td><td>1 to 5</td><td>36</td><td>11</td><td rowspan="2">>1</td><td rowspan="2">>3</td><td rowspan="2">>1.5</td><td rowspan="2">>1</td><td>>4</td><td rowspan="2">>1.5</td><td>75</td><td>7</td></tr>
<tr><td>Single</td><td>1 to 5</td><td>16</td><td>4.9</td><td>–</td><td>35</td><td>3.3</td></tr>
</table>

*Dimensions are always measured to pipe centerline.
**Preferred location.

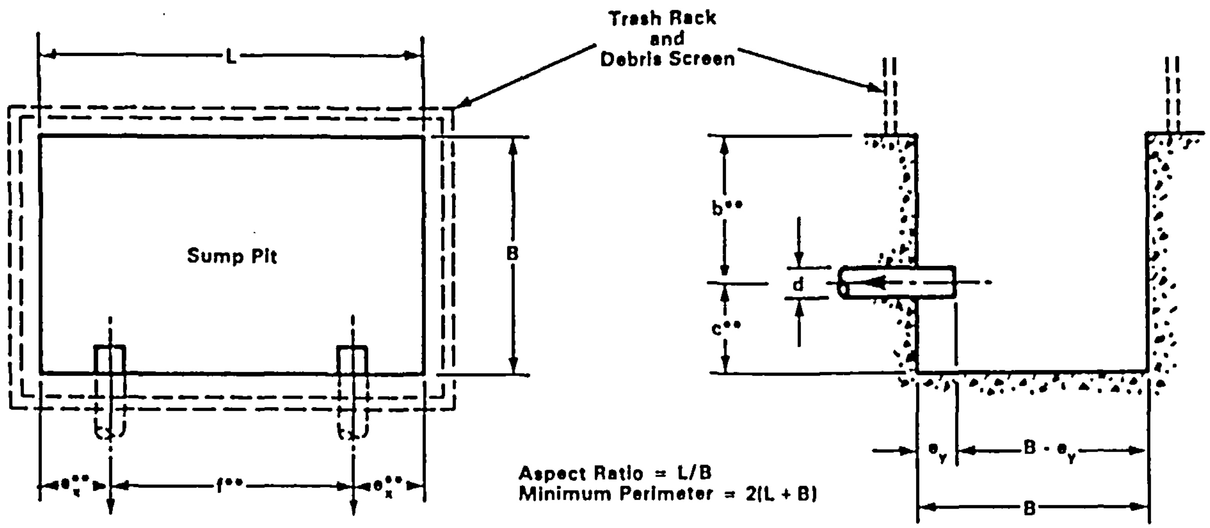
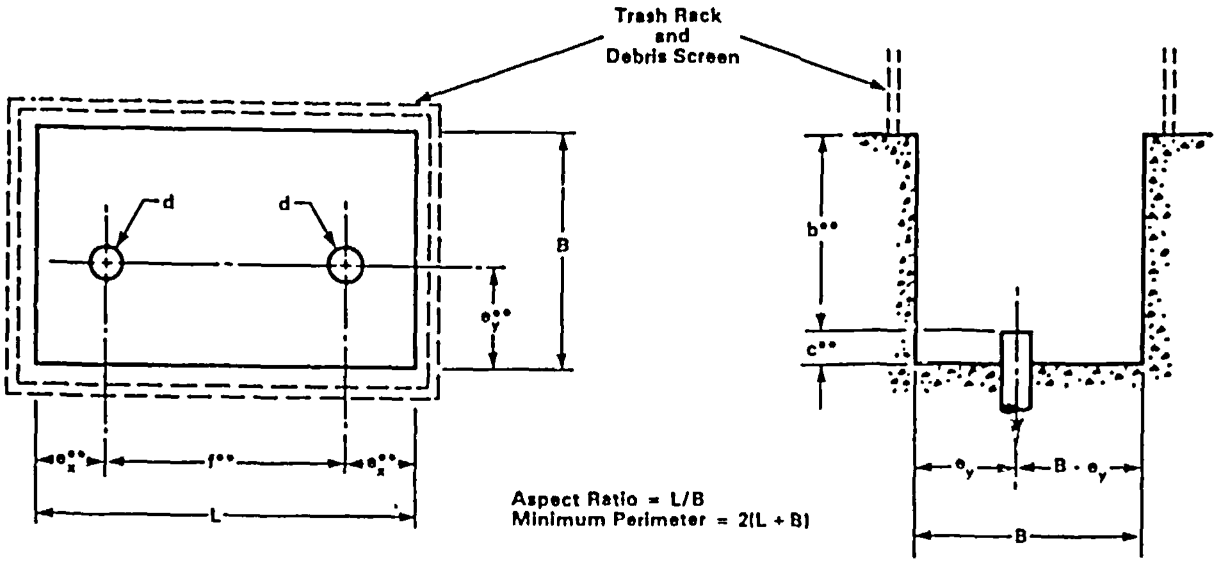


TABLE A-3.2

GEOMETRIC DESIGN ENVELOPE GUIDELINES FOR VERTICAL SUCTION OUTLETS*

<table>
<tr><th rowspan="2">Sump Outlet</th><th colspan="3">Size</th><th colspan="6">Sump Outlet Position**</th><th colspan="2">Screen</th></tr>
<tr><th>Aspect Ratio</th><th>Min. Perimeter (ft)</th><th>(m)</th><th>e_y/d</th><th>$(B - e_y)/d$</th><th>c/d</th><th>b/d</th><th>f/d</th><th>e_x/d</th><th>Min. Area (ft^2)</th><th>(m^2)</th></tr>
<tr><td>Dual</td><td>1 to 5</td><td>36</td><td>11</td><td rowspan="2">>1</td><td rowspan="2">>1</td><td>>0</td><td rowspan="2">>1</td><td>>4</td><td rowspan="2">>1.5</td><td>75</td><td>7</td></tr>
<tr><td>Single</td><td>1 to 5</td><td>16</td><td>4.9</td><td><1.5</td><td>–</td><td>35</td><td>3.3</td></tr>
</table>

*Dimensions are always measured to pipe centerline.
**Preferred location.



Aspect Ratio = L/B
Minimum Perimeter = 2(L + B)

TABLE A-4

ADDITIONAL GUIDELINES RELATED TO SUMP SIZE AND PLACEMENT

1. The clearance between the trash rack and any wall or obstruction of length ℓ equal to or greater than the length of the adjacent screen/grate (B_s or L_s) should be at least 4 ft (1.2 m).

2. A solid wall or large obstruction may form the boundary of the sump on one side only, i.e., the sump must have three sides open to the approach flow.

3. These additional guidelines should be followed to ensure the validity of the data in Tables A-1, A-2, A-3.1, and A-3.2.

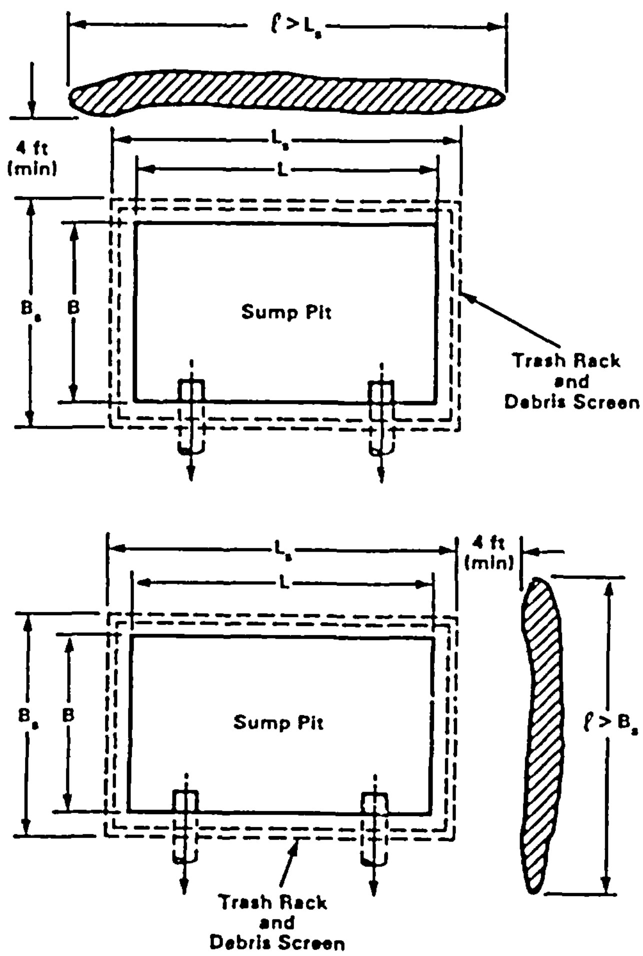


TABLE A-5

DESIGN GUIDELINES* FOR INTERCEPTORS AND COVER PLATE

1. Screen area should be obtained from Table A-3.1 and A-3.2.

2. Minimum height of interceptors should be 2 feet (0.61 m).

3. Distance from sump side to screens, g_s, may be any reasonable value.

4. Screen mesh should be ¼ inch (6.4 mm) or finer.

5. Trash racks should be vertically oriented 1- to 1½-inch (25- to 38-mm) standard floor grate or equivalent.

6. The distance between the debris screens and trash racks should be 6 inches (15.2 cm) or less.

7. A solid cover plate should be mounted above the sump and should fully cover the trash rack. The cover plate should be designed to ensure the release of air trapped below the plate (a plate located below the minimum water level is preferable).

*See Reference 1.

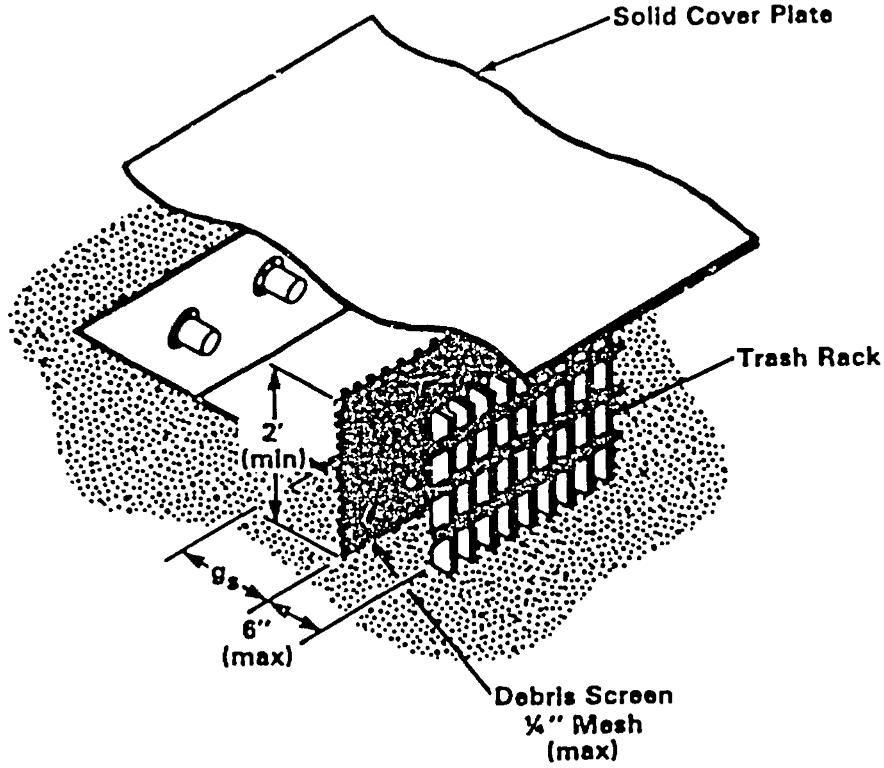


TABLE A-6

GUIDELINES FOR SELECTED VORTEX SUPPRESSORS*

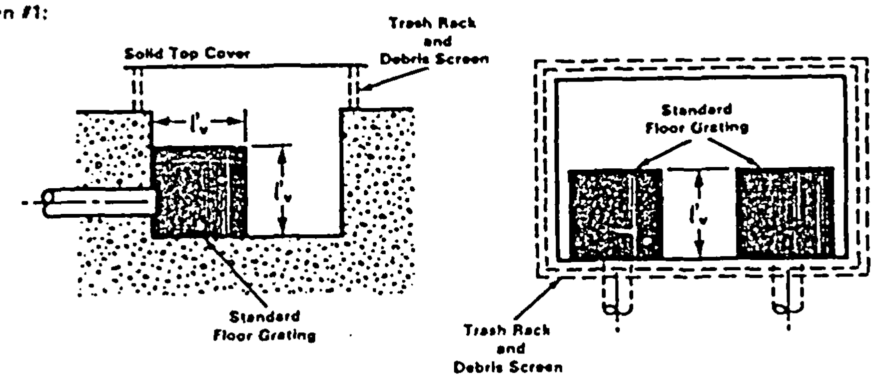
1. Cubic arrangement of standard 1½-inch (30-mm) deep or deeper floor grating (or its equivalent) with a characteristic length, ℓ_v, that is at least 3 pipe diameters and with the top of the cube submerged at least 6 inches (15.2 cm) below the minimum water level. Noncubic designs with ℓ_v > 3 pipe diameters for the horizontal upper grate and satisfying the depth and distances to the minimum water level given for cubic designs are acceptable.

2. Standard 1½-inch (38-mm) or deeper floor grating (or its equivalent) located horizontally over the entire sump and containment floor inside the screens and located below the lip of the sump pit.

---

*Tests on these types of vortex suppressors at Alden Research Laboratory have demonstrated their capability to reduce air ingestion to zero even under the most adverse conditions simulated.

Design #1:



Design #2:

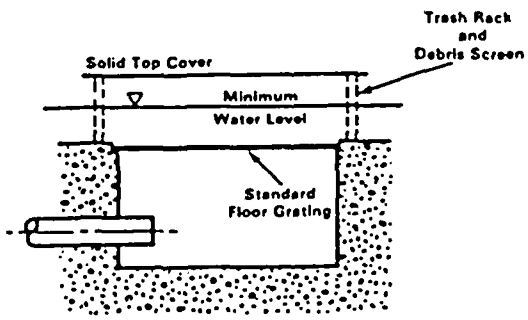


TABLE A-7

DEBRIS ASSESSMENT

CONSIDERATION	EVALUATE
1. Debris generator (pipe breaks and location as identified in SRP Section 3.6.2)	• Major pipe breaks and location • Pipe whip and pipe impact • Break jet expansion envelope (This is the *major* debris generator)
2. Expanding jets	• Jet expansion envelope • Piping and plant components targeted (i.e., steam generators) • Jet forces on insulation • Insulation that can be destroyed or dislodged by blowdown jets • Survivability under jet loading
3. Short-term debris transport (transport by blowdown jet forces)	• Jet/equipment interaction • Jet/crane wall interaction • Sump location relative to expanding break jet
4. Long-term debris transport (transport to the sump during the recirculation phase)	• Containment layout and sump (or suction) locations • Debris physical characteristics • Recirculation velocity • Debris transport velocity
5. Screen or sump outlet blockage effects (impairment of flow and/or NPSH margin)	• Screen or outlet area • Water level under post-LOCA conditions • Recirculation flow requirements • Head loss across blocked screen or outlet
6. Downstream blockage (effects of debris deposition and recirculation)	• Core coolant channels • Spray nozzles • Pump clearances
Key elements for assessment of debris effects	• Estimated amount and types of debris that can reach sump • Predicted screen or outlet blockage • ΔP across blocked screens or outlets • NPSH required vs NPSH available

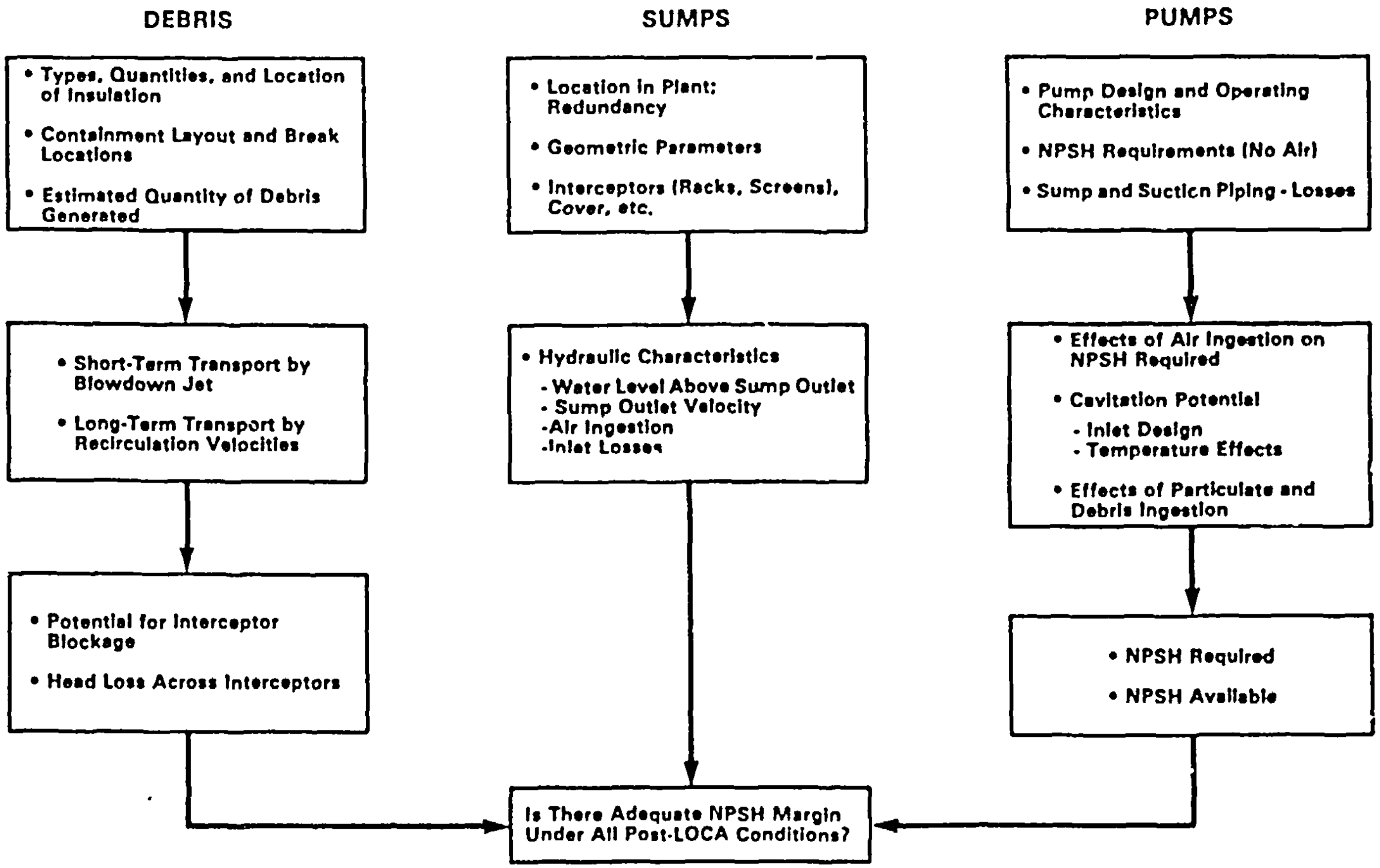


FIGURE A-1. Technical Consideration Relevant to ECC Sump Performance

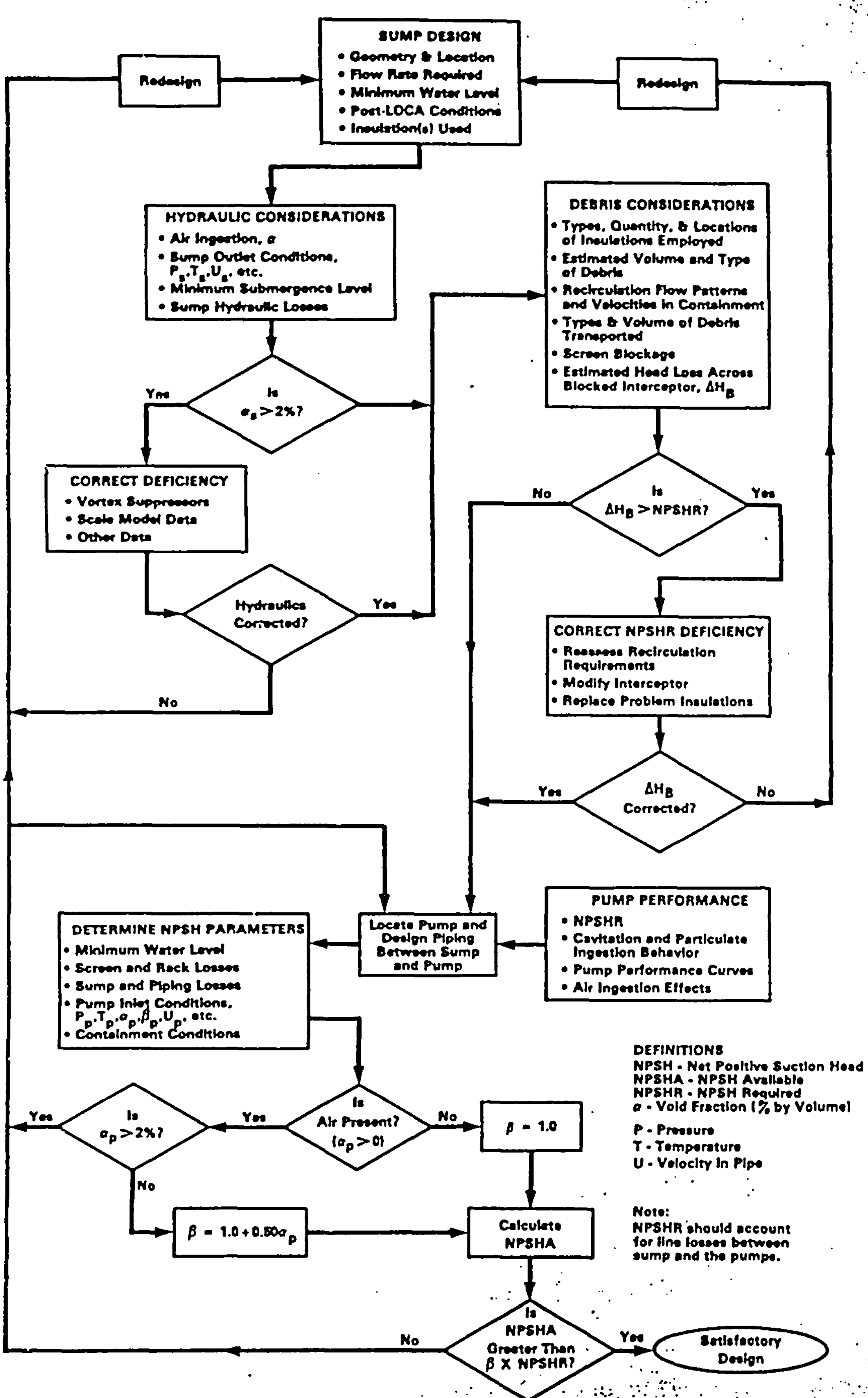


FIGURE A-2. Combined Technical Considerations for Sump Performance

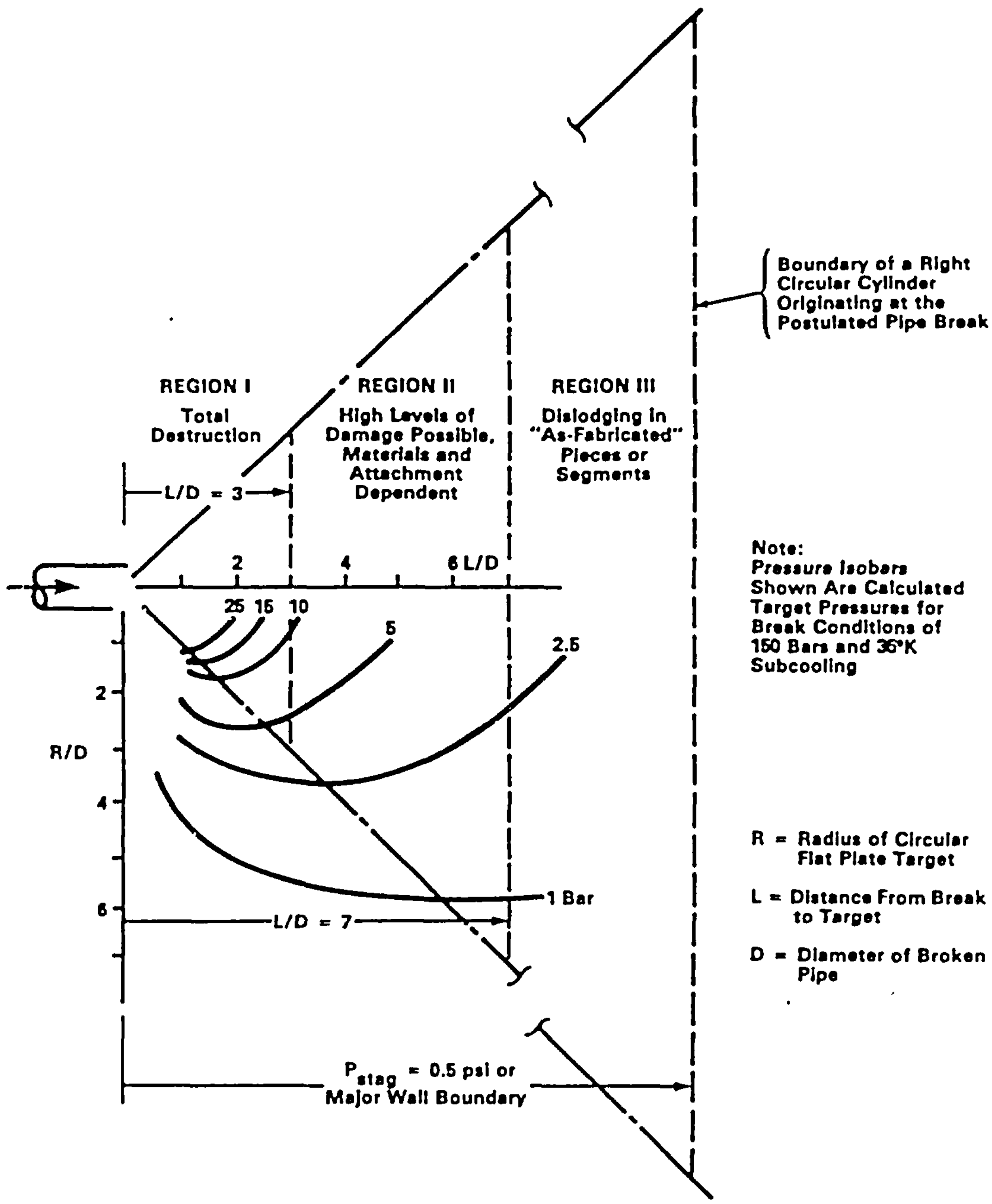


FIGURE A-3. Multiple Region Insulation Debris Model
(A discussion of the model is provided in Ref. 1)

REFERENCES

1. U.S. Nuclear Regulatory Commission, "Containment Emergency Sump Performance (Technical Findings Related to Unresolved Safety Issue A-43)," NUREG-0897, Revision 1, October 1985.

2. U.S. Nuclear Regulatory Commission, "Methodology for Evaluation of Insulation Debris Effects," NUREG/CR-2791 (SAND82-7067), September 1982.

3. U.S. Nuclear Regulatory Commission, "A Parametric Study of Containment Emergency Sump Performance," NUREG/CR-2758 (SAND82-0624), July 1982.

4. U.S. Nuclear Regulatory Commission, "A Parametric Study of Containment Emergency Sump Performance: Results of Vertical Outlet Sump Tests," NUREG/CR-2759 (SAND82-7062), October 1982.

5. U.S. Nuclear Regulatory Commission, "Assessment of Scale Effects on Vortexing, Swirl and Inlet Losses in Large Scale Sump Models," NUREG/CR-2760 (SAND82-7063), June 1982.

6. U.S. Nuclear Regulatory Commission, "Results of Vortex Suppressor Tests, Single Outlet Sump Tests and Miscellaneous Sensitivity Tests," NUREG/CR-2761 (SAND82-7065), September 1982.

7. U.S. Nuclear Regulatory Commission, "Hydraulic Performance of Pump Suction Inlets for Emergency Core Cooling Systems in Boiling Water Reactors," NUREG/CR-2772 (SAND82-7064), June 1982.

8. U.S. Nuclear Regulatory Commission, "An Assessment of Residual Heat Removal and Containment Spray Pump Performance Under Air and Debris Ingesting Conditions," NUREG/CR-2792 (CREARE TM-825), September 1982.

9. U.S. Nuclear Regulatory Commission, "Buoyancy, Transport, and Head Loss of Fibrous Reactor Insulation," NUREG/CR-2982 (SAND82-7205), Revision 1, July 1983.

10. U.S. Nuclear Regulatory Commission, "The Susceptibility of Fibrous Insulation Pillows to Debris Formation Under Exposure to Energetic Jet Flows," NUREG/CR-3170 (SAND83-7008), March 1983.

11. U.S. Nuclear Regulatory Commission, "Probabilistic Assessment of Recirculation Sump Blockage Due to Loss-of-Coolant Accidents," NUREG/CR-3394, Volumes 1 and 2 (SAND83-7116), July 1983.

12. U.S. Nuclear Regulatory Commission, "Transport and Screen Blockage Characteristics of Reflective Metallic Insulation Materials," NUREG/CR-3616 (SAND83-7471), January 1984.

NOTE: NUREG-series documents are available from the Superintendent of Documents, U.S. Government Printing Office, P.O. Box 37082, Washington, D.C. 20013-7982. Information on ordering may be obtained by calling (202) 275-2060 or 2171.

REGULATORY ANALYSIS

A separate regulatory analysis has not been prepared for the revision to this regulatory guide. The changes were made as a result of the resolution of unresolved safety issue (USI) A-43, "Containment Emergency Sump Performance." A regulatory analysis (NUREG 0869, Revision 1, October 1985) prepared for the resolution of USI A-43 was made available in the Commission's Public Document Room, 1717 H Street NW., Washington, D.C., at the time of its publication. This analysis is appropriate for Revision 1 to Regulatory Guide 1.82.

NUREG-0897
Rev. 1

Containment Emergency Sump Performance

Technical Findings Related to
Unresolved Safety Issue A-43

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

A. W. Serkiz, Task Manager

NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 1717 H Street, N.W. Washington, DC 20555
2. The Superintendent of Documents, U.S. Government Printing Office, Post Office Box 37082, Washington, DC 20013-7082
3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC Office of Inspection and Enforcement bulletins, circulars, information notices, inspection and investigation notices; Licensee Event Reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, and NRC booklets and brochures. Also available are Regulatory Guides, NRC regulations in the *Code of Federal Regulations,* and *Nuclear Regulatory Commission Issuances.*

Documents available from the National Technical Information Service include NUREG series reports and technical reports prepared by other federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal and periodical articles, and transactions. *Federal Register* notices, federal and state legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, and are available there for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

NUREG-0897
Rev. 1

Containment Emergency Sump Performance

Technical Findings Related to Unresolved Safety Issue A-43

Manuscript Completed: March 1985
Date Published: October 1985

A. W. Serkiz, Task Manager

Division of Safety Technology
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ABSTRACT

This report summarizes key technical findings related to Unresolved Safety Issue (USI) A-43, Containment Emergency Sump Performance. Although this issue was formulated considering pressurized water reactor (PWR) sumps, the generic safety questions apply to both boiling water reactors (BWRs) and PWRs. Hence, both BWRs and PWRs are considered in this report.

Emergency core cooling systems require a clean, reliable water source to maintain long-term recirculation following a loss-of-coolant accident (LOCA). PWRs rely on the containment emergency sump to provide such a water supply to residual heat removal pumps and containment spray pumps. BWRs rely on pump suction intakes in the suppression pool or wet well to provide water to residual heat removal and core spray systems.

Thus, the technical findings in this report provide information on post-LOCA recirculation. These findings have been derived from extensive experimental studies, generic plant studies, and assessments of sumps used for long-term cooling. The results of hydraulic tests have shown that the potential for air ingestion is less severe than previously hypothesized. The effects of debris blockage on NPSH margin must be dealt with on a plant-specific basis. These findings have been used to develop revisions to Regulatory Guide 1.82 and Standard Review Plan Section 6.2.2 (NUREG-0800).

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

LIST OF FIGURES

LIST OF FIGURES (Continued)

LIST OF FIGURES (Continued)

LIST OF TABLES

FOREWORD

This report has been prepared to provide a concise and self-contained reference that summarizes technical findings relevant to Unresolved Safety Issue A-43, Containment Emergency Sump Performance. This report was originally issued for public comment in May 1983; comments received were reviewed, and those of substantive technical or informational content have been incorporated into this Revision 1. It should be clearly noted that this report is not a substitute for requirements set forth in General Design Criteria 16, 35, 36, 38, 40, and 50 in Appendix A of Title 10 of the Code of Federal Regulations Part 50, nor is this document a substitute for guidelines set forth in NRC's Standard Review Plan (SRP, NUREG-0800), regulatory guides, or other regulatory directives. The information contained herein is of a technical nature and can be used as reference material relevant to the revised SRP Section 6.2.2 (Revision 4) and Regulatory Guide 1.82, Revision 1.

ACKNOWLEDGEMENTS

The technical findings relevant to Unresolved Safety Issue A-43, Containment Emergency Sump Performance, set forth in this report are the result of the combined efforts of the staff of the Nuclear Regulatory Commission (NRC), the Department of Energy (DOE), Sandia National Laboratories (SNL), Alden Research Laboratory (ARL), Burns & Roe (B&R), Inc., and Creare, Inc. The following persons deserve special mention for their participation and contributions:

W. Butler, NRC/DSI
W. Durgin, ARL
G. Hecker, ARL
D. Jaffee, NRC/DL
J. Kudrick, NRC/DSI
A. Mullunzi, DOE
P. Norian, NRC/DST
F. Orr, NRC/DST
G. Otey, SNL
M. Padmanabhan, ARL
S.K. Shaukat, NRC/DST
P. Strom, SNL
W. Swift, Creare
G. Weigand, SNL
F. Wind, HDR Project
J. Wysocki, B&R

Acknowledgement is also given to persons whose efforts are referenced herein, and to those persons who participated in peer reviews of the results obtained and the application of such results. Particular acknowledgement is given to Gilbert Weigand (SNL), who played a major role in developing investigative approaches and maintaining technical quality and continuity in those early efforts, and to G. Hecker (ARL), for his keen insight and questioning regarding the application of the results obtained, particularly in the concluding phases of resolving this USI. In addition, acknowledgement is given to the Diamond Power Speciality Company and Owens Corning Fiberglass, Inc. for providing the technical data and other product line information that have been included in this report.

A. W. Serkiz
Task Manager

1 INTRODUCTION

1.1 Safety Significance

After a loss-of-coolant accident (LOCA) in a pressurized water reactor (PWR), water discharged from the break will collect on the containment floor and within the containment emergency sump. PWR emergency core cooling systems (ECCS) and containment spray systems (CSS) initially draw water from the refueling water storage tank (RWST); long-term cooling is implemented by realignment of these ECCS pumps to the containment emergency sump. In boiling water reactors (BWRs), the break flow collects in the suppression pool (or torus), and the residual heat removal (RHR) and core spray (CS) systems take suction from intakes located in the suppression pool. Thus successful long-term recirculation depends on the PWR sump design--or BWR suction intake design--to provide adequate, debris-free water to the RHR recirculation pumps for extended periods of time.

The primary areas of safety concern addressed in this report are as follows:

(1) post-LOCA hydraulic effects (i.e., air ingestion potential)

(2) generation of insulation debris as a result of a LOCA, with subsequent transport of the debris to PWR sump screens (or BWR suction strainers) and blockage thereof

(3) the combined effects of (1) and (2) on the required recirculation pumping capacity (i.e., impact on net positive suction head (NPSH) of the recirculation pumps)

1.2 Background

The importance of the ECCS sump and the safety considerations associated with its design were early considerations in PWR containment design. Net positive suction head (NPSH) requirements, operational verification, and sump design requirements are issues that have evolved and are addressed in the following Nuclear Regulatory Commission (NRC) regulatory guides (RGs):

RG 1.1	Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Systems Pumps, 1970
RG 1.79	Preoperational Testing of Emergency Core Cooling Systems for PWRs, 1974
RG 1.82	Sumps for Emergency Cooling and Containment Sprays Systems, 1974

Review of these regulatory guides reveals that the concerns of the NRC staff regarding emergency sump performance evolved over time. Initially, in-plant tests were called for in RG 1.79. Then, in the mid-1970s, there was a transition to containment and PWR sump model tests. During these model tests, considerable emphasis was placed on "adequate" sump hydraulic performance, and vortex formation was identified as the key determinant. The staff's main concern was that formation of an air-core vortex would result in unacceptable levels of air ingestion and severely degraded pump performance. There was also concern about sump damage or blockage of the flow as a result of insulation debris generated by LOCAs, missiles, and break jet loads. These concerns led to the formulation of some of the guidelines set forth in RG 1.82 (those relating to cover plates, debris screen, and a 50% screen blockage criterion).

In 1979, as a result of continued staff concern about the safe operation of ECCS sumps, the Commission designated the issue as Unresolved Safety Issue (USI) A-43, Containment Emergency Sump Performance. To assist in the resolution of this issue, the Department of Energy (DOE) provided funding for construction of a full-scale sump hydraulic test facility at the Alden Research Laboratory (ARL) of Worcester Polytechnic Institute (WPI) (Durgin, Padmanabhan, and Janik, 1980). At about the same time, an NRC Task Action Plan (TAP) A-43 was developed to address all aspects of this safety issue. Potential debris effects were investigated through plant insulation surveys, sample plant calculations, and supplemental experiments conducted at ARL to determine the transport characteristics of various types of insulation debris and attendant screen blockage head losses.

1.3 Technical Issues

The principal concern is summarized in the following question:

> In the recirculation mode following a LOCA, will the pumps receive water sufficiently free of debris and air and at sufficient inlet pressure to satisfy NPSH requirements so that pump performance is not degraded to the point that long-term recirculation requirements cannot be met?

This concern can be divided into three areas for technical consideration: sump (or suction intake) hydraulic design, insulation debris effects, and pump performance. The three areas are not independent, and certain combinations of effects must be considered as well.

This report presents the technical findings derived from extensive, full-scale experimental measurements, generic plant surveys, sample plant calculations, assessment of the performance of residual heat removal pumps, and public comments received. Public comments received and the staff response to them are in Appendix A. These technical findings provide a basis for technically resolving USI A-43 and for developing revisions to RG 1.82 and Section 6.2.2 of the NRC Standard Review Plan (SRP, NUREG-0800).

1.4 Summary of Technical Findings

The following key determinations are derived from the technical findings presented in Section 3 below:

(1) Visual observations of vortex formation cannot be used to quantify levels of air ingestion. Full-scale PWR sump experiments and BWR suction inlet experiments have shown that levels of measured air ingestion were generally less than 2% under a wide range of simulated post-LOCA conditions. On the other hand, the absence of air-entraining vortices can be used to infer zero air ingestion.

(2) Air ingestion levels have been correlated with the Froude number (Fr) that embodies suction submergence level and suction inlet flow velocity. Full-scale experiments have shown zero air ingestion in PWR sumps for $Fr \leq 0.2$ and zero air ingestion for BWR suction inlet designs up to $Fr \leq 0.8$. Envelope, or bounding, plots for estimating air ingestion levels as a function of Froude number are presented in Section 3.4.

(3) Excessive air ingestion levels (i.e., > 2 to 4 volume %) can lead to degradation of pumping capacity (see Section 3.2). Use of vortex suppressors (fabricated from floor grating materials) can effectively reduce air ingestion to zero (see Section 3.4). For BWR suction inlets, the inlet strainer appears to act as a vortex suppressor and retardant to air ingestion.

(4) RHR recirculation pump operation can be assessed using the findings and methods provided in Section 3.2. As noted above, low levels of air ingestion can be tolerated. However, pumping performance should be based on calculated pump inlet conditions for the postulated LOCA, including adjustment of the net positive suction head requirements (NPSHR) for low levels of air ingestion (see Section 3.2).

(5) Ingestion of small particulates that result from erosion does not appear to pose a pumping problem for the post-LOCA circulating pumps in either PWR or BWR plants because of the materials of construction used in the impellers and casings. Pump seal systems should be reviewed from the viewpoint of possible clogging. Catastrophic failure of shaft seals (as a result of debris generation) is unlikely because of the safety bushings built into pump seal assemblies. If water-lubricated bearings are specified or used in any of the post-LOCA circulating pumps (e.g., in multistage RHR, reactor core isolation cooling (RCIC), high pressure coolant injection (HPCI), or high pressure core spray (HPCS) pumps in some BWRs), the seal system should be carefully reviewed. Particulate ingestion may be sufficient to cause seal failure and/or bearing seizure in these cases.

(6) Surveys of plant insulation materials have shown a wide variability in the types and quantities of insulations employed in nuclear power plants (see Section 3.3). Furthermore, feedback received during the "for comment" period has shown that the types and quantities of insulation have changed over time and with replacement changes made in operating plants. Thus, because of the nature and quantities of insulation materials used, debris blockage assessments become very plant specific and time dependent.

(7) Estimating the effects of debris blockage requires an estimation of (a) the quantity of debris that might be generated by a LOCA, (b) the transport of such insulation debris to the PWR sump screen (or BWR suction strainer), and (c) the potential blockage as a result of flow entrainment of debris to the screen (or strainer) surface. Plant-specific studies have shown that there is a strong dependance on plant layout (which affects migration of debris) and on PWR sump design features (or BWR suction intake design). Appendix B provides illustrative sump designs and containment layout.

(8) The destructive power of a LOCA jet has been demonstrated in HDR* blowdown experiments, particularly from the viewpoint of destruction of fibrous insulation materials. Because finely shredded insulation can be transported at low recirculation flow velocities (i.e., 0.2 ft/sec) (see Appendix D) and distribute uniformly over debris screens, or suction strainers, such insulations must be closely considered in estimating the effects of post-LOCA blockage on pump NPSH margin. Experiments have also shown (a) that reflective metallic insulations can suffer severe damage from LOCA jets (see Appendix E) and (b) that undeformed thin foils (such as those used internally in reflective metallic assemblies) can be transported at low velocities (e.g., 0.2 to 0.4 ft/sec). Information on the transport characteristics of simulated insulation debris and debris generation is in Section 3.3.

(9) Sample plant analyses and experiments have shown that the uniform 50% blockage criterion in RG 1.82 is not adequate, for the reasons noted above. Sump screen blockage (or suction strainer blockage) should be evaluated on a plant-specific basis on the basis of the insulation materials employed, and a plant-specific assessment of potential debris transport and debris screen blockage should be made. Therefore, RG 1.82 has been revised accordingly.

(10) The technical findings in Section 3 have been further refined to develop PWR sump and BWR suction inlet evaluation guidelines. These guidelines are in Section 5.

(11) Methods for estimation of debris generation and transport developed in NUREG/CR-2791 (published in September 1982) are superseded by those outlined in Sections 3.3 and 5.3 of this NUREG.

NUREG/CR-2791 (published in September 1982) should be reviewed from the viewpoint of later information (such as that contained in Sections 3.3 and 5.3 of this NUREG). Certain assumptions made in NUREG/CR-2791 (i.e., that insulation damage effects extending outward to a stagnation pressure level of 0.5 psi) are not supported by more recent evaluations.

*The Heissdampfreaktor or superheated steam reactor, in the Federal Republic of Germany; see Appendix C.

2 SUMMARY OF KEY FINDINGS

2.1 Pump Performance

Sustained operation of PWR RHR and CSS pumps, or BWR RHR pumps, in the recirculating mode presents two principal areas of concern:

(1) possible degradation of the hydraulic performance of the pump (inability of the pump to maintain sufficient recirculation flow as a result of sump screen blockage, cavitation, or air ingestion effects)

(2) possible degradation of pump performance over the long- or short-term because of mechanical problems (material erosion due to particulates or severe cavitation, shaft or bearing failure due to unbalanced loads, and shaft or impeller seizure due to particulates)

Pumps used in PWR RHR and CSS systems are primarily single-stage centrifugal designs of low specific speed. PWR CSS pumps are generally rated at flows of about 1500 gpm, with heads of 400 feet, and require about 20 feet of NPSH at their inlet; PWR RHR pumps are generally rated at about 3000 gpm, with heads of 300 feet, and require about 20 feet NPSH at maximum flow. Rating points and submergence requirements for the pumps are plant specific. Pump impeller materials are generally highly resistant to erosion, corrosion, and cavitation damage.

Experimental results show that under normal flow conditions and in the absence of cavitation effects, pumping performance is only slightly degraded when air ingestion is less than 2%. This value would be a conservative estimate for acceptable performance and is dependent on many variables. However, air ingestion greater than 15% almost completely degrades the performance of pumps of this type.

Submergence or NPSH requirements for RHR and CSS pumps (routinely determined by manufacturers' tests) are established on the basis of percent of degradation in pump output pressure. Individual pump specifications determine that NPSH required be set according to a 1% or 3% degradation criterion. No industry standard exists for the percent of degradation criterion, nor for the margin between available NPSH and that required in setting RHR and CSS pump submergence criteria. Air ingestion affects NPSHR for pumps. Test data on the combined effects of air ingestion and cavitation are limited, but the combined effects of both increase the NPSH required. A value of 3% degradation in pump output pressure for the combined effects of air ingestion and cavitation appears to be realistic for assessing recirculation pump performance.

The types and quantities of debris small enough to pass through screens (or suction strainers) and reach the pump impeller should not impair long-term hydraulic performance. In pumps with mechanical shaft seals, accumulated quantities of soft or abrasive debris in the seal flow passages may result in clogging or excessive wear, either of which, in turn, may lead to increased seal leakage. Catastrophic failure of a shaft seal in the post-LOCA circulation pumps in either PWR or BWR systems as a result of debris ingestion is considered unlikely. In the event of complete failure of shaft seals, pump leakage would be restricted by the throttle or safety bushing incorporated in these seals.

The spectrum of both design features and rated performance values for centrifugal pumps used in BWR safety systems is much broader than that for pumps used in PWR systems. Although there is a wider variation in BWR pumping capacities, the pumps in BWR systems have low to medium specific speed designs. Their performance characteristics are very similar to those of pumps used in PWRs. Pumps in BWRs and PWRs should be subject to the same technical considerations regarding hydraulic performance (i.e., the criteria used in calculation of NPSH and in considering the quantities of air will apply directly to the BWR pumps).

The main bearings for BWR safety pumps are similar in construction and protection details to those of their PWR equivalents. That is, the main bearings are rolling element or ball bearings, either grease or oil lubricated. These bearings are generally protected from damage as a result of pump leakage by mechanical shaft seals equipped with safety bushings and, in some cases, downstream deflectors. This is true for multistage pumps as well as conventional single-stage pumps. As is the case for comparable PWR pumps, even a complete mechanical seal failure produces only a limited amount of leakage. The outboard ball bearings for these pumps are protected by disaster bushings and deflector disks, and, therefore, total failure of these bearings is not likely.

The BWR pumps are distinguished from PWR safety system pumps principally by the fact that multistage pumps are frequently used in BWR safety systems. When multistage pumps are used, one must be concerned about the effects of particulates and debris on the interstage bushings.

In multistage pumps, interstage bushings are generally cooled and lubricated by the pumped fluid. For plants where it has been determined that significant amounts of abrasive particulates or fiberous debris may be transmitted from the pump inlet screen into the pumps themselves, the interstage bushing systems should be evaluated to determine whether external pressurized cooling or flushing is needed to prevent damage as a result of wear or clogging. Plant operational experience (based on periodic startup and verification of safety system operation) has shown no problems with interstage bushing assemblies even though the suppression pool water quality is less than that used for reactor recirculation.

2.2 Effects of Debris on Recirculation Capability

The safety concerns related to the effects of LOCA-generated insulation debris on RHR recirculation requirements depend on the following:

(1) the types and quantities of insulation employed (dependent on plant design and installation)

(2) the potential for a high pressure system break to severely damage or destroy large quantities of insulation (dependent on plant layout and insulation distribution, and on break-targeted insulations)

(3) the potential for LOCA-generated insulation debris to be transported to the PWR sump screen or BWR suction strainer (dependent on plant layout and recirculation velocity)

(4) the extent to which such transported debris would result in blockage of the sump screen or suction strainer (dependent on screen design and size)

(5) the blocked screen head loss impact on RHR recirculation pump available NPSH (dependent on the material and blockage characteristics of the debris transported to the screen)

The variability of plant layout, sump design, insulation employed, and recirculation requirements make debris assessments very plant specific. The results of debris considerations studied can be summarized as follows:

(1) Types of insulation vary from plant to plant and are subject to change with time (i.e., replacement insulation may be different from the original installation).

(2) Generally speaking, insulations can be categorized as

  (a) reflective metallic insulation (both stainless steel and aluminum are utilized)

  (b) encapsulated, by metallic or other types of coverings, but with various core materials (typical core materials are calcium silicate, fiberglass, mineral wool Cerablanket™, and Unibestos™)

  (c) nonencapsulated insulations, which are typically fabricated as "blankets" or "pillows" and in which the core materials noted in (b) are used, with varying methods of attachment

  (d) molded insulations with closed-cell structure (e.g., foam-glass)

  (e) antisweat insulations (typically fiberglass, urethane and polyurethane foams, and closed-cell rubber)

Although encapsulation can afford protection from high pressure jet loads and missile impacts, encapsulated structures must be reviewed to assess the real degree of protection that is afforded. The characterization "totally encapsulated" can be misleading because of the variability of encapsulations and attachment mechanisms provided. Thus insulation should be carefully assessed to determine whether it is totally or partially encapsulated.

Insulation surveys conducted in 1982 (see Section 3.3) indicated a decreasing trend in the use of insulations such as fiberglass, mineral wool, and calcium silicate, with licensees of newer plants appearing to elect to install reflective metallic insulation. However, feedback received during the "for comment" period (June-July 1983) reversed this finding. More recently, some licensees of operating plants have elected to replace old insulation with fiberglass, and applicants for plants in the operating license (OL) review stage also have selected fiberglass. The more extensive use of fiberglass should be reviewed on a plant-specific basis to assess the screen blockage impact.

LOCA jets are capable of high levels of insulation destruction, as evidenced by the HDR blowdown experiments (see Appendix C). In these HDR experiments, all glass fiber insulation, within 2 to 4 meters of the break nozzle of diameters up to 450 mm was destroyed and distributed throughout the containment as very fine particles. In addition, Sandia National Laboratory (SNL) has analyzed two-dimensional-break jet expansion phenomena and target pressure loads. The SNL calculations correlate well with the HDR data and show that significant jet loads occur within 3 to 5 L/D's* of the pipe break location. More recent HDR experiments (see Appendix E) illustrate the level of damage that than be incurred by reflective metallic insulation. These experiments reveal severe damage near the break location and much less damage 7 L/D's from the break. Debris generation is discussed in Section 3.3.3.

Insulation debris transport tests at Alden Research Laboratory (ARL) show that severely damaged or fragmented insulation can be transported at low velocities (0.2 to 0.5 ft/sec). Both fiberglass shreds and thin (0.0025- to 0.004-inch) metallic foils (if undeformed) can be transported at these low velocities. Therefore, the level of damage near the postulated break location(s) becomes a dominant consideration in assessing the type and volume of debris generated, as well as in estimating transport probability. Larger or intact pieces require much higher transport velocities (> 1.0 ft/sec). Thus determination of recirculation flow velocities within containment is an important factor in assessing debris transport (See Appendix D). In PWR containments, recirculation flow velocities on the order of 0.2 to 0.6 ft/sec can be calculated; hence, the transport of large pieces of debris is less likely. However, such assessments become highly plant dependent because the types of insulation used, levels of damage, available recirculation paths, and the location of the sump versus the location of the break are controlling considerations.

Assessment of the probabilities for PWR sump failure (NUREG/CR-3394) has also revealed that:

(1) Principal attention should be given to insulation on the primary coolant system piping and lower half of the steam generators, because insulation on these components is the major source of potential debris, based on postulated break locations and possible break jet targets.

(2) Piping less than 10 inches in diameter is of secondary importance because smaller diameter breaks generate lower quantities of debris. The jet envelope and target area are less for these sizes.

Although these findings should not be applied unilaterally, these trends are applicable to PWRs for initial debris assessments. Thus they provide a means to scope the magnitude of the debris generation potential.

Low density insulations with a closed cell structure will float and are not likely to impede flow through the sump screens, except where the screens are not totally submerged. Low density hygroscopic insulations with submerged densities greater than water must be assessed on a plant-specific basis, as must nonencapsulated insulation (particularly mineral fiber, fiberglass, or mineral wool blanket), to determine the potential for sump screen blockage. If reflective metallic insulation is damaged to the extent that interior foils are released, transport and potential screen blockage must be assessed on a plant-specific basis. In summary, all insulations should be evaluated on a plant-specific basis.

*Here L is the centerline axial distance from the mean to the target and D is pipe break diameter.

Conservative methods have been developed for estimating quantities of debris, break sources, transport mechanisms, and blockage effects based on the findings summarized above. These methods are detailed in Section 3.3 and summarized in Section 5.3.

2.3 Sump Hydraulic Performance Findings

Data obtained from full-scale sump tests provide a sound base for assessing sump hydraulic performance. Both side-suction and bottom-suction designs were tested over a wide range of design parameters, and the effects of elevated water temperatures were also assessed. Scaling experiments (1:4, 1:2, 1:1) were also conducted to provide a means for assessing the validity of previous scale-model tests. The effectiveness of certain vortex suppression devices was also evaluated. For completeness, plant-specific and LOCA-introduced effects (ice condenser drain flow, break flow impingement, large swirl and sump circulation effects, and sump screen blockage) were evaluated at full scale. In addition, a limited number of BWR suction tests were performed.

The results of this test program can be summarized as follows:

(1) The broad data base from the sump studies resulted in the development of envelope curves for reliably quantifying the expected upper bound for the hydraulic performance of any given sump whose essential features fall approximately within the flow and geometric ranges tested.

(2) Vortices are unstable, randomly formed, and, for cases where air ingestion occurs, cannot be used to quantify air ingestion levels, suction inlet losses, or intake pipe fluid swirl. The full-scale tests show that at water submergences deeper than 9 feet and inlet water velocities of less than 4 ft/sec, significant vortex activity disappears. Correspondingly, air ingestion is negligible or non-existant.

(3) Based on void fraction measurements, air ingestion was found to be less than 2% in most cases. A few test conditions resulted in higher air ingestion, 2% to 8%, with or without perturbations of the approach flow. Maximum air ingestion rates of 8% to 15% were recorded for only short periods with deliberately induced adverse-approach flow conditions of severely blocked screens. These tests revealed the importance of measuring void fraction and demonstrated the ineffectiveness of visual observations of vortices as a means of quantitatively evaluating air entrainment.

(4) Swirl angles in suction pipes were generally found to have decreased to about 4° at a distance 14 pipe diameters from inlets. Swirl angles of up to 7° at a distance 14 pipe diameters from inlets were observed in some sump tests at low submergence with induced flow perturbations.

(5) Hydraulic grade line measurements for all experiments revealed that the sump intake loss coefficient was insensitive to overall sump design variation. Loss coefficients are basically a function of local intake geometry, and the measured values are consistent with those obtained from standard hydraulic handbooks.

(6) Testing over the temperature range of 70°F to 165°F revealed that water temperature (or previously hypothesized Reynolds number effects) had no

measurable effect on surface vortexing, air ingestion, pipe swirl, or loss coefficient.

(7) Vortex suppressor testing for PWR applications revealed that cage-type and submerged-grid-type designs generally (a) reduced surface vortexing from a full air-core vortex to surface swirl only; (b) reduced air ingestion to essentially zero; (c) reduced pipe swirl to less than 5°; and (d) had no significant effect on the loss coefficient. These vortex suppression structures were fabricated from floor grating materials typically used for walkways.

(8) There were no major differences between the hydraulic performance of vertical outlet sumps and that of horizontal outlet sumps of similar design geometry and similar flow conditions.

(9) Comparison of the results of different scale models showed that scale modeling down to 1:4 scale using Froude number similitude adequately predicted the sump hydraulic performance variables (void fraction, vortex type, swirl, and loss coefficient) of full-scale tests. Tests on 1:4-, 1:2-, and 1:1-scale versions of the same sump under comparable operating conditions showed no significant scale effects in the modeling of air withdrawal because of surface vortices or in free-surface vortex behavior. Additionally, model tests accurately predicted swirl and inlet losses if specified Reynolds number criteria were maintained.

(10) A parametric assessment of nonuniform approach flow into the sump as a result of specific structural features did not reveal any significant adverse effects (see also Section 3.4).

(11) Drain flow impingement on the sump water surface resulted in extensive turbulence that tended to reduce vortexing and did not lead to increased air ingestion.

(12) Break flow impingement tests produced considerable air entrainment at the water surface, but void fractions of the pipe flow were generally small, less than 1%. In one case, a considerably higher void fraction was recorded, 6%, because of a change in approach flow to the sump caused by the break flow.

(13) PWR sump screen blockage tests sometimes revealed slight increases in air ingestion and some degradation of the hydraulic performance of the sump, depending on the sump configuration and test conditions. However, no significant changes were noted. In each case where air-core vortices were generated, the use of a vortex suppressor eliminated the air-core vortex and reduced the air ingestion to zero or negligible levels. Thus, the effectiveness of vortex suppressors (such as submerged floor grating designs) has been demonstrated.

(14) BWR suction intake tests (see Section 3.4.6) revealed that air ingestion was essentially zero for Froude numbers less than 0.6. The suction strainers typically utilized in BWR installations appear to act as vortex suppressors, thereby inhibiting air ingestion (even though air core vortices were observed at lower Froude numbers).

Thus the full-scale sump hydraulic test program conducted at ARL has resulted in an extensive data base that has broad applicability and can be used in lieu of model tests or inplant tests (if the sump design being evaluated falls within the design and flow envelope investigated). Sump hydraulic design guidelines and criteria for assessing air ingestion potential are in Section 5.

3 TECHNICAL FINDINGS

3.1 Introduction

Before a plan for the resolution of Unresolved Safety Issue A-43 was developed, the following key safety questions were identified:

(1) What are the performance capabilities of pumps used in containment recirculation systems, and how tolerant are such pumps to air entrainment, cavitation, and the potential ingestion of debris and particulates that may pass through screens?

(2) Were a LOCA to occur, would the amount and type of debris generated from containment insulation (and its subsequent transport within containment) cause significant sump screen blockage and, if so, would such blockage be of sufficient magnitude to reduce the NPSH available below the NPSH required?

(3) Can geometric and hydraulic sump system designs be established for which acceptable sump performance can be ensured?

It was recognized that resolution of USI A-43 depended upon the responses to these questions. The effort to resolve these questions was undertaken in three parallel tasks, each designed to respond to one of the key safety questions.

The first question was addressed through an evaluation of the general physical and performance characteristics of RHR and CSS pumps used in existing plants. Conditions likely to cause degraded performance or damage to pumps performance were evaluated. The investigation of pump cavitation, air ingestion, particulate ingestion, and swirl is reported in NUREG/CR-2792 and Creare Technical Memorandum 962. It is summarized in Section 3.2 below.

To address the second question, 19 power reactor plants were surveyed concerning the quantity, types, and location of insulation used within containment (see NUREG/CR-2403 and its Supplement 1). Then, calculational methods were developed for estimating (1) the quantities and sources of debris that could be generated during a LOCA, (2) the transport of such debris, (3) the quantities and properties of insulation debris that could potentially be transported to sump screens, and (4) head losses as a result of debris buildup on sump screens (NUREG/CR-2791). Many of the methods for the assessment of debris blockage in NUREG/CR-2791 are superseded by those described in this report. Experiments were conducted to estimate the onset of jet erosion damage to fibrous insulations (NUREG/CR-3170) and to determine the transport and screen blockage head losses associated with fibrous insulations (NUREG/CR-2982, Rev. 1). The transport and blockage characteristics of reflective metallic insulations are reported in NUREG/CR-3616.

The third key safety question was addressed in an investigation of the behavior of ECCS sumps under diverse flow conditions that might occur during a LOCA. The test program was designed to cover a broad range of geometric and flow variables representative of emergency sump designs. The results are reported in NUREG/CR-2758, NUREG/CR-2759, NUREG/CR-2760, NUREG/CR-2761, and NUREG/CR-2772.

3.2 Performance of Emergency Core Cooling System Pumps

This section summarizes the general physical and performance characteristics of RHR and CSS pumps used in PWRs and RHR, CS, and CI pumps used in BWRs. The summary characteristics are based on information from 12 PWRs and 7 BWRs that were sampled in the study. Effects likely to cause degraded performance or damage are identified, and the results of an analysis of these effects on pump performance are presented.

3.2.1 Characteristics of Pumps Used for Emergency Core Cooling Systems

The pumps used in PWR and BWR systems have different characteristics.

3.2.1.1 RHR and CSS Pumps Used in PWRs

A study of pumps used in 12 PWR plants has shown that although individual pump details are plant specific, the pumps used in RHR and CSS services are similar in type, mechanical construction, and performance.

Similarities in the types of pumps are shown in Table 3.1; the table lists the manufacturer, model number, and rated conditions for each of the pumps used in the plants surveyed. The column labeled "Specific Speed" provides a parameter conventionally used by pump manufacturers to specify hydraulic characteristics and, hence, the overall design configuration of a pump. As the table shows, all pumps are relatively high-speed, centrifugal pumps and are in the specific speed range of 800 to 1600 rpm, with specific speed defined as N_s = (speed) (volumetric flow)$^{1/2}$/(head)$^{3/4}$.

The pumps used for RHR and CSS service have the following similarities in mechanical construction:

(1) Impellers and casings are usually austenitic stainless steel, highly resistant to damage by cavitation.

(2) Impellers are shrouded with wear rings to minimize leakage.

(3) Shaft seals are the mechanical type.

(4) Bearings are grease- or oil-lubricated ball type.

A pump assembly typical of pumps used for RHR and CSS service is shown in cross-section in Figure 3.1.

Similarities in the performance of pumps used in RHR and CSS service are shown in Figures 3.2 and 3.3. Performance and cavitation data from each of the pumps listed in Table 3.1 have been plotted for comparison. Performance data are given in terms of normalized head versus normalized flow rate where the best-efficiency-point head and flow are used for the reference values. Cavitation data are given in terms of NPSH required.

3.2.1.2 RHR, CS, and CI Pumps Used in BWRs

There is a wider variation in rating conditions for pumps used in BWR safety systems than for their counterparts in PWRs. Table 3.2 lists rating points,

Table 3.1 RHR and CSS pump data

Plant	Manufacturer*/Model RHR	Manufacturer*/Model CSS	Rated Conditions (RPM) Speed	Rated Conditions (FT) Head	Rated Conditions (GPM) Flow	Rated Conditions Specific Speed
Arkansas Unit #2	I-R 6x23 WD		1800	350	3100	1238
		I-R 8x20 WD	1800	525	2200	851
Calvert Cliffs 1&2	I-R 8x21 AL		1780	360	3000	1205
		B&W 6x8x11 HSMJ	3580	375	1350	1544
Crystal River #3	W 8HN-184		1780	350	3000	1205
		W6HND-134	3550	450	1500	1407
Ginna	Pac 6" SVC		1770	280	1560	1016
Haddom Neck	Pac 8" LX		1770	300	2200	1152
		Pac 8" LX	1770	300	2200	1152
Kewaunee	B-J 6x10x18 VDSM		1770	260	2000	1222
		I-B 4x11 AN	3550	475	1300	1257
McGuire 1&2	I-R 8x20 WD		1780	375	3000	1144
		I-R 8x20 WD	1780	380	3400	1205
Midland #2	B&W 10x12x21 ASMX		1780	370	3000	1156
		B&W 6x8x135 MX	3550	387	1300	1467
Millstone Unit 2	I-R (No Model #)		1770	350	3000	1198
		G3736-4x6-13DV	3560	477	1400	1370
Oconee #3	I-R 8x21 AL		1780	360	3000	1180
		I-R 4x11 A	3550	460	1490	1380
Prairie Island	B-J 6x10x18 VDSM		1770	285	2000	1141
		I-R 4x11 AN	3550	500	1300	1210
Prairie Island 1&2	B-J 6x10x18 VDSM	I-R 4x11 AN	1780	280	2000	1156
			3550	510	1300	1210
Salem #1	I-R 8x20W		1780	350	3000	1205
		G 3415 8x10-22	1780	450	2600	929

*Pac -- Pacific
I-R -- Ingersoll-Rand
W -- Worthington
G -- Gould
B&W -- Babcock & Wilcox
B-J -- Byron Jackson

Specific Speed is defined as N_s = Speed $(\text{Flow})^{1/2}/(\text{Head})^{3/4}$

In this definition: Speed is in rpm, flow in gpm and head in ft.

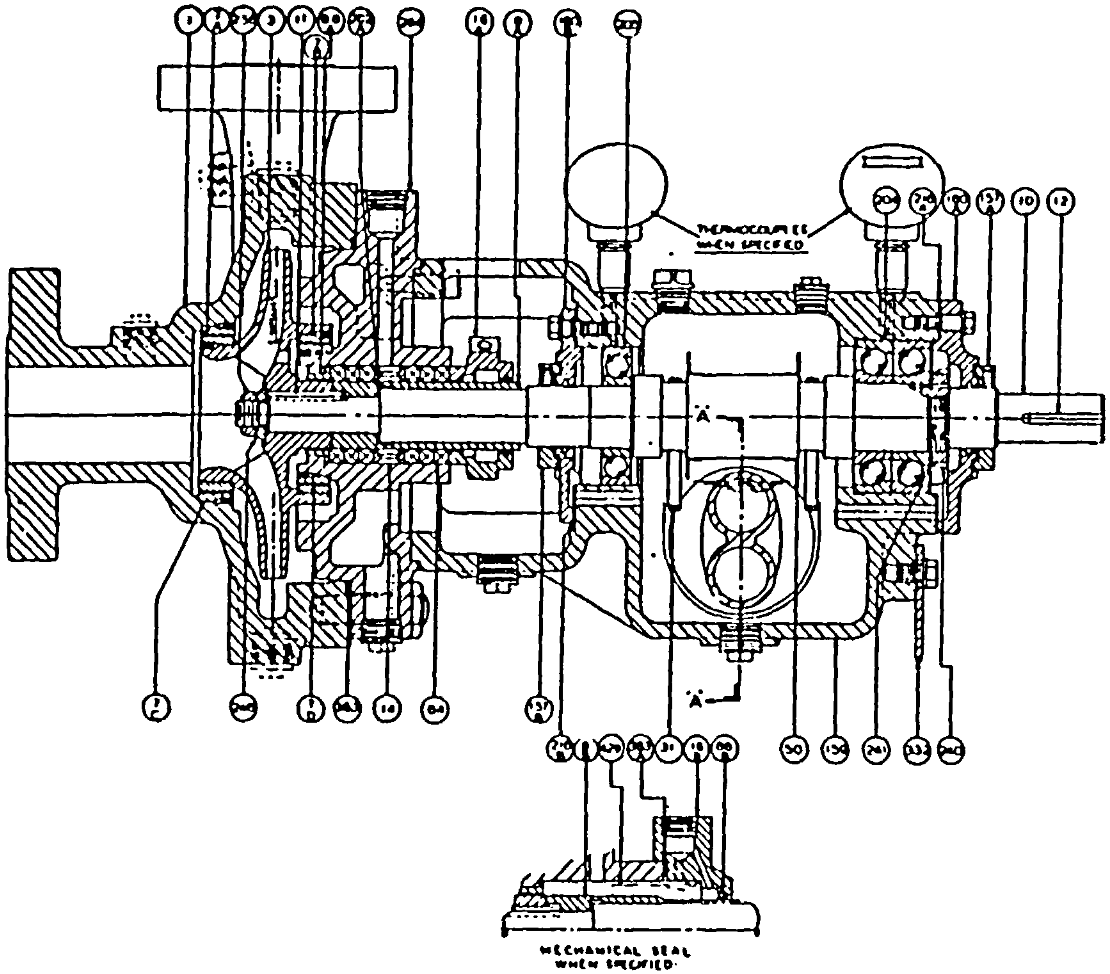


Figure 3.1 Assembly schematic of centrifugal pump typical of those used for RHR or CSS service

RHR Pumps

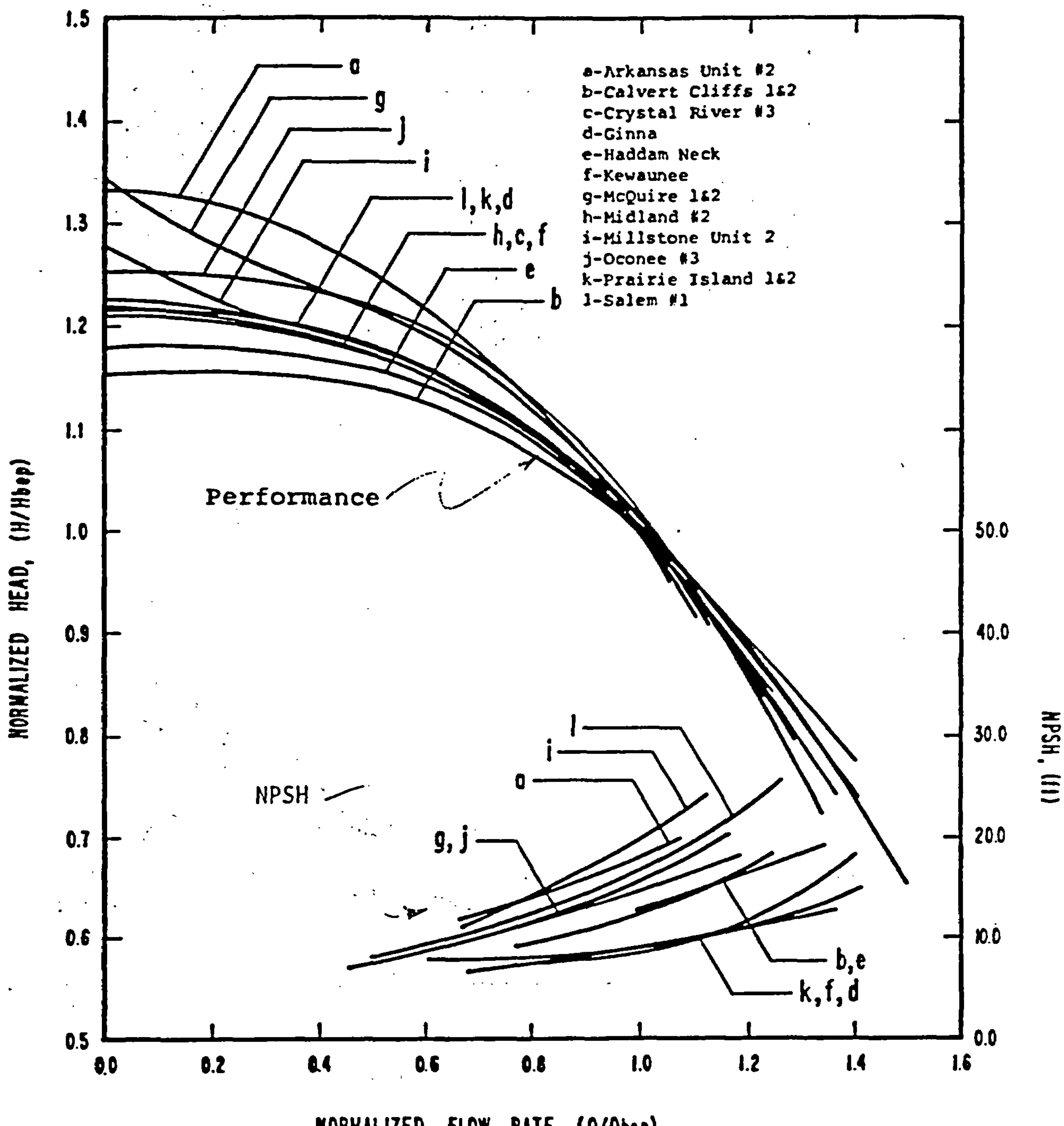


Figure 3.2 Performance and NPSH curves for RHR pumps, head versus flow rate data normalized by individual best-efficiency-point values

CSS Pumps

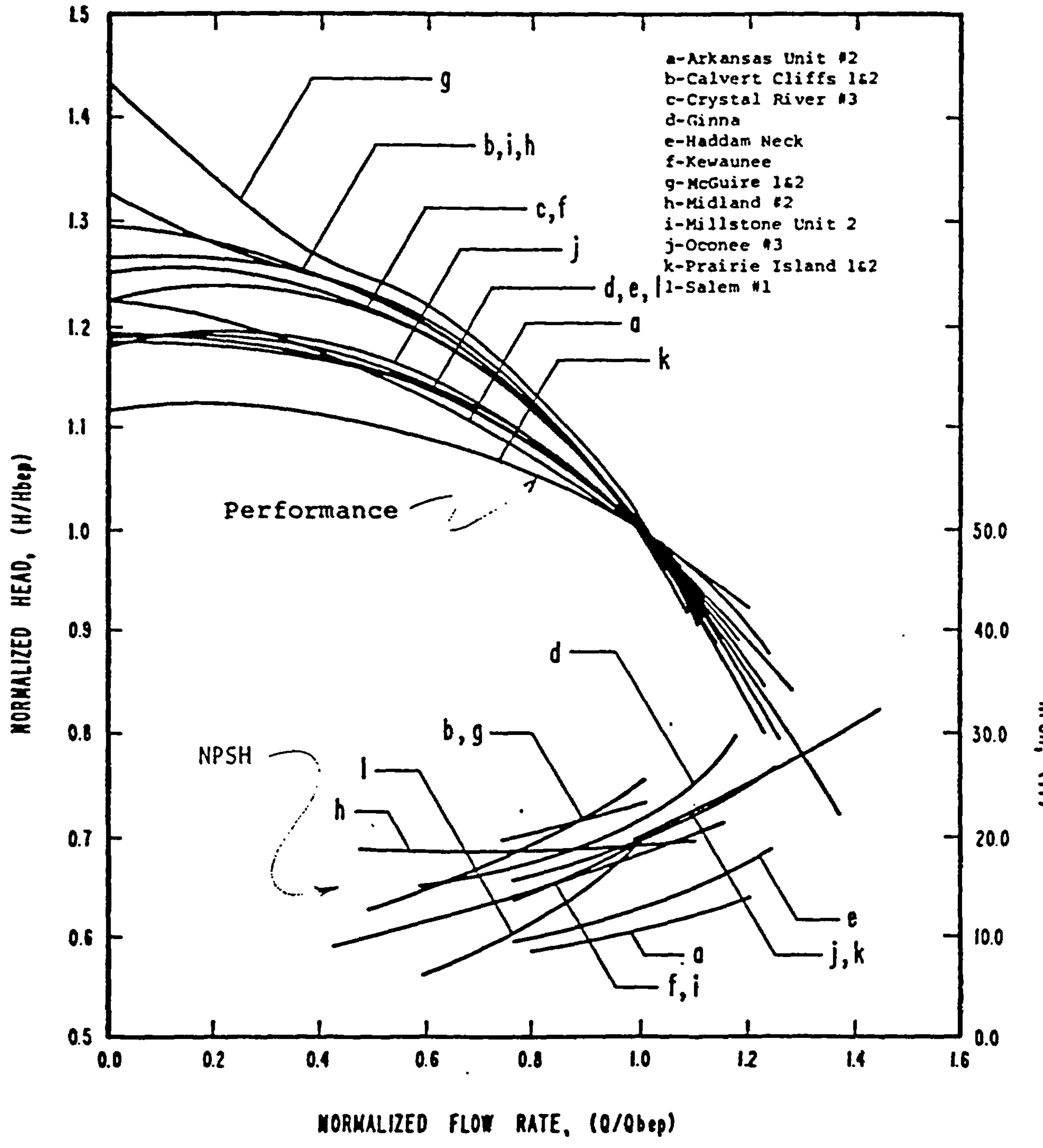


Figure 3.3 Performance and NPSH curves for CSS pumps, head versus flow rate data normalized by individual best-efficiency-point values

Table 3.2 RHR, CS, and CI pump data for BWRs

PLANT	ECCS MODE	PUMP TYPE*	RATED CONDITIONS SPEED (rpm)	HEAD (ft)	FLOW (gpm)	SPECIFIC SPEED
Cooper	CS	VSS				
	LPCI	VSS	1760	420	7800	1675
	HPCI	STD				
Dresden (2)	CS	VSS	3560	585	4700	2052
	LPCI	VSS	3560	570	2700	1585
	HPCI	STD				
Edwin Hatch (1&2)	CS	VMS	1780	670	4700	982
	RHR	VMS	1780	420	7700	1684
	HPCI	STD				
LaSalle (1&2)	LPCS	VMS	1780	725	6350	1015
	HPCS	VMS	1780	1569	6942	595
	RHR	VMS	1780	280	7450	2244
Limerick (1&2)	CS	VMS	1780	668	3175	763
	RHR	VMS	1180	525	10000	1076
Susquehana (1&2)	CS	VMS	1780	668	3175	763
	RHR	VMS	1180	600	10000	973
	HPCI	STD	Varies	525/ 2940	5070	770
Zimmer (1)	LPCS	VMS	1780	690	4750	911
	HPCS	VMS	1780	1347	5142	574
	RHR	VMS	1780	270	5050	1900

* STD - Steam Turbine Drive
VSS - Vertical Single Stage
VMS - Vertical Multistage

pump types, and specific speeds for a sample of seven BWR plants. Flow rates and rated heads for the BWR pumps are in many cases significantly larger than those conditions for PWR pumps discussed in Section 3.2.1.1. In spite of these plant-specific differences, the pumps have all low to medium specific speed designs with performance characteristics similar to those used in PWRs.

Many of the pumps used in BWR ECC systems are multistage designs. Both the single-stage and multistage design pumps used in BWR systems have many of the following construction features similar to those for PWR pumps:

(1) Impellers are usually austenitic stainless steel with high resistance to damage from cavitation.

(2) Impellers are shrouded with wear rings to minimize leakage.

(3) External shaft seals are mechanical.

(4) Main bearings may be grease- or oil-lubricated ball types or oil-lubricated sleeve bearings. In the multistage designs, internal sleeve bushings may be used between stages to provide additional support to the shaft.

The technical considerations relative to hydraulic performance (i.e., cavitation, air ingestion) are the same for single-stage or multistage designs. However, because of the differences in construction details between the two types of pumps, the effects of particulates may be significantly different for each design. Figure 3.4 illustrates the main features of a multistage design typical of pumps found in BWR emergency cooling systems. These pumps use interstage shaft bushings that are lubricated by the pumped water and are therefore subject to wear or clogging from debris.

3.2.2 Effects of Cavitation, Air or Particulate Ingestion, and Swirl on Pump Performance

Several items have been identified as potential causes of long- or short-term degradation of emergency cooling pumps in PWRs and BWRs. They are

(1) cavitation, which may cause head degradation and damage to impellers
(2) air ingestion, which may cause head degradation
(3) particulate ingestion, which may cause damage to internal parts
(4) swirl at the pump inlet, which may cause head degradation

All of these effects also have the potential for inducing hydraulically or mechanically unbalanced loads. They are discussed below.

3.2.2.1 Cavitation

Net positive suction head (NPSH) is defined as the total pressure at the pump inlet above vapor pressure at the liquid temperature, expressed in terms of liquid head (pressure/specific weight); it is equivalent to the amount of subcooling at the pump inlet. If the NPSH available at the pump is less than the NPSH required, some degree of cavitation is ensured and some degradation of performance and perhaps material erosion are likely.

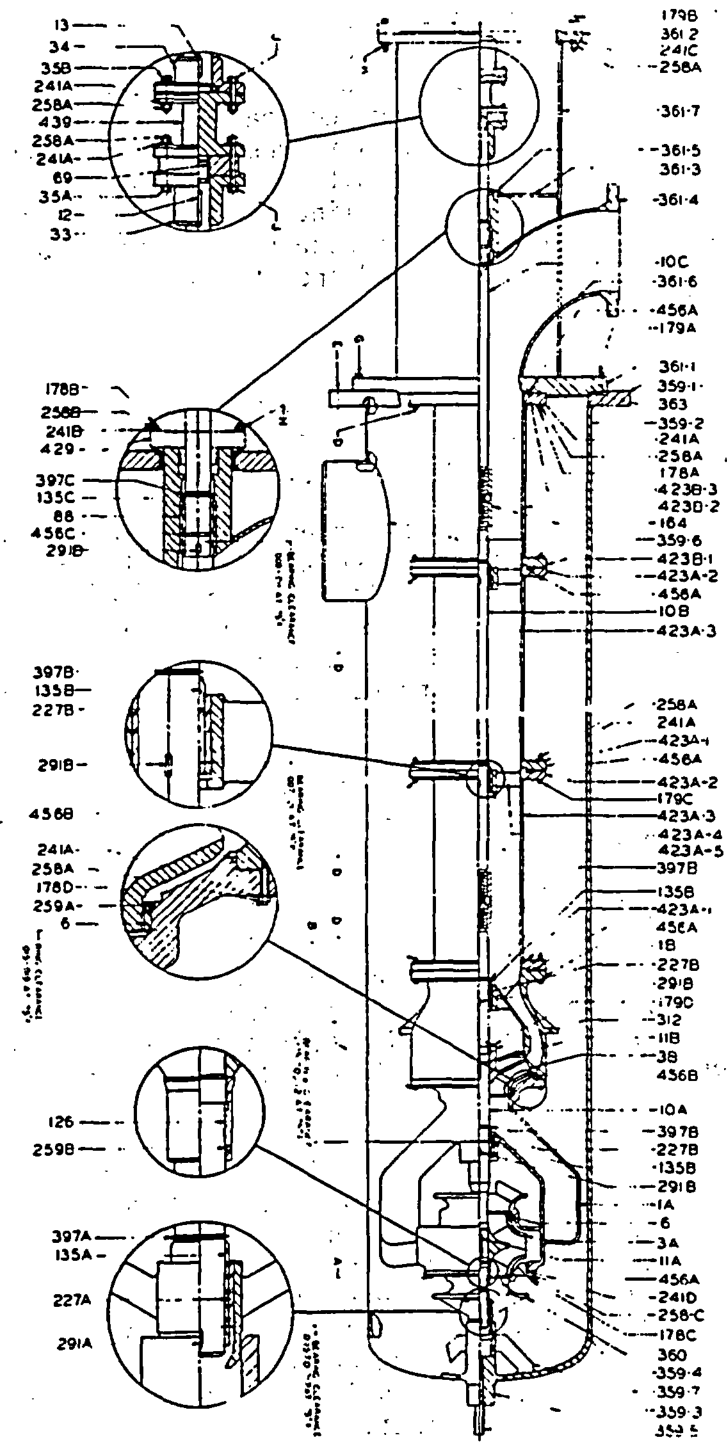


Figure 3.4 Assembly schematic of multistage pump used in BWR emergency cooling systems

There is no standard for identifying the NPSH required for a given pump. Unless there is a stipulation in the specifications, manufacturers have used some percentage (1% to 3%) in head degradation as the criterion for establishing the NPSH required at some flow condition. These are empirically established values for which very rapid degradation occurs (see Figure 3.4) and when cavitation occurs severe erosion is likely to happen. Figure 3.5 illustrates the changes in pump performance at several flow rates as a function of NPSH; these curves are typical of those provided by pump manufacturers to define the NPSH required for their pumps. Because NPSH is reduced for each flow rate shown (Q1-Q4), a point is reached below the 3% limit at which substantial degradation begins. When designing emergency core cooling systems, fluid system designers may choose to apply some margin to the NPSH requirements for a pump but currently no standard margin between NPSH required and NPSH available has been established by NRC regulations.

Some conservatism may be introduced in the calculation of NPSH following guidelines established in RG 1.1 where no credit is allowed for increased containment pressure. However, RG 1.1 does not address subatmospheric conditions in containment with respect to NPSH.

The cavitation behavior of pumps changes at elevated liquid temperatures. Figure 3.6, which is extracted from the Hydraulic Institute Standards (Hydraulic Institute, 1975), shows that as liquid temperatures increase, less NPSH is required by the pump. As a result, increases in liquid temperature have two effects on NPSH: (1) the vapor pressure increases, which reduces NPSH available, and (2) the NPSH required is reduced by an amount, as given in Figure 3.6.

The austenitic stainless steels specified for impellers and casings in these pumps are highly resistant to erosion damage caused by cavitation. Erosion rates for extended operation are not significant as long as the NPSH available exceeds the NPSH requirement of the pump.

3.2.2.2 Air Ingestion

The key findings derived for emergency cooling pumps with respect to air ingestion are based primarily on data from carefully conducted tests in air/water mixtures on pumps of a scale and specific speed range comparable to emergency cooling pumps.* Test data from independent programs on different pumps have

---

*All relevant test data were gathered through reviews of technical papers and interviews with pump manufacturers. Manufacturers' test data on air/water performance of pumps are sparse, and apply primarily to the development of commercial pumps for the paper industry. Although these pumps are similar to those used for emergency cooling service, test methods and results are generally poorly documented. Therefore, manufacturers' data have not been used to establish the air/water performance characteristics of pumps in this report. (Manufacturers' data and testimonials do, however, corroborate published data.) Only sources of information meeting the following criteria were used:

- Pumps must be low specific speed (N_s = 800 to 2000 rpm).
- Pumps must be of reasonable design (with efficiencies $\geq$ 60% and impellers diameter > 6-inch).

(Continued)

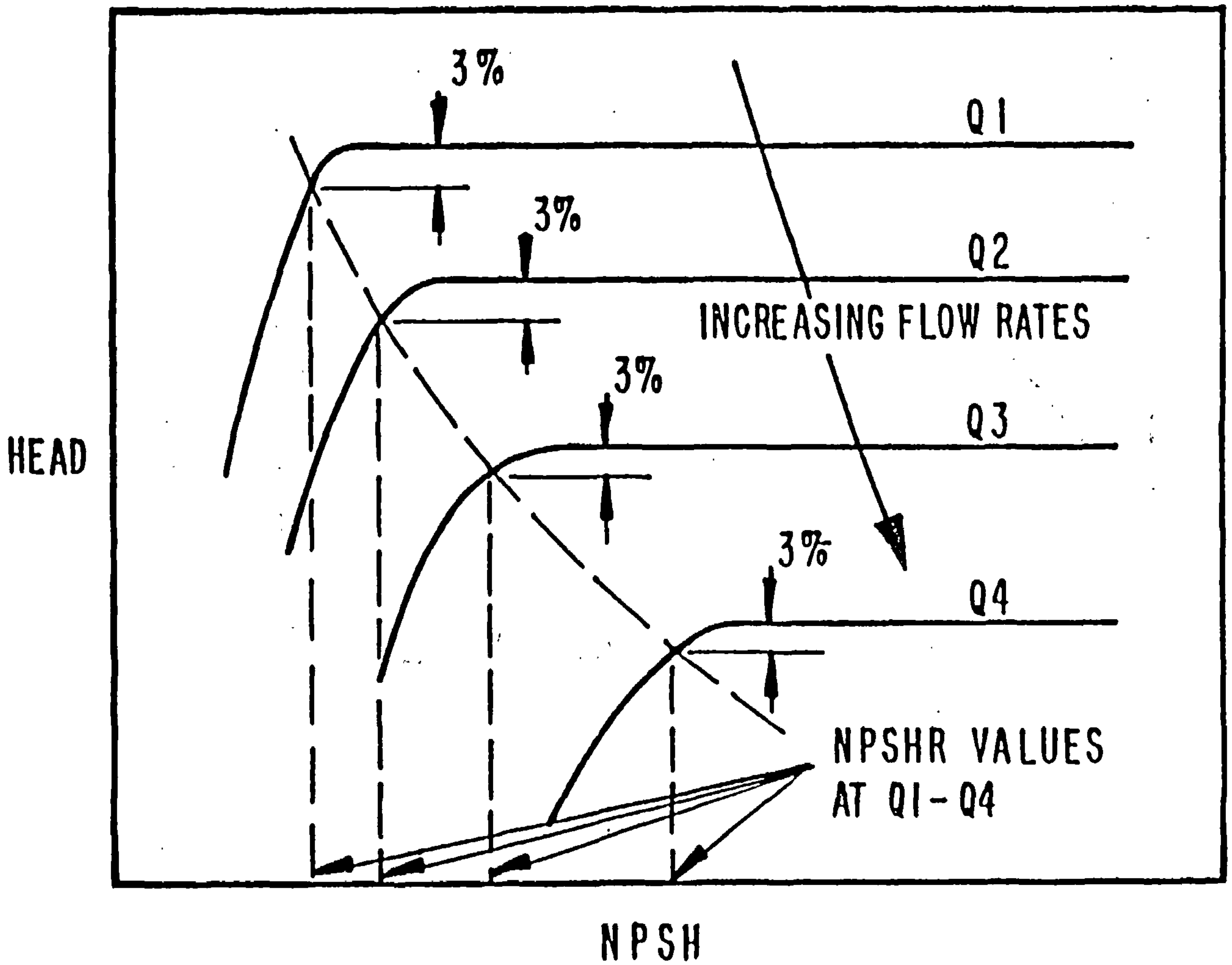


Figure 3.5 Typical head degradation curves due to cavitation at four flow rates (Q1, Q2, Q3, and Q4)

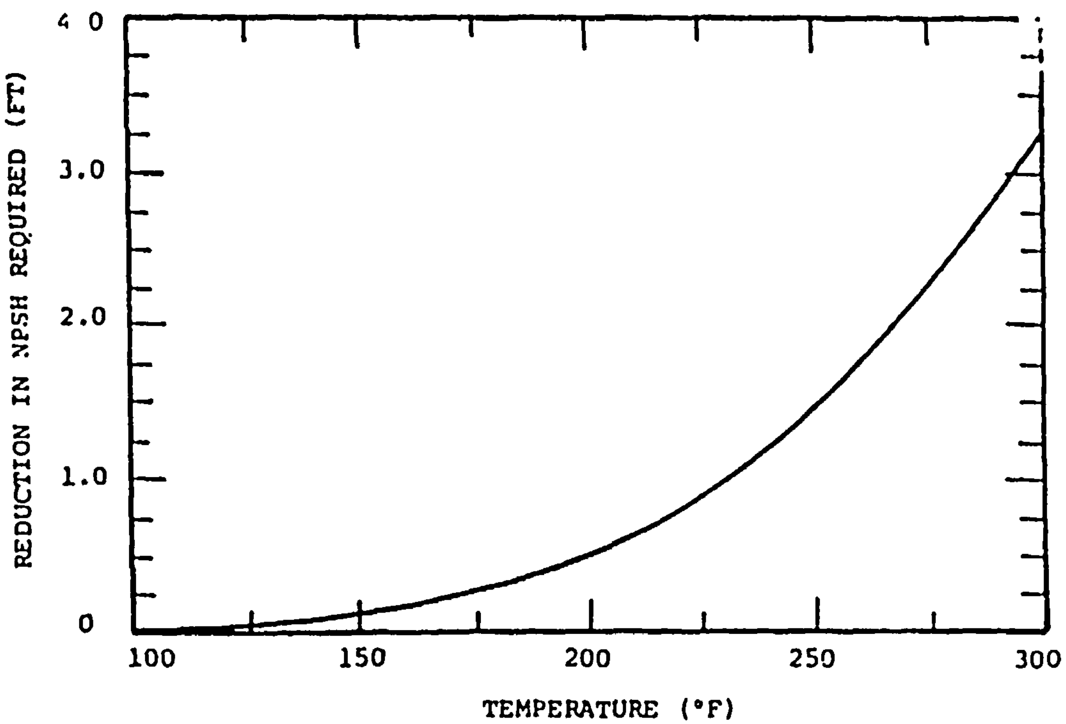


Figure 3.6 Reduction in pump NPSH requirements as a function of liquid temperature (Murakami and Minemura, 1977)

been plotted in Figure 3.7 to illustrate the degradation in head at different levels of air ingestion (percent by volume). Performance degradation is indicated by the ratio of the two-phase (air/water) pressure rise to the single-phase (water) pressure rise.

Figure 3.7 shows that for low levels of air ingestion, the degradation in pump head follows the curve (dashed line) predicted by the change in average fluid density due to the air content. Above 2% void fraction, the data depart from this theoretical line, and the rate of degradation increases. The data in the figure are shown for tests on single-stage pumps. Similar tests show that multistage pumps degrade less in performance for comparable quantities of air.

Above void fractions of about 15%, pump performance is almost totally degraded. The degradation process between 2% and 15% void fraction is dependent on operating conditions, pump design, and other unidentified variables. These findings closely approximate the guidelines empirically established by pump manufacturers: at air ingestion levels of less than 3%, degradation is generally not a concern; for air ingestion levels of approximately 5%, performance is pump and site dependent; for air ingestion greater than 15%, the performance of most centrifugal pumps is fully degraded.

For emergency cooling pump operation at very low flow rates (< about 25% of best efficiency point), even small quantities of air may accumulate, resulting in air binding and complete degradation of pump performance.

3.2.2.3 Combined Effects of Cavitation and Air Ingestion

Few data on the combined effects of cavitation and air ingestion are available. Figure 3.8, which uses test results from Merry (1976), shows that as the air ingestion rate increases, the NPSH requirement for a pump also increases. The curves for this particular pump show that air ingestion levels of about 2% result in a 60% increase in the NPSH required (allowed head degradation based upon 3% degradation from the liquid head performance).

3.2.2.4 Particulate Ingestion

The assessment of pump performance under particulate-ingesting conditions is based on estimates of the type and concentrations of debris likely to be transported through the screens to the pump inlet. In the absense of comprehensive test data to quantify types and concentrations of debris that will reach the pumps, it has been estimated that concentrations of fine, abrasive-precipitated hydroxides are of the order of 0.1% by mass, and concentrations of fibrous debris

---

(Continued)

- Reasonable care must have been used in experimental techniques and in the documentation of results.
- (It should also be noted that the quantities of water recirculated in BWRs are significantly larger than those in PWRs.)

Test results meeting these criteria were then reduced to common, normalizing parameters and plotted for comparison.

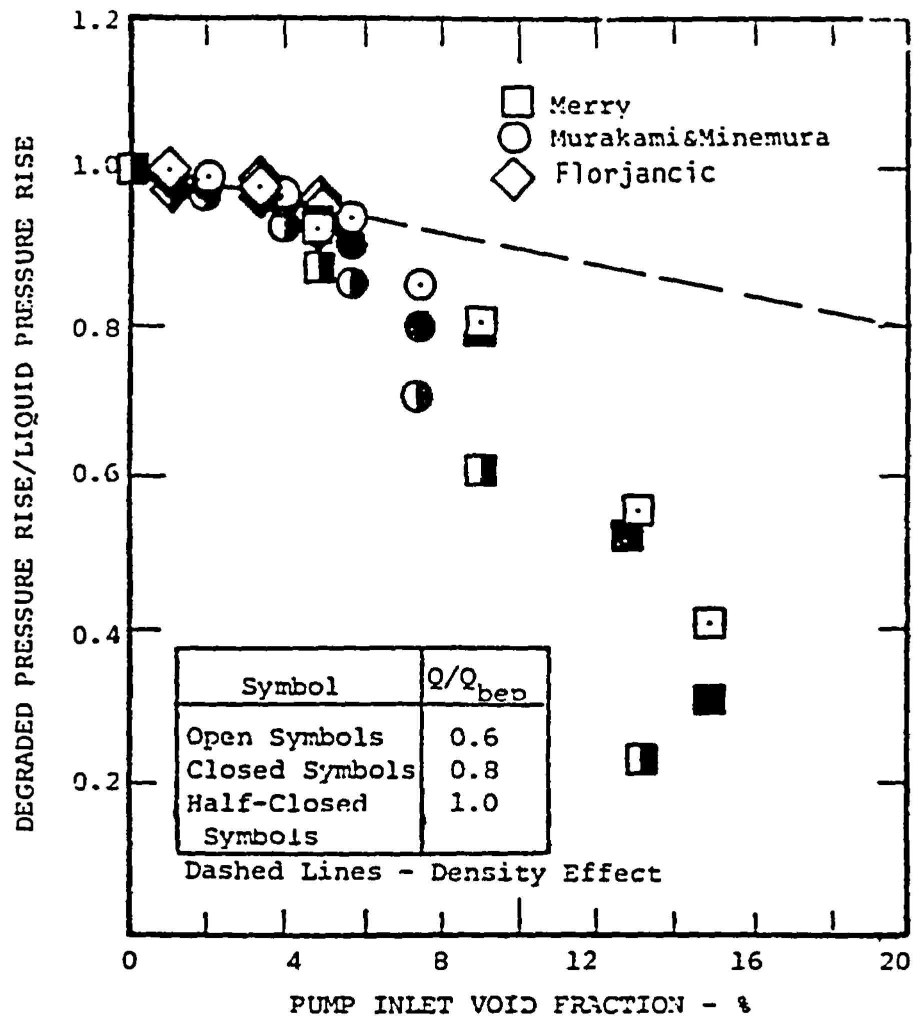


Figure 3.7 Head degradation under air ingesting conditions as a function of inlet void fraction (% of total flow rate by volume)

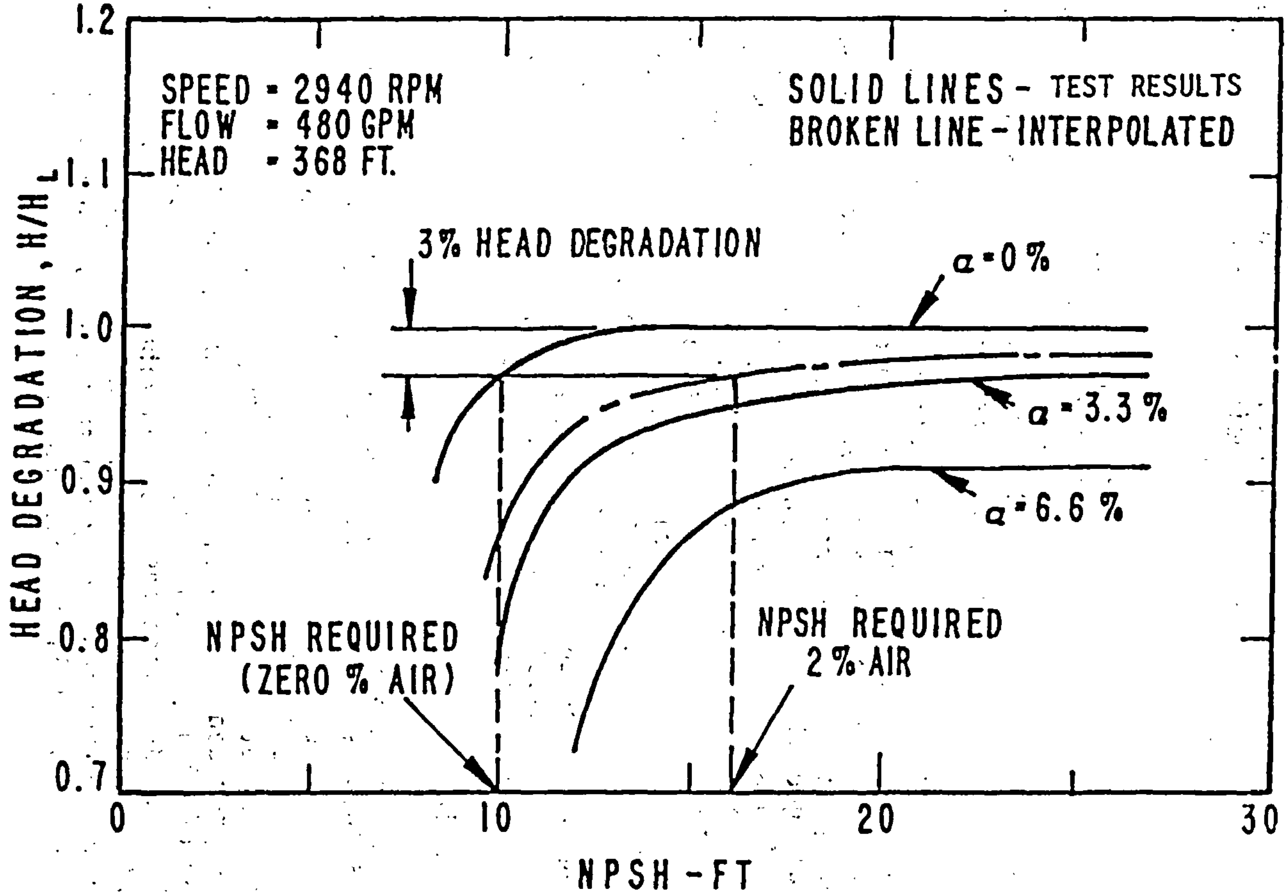


Figure 3.8 Effect of air ingestion on NPSH requirements for a centrifugal pump

are of the order of 1% by volume.* The effects of particulates in these quantities have been assessed on the basis of known behavior of this type pump under similar operating circumstances.

Ingestion of particulates through pumps is not likely to cause performance degradation for the quantities and types of debris estimated above. Because of the upstream screens, particulates likely to reach the pumps should be small enough to pass directly through the minimum cross-section passages of the pumps. Because of generally low pipe velocities on the pump suction side, particulates reaching the pumps should be of near-neutral buoyancy and, therefore, behave like the pump fluid.

Manufacturers' tests and experience with these types of pumps have shown that abrasive slurry mixtures up to concentrations of 1% by mass should cause no serious degradation in performance. Tests on single-stage pumps similar in construction to those used in RHR service have shown that quantities up to 4% of fiber paper stock by mass could be handled without appreciable degradation.

A major concern regarding the effects of particulates on pump performance and operability has been the effects of fibrous or other debris (such as paint chips) on pump seal and bearing systems. Porting within cyclone separators and the flush ports for mechanical shaft seals or water-lubricated bearings may become clogged with debris. In such an event, seal or bearing failure is likely. In the PWR plants that were reviewed, pumps used oil-lubricated or permanently lubricated bearings and mechanical shaft seals. For these configurations, the seals may be subject to failure because of clogging, but the bearings are not. The construction of mechanical face seals used in these pumps is such that complete pump degradation or failure is not likely, even in the event of seal failure. In many of the applications in BWRs, multistage pumps incorporate interstage bushings that are lubricated by the pumped fluid. In these applications, it is possible that excessive wear or clogging due to the presence of particulates or debris may cause bearing failure.

3.2.2.5 Swirl

The effects on pump performance because of swirl resulting from sump vortices are negligible if the pumps are located at significant distances from sumps. Test results discussed in Section 3.4 indicate that swirl angles in the suction pipe were typically 4° in PWR sump configurations (measured at 14 pipe diameters from the sump outlet) and 0 to 7° in BWR configurations. RHR and CSS pumps are generally preceded by valves, elbows, and piping with characteristic lengths on the order of 40 or more pipe diameters. This system of piping components is more likely to determine the flow distribu-tions (swirl) at the pump inlet than the swirl caused by sump hydraulics. However, for swirl angle > 10° it should be noted that swirl induced by the sump causes a higher friction loss than is the case with nonswirling flow. For pumps with inlet bells directly in the

---

*The concentration for abrasive AlO(H) was obtained from Niyogi and Lunt (1981) in which it was estimated that 3000 pounds of precipitate would develop in 30 days and recirculate with 3.7 million pounds of water. The 1% by volume concentration of fibrous debris is based on the quantity of fibrous insulation reaching the sump screens typical of a PWR (see Table 3.4 below), mixing with 200,000 gallons from the refueling water storage tank and being recirculated through the pumps.

sumps, vortices and accompanying swirl in the inlet bell can cause severe problems, because of asymmetric hydraulic loads in the impeller. Hence, this type of installation should be avoided.

3.2.3 Calculation of Pump Inlet Conditions

The steps given below delineate the calculational procedure for assessing the inlet conditions to the pump, based on the findings noted above. The procedure follows routine calculation methods used for estimating NPSH available, except that the procedures incorporate steps to allow for air ingestion effects.

Figure 3.9 shows a schematic of the pump suction system with appropriate nomenclature. The procedure is as follows:

(1) Determine the hydrostatic water pressure (gage), P_{sg}, at the sump suction inlet centerline, accounting for temperature dependency and minimum sump water level. An important factor to include in determining the maximum sump water level is pressure head loss across the sump screen (see Section 3.3).

(2) Based on the sump hydraulic assessment, determine the potential level of air ingestion at the sump suction pipe, α_s, as discussed in Section 5.2.

(3) Calculate the pressure losses in the suction pipe between the sump and the pump inlet flange. Pressure losses are calculated for each suction piping element (inlet loss, elbow loss, valves, pipe friction) using the average velocity through each element, V_i, and a loss coefficient, K_i, for each element. The total pressure losses are then

$$P_{\ell} = (\gamma/144) \sum_{i=1}^{N} K_i \; V_i^2/2g$$

where γ is the specific weight of water (lb/ft³), 144 is the conversion from psf to psi and N is the number of elements.

The loss coefficients are defined as

$$K_i = \frac{h_{\ell i}}{V_i^2/2g}$$

where

$h_{\ell i}$ is the head loss in ft of water in element i

g is the acceleration due to gravity

V_i is the average velocity in element i in fps

Loss coefficients can be found in standard hydraulic data references such as Hydraulic Institute Standards (1975).

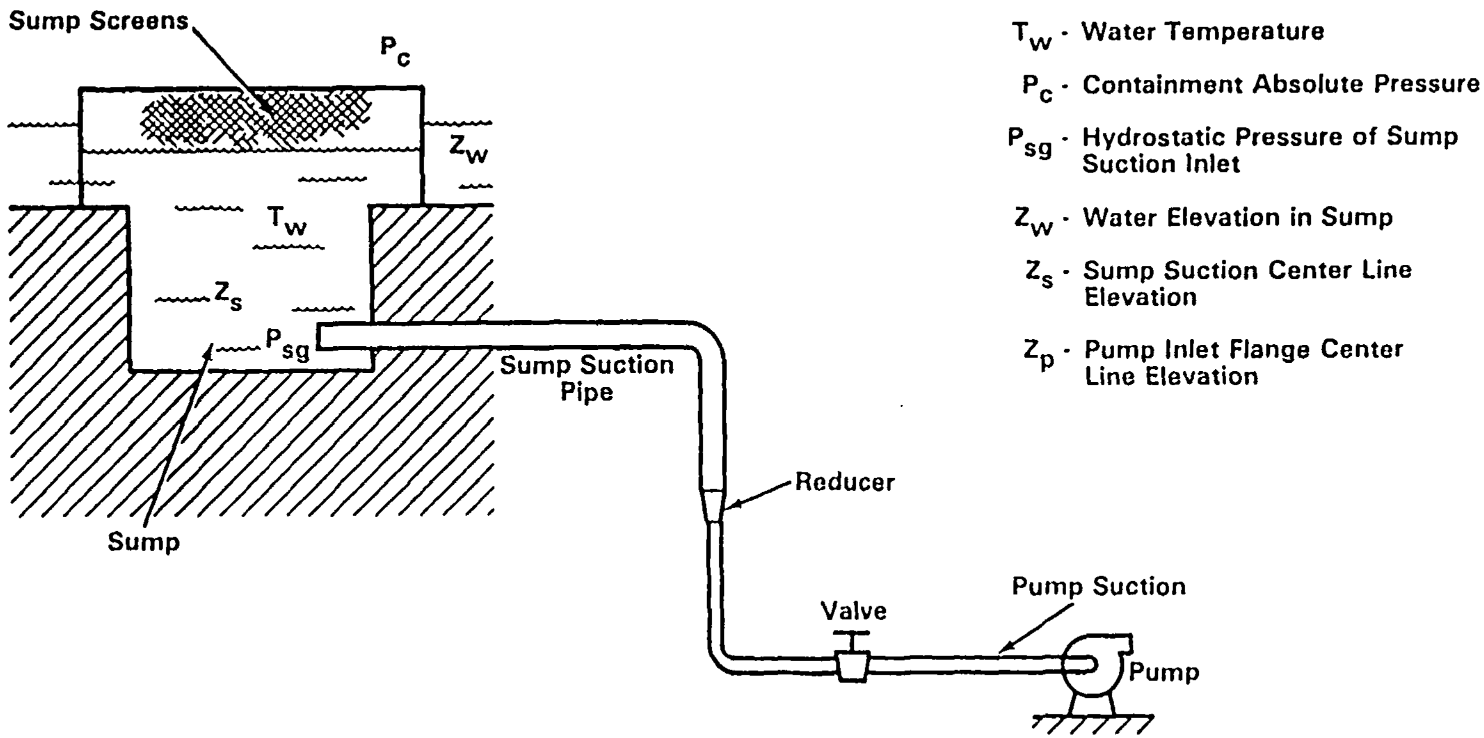


Figure 3.9 Schematic of suction systems for a centrifugal pump

(4) Calculate a value for P_p that will be used to correct the volumetric flow rate of air at the sump suction pipe for density changes (If air ingestion is zero, Steps 4, 5, and 6 can be ignored):

$$P_p = P_{sa} - P_\ell + P_h - P_d$$

where

P_{sa} = the total absolute pressure at the sump suction pipe centerline, which is the sum of the hydrostatic pressure, P_{sg}, and the containment absolute pressure, P_c (determined in accordance with RG 1.1 and 1.82 for NPSH determination)

P_ℓ = the pressure loss determined in Step 3,

P_h = the hydrostatic pressure due to the elevation difference between the sump suction pipe centerline, Z_s, and the pump inlet flange centerline, Z_p

P_h = $(\gamma/144)\ (Z_s - Z_p)$

P_d = the dynamic pressure at the pump inlet flange using the average velocity at the pump suction flange, V_p

$$P_d = \frac{\gamma}{144}\frac{(V_p)^2}{2g}$$

(5) Calculate the corrected air volume flow rate at the pump inlet flange, α_p, based on perfect gas, isothermal process

$$\alpha_p = (P_{sa}/P_p)\alpha_s$$

(6) If α_p is greater than 2%, inlet conditions are not acceptable.

(7) Calculate NPSH at the pump inlet flange, taking into account the requirements of RG 1.1 and 1.82, as follows:

$$NPSH = (P_c + P_{sg} - P_\ell + P_h - P_{vp})\ (144/\gamma)$$

where

P_{vp} = the vapor pressure of the water at evaluation temperature and the other terms are as defined in Steps 1, 3, and 4 above.

(8) If air ingestion is not zero, the NPSH required from the pump manufacturer's curves must be modified to account for air ingestion as follows:

$$\beta = 0.50\ (\alpha_p) + 1.0$$

where

α_p = the air ingestion level percent by volume at the pump inlet flange.

Then

NPSH required (air/water) = βx (NPSH required for water)

The expression for β is empirical. It has been selected because it provides a reasonable amount of conservatism in predicting NPSH requirements in the presence of less than 2% air ingestion at the pump inlet. However, the data on which this conclusion is based are limited mainly to the tests of Merry (1976), and the test data scatter mentioned in the published report are not quantified. Therefore, it is important that good judgment be used in the application of the correct factor β to plant calculations. In particular, the conservatisms in assumptions for calculating the pump inlet conditions should be weighed carefully if the calculated NPSH available for air/water operation is marginal with respect to the required NPSH.

(9) If NPSH available from Step 7 is greater than NPSH required from Step 8, pump inlet conditions should be satisfactory.

3.3 Debris Assessment

The safety concerns related to the generation of thermal insulation debris as the result of a LOCA and the potential for sump screen blockage were addressed generically as follows:

(1) Nineteen reactor power plants were surveyed in 1982 to identify insulation types used, quantities and distribution of insulation, methods of attachment, components and piping insulated, variability of plant layouts, and sump designs and locations. Additional information was contributed during a public comment period in 1983.

(2) Experiments were conducted to establish the pressure conditions leading to the onset of damage to typical nonencapsulated mineral wool and fiberglass insulations, and attendant debris generation. The buoyancy and transport characteristics of both fibrous and reflective metallic insulations were investigated, along with screen blockage and head loss.

3.3.1. Overview

Assessing LOCA-generated insulation debris requires consideration of the following elements:

(1) The type and quantities of insulation employed. These are important because the potential for transport and blockage depends upon the insulation material employed. Identification of insulations employed and their distribution on piping and major components is important, as is the identification of methods of attachment.

(2) Long-term cooling. For both PWRs and BWRs, the maintenance of long-term recirculation cooling is the underlying safety concern, and breaks (or LOCAs) requiring long-term cooling must be assessed. For PWRs, breaks in the primary coolant system are of principal concern, and evaluations of potential break locations (and size) should be the basis for estimating quantities of debris generated. For BWRs, potential breaks in the feedwater and recirculation loop piping and steamline breaks constitute the

LOCAs that necessitate long-term cooling. SRP Section 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping," should be used to identify potential break locations.

(3) Possible break-target combinations. On the basis of the break locations identified in Step 2, possible break-target combinations must be assessed. Sections 3.3.3 and 3.3.4 provide guidance for defining the break jet envelope. Analyses should consider the effects in close proximity of the break (within $\leq$ 7 L/D's of the break) where insulation destruction will be highest. Beyond 7 L/D's, insulation could be dislodged in the as-fabricated state, depending on the methods of attachment.

(4) Level of insulation damage and volume of LOCA-generated insulation debris. The level of damage can be severe, partly damaged, or dislogement of as-fabricated insulation segments. Insights regarding potential levels of destruction can be derived from the HDR (Heissdampfreaktor or superheated steam reactor) experiments (see Appendix C). In those experiments, destruction of insulation (particularly fiberglass insulation material) within 2 to 4 meters of the break was very severe.

Analytical studies (see Section 3.3.4) of expanding two-phase jets also show very high stagnation pressures near the break location (within 3 to 5 L/D's). The insulations and coverings within this region will be subjected to stagnation pressures on the order of 10 to 50 bars.

Small-scale experimental studies on some typical fibercloth-jacketed insulation pillows (see Section 3.3.3) revealed that the onset of destruction (the start of tearing of the fibercloth jacket) occurred at stagnation pressures of 20 to 35 psi.

Thus the estimation of debris generation is complex and material dependent. Sections 3.3.3 and 3.3.4 provide means for making such estimates.

(5) Transport characteristics. The transport of LOCA-generated insulation debris will be controlled initially by the blowdown phase (when the jet forces will distribute debris). Long-term transport will occur during the recirculation phase when containment-flow forces (or velocities) control the transport of debris. This long-term transport depends on the type of insulation, level of damage and flow velocity. Both fibrous insulation and reflective metallic insulation (RMI) debris fragments transport at low velocities (0.2 to 0.5 ft/sec). RMI debris generally accumulates at the lower portion of debris screen, while fibrous insulation debris builds up uniformly on the screen. Thus, highly damaged insulation debris will exhibit transport characteristics significantly different from the as-fabricated insulation segments (e.g., transport can occur at low velocities).

The plant layout, particularly for PWRs, is an important consideration in the initial transport (or blowdown) phase. If the sump and break locations are such that the break jet can target the sump region directly, direct transport to the vicinity of the sump screen can be postulated immediately. Moreover, if the break jet can target the sump screen, screen survivability relative to jet loads should be assessed.

(6) Screen blockage (or suction strainer blockage). This blockage is dependent on the material characteristics of the debris transported to the screen and on the local velocities, which can pull such debris to the screen, as well as on the findings obtained for the transport of fibrous and metallic materials and as-fabricated sections of typical insulation materials.

There are two parts to this element:

(a) Will the debris be transported? Transport is dependent on recirculation flow velocities within containment.

(b) Will blockage occur? Blockage is dependent on the approach velocities near the screen or suction strainer, and the approach velocity will establish the blockage patterns that will occur.

Shredded fibrous debris is transported at near-neutral buoyancy conditions and is deposited (in a general sense) uniformly across a screen structure. Metallic foils (such as those used internally in reflective metallic insulations) exhibit transport characteristics and screen blockage patterns that are a function of foil thickness (or rigidity) and screen-approach velocities. Development of a blockage model for foils is more difficult than it is for fibrous debris.

(7) Head loss as a result of the estimated screen blockage. The results of Step (6) dictate the estimating methods applicable. Results of experiments have shown that blockage losses for fibrous insulation materials can be described as a power function such as

$$\Delta H = a\, U^b t^c$$

where

a, b, and c are coefficients that should be derived from experimental data

t(thickness) = volume of debris/effective screen area

U = approach velocity

Head losses that result from impervious materials (such as metallic sheets) are dependent on the potential blockage patterns resulting from the plant-specific reviews. For example, a PWR sump with a horizontal debris screen will incur a different type of blockage than will a sump with high vertical debris screens. Sections 3.3.5 and 5.3 provide additional information relative to these considerations.

(8) Accurate predictions of recirculation flow velocities within the containment during the long-term cooling mode. These are as important as the experimentally derived debris transport velocities discussed above. If predicted recirculation velocities exceed transport velocities, debris will move toward the sump. An analytic method that permits estimation of velocities within containment is reported in NUREG/CR-2791. However, because of simplifications inherent in that modelling technique, a more refined analysis may be warranted if the predicted fluid velocities are within a factor of two of the transport velocity determined experimentally

for each of the insulation types. That is to say, although the recirculation flow velocities discussed in Appendix D would predict one-half of the critical transport velocity (thereby indicating zero transport), transport might actually occur because of flow field variabilities within containment that are not accounted for.

3.3.2 Types of Insulations Employed

Insulations utilized in nuclear power plants can be categorized in two major groups, as discussed in the following paragraphs.

(1) Reflective Metallic Insulation

Reflective metallic insulation (RMI) is an all-metallic insulation design based on the concept of utilizing a series of highly reflective foils to retard heat transfer. RMI is generally constructed from stainless steel, although aluminum interior foils have been used in conjunction with stainless steel inner and outer liners. Figure 3.10 provides details of (1) typical, as-fabricated RMI segments and (2) their internal foil construction. Generally RMI is manufactured in half-shell segments or other geometric shapes that are prefabricated to fit piping or other major components (reactor vessels, steam generators, and the like) and that use snap-on latching for attachment.

There are currently at least four different manufacturers of RMI: Diamond Power Speciality Company, TRANSCO, Johns-Manville, and ROMET. All vendor designs vary. Some designs have open ends; others have sides sealed with foils. Interior foils range in thickness from 0.0025 inch to 0.010 inch. Inner and outer liners are generally thicker (on the order of 0.030 inch to 0.040 inch) and may be flat, corrugated, or dimpled.

(2) Conventional or Mass-Type Insulation

Mass-type insulation is an industry-derived term that encompasses a wide range of insulation materials and differentiates them from RMI.

In mass-type insulation, the materials used as the insulation filler are from one of two broad categories, fibrous and others.

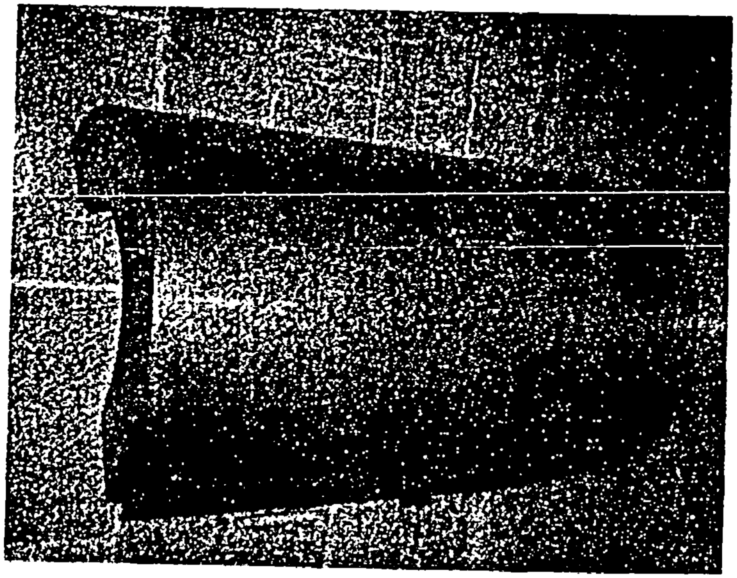
Fibrous insulations include:

Calcium Silicate Molded Block

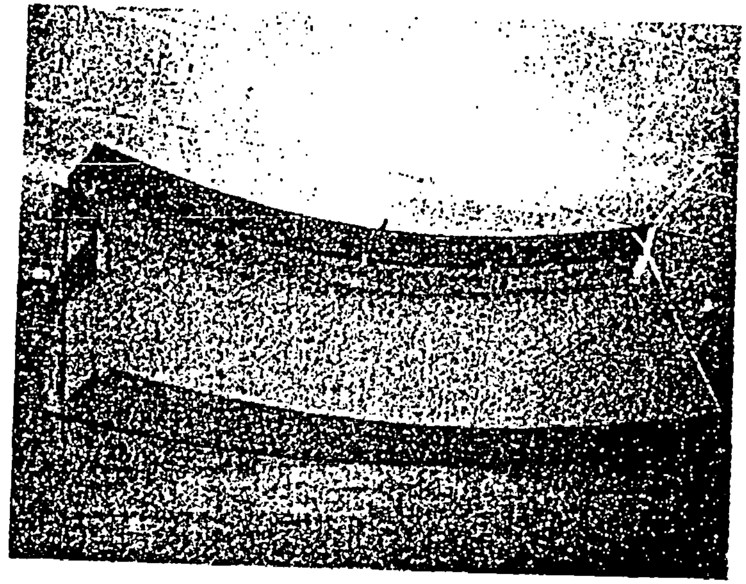
Calcium silicate molded block insulation is a molded, high-temperature pipe and block insulation composed of hydrous calcium silicate. It is light weight, has low thermal conductivity and high structural strength, and is insoluble in water. Its density (dry) is 13 to 14 pounds per cubic foot. Its compressive strength (based on 1-1/2 inch thickness) is 60 to 250 psi. The molded blocks are provided in thicknesses of up to 4 inches and lengths of up to 3 feet.

Expanded Perlite Molded Block

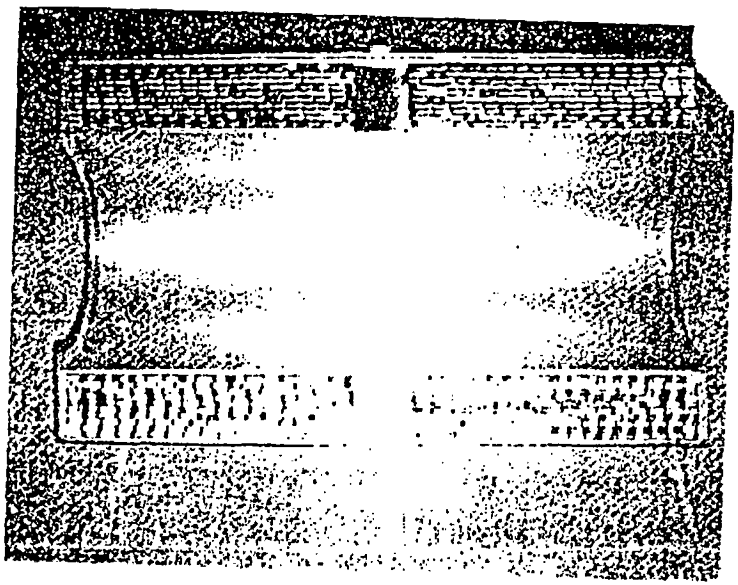
Expanded perlite molded block insulation is composed of expanded perlite with reinforced mineral fiber and inorganic binders. It is an insulating



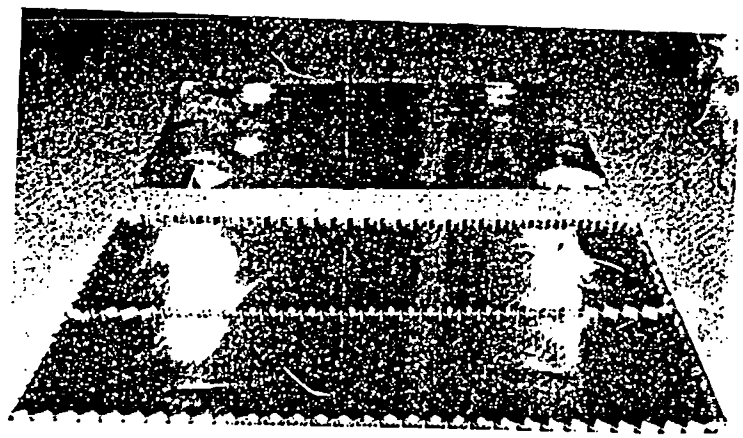
HALF SHELL (OUT-SIDE)



TYPICAL OUTER SHELL



HALF SHELL (IN-SIDE)



INNER FOILS (.0025 inches thick)

Figure 3.10 As fabricated, reflective metallic insulation components material with properties similar to those of calcium silicate insulation. The average maximum density is 14 pounds per cubic foot. Its flexural strength should be not less than 35 psi, and its compressive strength dry is 60 psi and wet is 25 psi.

- Fiberglass Molded Block

  Fiberglass molded block insulation is composed of glass that has been foamed or cellulated under molten conditions, annealed, and set to form a rigid incombustible material with hermetically sealed cells. The density is between 7.0 and 9.5 pounds per cubic foot. Its flexural strength is 60 psi, and its compressive strength is 75 psi.

- Nukon™ Fiberglass Blankets

  The leading manufacturer of this type insulation is Owens-Corning which makes thermal insulation system called NUKON™ for use in the containment areas of light water nuclear power plants. NUKON™ is a blanket insulation consisting of fiberglass insulating wool reinforced with fiberglass scrim and sewn with fiberglass thread. The blanket may have secondary holding straps attached to it and wrapped completely around it. This material has a low density (2 to 4 pounds per cubic foot). Figure 3.11 shows this type of insulation as fiberglass core material.

- Mineral Wool Fiber Block

  Mineral wool fiber block insulation is made of a mineral substance, such as rock, slag, or glass processed from a molten state into fibrous form. The density, depending on kind, ranges from 10 to 20 pounds per cubic foot. The strength varies considerably with the classes of insulation. The moisture is less than 1.0% by volume.

Other insulations include.

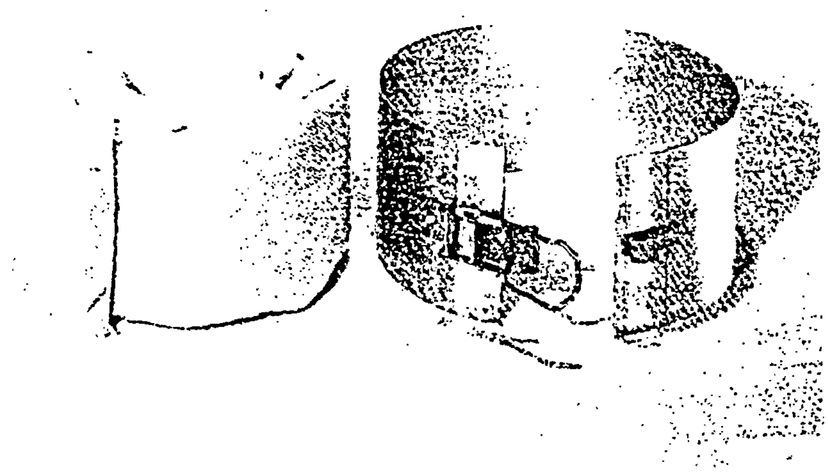
- Cerablanket™

  Cerablanket™, manufactured by Johns-Manville, is a ceramic fibrous insulation material with a density of 6 pounds per cubic foot. The Cerablanket is enclosed in 0.006-inch metal foil and then encapsulated in a reflective insulation structure.

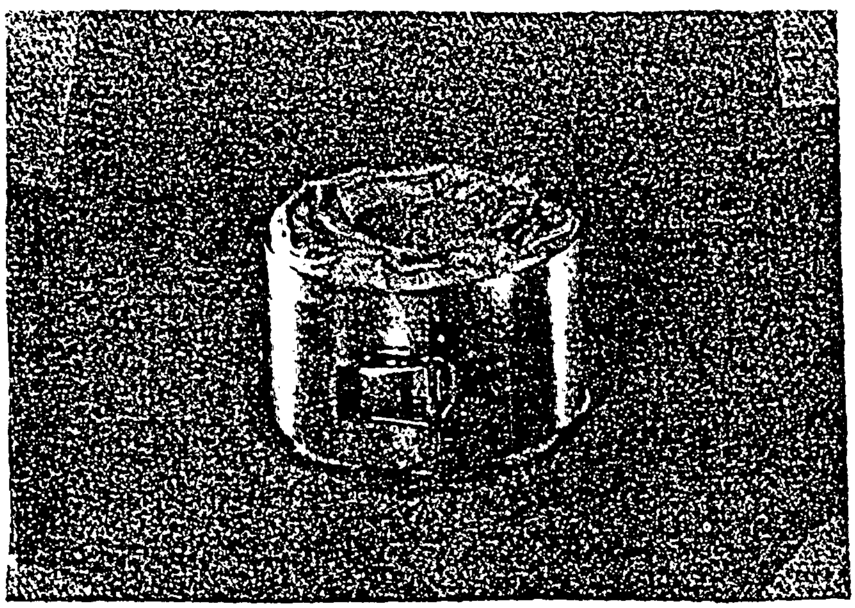
- Unibestos™

  Unibestos™ insulation is composed of lime and diatomaceous silica taken from natural deposits. These basic ingredients are bonded with asbestos fiber possessing the tensile strength of piano wire. This composition is then encased in stainless steel sheet.

Figure 3.12 illustrates a variety of materials of the mass-type insulation. (NUREG/CR-2403 provides a more extensive description of insulations employed, particularly those used in older plants.)

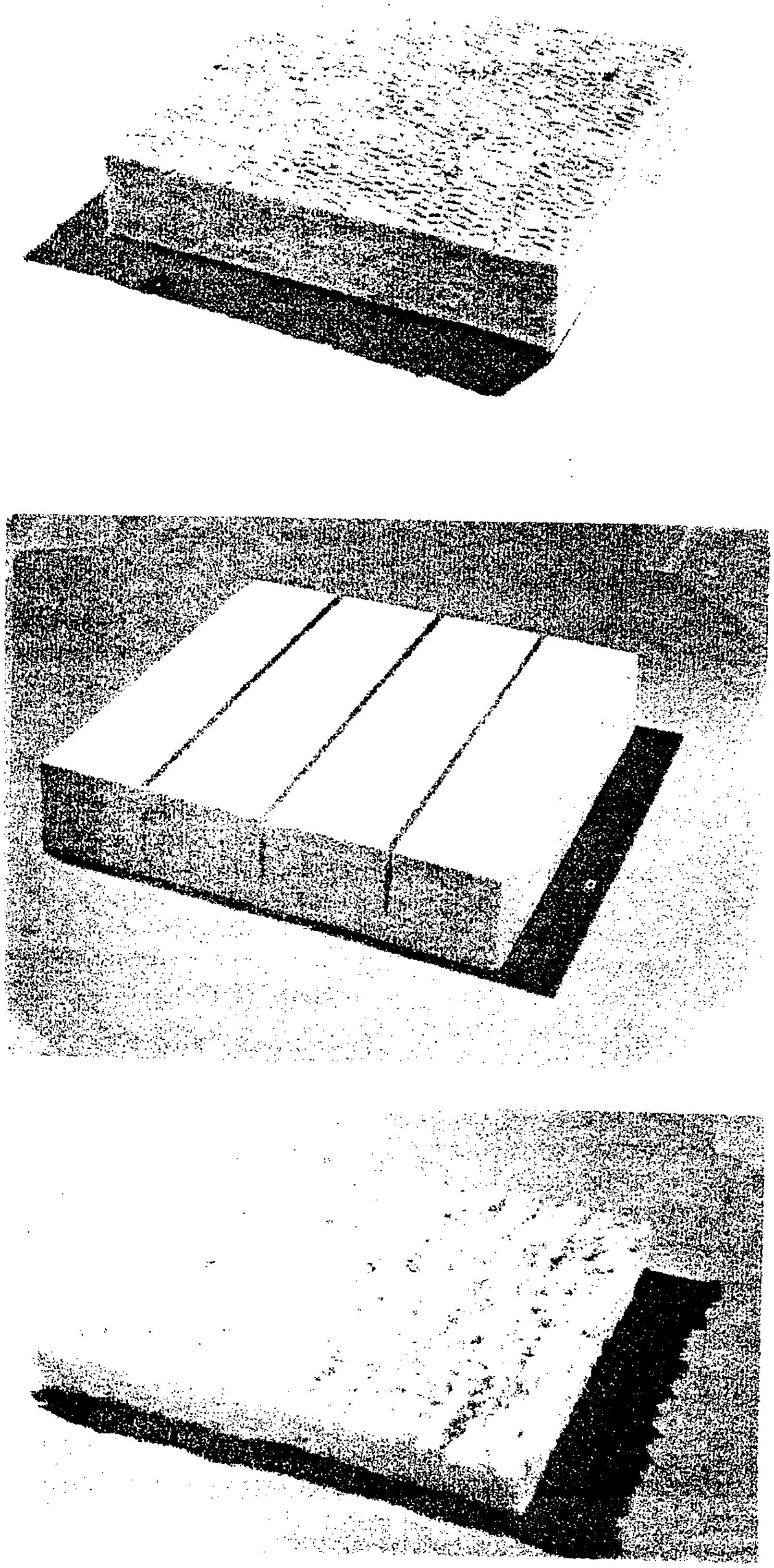


FIBERGLASS CORE MATERIAL   JACKET AND LATCH



JACKETED ASSEMBLY

Figure 3.11 Jacketed insulation assemblies



MINERALWOOL

CALCIUM SILICATE

NEEDLED FIBERGLASS
(E - TYPE)

Figure 3.12 Mass-type insulations

Any of the above described mass-type insulations can sometimes be enclosed in an outer shell or jacket or cloth covers. The following categories are currently being used by the industry:

- Totally Encapsulated or Semi-Encapsulated Insulation

  Internal insulation in the totally encapsulated or semi-encapsulated category can be mass-type materials that act as the principal heat barrier. The outer shell is generally made of sheet metal and in some cases the ends are closed. The encapsulation is used to contain the mass insulation and to ease installation and removal.

  Caution is recommended in assessing encapsulated insulation because of the generalized use of this category and wide variability of designs procured and installed in plants. Figure 3.13 illustrates some encapsulated insulations. Survivability under break jet loads requires assessment of the specific insulation employed and the structural capability of the encapsulation provided.

  The construction of semi-encapsulated insulation modules is exactly the same as that of totally encapsulated ones, except that semi-encapsulated modules are assembled in the field and clamped, not welded, together.

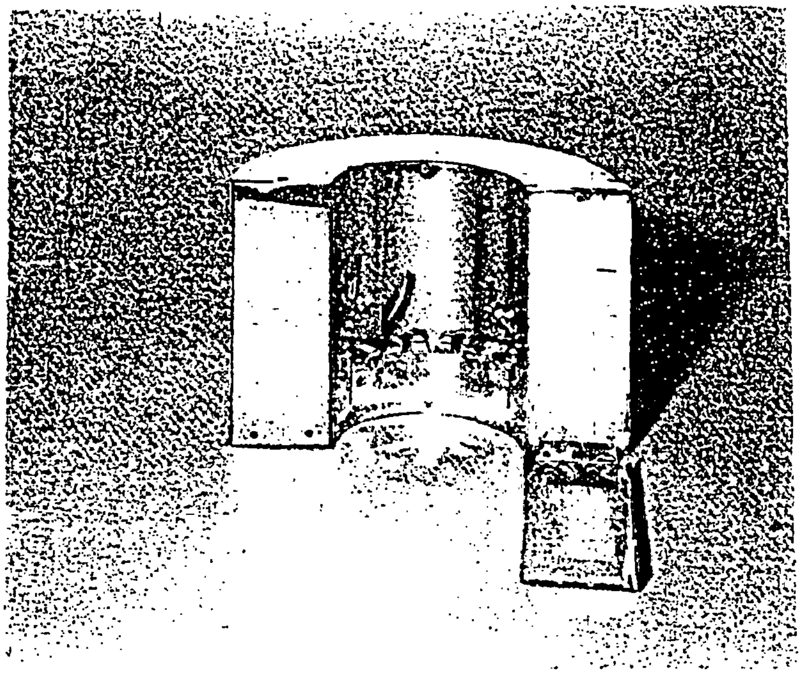
- Jacketed Insulations

  In the jacketed insulation category, the principal heat barrier (internal insulation) is the same as it is for mass-type insulation. The jacket (which is usually a separate outer metal cover such as a stainless steel sheet, asbestos cloth, fiberglass cloth, or aluminum) is simply an outer cover to protect the core material. Thus jacketed insulations are an intermediate arrangement between encapsulated and nonencapsulated insulation. Generally banding or latching mechanisms are employed for jacketed insulations such as shown on Figure 3.11.

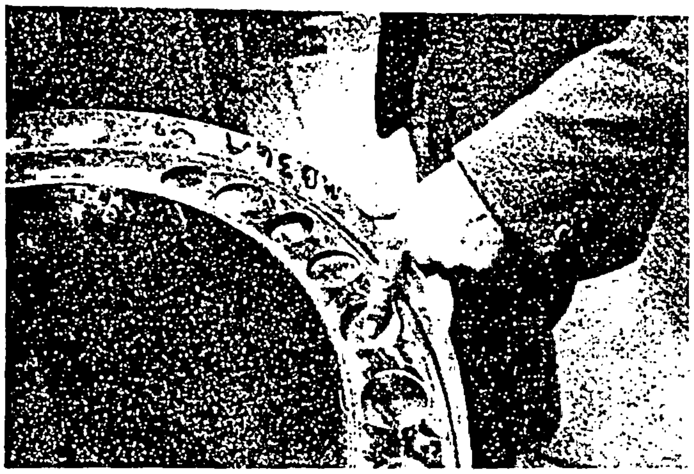
Urethane and polyurethane foam antisweat is another jacketed-type insulation. It is a rigid cellular foam plastic that combines light weight and strength with exceptional thermal insulating efficiency. The foam is a vast cross-linked network of closed cells; each cell is a tiny bubble full of gas that accounts for 90% of its volume. Its density ranges from 1.8 to 4.0 pounds per cubic foot. The insulation is sealed with a vapor barrier of aluminum foil or a metal jacket.

Regardless of the type of insulation employed, the assessment of debris effects must focus on types and quantities of materials present and their survivability during a LOCA, as discussed in Sections 3.3.3 and 3.3.4.

Plant insulation surveys were performed in 1981 and 1982, and the results are summarized in Table 3.3. (The details associated with these surveys are in NUREG/CR-2403 and its Supplement 1.) These surveys showed that there was a wide variability in types of insulations employed, but that the newer plants were electing to utilize RMI. Moreover, based on the two BWRs surveyed, the trend appeared to be total use of RMI or totally encapsulated insulation.



ENCAPSULATED FIBERGLASS



ENCAPSULATED REFLECTIVE METALLIC

Figure 3.13 Encapsulated insulation assemblies

Table 3.3 Types and percentages of insulation used within the primary coolant system shield wall in plants surveyed

Plant	Types of Insulation and Percentage*					
	Reflective Metallic	Totally Encapsulated	Mineral Fiber/Wool Blanket	Calcium Silicate Block	Unibestos Block	Fiberglass
Oconee Unit 3	98	--	--	--	--	2
Crystal River Unit 3	94	5	1	--	--	--
Midland Unit 2	78	--	--	--	--	22
Haddam Neck	3	--	--	--	95¶	1
Robert E. Ginna	--	--	5	80	10	--
H. B. Robinson	--	--	--	15	85	--
Prairie Island Units 1 & 2	98	--	--	--	--	2
Kewaunee	61	--	--	--	39	--
Salem Unit 1	39	8	53**	--	--	--
McGuire Units 1 & 2	100	--	--	--	--	--
Sequoyah Unit 2	100	--	--	--	--	--
Maine Yankee	13	--	48	25	13	1
Millstone Unit 2	25	35	5	30	--	--
St. Lucie Unit 1	10	--	--	90	--	--
Calvert Cliffs Units 1 & 2	41	59	--	--	--	--
Arkansas Unit 2	46	53	--	--	--	1
Waterford Unit 3	15	85	--	--	--	--
Cooper	30	70	--	--	--	--
WPPSS Unit 2	100	--	--	--	--	--

*Tolerance is ± 20 percent

**Both totally and semi-encapsulated Cerablanket is used, however, inside containment only totally encapsulated is employed.

¶Unibestos is currently being replaced by Calcium Silicate. However, both types of insulation have the same sump blockage characteristics.

However, comments received during the public "for comment" period associated with USI A-43 (June-July 1983) presented a changing picture (see Table 3.4). Some older operating plants (e.g., Monticello) have been reinsulated with fibrous insulation. Newer BWRs (e.g., Limerick) are being insulated with fiberglass, and the increasing use of fiberglass is evident. Replacement of selected insulation also occurs during, or following, inservice inspections. These recent observations re-emphasize the large varibility of insulations employed, the plant-specific aspects associated with insulations used (licensees handle insulation on a site-specific basis and changes need not be reported), and the time dependency factor. As new insulation products are developed, new materials are introduced into nuclear plants.

3.3.3 Insulation Debris Generation

Jet impingement forces are the dominant insulation debris generator. Other contributors, such as pipe whip and impact, have been studied and shown to be of secondary importance (NUREG/CR-2791).

The criteria for defining break or rupture locations should be consistent with the requirements of SRP Section 3.6.2, which provides guidance for selecting the number, orientation, and location of postulated ruptures within a containment.

The safety concerns associated with debris relate to ensuring long-term recirculation capability. Therefore, for PWRs, the postulated breaks of concern are those in the primary coolant system and in components (or other systems) that are connected to the primary coolant system. For BWRs, the postulated breaks of concern are in the feedwater and recirculation systems and in the steam lines.

The destructive nature of high pressure break jets has been experimentally demonstrated in blowdown experiments conducted in the HDR facility (see Appendix C). Figures 3.14 and 3.15 show damage to reinforced concrete structures in the HDR. Figures 3.16, 3.17, and 3.18 show the damage to insulation and insulated components in the HDR.

These blowdown tests (blowdown was from 110 bars and 280°C to 315°C, under steam and subcooled water conditions) revealed that all glass fiber insulation was destroyed within 2 meters of the break nozzle and distributed throughout the HDR containment as very fine particles. In addition, iron wrappers were thrown away from vessels within 4 to 6 meters of the break nozzle, with glass fiber untouched. With enforced shieldings (steel bandages) around the vessels, the damage was reduced. Mineral wool insulation that was encapsulated in iron plate withstood the rough blowdown conditions well. Break sizes of 200-mm, 350-mm, and 430-mm diameter were investigated.

3.3.4 Two-Phase Jet Loads Under LOCA Conditions

Determination of the extent of potential damage requires estimation of pressure and flow field forces resulting from the expanding jet. On the other hand, the flow field for a two-phase jet is extremely complicated and multidimensional. The jet impingement model discussed in this section is based on a study of HDR experimental data by Sandia National Laboratory (SNL). This model is under peer review by the ANS-58.2 Committee on Pipe Rupture and has not yet been incorporated in

Table 3.4 Insulation types used on nuclear plant components*

	Vessel	Coolant Piping	Coolant Pumps	S. G. (less bottom head)	S. G. Bottom	Pressurizer
PWRs						
Haddam Neck	Rm	C	C	C	C	C
IP-2 & IP-3	Rm	C	C	C	C	C
Maine Yankee	Rm	C	C	C	C	C
Millstone-3	Rm	C	C	C	C	C
Yankee	C	C	C	C	C	C
Palisades	Rm	C	C	C	C	C
Wolf Creek	Rm	C	C	C	C	C
Ft. Calhoun	Rm	C	C	C	C	C
Callaway	Rm	C	C	C	C	C
Robinson-2	Rm	C	C	C	C	C
Turkey Pt-3	Rm	C	C	Rm	Rm	C
Turkey Pt-4	Rm	C	C	C	C	C
St. Lucie-2	Rm	C	C	C	C	C
Waterford-3	Rm	E	E	E	E	E
South Texas 1&2	Rm	C	C	C	C	C
San Onofre-1	Rm	C	C	C	C	C
Ginna	Rm	C	Rm	Rm	C	C
Marble Hill	Rm	Rm	Rm	C	Rm	Rm & C
ANO-2	Rm	Rm	Rm	C	Rm	Rm
BWRs						
Limerick 1&2	Rm	C	C	N/A	N/A	N/A
Fitzpatrick	Rm	C	C	N/A	N/A	N/A
Perry 1&2	Rm	C	C	N/A	N/A	N/A
Monticello	Rm	C	C	N/A	N/A	N/A
Hatch-1	Rm	C	C	N/A	N/A	N/A

Insulation Legend:

Rm - Reflective Metallic Insulation
C - Conventional Insulations (e.g., fibrous & mass materials)
E - Encapsulated Insulation

*Based on material obtained during a public comment period may be obtained by writing to Generic Issues Branch, NRC, Washington, DC 20555.

Figure 3.14 Structural damage to railing and walls in the HDR facility following a blowdown experiment

Figure 3.15 Erosion of reinforced concrete in the HDR facility due to direct break jet impingement

Figure 3.16 Blowdown damage to fiberglass insulation covering the HDR pressure vessel

Figure 3.17 Distribution of fiberglass insulation after an initial HDR blowdown test

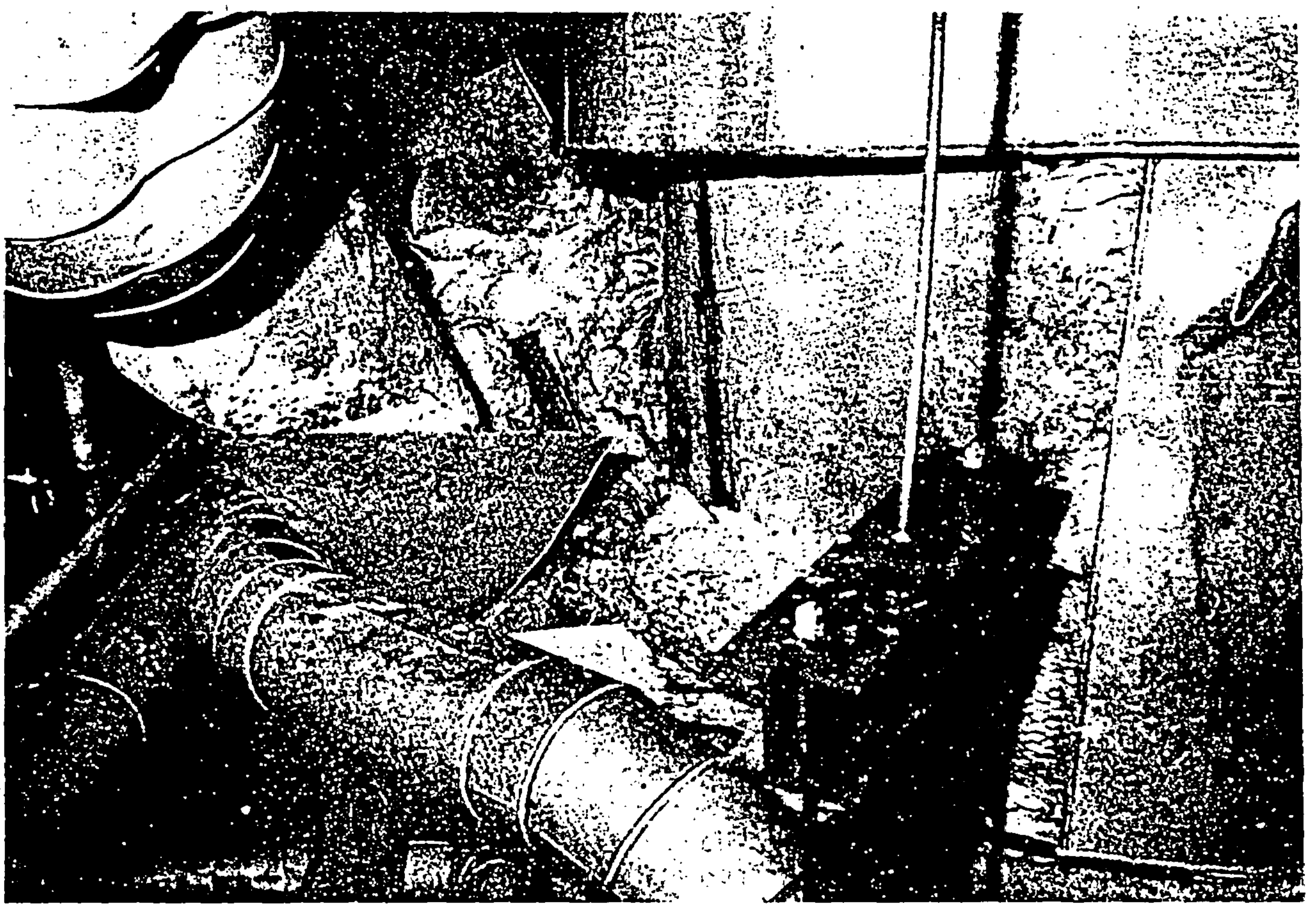


Figure 3.18 Blowdown damage to jacketed (sheet metal cover) reinforced (with wire mesh) fiberglass in the HDR blowdown compartment

SRP 3.6.2 as an endorsed approach. SNL has analytically studied two-phase jet impingement on targets over a range of pressures and temperatures representative of postulated LOCAs for BWRs and PWRs; those results are reported in NUREG/CR-2913.

In the expanding jet flow field, there are three natural divisions of the field (see Figure 3.19). There is a nozzle (or break) region where the flow chokes. In this region, there is a core at choked flow thermodynamic properties that projects downstream of the nozzle at distances that depend on the degree of subcooling. Downstream of this region there is the free jet region. Here the jet expands almost as a free, isentropic expansion; the flow is supersonic throughout this entire region. The free jet region terminates at a stationary shock wave near the target. This shock wave arises because the target propagates pressure waves upstream and, thus, produces a pressure gradient that will direct the fluid around the target. Downstream of the shock is the target region where the local flow field imposes a pressure loading on the target. Depending on the upstream flow conditions and the L/D's of the target, there may be a substantial total pressure loss across the shock wave. This loss arises because of the irreversible physics that characterize the shock. The pressure loss across the shock and radial velocity components can lead to negative pressure loads across the target, which can lift away materials (such as insulation segments) from targeted components. The HDR tests revealed evidence of such loadings.

NUREG/CR-2913 addresses the centerline behavior of two-phase jets and the radial loading for axisymmetric impinging two-phase jets. The method developed for calculating centerline behavior indicates that the jet stagnation pressure at a given target distance from the break (in terms of L/D) is a function of the stagnation pressure and steam quality or the degree of subcooling in the vessel. This functional dependence (on pressure and subcooling) largely disappears at about 5 L/D's from the break. At approximately 7 L/D's downstream of the jet origin along the centerline of the jet, stagnation pressure falls to roughly 20 psig regardless of the break thermodynamic conditions.

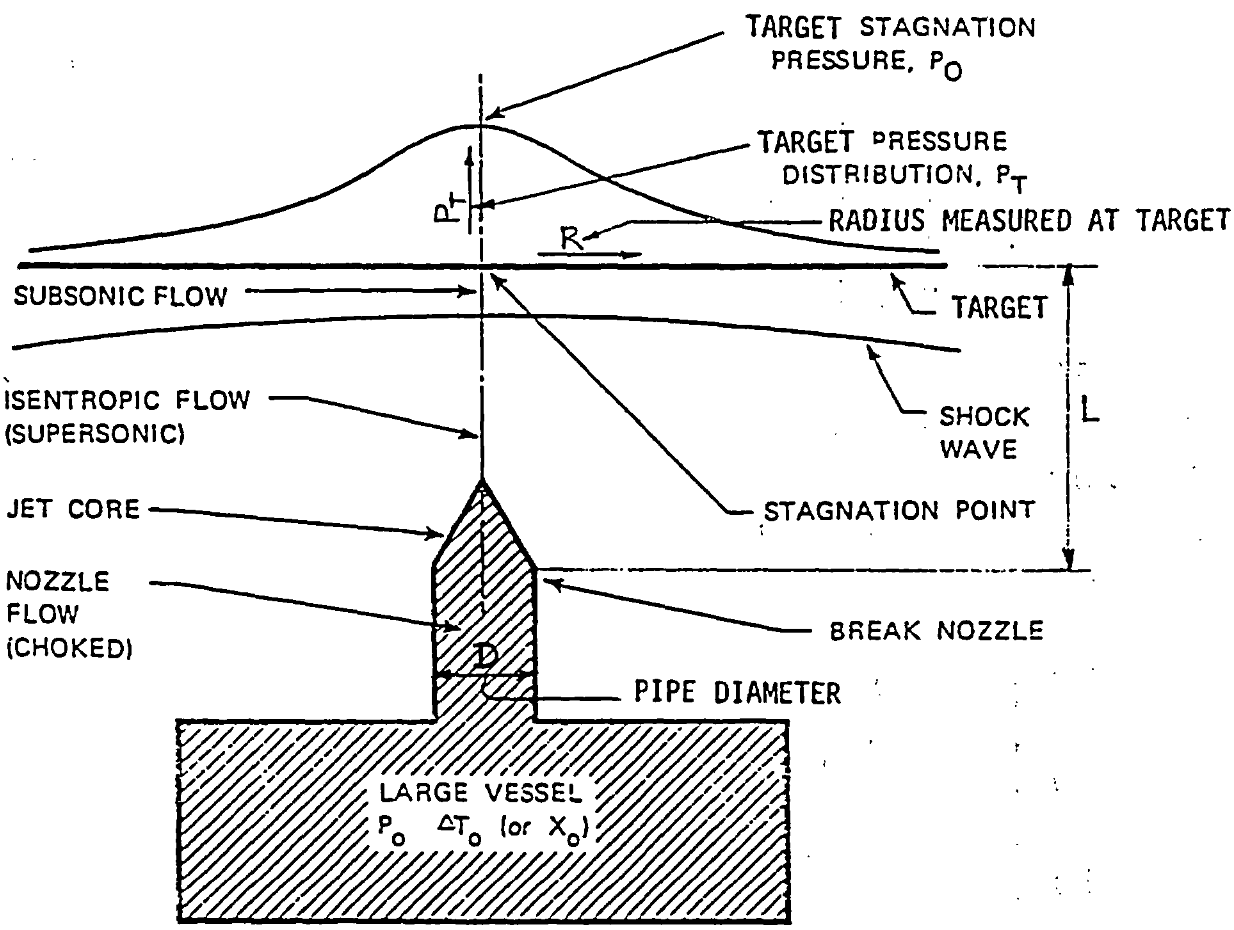
Two-dimensional pressure distributions were calculated and are reported in NUREG/CR-2913. These results indicate that the region targeted by an impinging two-phase jet is highly dependent on the thermodynamic conditions at the break. The constant pressure contours (as a function of target L/D) form complex shapes in space. Figures 3.20 through 3.23, which are reproduced from NUREG/CR-2913, illustrate axial and radial pressure distributions of an expanding jet representative of PWR and BWR blowdown conditions. Figure 3.24 is a comparison of Sandia calculations (taken from NUREG-2913) with HDR experiment V21.1.

The significant findings to be derived from the calculations contained in NUREG/CR-2913 are as follows:

(1) Target pressure loadings increase asymptotically at L/D's less than 3.0 to break exit pressures. At L/D's less than 3, survivability of insulation materials is highly unlikely.

(2) At L/D's from 5 to 7, the centerline stagnation pressure becomes essentially constant at approximately 2 ± 1 bars.

(3) The multidimension pressure field loads the target over a large region (see Figures 3.22 and 3.23); this region may be approximated by a 90° jet cone expansion model. A hemispherical expansion model could be another



P_o = Stagnation pressure at break
ΔT_o = Subcooling of stagnation temperature at break
X_o = Stagnation quality at break

Figure 3.19 Schematic of jet impinging on target

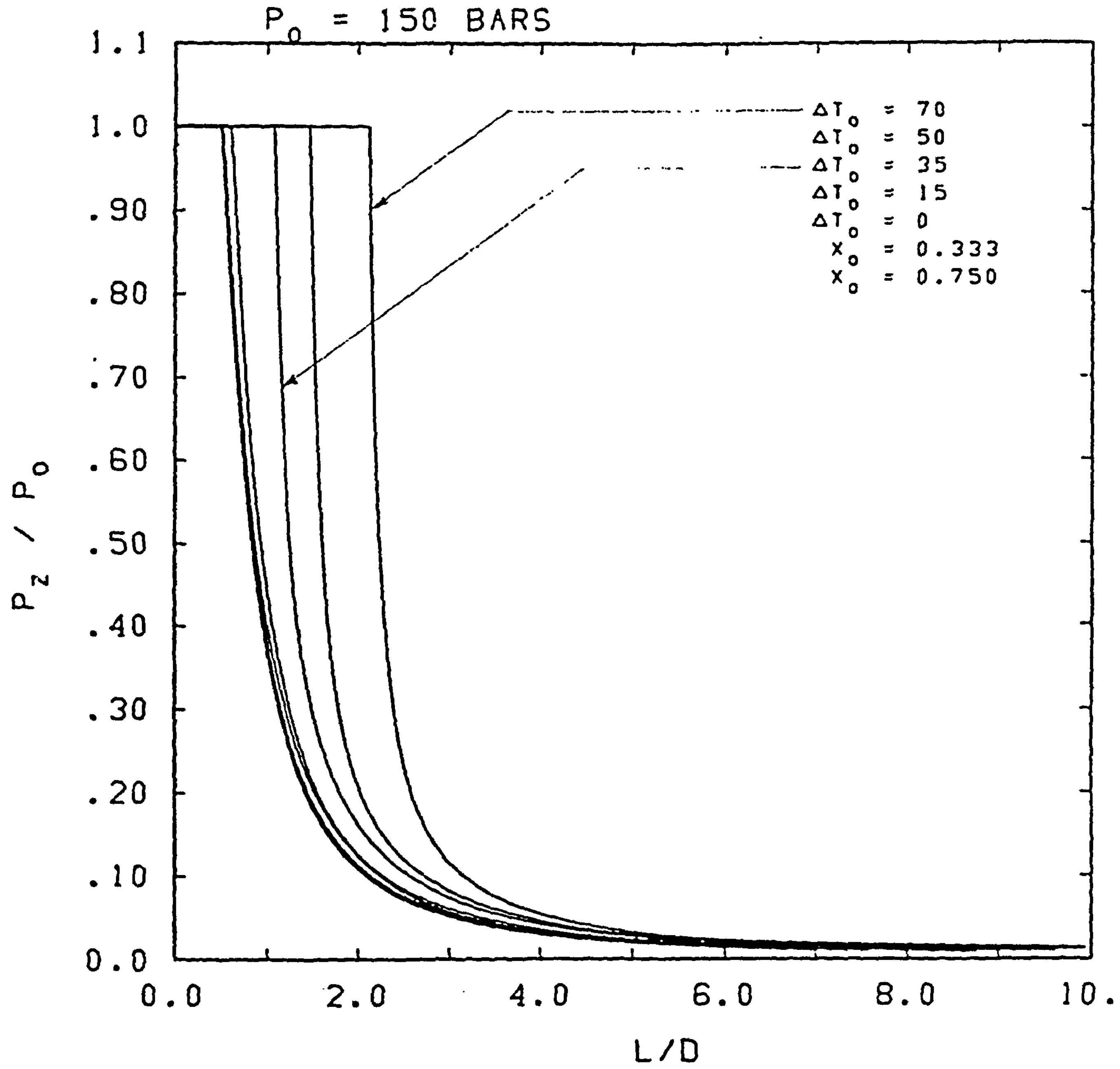


Figure 3.20 Centerline target pressure as a function of axial target position (L/D) for break stagnation conditions of 150 bars and various subcoolings and qualities. L is the target position, D is the pipe diameter, P_z is the centerline pressure, and P_0 is the stagnation pressure at the break.

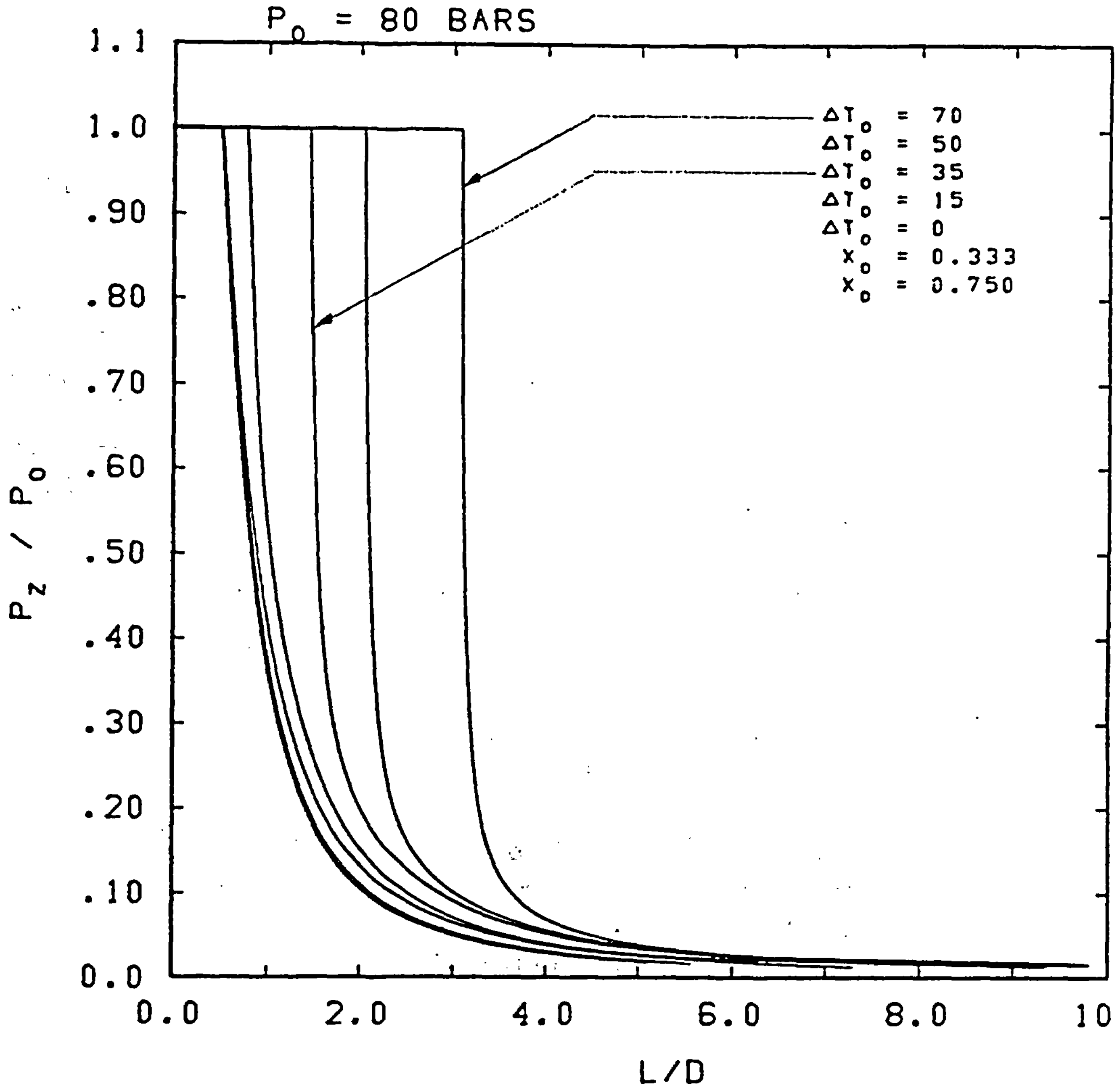


Figure 3.21 Centerline target pressure as a function of axial position (L/D) for break stagnation conditions of 80 bars and various subcoolings and qualities. L is the target position, D is the pipe diameter, P_z is the centerline pressure, and P_o is the stagnation pressure at the break.

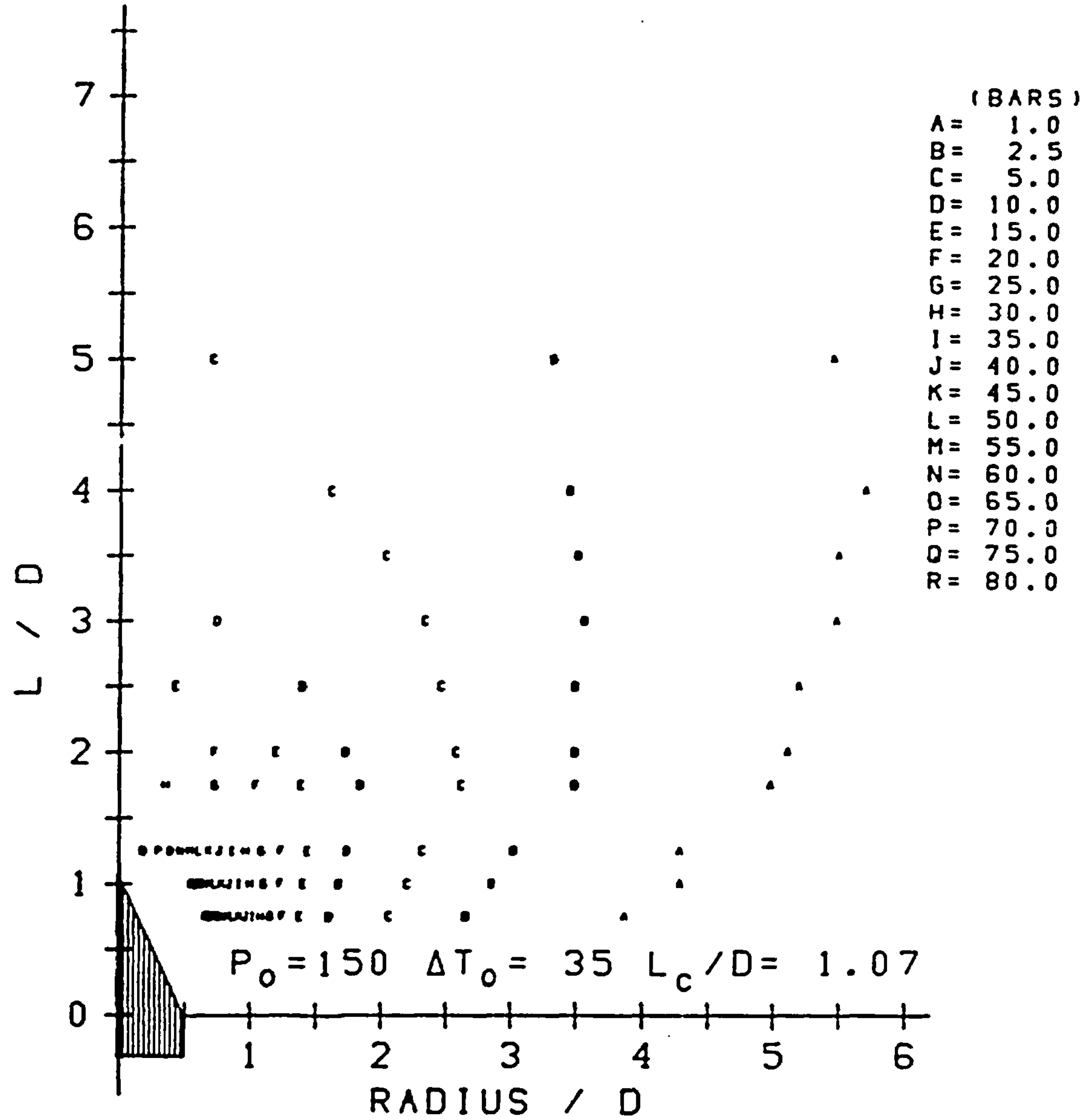


Figure 3.22 Composite target pressure contours as a function of target length/jet diameter (L/D) and target radius/jet diameter (RADIUS/D) for stagnation conditions of P_o = 150 bars and 35° of subcooling. Smooth lines connecting like alphabetic letters form an approximate pressure contour corresponding, in bars, to the pressure versus alphabetic letter key. This countour is approximate and is only informational.

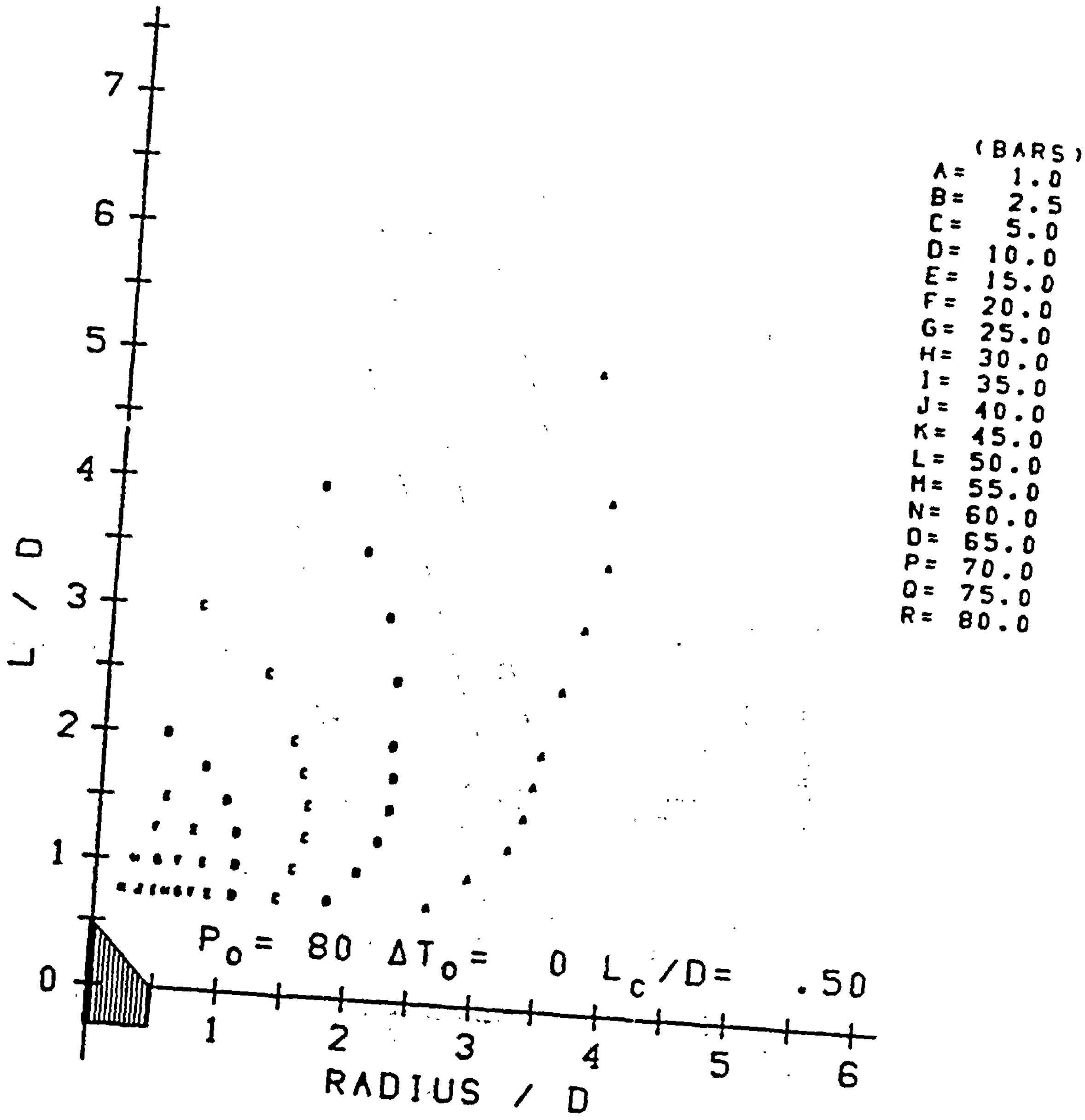


Figure 3.23 Composite target pressure contours as a function of target length/jet diameter (L/D) and target radius/jet diameter (RADIUS/D) for stagnation conditions of P_o = 80 bars and saturated liquid. Smooth lines connecting like alphabetic letters form an approximate pressure contour corresponding, in bars, to the pressure versus alphabetic letter key. This contour is approximate and is only informational.

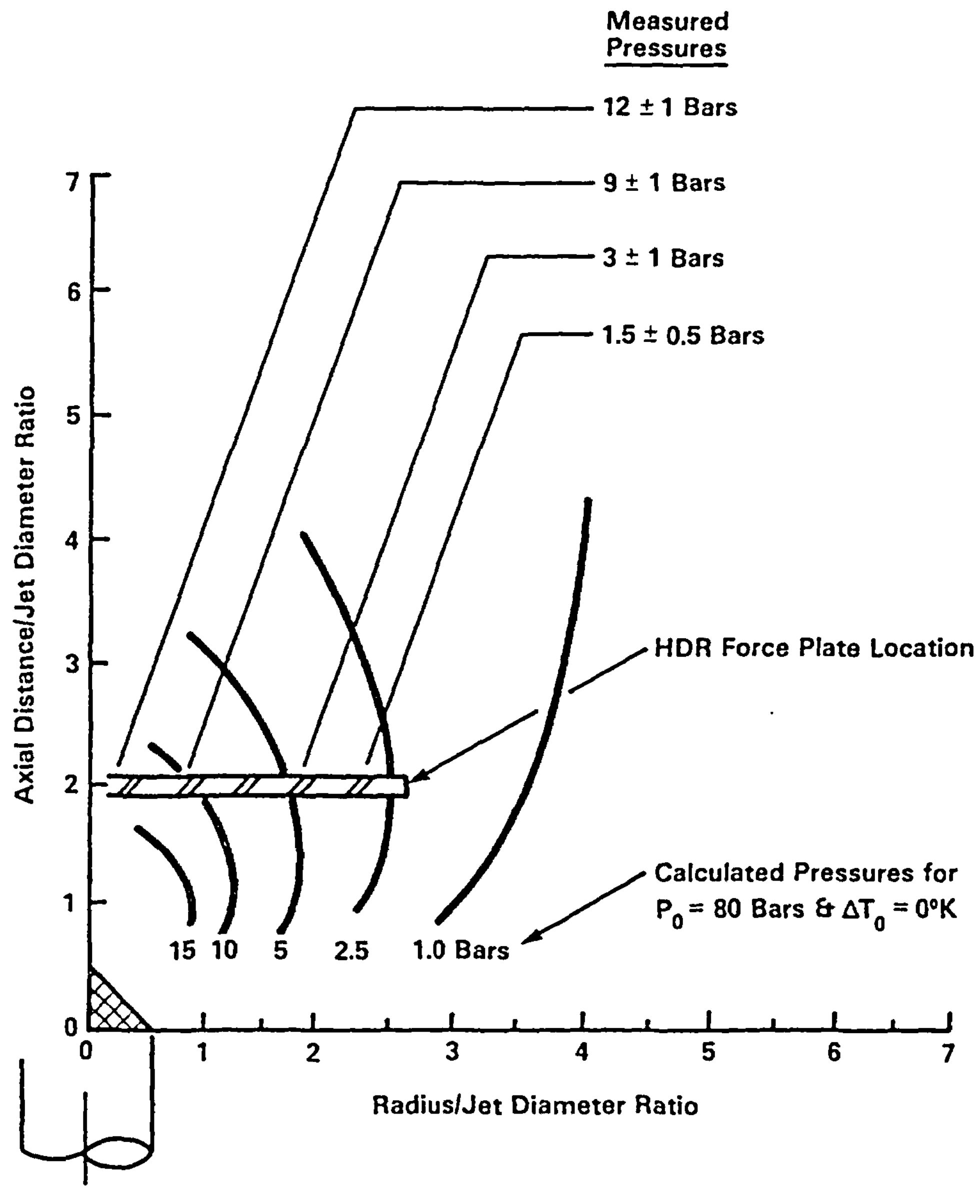


Figure 3.24 Comparison of calculated target pressures with HDR experiment V21.1

approximation for this expanding pressure field. These two-dimensional calculations do not support the use of the Moody jet model (a narrow jet cone) for target close to the break locations.

The two-phase jet modelling results and the levels of insulation damage evidenced by the HDR experiments lead to the development of a three-region jet-debris-generation model, which is shown in Figure 3.25. In Region I ($\leq$ 3 L/D's from the break) extremely high levels of destruction would occur due to the very high break jet pressures (see also Figures 3.20 and 3.21) and total destruction can be assumed to occur. Region II (3 < L/D's < 7 from the break) is a zone where high levels of damage (or destruction) are possible; but with the recognition that the types of insulation employed (reflective metallic, fibrous, foam-glass, etc.), methods of attachment, whether the materials are encapsulated, etc. are factors that should be considered in estimating the types and volumes of debris generated in Region II. Region III (L/D's $\geq$ 7's from the break) is a zone where destruction (or damage) is likely to be dislodgement of insulation in the as-fabricated mode, or as modules. Beyond 7 L/D's, break jet pressures decay to 1 to 2 bars. It should also be noted that the superimposed pressure field on Figure 3.25 is representative of a PWR primary coolant system break. BWR jet expansion fields decay more rapidly (see pressures in Figure 3.21 versus those in Figure 3.20).

Despite the calculational simplification afforded by a three-region model, determination of the types and quantities of insulation debris will always be material (or type) dependent. Figure 3.26 has been constructed to illustrate the possible variation of debris types as a function of distance from the break jet and the relative quantities of different types of possible debris. A quantified debris distribution model would require extensive experiments designed to develop such data; these do not exist. On the other hand, results from HDR experiments (see Appendices C,E, and F) do provide insights regarding debris generation and were used to construct Figure 3.26.

First of all, the assumption of severe damage or total fragmentation within 3 L/D's is supported by experiments and is applicable to both RMI assemblies and fibrous insulation assemblies. However, the hypothesis of "exploded" RMI assemblies releasing free, or undamaged, interior foils (which can transport at very low velocities) is not supported by the experimental evidence reported in Appendix E.

Pursuing those potential levels of damage expected in Region II (see Figure 3.25), it appears that the RMI debris could consist of damaged inner foils and damaged assemblies or components that were the result of further LOCA damage. Experimental data available for fibrous insulations indicate that shredding and damage can extend into Region II, with such damage decreasing with distance from the jet. However, if the "inner core" of fibrous insulation is exposed to the break jet (as would occur if the cover blanket were breached), blowdown transport of this material would be expected to extend for distances much greater than 7 L/D's. Jacketing of fibrous insulations does appear to provide some protection, provided such jackets are not blown away by the initial blowdown jet forces, as demonstrated by HDR blowdown tests (Appendix F) where unjacketed fibrous insulations or insulations covered by a metal mesh are nearly totally destroyed within 3L/D's, with some damaged and partially destroyed segments within 7 L/D's. But the same blankets enclosed in stainless steel jacket withstand the blast better (see Appendix F). Figure 3.26 illustrates examples

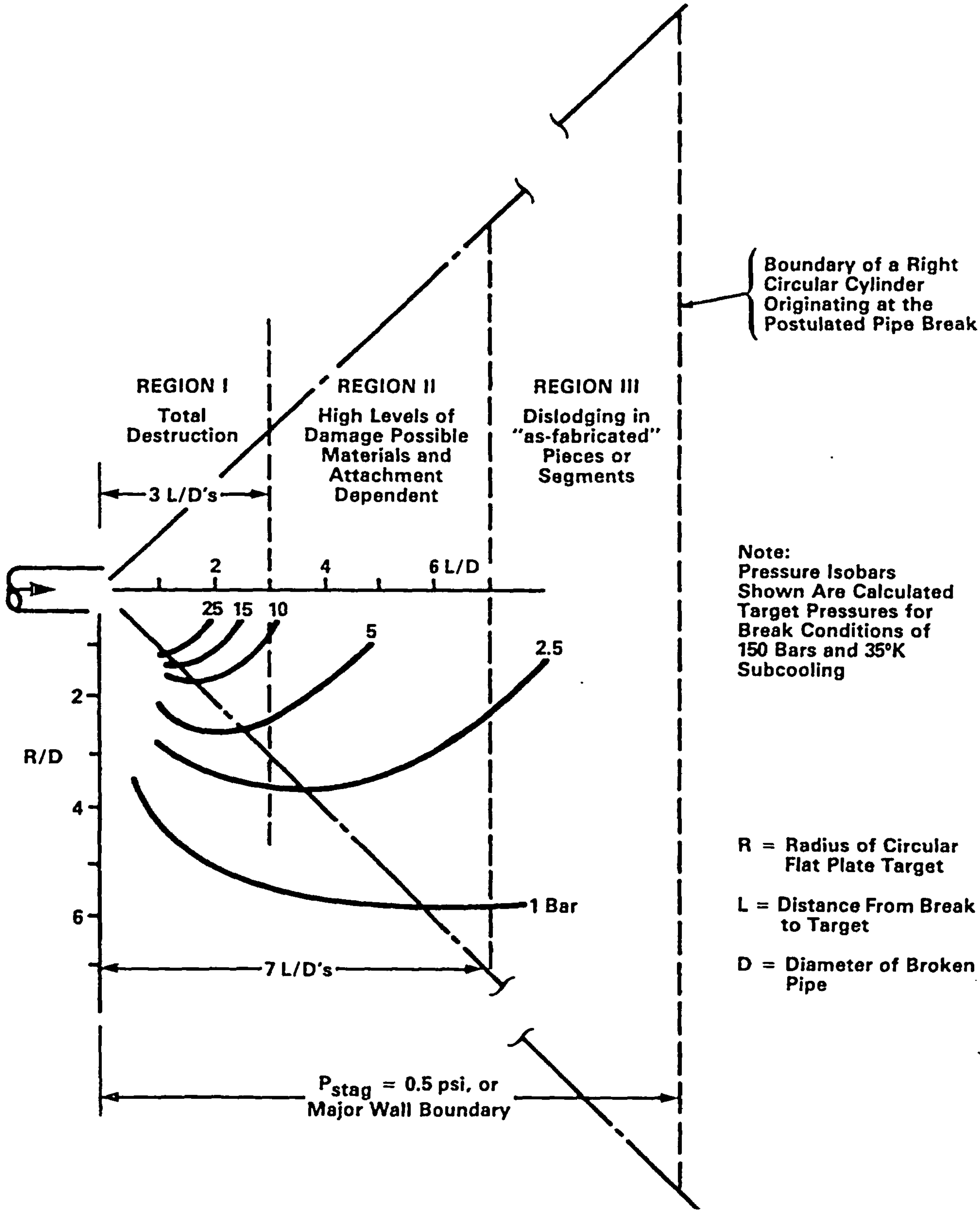


Figure 3.25 Multiple region insulation debris generation model

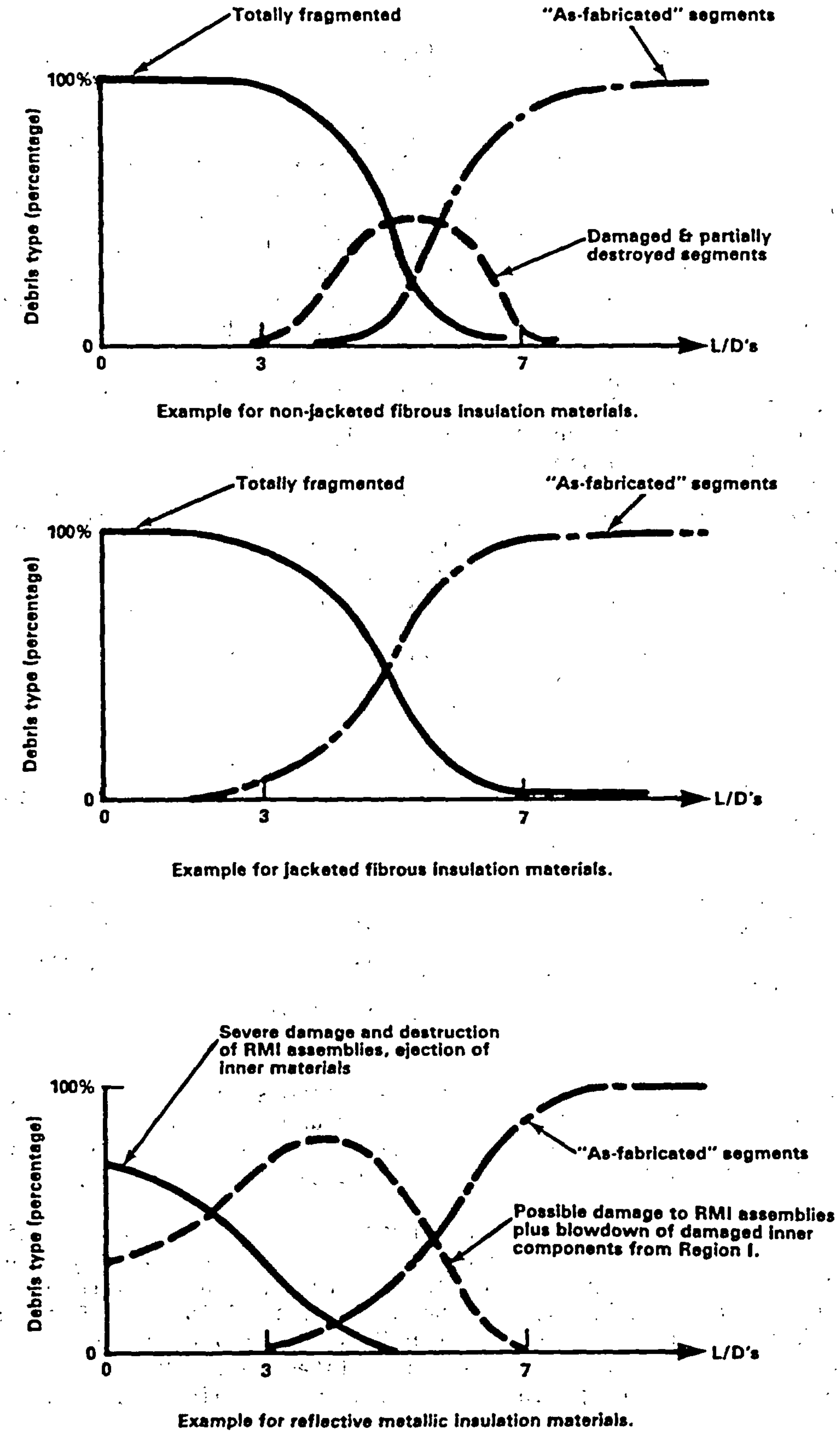


Figure 3.26 Possible variation of debris types and relative quantities in regions of the three-region jet model (see Figure 3.25)

of debris generation for RMI, fibrous jacketed insulation materials and fibrous non-jacketed insulation materials.

Thus, debris generation in Region II can be very complex, and generic conclusions should not be drawn nor extrapolated to cover different materials or conditions. The specific materials and products used as insulation should be carefully reviewed in light of the data base available as results of tests (see Appendices C, E, and F). The assessment of the volume of debris generation, transport, and screen blockage should be made on a plant-specific basis. If such a determination shows that estimated blockage head losses do not exceed the NPSH margin, a conservative safety assessment has been made.

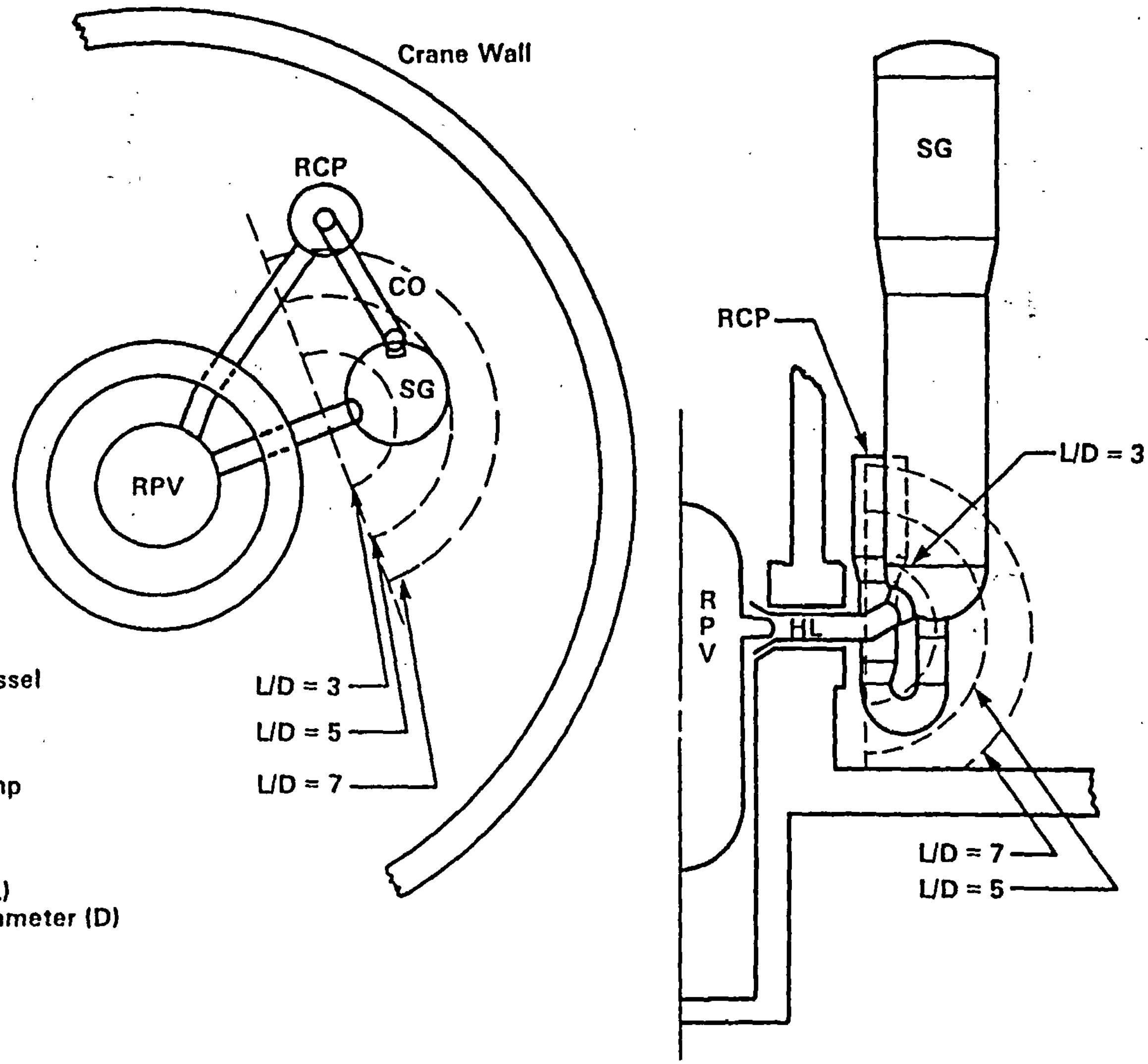
The size of the third volume (Region III) was established using the Moody jet analysis as modified and discussed in NUREG/CR-2791. It begins at L/D = 7 and extends to an axial position in the jet where the jet thrust (as calculated by the Moody jet expansion model) would be equal to 0.5 psig when calculated for a flat axisymmetric target. The Moody-type jet expansion model was selected for establishing the outer boundary of Region III because it always results in a larger L/D value for the boundary than the two-phase jet analysis in NUREG/CR-2913. This ensures that the effects of debris modeling uncertainties are mitigated by a conservative outer boundary selection.

Break location(s) and insulation(s) targeted by the break jet are the key factors in estimating debris generation. This is illustrated in Figure 3.27 for a typical PWR where the influence of an expanding jet is shown. A break in the primary coolant system piping will target large quantities of insulations located in the lower portions of the steam generators. Although break locations are identified in SRP Section 3.6.2, the reviewer (or analyst) should determine which breaks are most significant and estimate the extent (or volume) of insulation debris generation.

Such a detailed break evaluation was carried out for a reference PWR (Salem Unit 1) and is reported in NUREG/CR-3394. Although this study was primarily directed at estimating the probability of sump blockage, the analyses revealed that breaks in large diameter piping ($\geq$ 10-inch diameter) were the dominant contributors to debris generation (see Table 3.5). This finding can be used by the analyst in scoping the extent of LOCA debris generation.

Table 3.6, which illustrates typical volumes of insulation two typcial PWRs employed on the primary coolant system and related components, provides an insight regarding volumes of insulations employed and their distribution on the PWR primary coolant system and components.

Although a generic conclusion cannot be drawn from these studies because of plant variabilities, the results do indicate that PWR debris assessments should concentrate on the primary coolant system insulation within the crane wall region and for pipe breaks of pipe diameter $\geq$ 10 inches. Because such a detailed break study has not been done for BWRs, the reviewer should consider debris generation as occurring for breaks postulated in the BWR feedwater and recirculation piping and for postulated breaks in BWR main steam lines.



Where RPV — Reactor Pressure Vessel

SG — Steam Generator

RCP — Reactor Coolant Pump

HL — Hot Leg

L/D — Distance to Target (L) Divided by Break Diameter (D)

CO — Crossover

Figure 3.27 Zones of influence for debris generation

Table 3.5 Maximum LOCA-generated insulation debris summarized by break size

Pipe diameter (inches)	Total fibrous debris (ft^3)	Total all types (ft^3)
2	1	1
6	2	22
8	2	3
10	4	31
14	227	227
16	270	270
32	144	295
34	315	726
36	118	408

Notes:

(1) These values correspond to break locations in the primary system within the crane wall and represent the largest quantity of debris generated by a single break of a given pipe diameter.

(2) The insulation types and distribution within containment are those used in Salem 1. All insulation within 7 L/D's of a break location is assumed to be destroyed and released as fragmented debris.

(3) For reference see NUREG/CR-3394.

Table 3.6 Typical volumes of primary system insulation employed[1]

Component	Salem Volume (ft^3)	Salem Type of insulation	Maine Yankee Volume (ft^3)	Maine Yankee Type of insulation
Steam generator	1284	reflective metallic/ fibrous	1144	calcium silicate/ fibrous
Hot leg	160	reflective metallic	149	fibrous
Cold leg	144	reflective metallic	156	fibrous
Crossover	60	reflective metallic	279	fibrous
Pressurizer surge line	129	reflective metallic	302	calcium silicate/ fibrous
RCP	570	reflective metallic	149	calcium silicate/ fibrous
Bypass	N/A	N/A	88	fibrous
TOTAL[2]	2507		2324	
SUBTOTAL[3] (excluding RMI and calcium silicate)	1284 (4402 ft^2)		1527 (5234 ft^2)	

[1]This table is based on information provided by the operators in 1981. Plant changes since 1981 have made the data less accurate for these two specific reactors. However, as representative data for reactors in general, the table is still valid.

[2]This volume includes all of the insulation that could be hit by a water jet from a LOCA pipe break (in pipes $\geq$10 inches diameters). If the volume were restricted to only insulation within 7 L/D's of a break, it might be significantly smaller.

[3]To be conservative, Salem's steam generator is assumed to be covered entirely with fibrous insulation. In all, 50% of the insulation of Maine Yankee's steam generator, pressurizer, and reactor coolant pump is assumed to be fibrous.

3.3.5 Transport and Screen Blockage Potential for Reflective Metallic Insulation Materials

A limited amount of testing has been conducted with RMI components to gain an insight into the transport and possible screen blockage configurations. The results are reported in NUREG/CR-3616. The thrust of these tests was to determine velocity levels that would transport various components, particularly thin foils that are used internally. As might be expected, intact units were not transported until flow velocities exceeded 1 ft/sec. On the other hand, very thin, stainless steel foil (0.0025-inch thick) materials were transported at low velocities (0.2 to 0.5 ft/sec) if such foils were in an uncrumpled and intact state. Table 3.7 summarizes experimental findings. In these tests, as the foil material became more rigid (increased thickness), the foil-type debris was transported by sliding along the floor, rather than in a tumbling mode, and higher velocities were required to flip the material into a vertical orientation against the debris screen.

Of more significance are the screen blockage patterns observed during these transport tests. Intact shells (or halves) can flip against a debris screen if velocities exceed 1 ft/sec (see Figure 3.28). On the other hand, free thin foil sheets tend to crumple, resulting in the blockage configurations shown in Figures 3.29 and 3.30. Multiple foil sheets can form a blockage pattern such as shown in Figure 3.31. Generally blockages occurred at the lower portion of the debris screen. Although enough sheet material to totally block the screen was introduced into the transport flume, total blockage did not occur (see Figure 3.29). The very thin foil material (when in large sheets) is transported with a tumbling, lifting-type motion; however, lack of structural rigidity results in transport deformations, as shown in Figure 3.29. Another significant finding was that none of the foil samples tested became water borne. This is particularly important in BWRs because the RHR suction intakes are generally 6 to 8 feet above the suppression pool floor.

Thus transport of metallic insulation debris at fairly low velocities cannot be discounted and plant-specific assessments should be made for those plants employing this type of insulation.

The transport and blockage findings discussed above can be used to estimate levels of potential blockage. Of equal importance is the severity of LOCA induced damage (see Section 3.3.4) and types of RMI debris generated (see Appendix E). The HDR tests discussed in Appendix E do not support a debris generation model consisting of free, undamaged interior foil materials being available for transport.

3.3.6 Buoyancy, Transport, and Screen Blockage Characteristics of Mass-Type Insulations

The buoyancy and transport characteristics of fibrous insulation materials are important because long-term screen blockage is a function of whether, and how, such debris material would be transported. Information regarding transport of shredded mineral wool insulation is provided in the Finnish tests conducted in the late 1970s (Imatran Voima Oy, "Model tests of the Loviisa Emergency Core Cooling System and Model Tests of Containment Sumps of the Emergency Core Cooling System"). These tests showed that shredded mineral wool would be transported

Table 3.7 Transport and blockage characteristics of reflective insulation materials (see also NUREG/CR-3616)

Sample description	Velocity to initiate motion (ft/sec)	Velocity to transport to screen (ft/sec)	Comments
Undamaged half jacket normal to flow			
Concave side up	1.0	1.0	Either flipped on screen (see Figure 3.28) or got stuck partially flipped
Concave side down	above 2.2		Never moved.
Outside cover (0.037 inches thick, diameter = 19 inches)			
Concave side up	0.7	0.8	Same blockage mode as undamaged half jackets.
Concave side down	above 1.8		
Inside cover (0.015 inches thick, diameter = 13 inches)			
Concave side up	0.7	0.8	With both initial positions, covers flipped against the screen on arrival and got flattened against it by the flow force.
Concave side down	1.1	1.6	
End covers	above 2		Never moved.
Single sheet inner foil (0.0025 inches thick, 36 x 25 inches) uncrumpled, with and without separating crimp	0.35	0.5	Moves in folding and tumbling mode. Flips vertically against screen when it reaches it (Figure 3.29). May be folded on screen (not cover full sheet area). Never covered screen higher than maximum sheet dimension, even for flow velocity of 2 ft/sec and water depth of 60 inches.
Single sheet inner foil (0.0025 inches thick, 36 x 25 inches)	0.20	0.25	Moves in folding and tumbling mode. Flips against screen when it reaches it; flattened on screen by current.

Table 3.7 (continued)

Sample description	Velocity to initiate motion (ft/sec)	Velocity transport to screen (ft/sec)	Comments
Four sheets inner foil (0.0025 inch thick, 36 x 25 inches), two crumpled, two uncrumpled	0.25	0.4 to 1.8	When numerous foil sheets are used they tend to jam up in piles that may need high velocity to unjam. Significant overlapping on screen.
Single cut-up sheet inner foil (0.0025 inch thick, 24 x 21 inches)			
Uncrumpled	0.20	0.25	Folding and tumbling transport mode. Flip vertically on screen upon arrival, sometimes folded.
Crumpled	0.20	0.25	Flip vertically on screen upon arrival, sometimes folded (see Figure 3.30).
Several cut-up sheets inner foil (0.0025 inch thick, (8 inches x 8 inches)			
Uncrumpled	0.5	1.2	Pieces not folded by flow as larger ones. Sliding transport mode. One piece reached screen at 0.5 ft/sec; all flipped vertically on arrival to screen (see Figure 3.31).
Crumpled	0.5	1.2	One piece reached screen at 0.9 ft/sec; all flipped vertically on arrival to screen.
Several cut-up sheets inner foil (0.0025 inch, 3 x 3 inches)			
Uncrumpled	0.8	2.0	Pieces not folded by flow as larger ones. Sliding transport mode.
Crumpled	0.6	1.0	Pieces flip vertically on screen unless a corner gets trapped under screen bottom, in which case the piece stays flat on bottom.

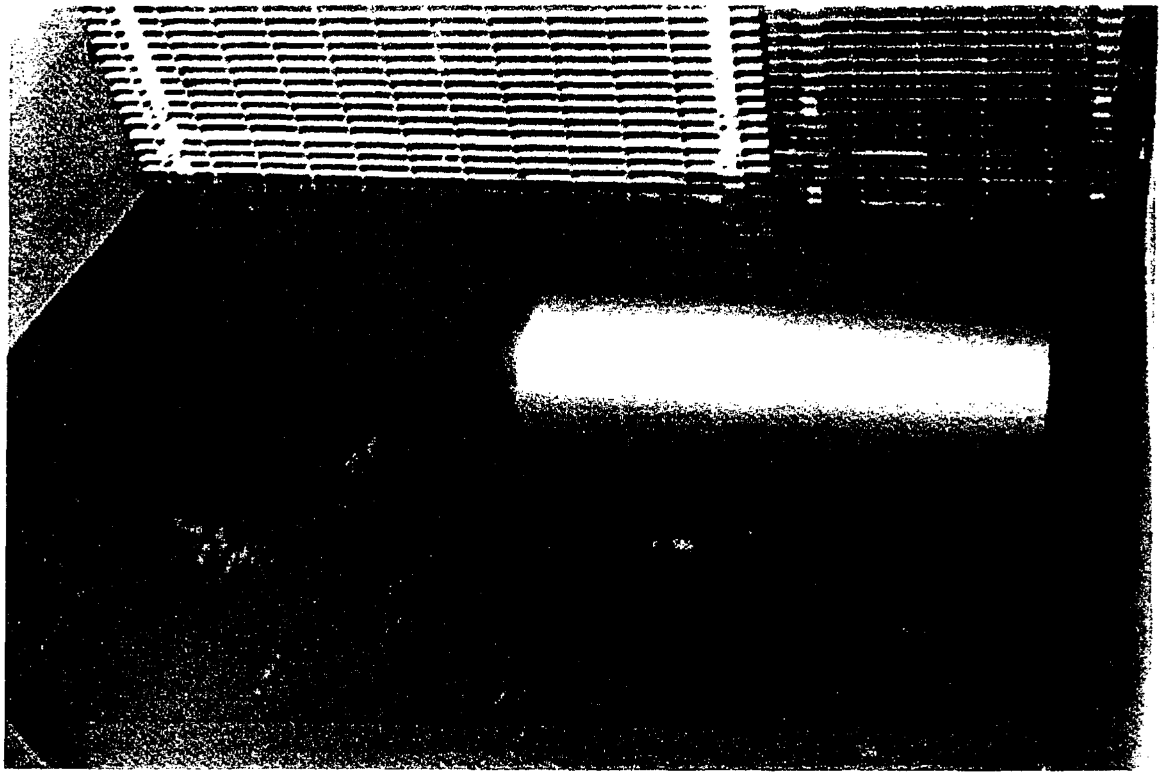


Figure 3.28 A half segment flipped onto screen

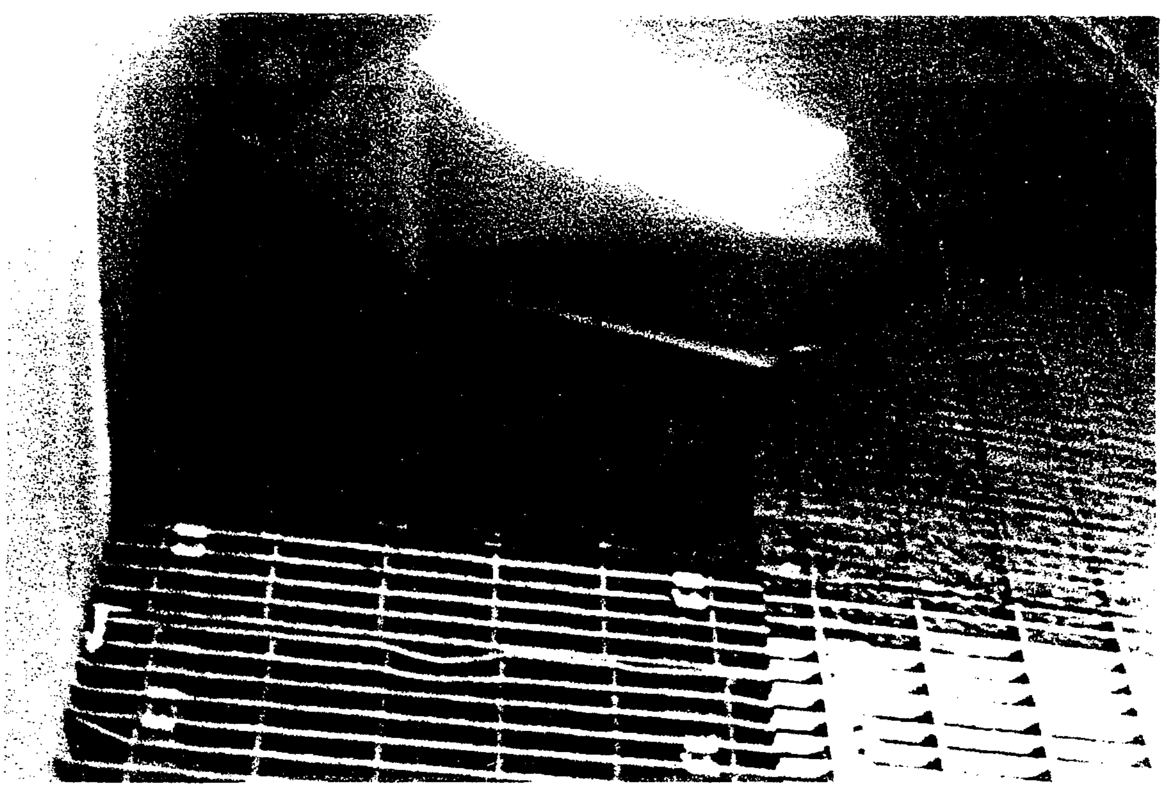


Figure 3.29 Uncrumpled foil sheet flipped vertically on screen
(flow velocity = 0.5 ft/sec)

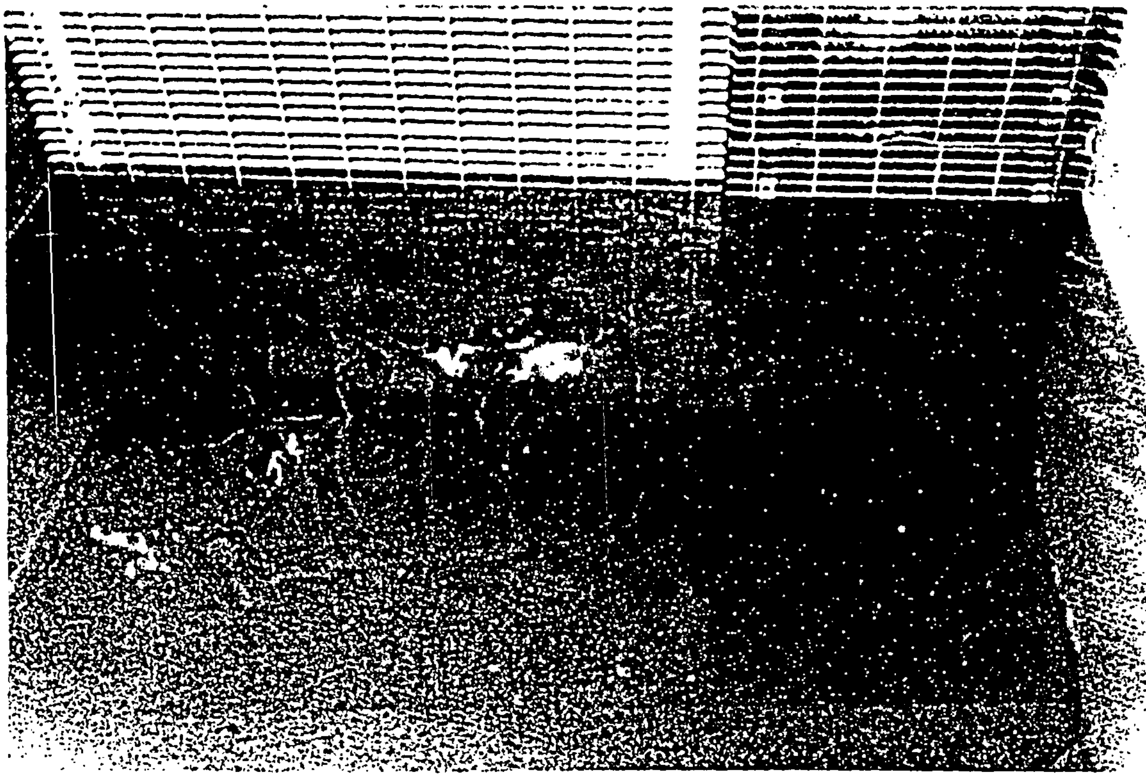


Figure 3.30 Crumpled foil sheet against screen (flow velocity = 0.3 ft/sec)

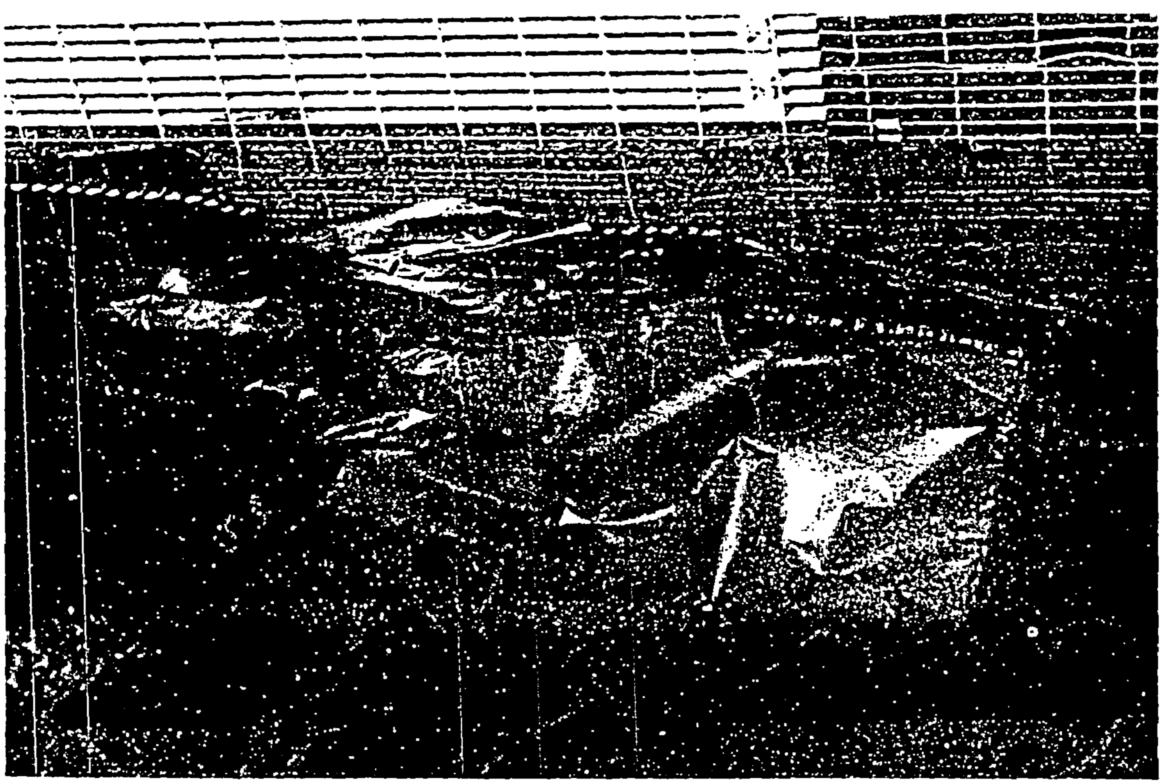


Figure 3.31 Several foil sheets on screen (flow velocity = 0.7 ft/sec)

at low velocities and build up uniformly on a debris screen, and thus could result in high head losses.

Similar tests were conducted under NRC sponsorship at ARL and are reported in NUREG/CR-2982, Revision 1. The results of those tests are summarized in the following paragraphs.

Buoyancy, transport, and head loss experiments were conducted with three types of as-fabricated insulation panels and with fragmented fibrous insulations. The three types of as-fabricated insulation panels were

Type 1: 4-inch mineral wool or refractory mineral fiber core mineral (6 pound density), covered with Uniroyal 6555 asbestos cloth coated with 1/2-mil Mylar.

Type 2: 4-inch Burlglass 1200, or 4 layers of 1-inch-thick Filomat D (fiberglass) core material, an inner covering of knitted stainless steel mesh, and an outer covering of Alpha Maritex silicone aluminum cloth, product 2619.

Type 3: Same insulation core materials as Type 2, but with an inner and outer covering of 18-ounce Alpha Maritex cloth, product 7371.

The fiberglass core material in Types 2 and 3 is a high density fiberglass ($\sim$10 lb/ft^3). Various types of fiberglass insulation are employed in nuclear plants, and, as evidenced by the data reported (Durgin and Noreika, 1983) for the Owens Corning Fiberglass product NUKON™, they can exhibit different characteristics. Therefore, evaluations should be based on the actual material(s) utilized in a given plant.

The buoyancy tests revealed

(1) In general, the time needed for both mineral wool and fiberglass insulation to sink was less at higher water temperatures.

(2) Mineral wool (Type 1) does not readily absorb water and can remain afloat for several days.

(3) Fiberglass insulation (Types 2 and 3) readily absorbs water, particularly hot water, and sinks rapidly (from 20 seconds to 30 seconds in 120°F water).

(4) Undamaged fiberglass pillows of Type 3 (and possibly also of Type 2) can trap air inside their covers and remain afloat for several days.

(5) Based on the observed sinking rates, it may be concluded that mineral wool pillows and some undamaged fiberglass pillows (those that trap air inside their cover) will remain afloat after activation of the containment recirculation system (approximately 20 minutes after the beginning of LOCA). Those floating pillows will move at any water velocity and can be transported to the sump before activation of the recirculation system.

The transportation tests revealed

(1) Water velocities needed to initiate the motion of insulation are on the order of 0.2 ft/sec for individual shreds, 0.5 to 0.7 ft/sec for individual small pieces (up to 4 inches on the side), and 0.9 to 1.5 ft/sec for individual large pieces (up to 2 feet on the side).

(2) For whole sunken pillows to flip vertically onto the screen, flow velocities of 1.1 ft/sec for Type 1 (mineral wool) and 1.6 to 2.4 ft/sec for Types 2 and 3 (fiberglass) are required.

(3) Whole floating pillows require a water velocity in excess of 2.3 ft/sec to flip vertically against the screen.

(4) Insulation pillows broken up in finite size sunken fragments tend to congregate near the bottom of the screen if there is no turbulence generator, and, depending on the water depth, unblocked space can remain near the top of the screen. With turbulence generators (vertical posts 2 feet upstream of the screen), some insulation fragments are lifted from the bottom and collect higher on the screen.

(5) Once insulation shreds are in motion, they tend to become suspended in the water column and collect over the entire screen area.

The head loss tests revealed

(1) The measured head loss across a vertical screen in a flume as a result of blockage by insulation released upstream varies from 7 to 10 times the approach velocity head, $U^2/2g$, for whole sunken pillows; from 13 to 36 times the approach velocity head as that for opened or broken up pillows; and more than 240 times the approach velocity head for shredded pillows. These results are for an equivalent volume for 50% screen blockage with the undamaged pillows.

    Opened pillows with separated, fragmented, or shredded insulation layers had enough area to block the entire screen. However, the screen was entirely (but not uniformly) covered only in the test with the shredded insulation. In the other tests, open space remained on the screen.

    For these conditions, the maximum measured head loss of 240 times the approach velocity head (for shredded pillows) would result in screen head losses of 0.15 foot to 0.60 foot for approach velocities of 0.2 ft/sec to 0.4 ft/sec.

(2) Measured head losses through beds of accumulated fragments or shreds of mineral wool or fiberglass insulation varied nonlinearly with approach velocity and bed thickness.

    For mineral wool fragments, the larger head losses were observed for the tests of larger fragments (3 x 2 to 4 x 1/8 inch). For an original insulation thickness of 1 inch, the maximum head loss was 0.4 foot at 0.2 ft/sec and 1.4 feet at 0.4 ft/sec.

For fiberglass insulation fragments and shreds, the larger head losses were observed for the shreds. For an original (as-fabricated) insulation thickness of 1 inch, the maximum head loss was 1.2 feet at 0.2 ft/sec and 6 feet at 0.4 ft/sec.

(3) The head loss through as-fabricated insulation material is higher, by a factor of up to 10, than that for accumulated fragments. For example, with water at 105° to 120°F and with an approach velocity of 0.2 ft/sec, the head loss through 2 inches of undisturbed mineral wool is about 3.5 feet, and the head loss through 1 inch of undisturbed fiberglass is about 20 feet. These head losses are for insulation samples sealed to the walls to prevent leakage. The head loss would be less if leakage occurred around the sample.

(4) In addition to the variables of insulation thickness and approach flow velocity, the actual head loss that may be expected across a sump screen depends critically on how the screen is blocked. If some unblocked screen area remains, or if water can flow between pieces of insulation, the head loss would be small; if the entire screen area is uniformly covered with mats of undisturbed insulation or accumulated fibers, the head loss can be many feet.

(5) Best-fit expressions for the head loss through shredded fibrous insulation, were derived as follows:

for mineral wool (Type 1): $\Delta H = 123U^{1.51}t^{1.36}$

for fiberglass (Types 2 and 3): $\Delta H = 1653U^{1.84}t^{1.54}$

where

U is the screen approach velocity (ft/sec)
t is the original (as fabricated) insulation debris thickness (ft)
ΔH is the head loss (ft H_2O)

Table 3.8 summarizes these transport and head loss characteristics.

The strong dependence on material characteristics cannot be overemphasized. Owens Corning Fiberglass conducted similar tests with fiberglass utilized in NUKON™ (a low density fiberglass, 2 lb/ft³). The transport characteristics were similar to those reported in NUREG/CR-2982, Revision 1, in that the transport of fragments occurred in the 0.2 to 0.3 ft/sec range. However, the screen blockage head loss correlation for fragments (experimentally derived) was

$$\Delta H = 68.3U^{1.79}t^{1.07}$$

This equation is significantly different from the two previous equations. These results are reported in ARL Report No. 110-83/M489F (Brocard, 1983). Thus, the reviewer should base evaluations on the particular type of insulation material(s) employed in a given plant application.

In summary, the following should be considered in determining fibrous insulation blockage effects:

Table 3.8 Summary of transport and screen blockage characteristics of high density fiberglass

Condition	Pillow Type	V_i (ft/sec)	V_s (ft/sec)	V_v (ft/sec)	ΔH (ft)	$\frac{\Delta H}{\frac{V^2}{2g}}$ *	Comments
Floating whole pillows	1	N/A	N/A	> 2.3			Never flipped
	2	N/A	N/A	N/A			Sunk while against screen; flipped vertical
	3	N/A	N/A	N/A			Sunk while against screen; flipped vertical
Sunken whole pillows	1	1.1	1.1	1.1	0.13		Only one pillow tested
		0.9	1.1	1.1	0.07		Only one pillow tested folded in half on screen
	2	1.2	1.8	2.0	0.44	7.1	
		1.4	1.6	2.4			
	3	1.5	1.7	2.0	0.60	9.4	
		1.1	1.6	1.6	0.33	8.3	Pillows on screens overlap by 2 inches
Sunken pillows with covers removed but included and separated insulation layers	1	1.1	1.1	1.1	0.67	36.0	
		0.9	1.5		0.96	27.5	Not all pieces vertical
	2 or 3	1.1	1.6				
		0.9	1.2	1.2	0.71	32.0	
Sunken pillows with covers and insulation layers in 5 pieces (see Figure 2.6)	1	1.0	1.9		1.4	25.0	Not all pieces vertical
		1.1	2.0		1.6	26.0	
	2 or 3	1.0	1.4	1.6	0.54	14.0	Significant overlap of pieces on screen

*For details in the size and amount of the insulation materials utilized in these tests see NUREG/CR-2982, Revision 1.

Table 3.8 continued

Condition	Pillow Type	V_i (ft/sec)	V_s (ft/sec)	V_v (ft/sec)	ΔH (ft)	$\frac{\Delta H}{\frac{V^2}{2g}}$	Comments
Sunken pillows in 4" x 4" x 1" fragments. Covers not included.	1	0.4	1.4	1.6	1.35	34.0	Fragments collect on bottom 1 ft of screen
		0.6	1.3	1.4	2.45	80.0	With turbulence generators. Fragments collect on bottom 3 ft of screen
	2 or 3	1.0	> 1.6				Not all pieces reached the screen. Collected near screen bottom, Figure 4.6
		1.0	> 1.6		0.72	18.1	With turbulence generators. Only about half the pieces on screen. Some pieces at mid-height.
Sunken pillows in shreds. Covers not included.	2 or 3	0.4	> 1.3	N/A	3.7 for 1.0 fps	240	Not all pieces on screen. Screen entirely but not uniformly covered.
Sunken single fragments							Tests conducted in 1 ft wide flume with 7 inch water depth
4"x4"x1"	1	0.6					
	2 or 3	0.7					
4"x1"x1"	1	0.3					
	2 or 3	0.5					
Shreds	1	0.3					
	2 or 3	0.2					

NOTATIONS: V_i = velocity needed to initiate motion of at least one piece of insulation (not including covers when separated from pillows)

V_s = velocity needed to bring all material on screen

V_v = velocity needed to flip all pieces vertically on screen

ΔH = head loss at V_v (or V_s if V_v not given)

(1) Recirculation velocities and break jet loads must be evaluated to determine whether they are high enough to transport debris to PWR sump screens or BWR suction strainers (See Appendix D). If not, blockage is not likely to occur.

(2) If the material can be shredded by the break jet, transport can occur at low velocities and a determination of screen head losses must be made, provided recirculation velocities are high enough to result in transport of the fragmented insulation debris.

3.3.7 Effects of Combined Blockage (Reflective Metallic and Mass Type Insulations)

Assessment of the effects of combined blockage, wherein both reflective metallic and mass type (fibrous) insulations are employed, is more difficult. As described above, both types of insulations can be transported at low velocities and block debris screens. Because metallic-type debris does not become water borne, blockages that can be ascribed to metal foils would occur at the lower (or bottom) portions of vertical screens. Fibrous insulation fragments can be transported at near-neutral buoyancy and do migrate to open flow passages. Therefore, a combined-effects model should be applied. Unfortunately, not enough experimental data are available to allow for development of a combined generic blockage model. Plant-specific evaluations should also consider the potential for this type of combined debris blockage.

3.4 Sump Hydraulic Performance

To investigate ECCS sump behavior under flow conditions that might occur during a LOCA, a test program was undertaken that covered a broad range of geometric and flow variables representative of PWR containment emergency sump designs. To avoid scaling uncertainties, a full-scale experimental facility at ARL was used. Scaling effects resulting from the use of reduced-scale hydraulic models were subsequently evaluated. The three broad areas of interest for ECCS sump design investigated were

(1) fundamental behavior of the sump with reasonably uniform approach flow conditions

(2) changes in the fundamental behavior of the sump as a result of potential accident conditions (screen blockage, break and drain flow, obstructions, nonuniform approach flow, etc.) that could cause degraded performance in the recirculation system

(3) design and operational items of special concern in ECCS sumps

Information from initial testing was used to plan or redirect later tests; hence, the tests were not necessarily conducted in the order listed below.

The tests performed may be divided into six series as follows:

(1) Factorial Tests

A fractional factorial matrix of tests was used to study primary sump flow and geometric variables. The factorial matrix provided a wide range of parameter variations and a method for effectively testing a large number of variables and determining their interdependencies.

(2) Secondary Geometric Variable Sensitivity Tests

The effects on sump performance of secondary geometric variables and design parameters of special concern in ECCS sumps were tested by holding all sump variables constant except one, for which several values were tested.

(3) Severe Flow Perturbations Tests

The behavior of selected sump geometries subjected to approach flow perturbations was investigated. Major flow disturbances considered were screen blockage (up to 75%), nonuniform approach velocity distribution, break-flow and drain-flow impingement, pump startup transients, and obstructions, as illustrated in Figures 3.32 and 3.33.

(4) Vortex Suppression Tests

The effectiveness of several types of vortex suppressors and inlet configurations was evaluated.

(5) Scale Tests

Scaling effects in geometrically scaled models using Froude number similitude and pipe velocity similitude were evaluated.

(6) BWR Suction Pipe Inlet Tests

The hydraulic performance of BWR suction pipe geometries typical of Mark I, II, and III RHR suction inlet designs was evaluated.

Data generated during the sump performance studies were analyzed using two approaches as follows:

(1) Functional Correlations of Dependent Variables

Correlations using response-surface regression analysis of nondimensional empirical data fitting were developed. Because of the extremely small values of the dependent variables and the complex time-varying nature of the three-dimensional flows in the sump, the use of functional correlations showed no consistent, or generally applicable, correlation between the dependent and independent variables. Thus, the hydraulic performance of a particular sump under given flow and submergence conditions could not be reliably predicted using this approach.

(2) Bounding Envelope Analysis

The broad data base that resulted from the sump studies made possible the use of envelope analysis for reliably predicting the expected upper bound for the hydraulic performance (void fraction, vortex type, swirl angle, and inlet loss coefficient) of any given sump whose flow and geometric features fall approximately within the ranges tested. The data boundary curves generated indicate

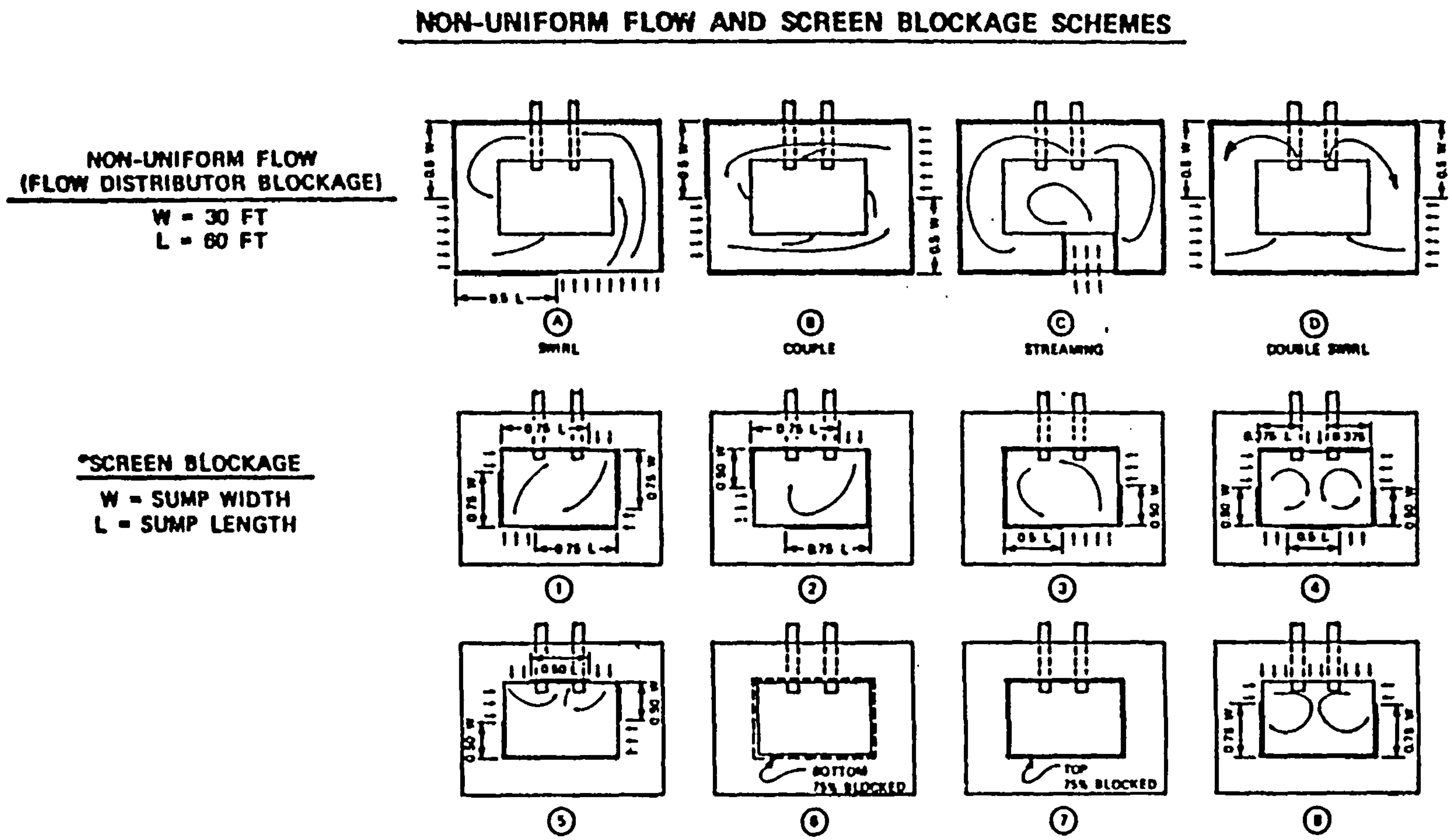


Figure 3.32 Approach flow perturbation and screen blockage schemes

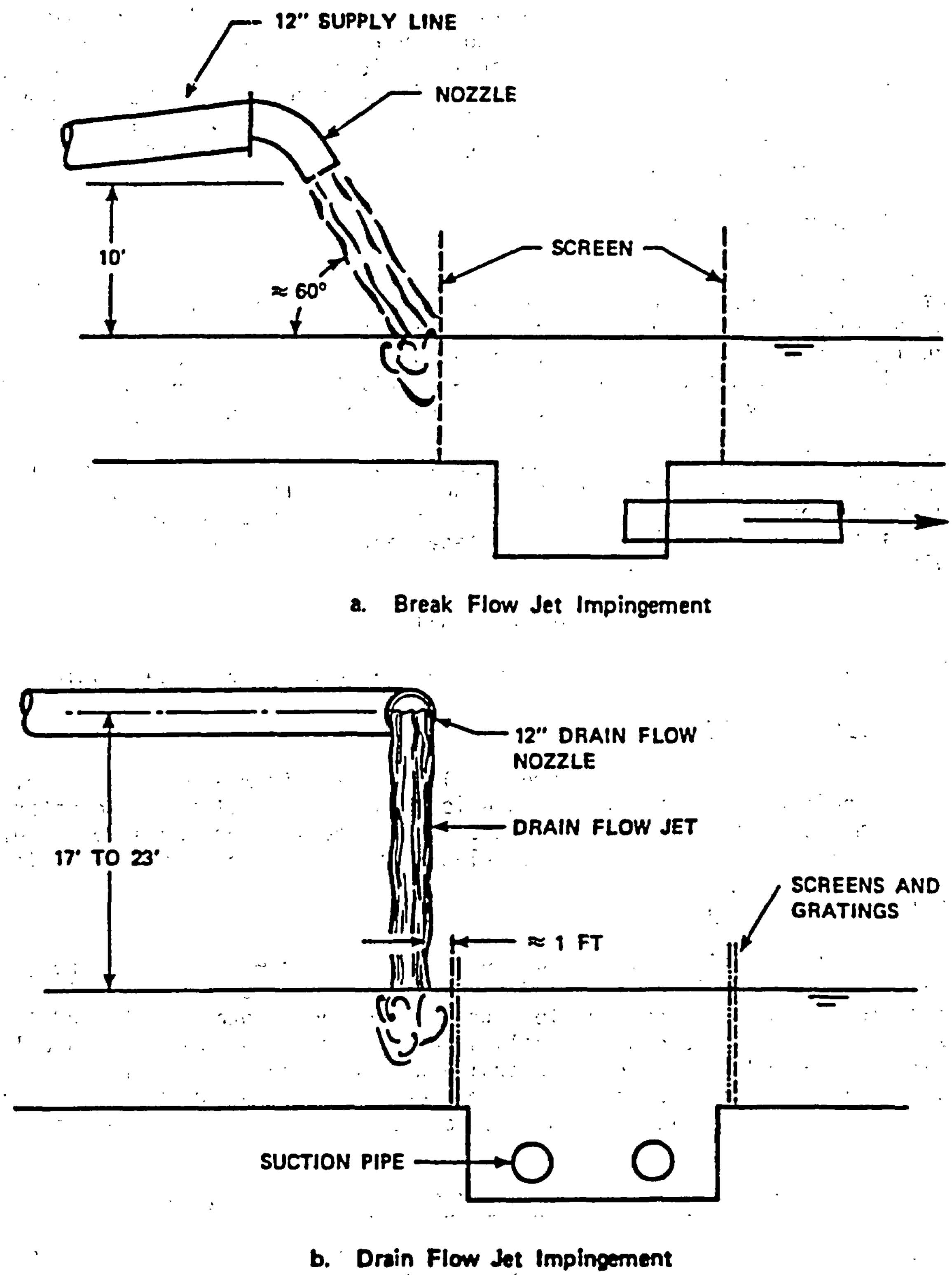


Figure 3.33 Break and drain flow impingement

the maximum response of the data for each of the hydraulic performance parameters as a function of the sump flow variables, particularly when plotted as a function of Froude number. Thus, the ability to describe the performance of PWR ECCS sumps, with or without flow perturbations, using bounding envelope curves was the most significant result of the ARL test program. The application of an envelope analysis to test data resulting from all the sump performance tests is discussed in Section 3.4.1. Findings of the sump performance tests are described in greater detail in subsequent sections.

3.4.1 Envelope Analysis

The sump performance test program generated a data base covering a broad range of ECCS geometric variables, flow conditions (including potential accident conditions), and design operations (horizontal or vertical inlets, single or dual pipes, etc.). An envelope analysis applied to this broad range of data resulted in boundary curves for vortex activity, swirl, and sump head loss as a function of key sump flow variables (Froude number, velocity, etc.).

Figures 3.34, 3.35, and 3.36 show typical envelope analysis curves for air ingestion, surface vortex activity, and swirl in PWR sumps with dual horizontal pump suction intakes. Figures 3.37, 3.38, and 3.39 show typical envelope analysis curves for air ingestion, surface vortex activity, and swirl in PWR sumps with dual vertical intakes.

3.4.2 General PWR Sump Performance (All Tests)

The following items were studied during the sump performance testing:

(1) Free Surface Vortices

Vortex size and type (see Figure 3.40) resulting from a given geometric and flow condition are difficult to predict and are not reliable indicators of sump performance. Performance parameters (void fraction, pressure loss coefficient, and swirl angle) are not well correlated with observed vortex formations.

(2) Air Ingestion

Measured levels of air ingestion, even with air core vortices, were generally less than 2%. Maximum values of air ingestion with deliberately induced swirl and blockage conditions were less than 7% for horizontal inlets and 12% for vertical inlets. These high levels always occurred for high flow and low submergence (Froude number (Fr) generally greater than 1.0). For submergences of 8 feet or more, none of the configurations tested indicated air-drawing vortices ingesting more than 1% over the entire flow range, even with severe flow perturbations.

(3) Swirl (Measured at a Distance 14 Diameters from Suction Inlet)

Flow swirl within the intake pipes, with or without flow perturbations, was very low. In almost all cases, the swirl angle was less than 4°, an acceptable value for RHR and CSS pumps. The maximum value for severely perturbed flows was about 8° and occurred during the screen blockage test series.

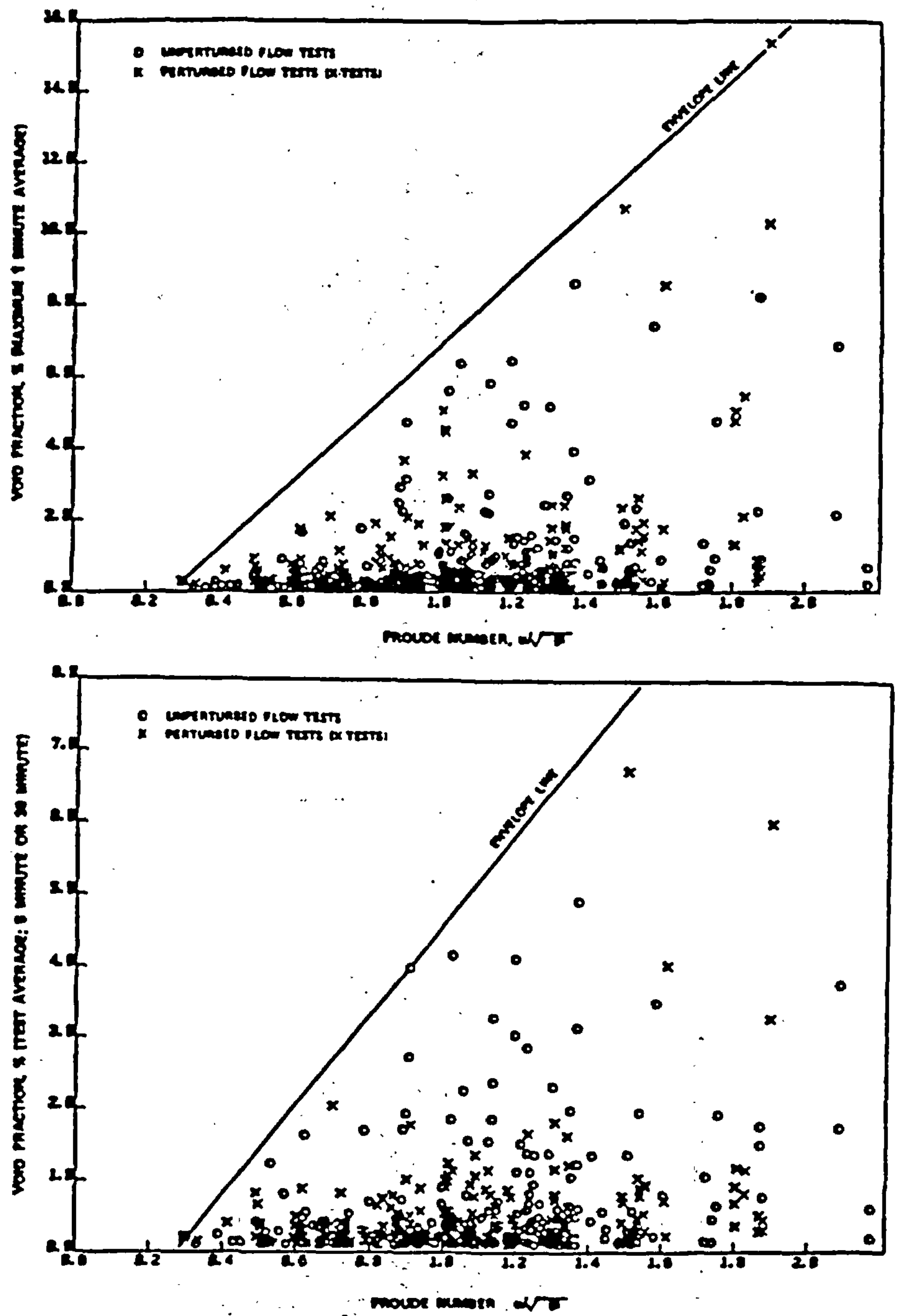


Figure 3.34 Void fraction (% by volume) as a function of Froude number; horizontal intake configuration; only data points indicating nonzero void fraction are plotted

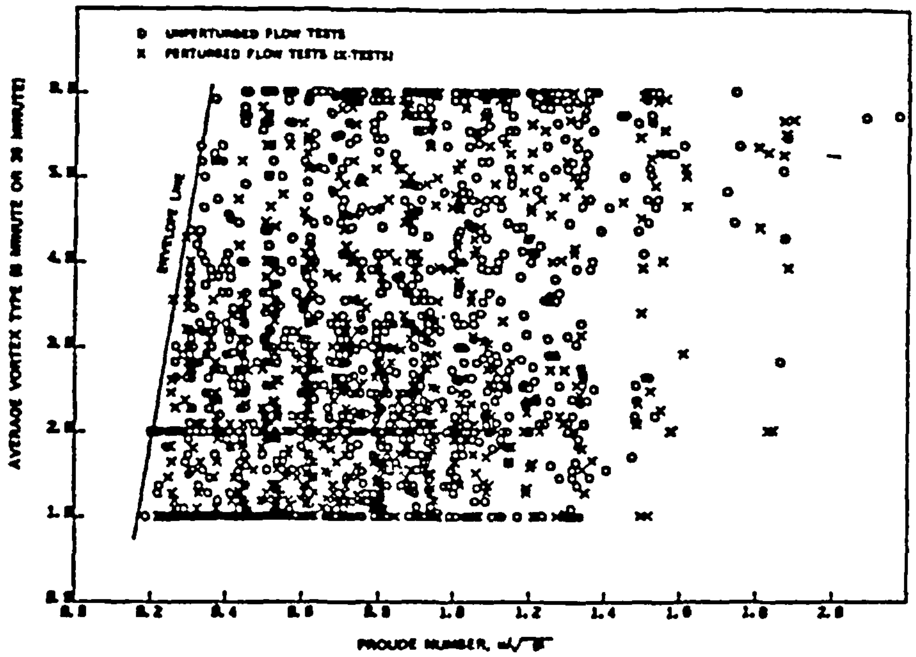


Figure 3.35 Surface vortex type as a function of Froude number, horizontal intake configuration (Type 1: surface swirl only; Type 6: full air core to intake)

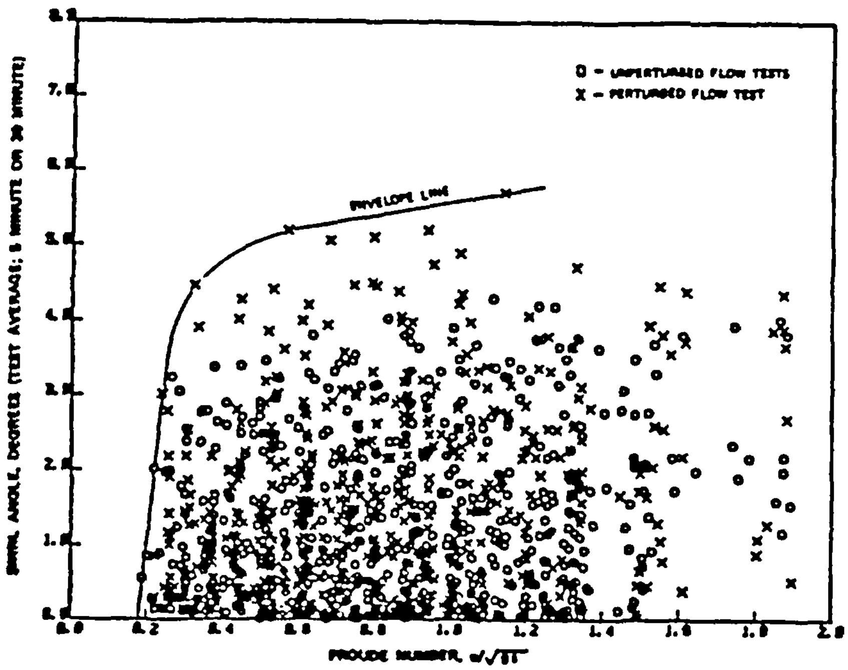


Figure 3.36 Swirl as a function Froude number; horizontal intake configuration

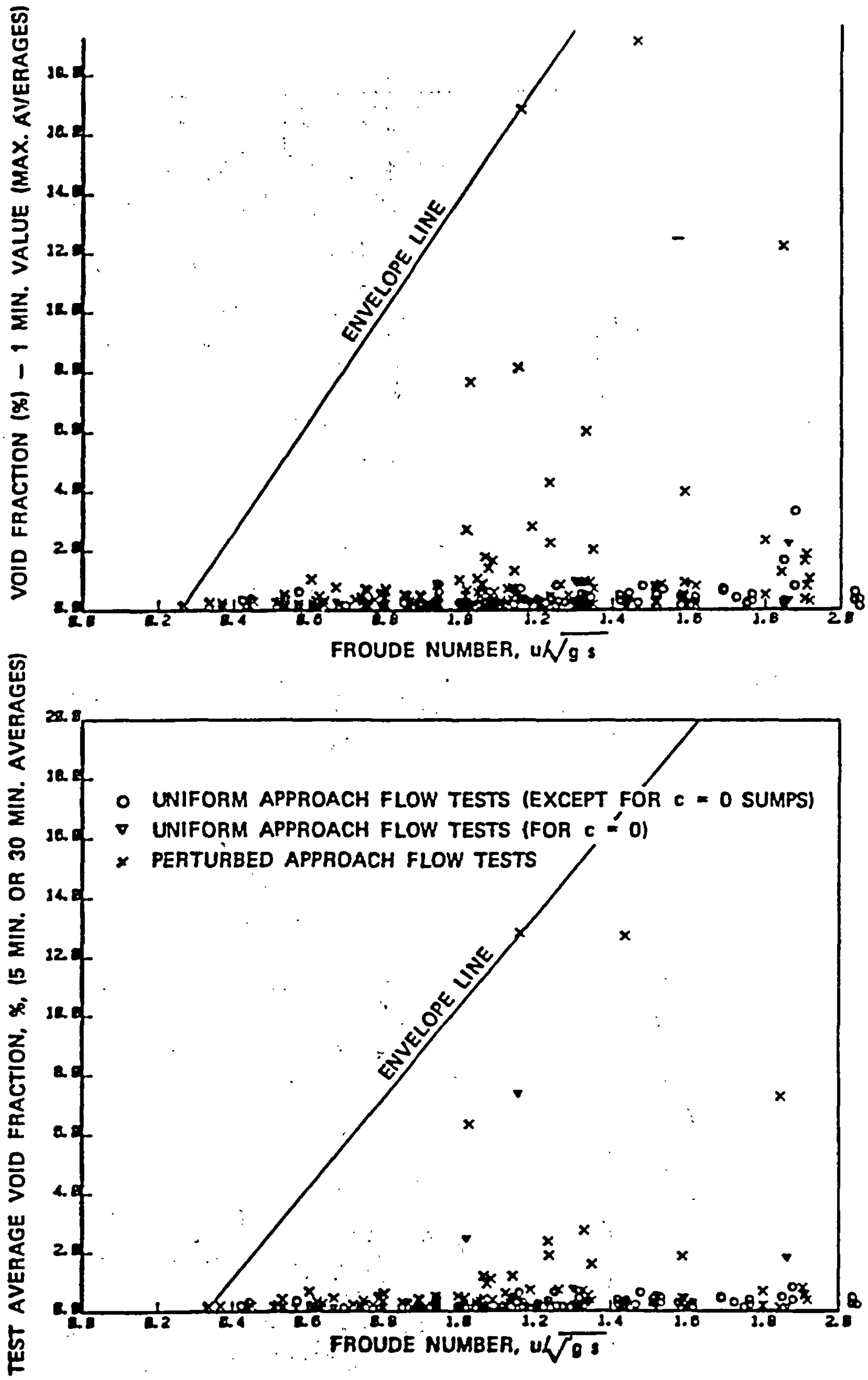


Figure 3.37 Void fraction data for various Froude numbers; vertical intake configuration; only data for nonzero void fraction plotted

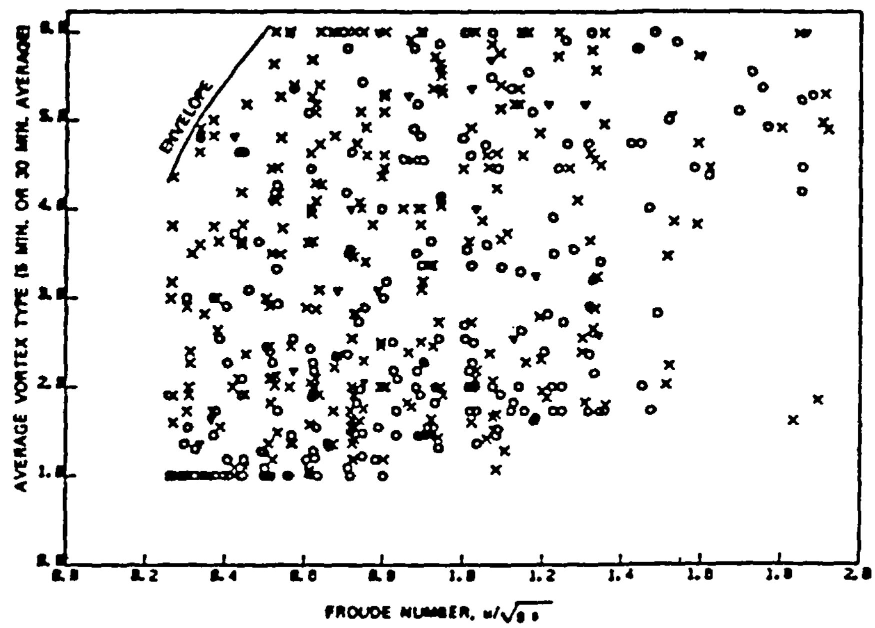


Figure 3.38 Surface vortex type as a function of Froude number, vertical intake configuration (Type 1: surface swirl only; Type 6: full air core to intake)

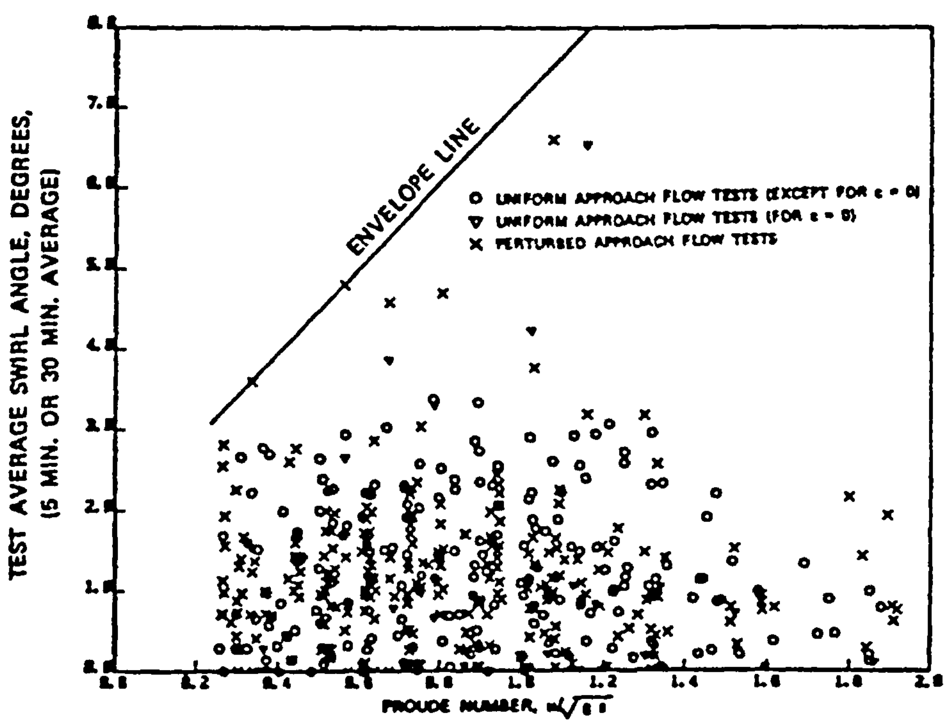


Figure 3.39 Swirl as a function of Froude number, vertical intake configuration

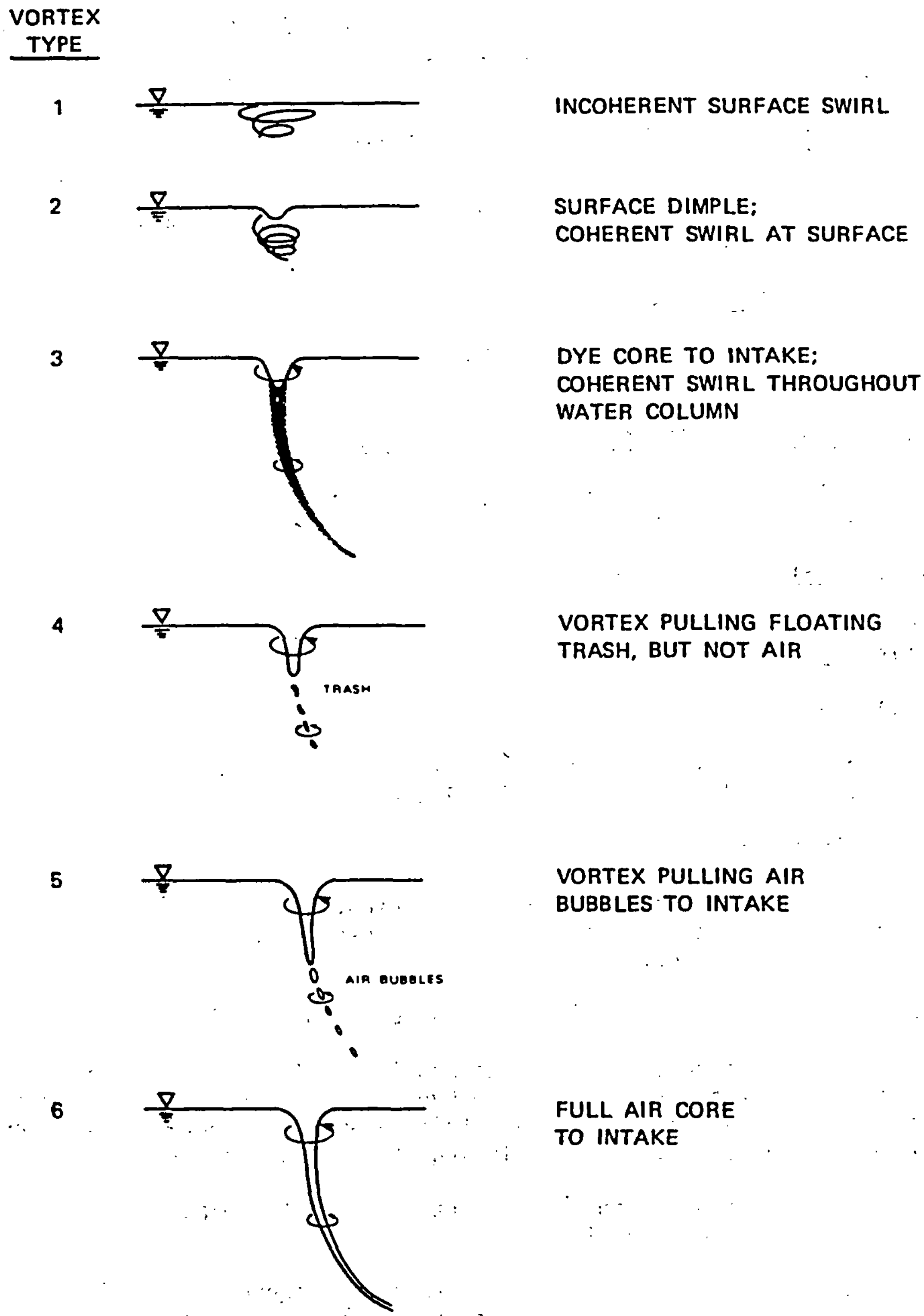


Figure 3.40 Vortex type classification

(4) <u>Sump Head Losses</u>

The suction pipe intake pressure loss coefficient for most of the tests, with and without flow perturbations, was in the range of 0.8 ± 0.2 and agreed with recommended values in standard hydraulic handbooks.

3.4.3 PWR Sump Performance During Simulated Accident Conditions (Perturbed Flow)

The following items were considered in evaluating sump performance under perturbed flow conditions:

(1) <u>Screen Blockage</u>

Screen blockages up to 75% of the sump screen resulted in air ingestion levels similar to those noted under 3.4.2(2) above.

(2) <u>Nonuniform Approach Flow Distributions</u>

Nonuniform approach flows, particularly streaming flow, generally increased surface vortexing and the associated void fraction.

(3) <u>Drain and Break Flow</u>

Drain and break flow effects were generally found not to cause any additional air ingestion. They reduced vortexing severities by surface wave action.

(4) <u>Obstructions</u>

Obstructions 2 feet or less in cross-section had no influence on vortexing, air withdrawals, swirl, or inlet losses.

(5) <u>Transients</u>

Under transient startup conditions, momentary vortices were strong, but no air-core vortices giving withdrawals exceeding 5% void fraction (1-minute average) were observed.

3.4.4 Geometric and Design Effects (Unperturbed Flow Tests)

In general, no consistent trends applicable for the entire range of tests were observed in the data between the hydraulic response of the sump (air withdrawal, swirl, etc.) and secondary geometric parameters. However, for some ranges of flow and submergence, the following observations are applicable:

(1) Greater depth from containment floor to the pipe centerline reduces surface vortexing and swirl.

(2) Lower approach flow depths with higher approach velocities may cause increased turbulence levels serving to dissipate surface vortexing.

(3) Suction pipe inlets located with less distance to the adjacent sump wall and greater pipe spacing reduces vortexing and swirl.

(4) There is no advantage in extending the suction pipe beyond 1 pipe diameter from the wall.

3.4.5 Design or Operational Items of Special Concern in PWR ECCS Sumps

(1) Pump Intake Orientation

Comparison of vertical intake data to corresponding horizontal intake data showed minor differences in hydraulic performance for sumps of the same geometry and flow conditions. Average vortex types agreed within ± 1 (types range from 1, incoherent surface swirl, to 6, full air core to pump intake); air withdrawals were somewhat higher for vertical intake sumps but usually within 1% (30-minute averages) to 4% (1- and 5-minute averages); swirl angles differed only within ± 1°. Both vertical and horizontal intake sumps performed better under perturbed flow when the pipe inlets were closer to an adjacent wall rather than at the center of the sump.

(2) Single Intake Sumps

Two sump configurations (4 x 4 feet and 7 x 5 feet in plan, both 4.5 feet deep with 12-inch-diameter intakes) were tested under unperturbed (uniform) and perturbed approach flows with screen blockages up to 75% of the screen area. For both the configurations, unperturbed flow tests indicated air withdrawals were always less than 1% by volume for the entire range of tested flows and submergences (Fr 0.3 to 1.6.). Even with perturbed flows, zero or near zero air withdrawals were measured in both sumps for Froude numbers less than 0.8, suggesting insignificant vortexing problems. For Froude numbers above 0.8, a few tests indicated significantly high air withdrawal (up to 17.4% air by volume; 1-minute average) especially for the smaller sized sump. Measured swirl values in the pipes were insignificant for both the tested sumps, in the range of 2 to 3 degrees, even with flow perturbations. The inlet loss coefficients for both sump configurations were in the expected ranges for such protruding inlets, 0.8 ± 0.2.

(3) Dual-Intake Sumps with Solid Partition Walls

Four dual-intake sump configurations (one 20 x 10-foot sump with 24-inch diameter intakes and three 8 x 10-foot sumps with 24-inch, 12-inch, and 6-inch intakes, respectively) were tested with solid partition walls in the sumps between the pipe inlets and with only one intake operational. None of the tests indicated any significant increases in vortexing, air withdrawal, swirl, or inlet losses compared to dual pipe operation without partition walls. Thus, a partition wall in a sump should not cause any additional problems when only one pipe is operating.

(4) Bellmouths at Pipe Entrance

Limited tests on a sump configuration were conducted with and without a bellmouth attachment to the 12-inch intakes. Adding bellmouths at the pipe entrances did not result in any significant changes in the vortex types, air withdrawals, and pipe swirl compared to those that otherwise existed under the same hydraulic conditions. An expected reduction of up to about 40% in inlet losses was noticed with the addition of a bellmouth.

(5) Cover Plate

A solid top cover plate over the sump was effective in suppressing vortices as long as the cover plate was submerged and proper venting of air from underneath was provided. No air drawing vortices were observed for the submerged cover plate tests, and no significant changes in swirl or loss coefficients occurred.

(6) Vortex Suppressors

Cage-shaped vortex suppressors made of floor grating in the form of cubes 3 and 4 feet on a side and single or multiple layers of horizontal floor gratings over the entire sump area were found to be effective in suppressing vortices and reducing air ingestion to zero. These suppressors were tested in sump configurations using 12-inch-diameter intake pipes, and with the water levels ranging from 0.5 foot to 6.5 feet above the top of the suppressors. Adverse screen blockages were imposed on these sump configurations, which produced considerable air ingestion and strong vortexing without the suppressors; thus, the effectiveness of the suppressors was tested when hydraulic conditions were least desirable. The suppressors also reduced pipe swirl and did not cause any significant increase in inlet losses. Both the cage-shaped grating suppressors and the horizontal floor grates were made of standard 1.5-inch floor grates.

Tests on a cage-shaped suppressor less than 3 feet on a side indicated the existence of air-core vortices for certain ranges of flow and submergences, even though air withdrawals were found to be reduced to insignificant levels.

Therefore, either properly sized cage-shaped suppressors made of floor grating or floor grating over the entire sump area may be used to reduce air-ingestion to zero in cases where the sump design and/or approach flow creates otherwise undesirable vortexing and air ingestion.

(7) Scale Model Tests

To evaluate the use of reduced-scale hydraulic models to determine the performance of containment emergency sumps and to investigate, in particular, possible scale effects in modeling the hydraulic phenomenon of concern, a test program involving two reduced-scale models (1:2 and 1:4) of a full-size sump (1:1) was undertaken (NUREG/CR-2760).

The test results show that the hydraulic models predicted the hydraulic performance of the full-sized sump; namely, vortexing, air ingestion from free surface vortices, pipe flow swirl, and the inlet loss coefficient. No scale effects on vortexing or air withdrawals were apparent within the tested range for both models. However, an accurate prediction of pipe flow swirl and inlet loss coefficient was found to require that the approach flow Reynolds number and the pipe Reynolds number be above certain limits.

Based on these results, it is concluded that properly designed and operated reduced-scale hydraulic models of geometric scales 1:4 or larger could be used to properly evaluate the hydraulic performance of a sump design. Evaluations of sump hydraulic model studies conducted in the past can be derived from this series of tests.

(8) Pump Overspeed Tests

Two 8 x 10 x 4.5-foot sumps (one with horizontal suction intakes and one with vertical suction intakes) were tested at higher flow rates to simulate pump overspeed or run out (to Froude number 1.6) conditions. No strong air-core vortices were observed with air-withdrawals greater than 1% (1-minute or 30-minute averages).

Maximum recorded pipe swirl angle was 0.9° (at 14.5 pipe diameters from entrance); inlet loss coefficients averaged 0.8 (NUREG/CR-2761).

(9) High Temperature Tests

A series of tests were performed on horizontal suction intake, and the conclusion was that changing water temperatures over the range from 40°F to 165°F had no significant effect on sump hydraulic performance parameters (NUREG/CR-2758, Section 4.6.).

3.4.6 BWR Suction Pipe Intakes

Because BWR plants do not have a sump or a floor depression with surrounding screens and gratings, typical residual heat removal system suction pipe inlet configurations applicable to Mark I, Mark II, and Mark III containment designs were investigated in full-scale flow experiments. Figure 3.41 shows the two inlet pipe and strainer configurations of the three designs under consideration.

Key parameters of interest were air-ingestion levels, vortex formation, suction pipe swirl, and the RHR inlet pressure loss coefficient. The tests were conducted with both perturbed and unperturbed approach flows to the inlets, as indicated in Figure 3.42. Flows ranged from 2000 to 12000 gpm per pipe, while submergences varied from 2 to 5 feet. The resulting Froude numbers ranged from about 0.2 to 1.1.

Figures 3.43 and 3.44 show the test-average (30-minute) and 1-minute void fractions for the two inlet configurations (A and B) and the various flow schemes examined. Essentially zero air withdrawal was measured for both configurations at Froude numbers $\leq$ 0.6 under all tested approach flows. For the double inlet or tee inlet design (configuration A), maximum air withdrawal was less than 0.5% at all Froude numbers examined. For the single inlet design (configuration B), air core vortices drawing up to 4% air by volume were observed to form at a Froude number above 0.6 under perturbed approach flows.

No air-core vortices were observed for either inlet configuration over the entire range of tested flows at submergences equal to or above 3.5 feet (Froude < 0.6). Swirl angle in the configuration B inlet pipe ranged from 0 to 3°, while the configuration A pipe swirl angle fell between 2 and 7° for the Froude numbers tested.

The measured inlet loss coefficients expressed in terms of suction pipe velocity head averaged to about 1.7 and 1.0 for configurations A and B respectively. The loss coefficients reflect entrance, strainer, and tee losses (if applicable).

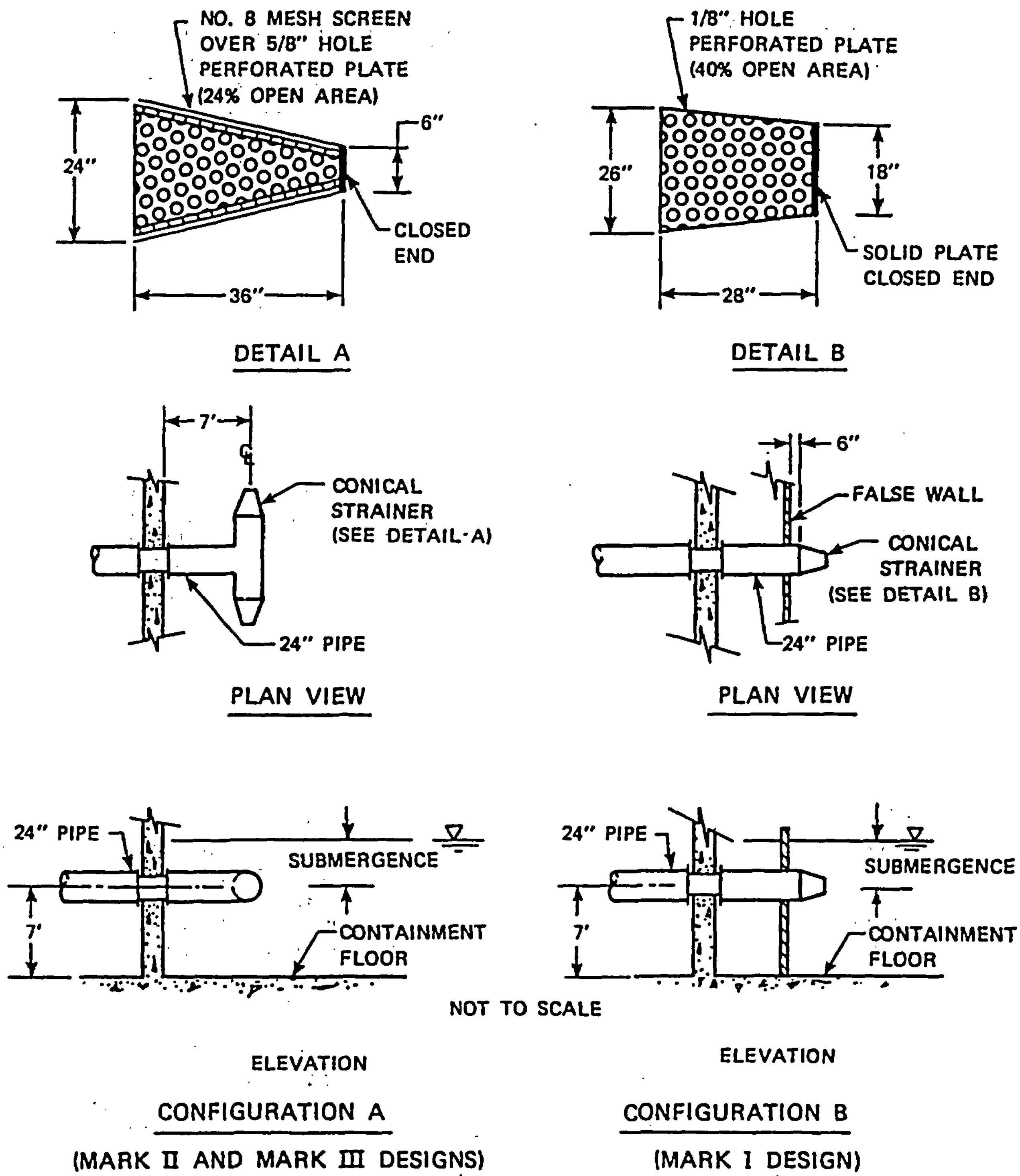


Figure 3.41 BWR pipe inlet configurations as built in full-size facility

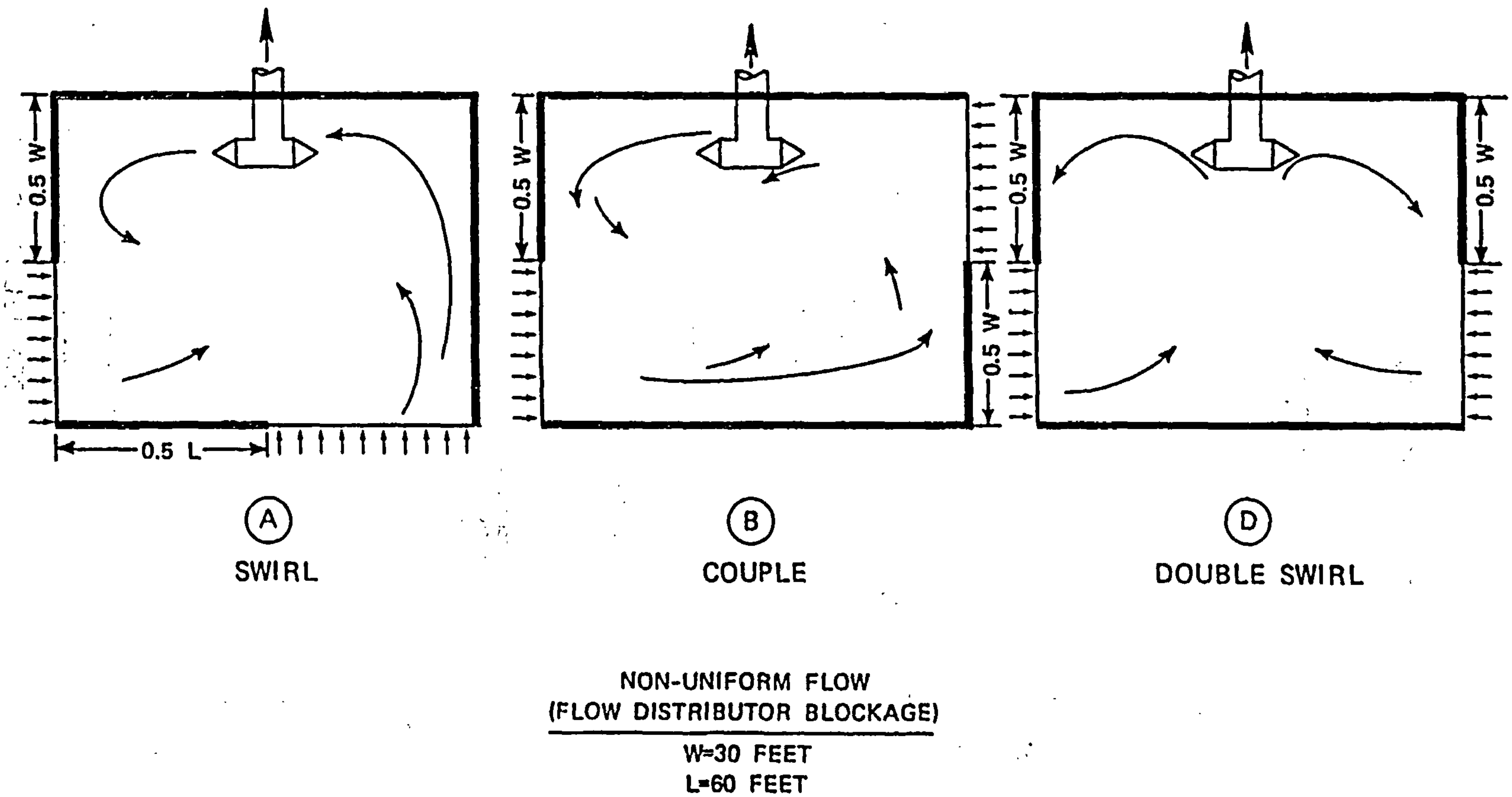
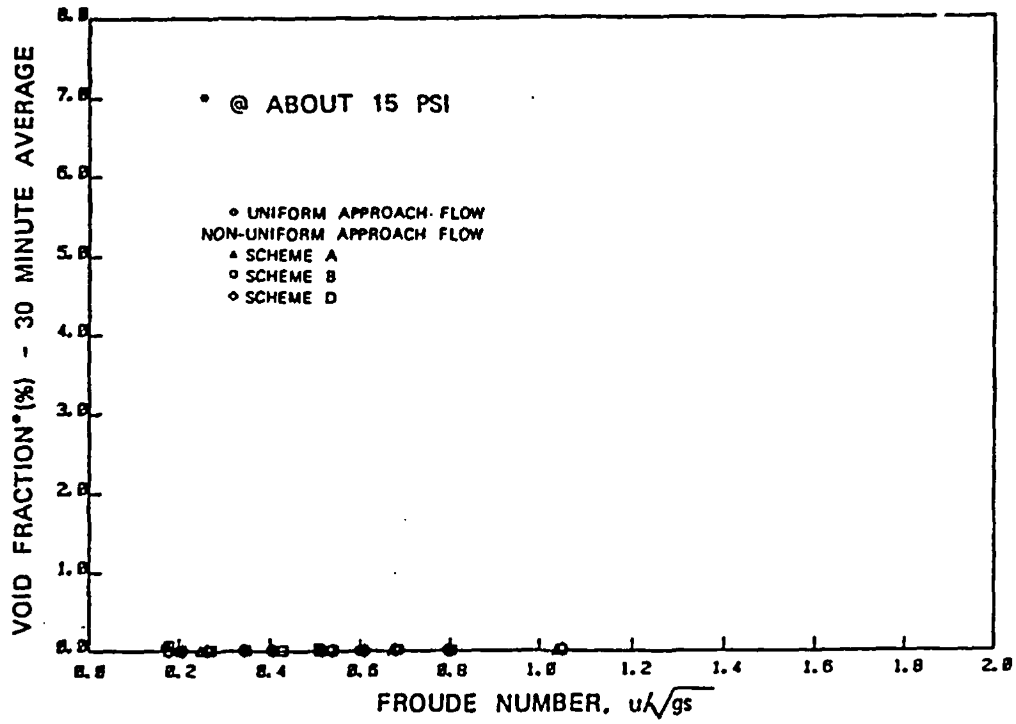
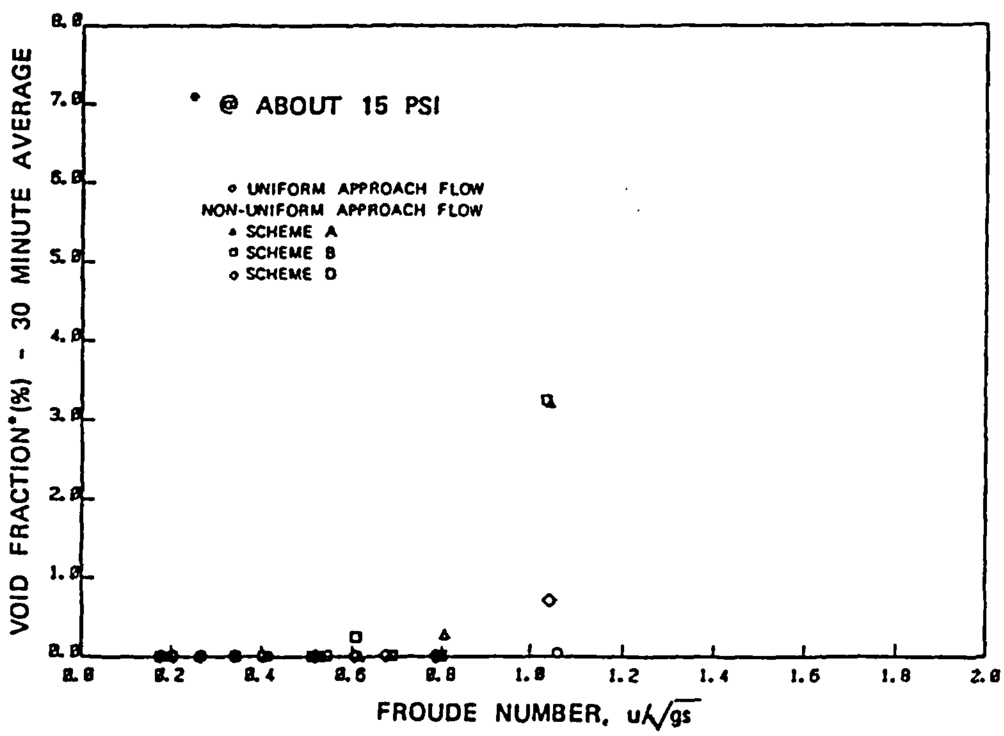


Figure 3.42 Perturbed flow schemes; schemes A, B, and D used for BWR inlet tests

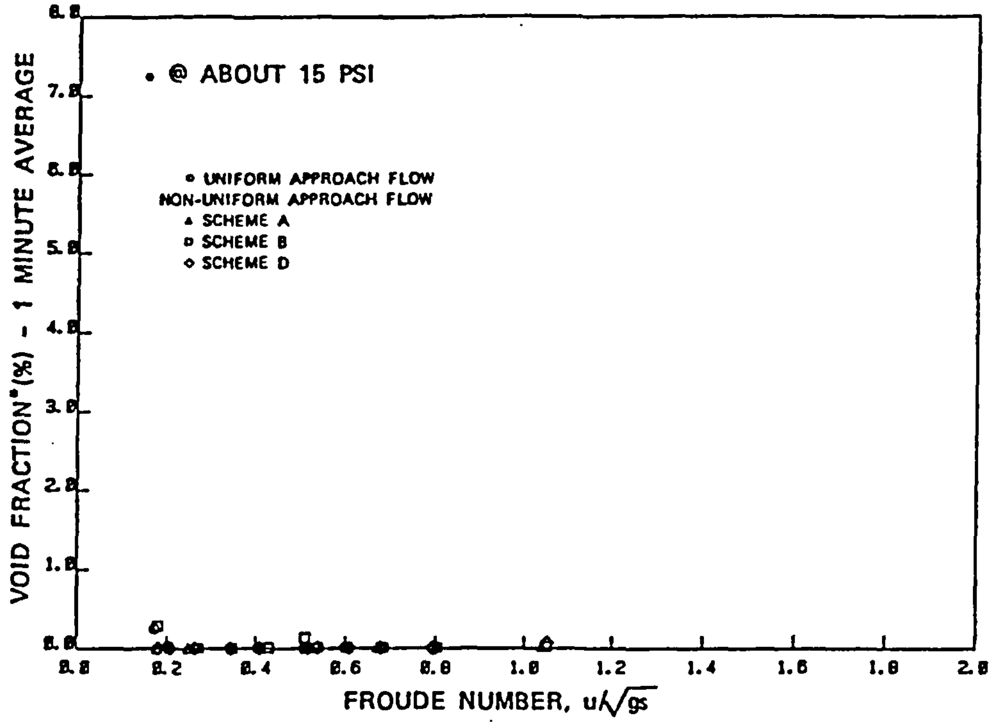


a. CONFIGURATION A

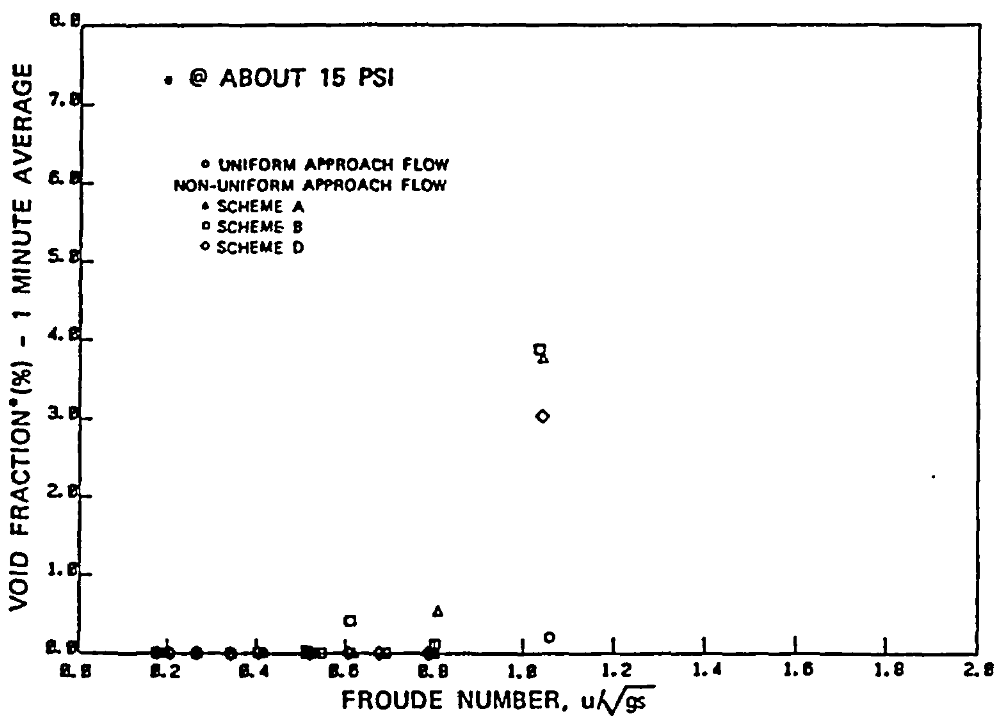


b. CONFIGURATION B

Figure 3.43 Test-average void fractions for tested BWR suction intakes



a. CONFIGURATION A



b. CONFIGURATION B

Figure 3.44 1-minute average void fraction for tested BWR suction intakes

4 INDEPENDENT PROGRAM TECHNICAL REVIEWS

Independent program technical reviews were conducted before and during key phases of the work reported in Section 3 to solicit comments and technical views about the program's direction and goals from experts not connected with the implementation and execution of Task Action Plan (TAP) A-43. The reviewers were selected from among the foremost authorities in each of the areas reviewed. Two reviews were conducted: sump hydraulic performance and insulation debris calculational methods effects.

4.1 Sump Hydraulic Performance Review

The sump hydraulic performance review consisted of two panel meetings,* held on March 17 and June 4, 1981. The primary purpose of the first meeting was to introduce in detail the program plan and initial test results. The second meeting was primarily for reviewer followup response and comment. Additionally, at both meetings the reviewers were provided with preliminary program redirections. They were asked to comment on results to date and give an analysis of the proposed future program plan. Overall, the reviewers approved of the program, the experimental test plan, its conduct, and data analysis. They concluded that the program was appropriate for resolving the sump hydraulic performance issues.

Divergent opinions emerged during the review concerning the potential for pump performance degradation when the fluid temperature was near saturation. Some concerns were expressed regarding the possibility of degraded pump performance as a result of cavitation or the release of dissolved air into the water in the suction lines leading to the pumps. Other opinions suggested that pump performance should be satisfactory at coolant temperatures near saturation, because the (1) solubility of air in water is low near saturation and, (2) if cavitation were not occurring in the pump, any voids would collapse as a result of the static pressure increase with depth in the sump. These collapsing bubbles would then form a turbulent environment and inhibit surface vortex activity. Although the pump issues raised by the reviewers are indirectly pertinent to the sump hydraulics program, they are a part of USI A-43 and have been addressed (see Section 3.2).

---

*Meetings were held on March 17, 1981, at Germantown, Maryland, and June 4, 1981, at Alden Research Laboratory of Worcester Polytechnic Institute, Holden, Massachusetts. Those attending and their affiliations were P. Tullis/Utah State University; D. Simons/Simons, Li and Associates; R. Gardiner/Western Canada Hydraulic Laboratories; D. Canup/Duke Power Company; W. Butler/NRC; S. Vigander/Tennessee Valley Authority (TVA); J. Kennedy/University of Iowa; and R. Letendre/Combustion Engineering, Inc. (R. Letendre did not attend the meeting of June 4, 1981.) Those attending were asked to provide formal written responses and comments at the close of the second meeting. Copies of the responses are available through the Office of Light Water Safety Research, Department of Energy, Washington, DC.

In direct response to reviewer comments, elevated temperature tests were performed immediately following the first 25 configurations, which was earlier in the program than originally planned. The experimental research program did not examine the effects on operation at temperatures near saturation conditions because the operational limits of the experimental facility (about 165°F). However, up to that limit, no significant or adverse temperature effects on sump system performance were detected.

An area of general peer review group agreement was that sump system performance with respect to air entrainment could be improved in most sump configurations by the addition of a vortex suppression device(s). One reviewer, however, commented that such a device(s) might be removed during some phase of reactor operations and not be replaced. Such a possibility, in his judgment, was sufficient justification for an experimental research program that would allow the development of adequate sump design guidelines that were based upon justifiable physical criteria (in the absence of vortex suppressors). The results of the studies provided in Section 3.4 confirm the effectiveness of vortex suppressors to reduce air ingestion to zero and provide hydraulic results for developing acceptable sump design guidelines.

The adequacy of recirculation sump pumps for performing reliably when ingesting air/water mixtures was a matter of some concern to the review group. These concerns have been resolved by the development of sump design guidelines that take into account pump performance specifications under such conditions.

4.2 Insulation Debris Effects Review

The purpose of the insulation debris effects review was to determine the adequacy of methods (described in Section 3.2 and in detail in NUREG/CR-2791) to conservatively estimate quantities of insulation debris that might be produced in containment, its transport, and its potential for sump screen blockage.

The review was conducted in two phases. In the initial phase, a draft report describing the methods was provided to peer panel and other reviewers* to solicit their comments. Reviewers provided highly useful criticisms and comments with recommendations for improvements in the physical basis and rigor of the development of the debris generation and transport models.

The draft document was then modified in response to the comments of the reviewers. The modified document was transmitted to the reviewers, who were then requested to prepare comments for a formal peer panel review, which was the second phase of the review process.

---

*The peer panel reviewers and their affiliations were R. Gardiner/Western Canada Hydraulic Laboratories; D. Simons/Simons, Li & Associates, Inc.; D. Canup/Duke Power Company; R. Mango/Combustion Engineering, Inc.; P. Tullis/Utah State Unviersity; J. Kennedy/University of Iowa; W. Butler/NRC; and S. Vigander/TVA. Other reviewers included G. Weigand/Sandia and R. Bosnak, G. Mazetis, and T. Speis/NRC. Their written review comments are available through the NRC Division of Safety Technology, NRC, Washington, DC 20555.

Formal peer panel review took place at NRC Headquarters on March 31, 1982. Panelists Kennedy and Canup were unable to attend the meeting; however, a number of other persons, in addition to peer panel members, participated in the review.* Questions that were raised during the meeting and their disposition are given below.

It was observed that, under some circumstances, the amount of debris generated with the potential to migrate to the sump could be greater than that estimated in the draft report. This concern was resolved by determining that the report would require the selection of those pipe break locations and jet targets that would generate the maximum quantities of potentially transportable debris without regard to initial blowdown and transport direction.

Questions were raised about (1) the applicability of the jet model used in the debris generation portion of the report, (2) the assumption of uniform distribution of debris across the face of the jet and, (3) the use of a 0.5-psi stagnation pressure cutoff for debris generation. Resolution of item (1) was arrived at by agreement that a modified Moody jet model (Moody, 1973) would be allowed to model the jet. It was agreed that the stripping of all insulation from plant and piping within the crane wall and within the jet represented a conservative treatment of insulation debris generation.

Discussions of item (2) concluded that a definite probability existed that debris distribution across the face of the jet would not be uniform. It was agreed that a distribution of debris across the jet face would be provided that would represent the geometric distribution of insulation targeted by the jet in the containment. In addition, because of uncertainties in jet transport to walls, it was agreed that the quantities of debris estimated to exit through crane wall openings would be doubled.

The use of a 0.5-psi stagnation pressure cut-off (item (3)) for insulation damage was questioned by a number of reviewers. An SNL staff member put forth technical views on the expected performance of jets under LOCA conditions. He stated that centerline stagnation pressures above 15 psig could be expected for at least 5 diameters downstream of high energy, high pressure breaks. An Atomic Energy Commission report (Glasstone, 1981) was cited by Burns and Roe as the origin of the cut-off estimate for debris generation. ARL personnel reported that preliminary experiments at ARL have shown that little insulation damage occurred to fibrous insulation assemblies up to 6.5-psi water jet pressures. It was agreed** that the 0.5-psi stagnation pressure represented a conservative treatment for the onset of insulation debris generation. It was further agreed that the assumption that all insulation within the jet cone would be transformed to insulation debris was conservative. This assumption was chosen to represent the volume within which insulation debris would be generated under the treatment provided in NUREG/CR-2791. The results of work performed subsequently on these issues are provided in Sections 3.3 and 5.3 of this report.

---

*Other attendees were: S. Hanauer, K. Kniel, C. Liang, P. Norian, F. Orr, A. Serkiz, J. Shapaker/NRC; G. Hecker/Alden Research Laboratory; E. Gahan, J. Wysocki/Burns and Roe; W. Swift/Creare, Inc.; and P. Strom and G. Weigand/Sandia.

**This decision has been superseded by information discussed in Section 3.3.

Discussions were held on the physical accuracy of the model in representing pipe whip, pipe impact, and the direction of motion of dislodged insulation and its trajectory. It was first pointed out that the quantity of insulation generated by this mechanism would amount to 10% or less of that generated by jet forces. It was further pointed out that the use of the treatment in the report would conservatively estimate the quantities of insulation debris produced by a minor contributor to debris production and, as such, was satisfactory.

Questions were raised on the treatment of long term transport following blowdown. These questions related to

(1) recirculation flow velocities within containment

(2) hydraulic lift provided to sunken debris

(3) drawdown of floating debris onto less than fully submerged sump screens (ice-jam effect)

(4) transport mechanisms of sunken debris, such as tumbling and sliding

In the resolution of question (1), agreement was reached to account for obstructions in flow paths and subsequent flow expansion (Appendix D and NUREG/CR-2791).

Agreement was reached on question (2) that, for horizontal orientation, lift would be approximated by drag for horizontal debris, would be zero for vertically oriented debris, and would be disregarded for tumbling debris.

Item (3) was recognized as a potentially important mechanism for screen blockage. It will be treated by established methods available as described in the literature, (Uzuner, 1977; NUREG/CR-2791).

Tumbling and other transport mechanisms (item (4)) could significantly affect the movement of debris towards screens. Panelists agreed to treatments that they considered to be conservative in dealing with debris transported by these mechanisms. Recent experiments at ARL have shown a wide variability of transport characteristics depending on the debris geometry (Section 3.3; NUREG/CRs-2982 and -3616).

Arguments were raised that a period of debris transport (intermediate-to short-term transport and long-term transport, as defined here) might exist. It was postulated that transport during such an interim period might seriously affect potential sump blockage. Because the report assumes that all floating debris reaches the sump, such an interim migration period would not affect the consequences of such transport. With respect to debris of density equal to or greater than unity and its transport, discussions brought out views that the likelihood of a significant effect during such an interim period would be minor, flow patterns would show no preferential transport toward the sump, and entrainment would be higher in the recirculation mode than in the interim period.

An issue that was not resolved concerned the behavior of fibrous insulation in its migration toward a sump and the potential for blockage by such material. Because this problem appears to exist at only a few plants, it is considered plant specific. Nevertheless, it was an open issue at the time of the meetings. Following the meetings, experimental studies were conducted at ARL to estimate stagnation pressures required for the onset of debris generation for nonencapsulated mineral wool and fiberglass insulations (NUREG/CR-3170), the transport characteristics of such debris, and the pressure losses at sump screens caused by the accumulation of fibrous debris on screens (NUREG/CR-2982). These findings are reflected in the findings provided in Sections 3.3 and 5.3 of this report.

All panelists, except S. Vigander of TVA, concluded that the use of the methods discussed would result in conservative estimates of sump screen blockage. Vigander commented that while he was of the opinion that the treatment would yield conservative, perhaps ultra-conservative, results, he could not with certainty arrive at that conclusion. He suggested that uncertainty analyses be conducted to establish the levels of conservatism (if any) that are provided in the development. Other panelists agreed that quantitative or qualitative error analyses would be desirable, although the needs for such analyses were deemed not to be immediate or pressing.

5 SUMMARY OF SUMP PERFORMANCE TECHNICAL FINDINGS

5.1 General Overview

Emergency core cooling systems require a clean and reliable water source for maintaining long-term recirculation following a LOCA. PWRs rely on the containment emergency sump to provide such a water supply to residual heat removal pumps and containment spray pumps. BWRs rely on pump suction intakes located in the suppression pool, or wet well, to provide a water source to residual heat removal pumps and core spray pumps. Thus, recirculation pump performance under post-LOCA conditions must be evaluated for both BWRs and PWRs.

Typical technical considerations are shown in Figure 5.1. Each major area of concern--pump performance, sump hydraulics, and debris generation potential--can be assessed separately, but the combined effects of all three areas should then be assessed to determine the overall effect on both the available and required NPSH requirements of the pumps. The sections below summarize technical findings and provide concise data sets.

5.2 Sump Hydraulic Performance

Full-scale tests show that adequate PWR sump (or BWR RHR suction intake) hydraulic performance is principally a function of depth of water (the submergence level of the suction pipe) and the rate of pumping (suction inlet water velocity). These variables can be combined to form a dimensionless quantity defined as the Froude number

$$\text{Froude number} = U/\sqrt{gs}$$

where

U = suction pipe mean velocity
s = submergence (water depth from surface to suction pipe centerline)
g = acceleration due to gravity

The extent of air ingestion is the principal parameter to be determined. Small amounts of air (less than 2% by volume) do not significantly degrade pumping capacity (Merry, 1976; Murakami and Minemura, 1977; and Florjancic, 1970). Generally speaking, full-scale tests revealed low levels of air ingestion (< 2%) over a wide range of Froude numbers despite the presence of air-core vortices. Other hydraulic effects, such as intake swirl, were found to be small, and inlet loss coefficients were in agreement with handbook values for similar intake geometries.

Section 3.4 summarizes the results of full-scale PWR sump hydraulic tests and BWR suction inlet tests. Figures 3.34 and 3.37 show typical void fraction data as a function of Froude number for PWR sumps; Figures 3.43 and 3.44 show void fraction data for BWR suction inlets. More detailed results are provided in NUREG/CR-2758; NUREG/CR-2759; NUREG/CR-2760; NUREG/CR-2761; and NUREG/CR-2772. Generally, sump (or suction intake) design acceptability should be based upon a $\leq$ 2% air ingestion criterion.

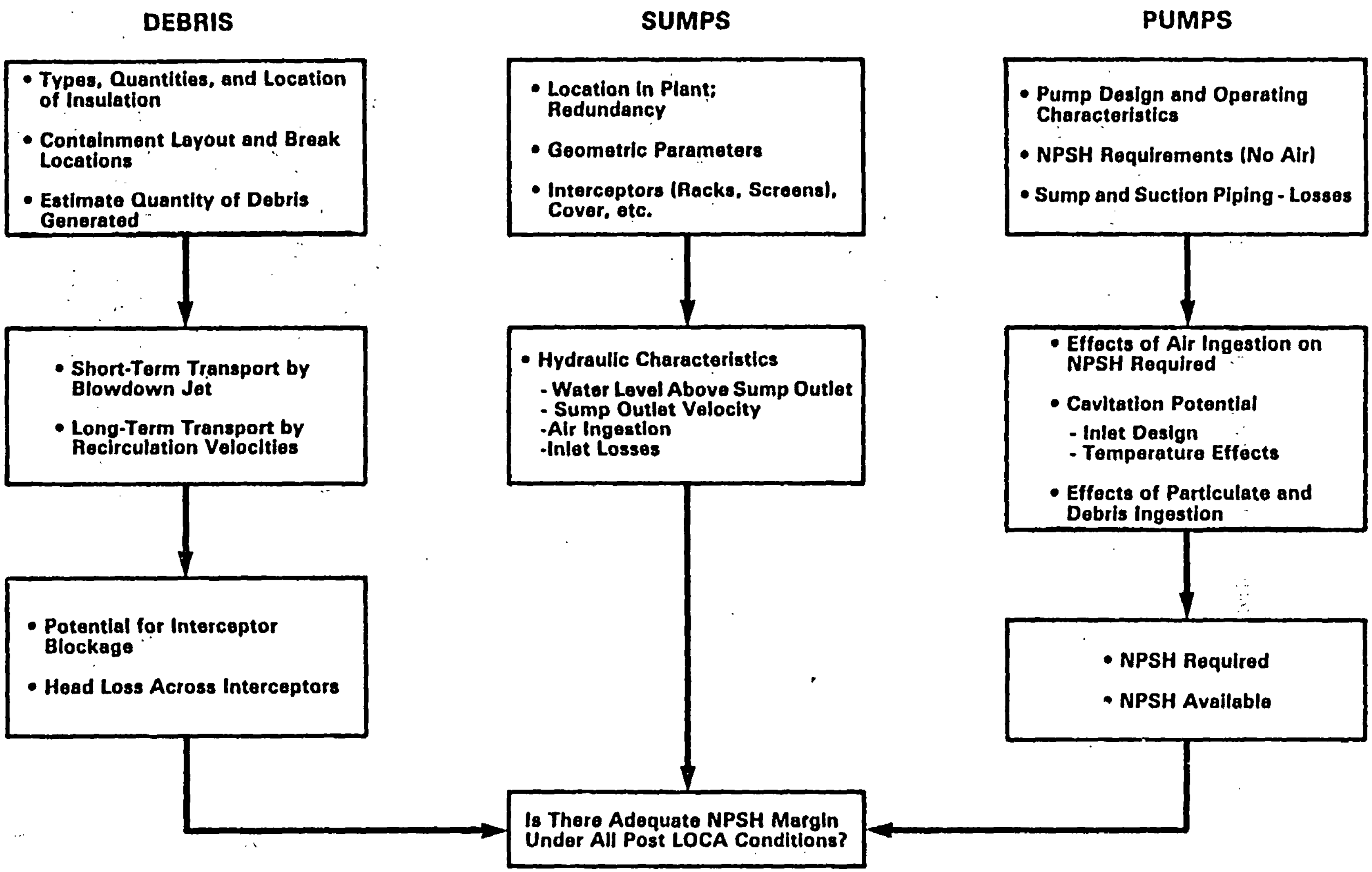


Figure 5.1 Technical considerations relevant to ECCS sump performance

PWR sump hydraulic performance can, therefore, be assessed as follows:

(1) Table 5.1 summarizes the conditions for PWR type sump designs where negligible (or zero) air ingestion would exist. Adequate submergence and low intake velocities are the key parameters derived from ARL tests.

(2) If the adequacy of the sump geometric design and hydraulic performance is to be based on air ingestion levels of $\leq$ 2%, such assessments can be made using Tables 5.2, 5.3, 5.4, and 5.5. Under such conditions, sump design features should be comparable with those sump geometries tested at ARL and as noted in these tables.

(3) Vortex suppressors provide a very effective means to achieve zero air ingestion. Vortex suppression devices such as those shown in Table 5.6 have been shown to reduce air ingestion measured levels to zero on PWR sump designs.

(4) Table 5.7 provides additional information pertinent to screens and grates that could affect PWR sump hydraulic performance and represents the types tested at ARL.

(5) Elevated water temperature has been shown to have negligible effect on sump hydraulic performance in full-scale tests conducted at temperatures up to 165°F.

BWR pump suction intake designs (employing suction strainers) that result in Fr $\leq$ 0.6 were found to have insignificant air ingestion. NUREG/CR-2772 reports experimental findings for Mark I, Mark II, and III intake designs.

5.3 Debris Assessments

Debris assessments should consider the initiating mechanisms (pipe break locations, orientations, and break jet energy content), the amount of debris that might be generated, short- and long-term transport, the potential for PWR sump screen or BWR suction strainer blockage, and head losses that could degrade available NPSH. In addition, an evaluation of the effects of small debris (or particulates) that can pass through screens or strainers should be made. Particulate effects on bearing and seal systems should be evaluated. Table 5.8 outlines key considerations requiring evaluation.

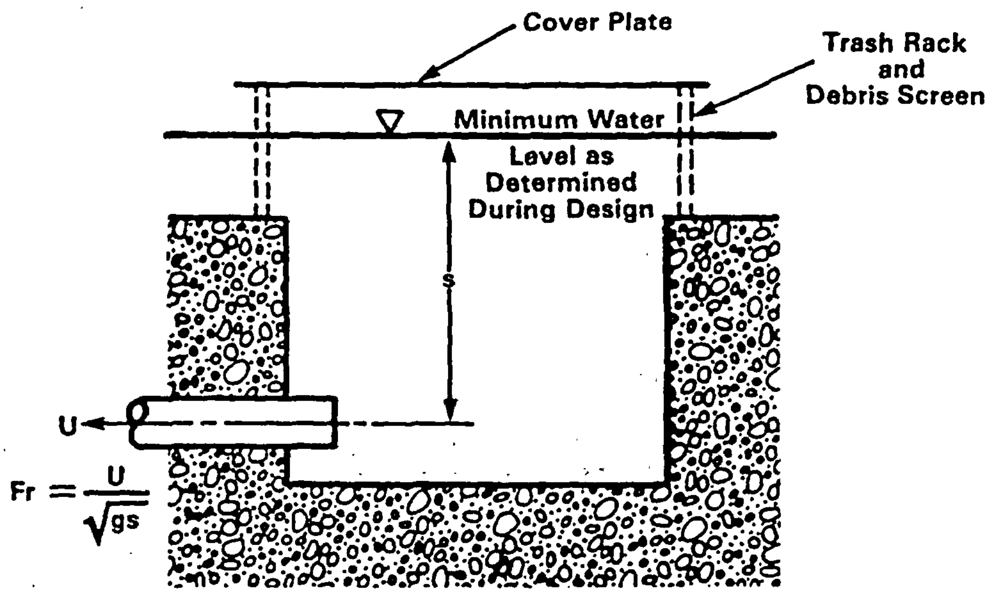
To evaluate potential debris effects, the following information is needed:

(1) Identification of major break locations (per SRP 3.6.2) and jet energy levels.

(2) Types and quantities of insulations employed, and methods of fabrication and installation (i.e., mechanical attachments). Material characteristics of the insulations utilized are important for determining transport and head loss characteristics. The primary and secondary system piping, reactor pressure vessel, and major components (PWR steam generators, reactor

Table 5.1 Hydraulic design findings* for zero air ingestion

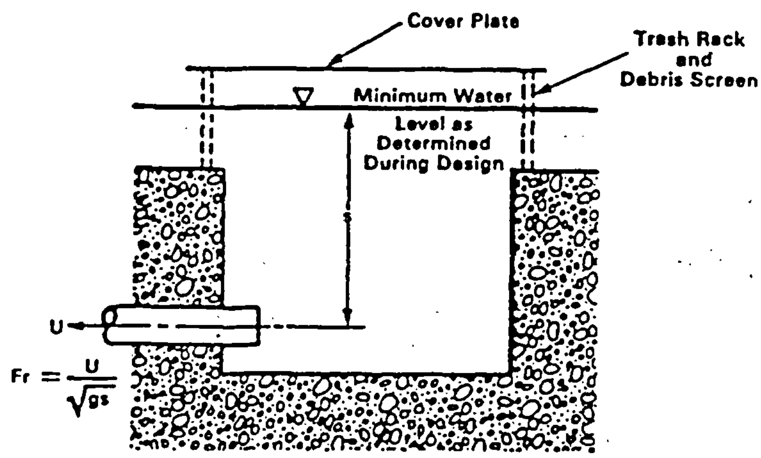
Item	Horizontal Outlets	Vertical Outlets
Minimum submergence, s (ft) (m)	9 2.7	9 2.7
Maximum Froude Number, Fr	0.25	0.25
Maximum Pipe Velocity, U (ft/s) (m/s)	4 1.2	4 1.2



*The hydraulic findings were established using experimental results from NUREG/CRs-2758, -2759, and -2760, and the variable ranges over which such data were taken for sump geometries which were of rectilinear design.

Table 5.2 Hydraulic design findings* for air ingestion $\leq$ 2%

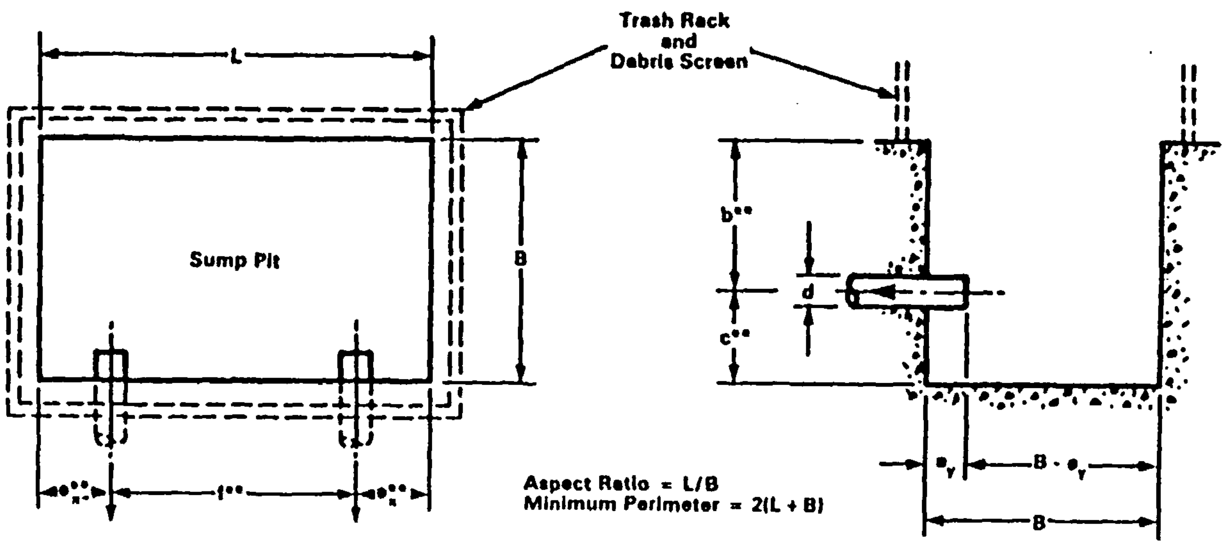
Item	Horizontal Outlets		Vertical Outlets	
	Dual	Single	Dual	Single**
Coefficient α_o	-2.47	-4.75	-4.75	-9.14
Coefficient α_1	9.38	18.04	18.69	35.95
Minimum Submergence, s (ft) (m)	7.5 2.3	8.0 2.4	7.5 2.3	10 3.1
Maximum Froude Number, Fr	0.5	0.4	0.4	0.3
Maximum Pipe Velocity, U (ft/s) (m/s)	7.0 2.1	6.5 2.0	6.0 1.8	5.5 1.7
Maximum Screen Face Velocity (blocked and minimum submergence) (ft/s) (m/s)	3.0 0.9	3.0 0.9	3.0 0.9	3.0 0.9
Maximum Approach Flow Velocity (ft/s) (m/s)	0.36 0.11	0.36 0.11	0.36 0.11	0.36 0.11
Maximum Sump Outlet Coefficient C_L	1.2	1.2	1.2	1.2



*See note on Table 5.1. Air ingestion α is empirically calculated as $\alpha = \alpha_o + \alpha_1 \times Fr$
where α_o and α_1 are coefficients derived from test results as given in the table below
**These numbers are not from test data, but are extrapolated.

Table 5.3 Geometric design envelope guidelines for horizontal suction outlets +

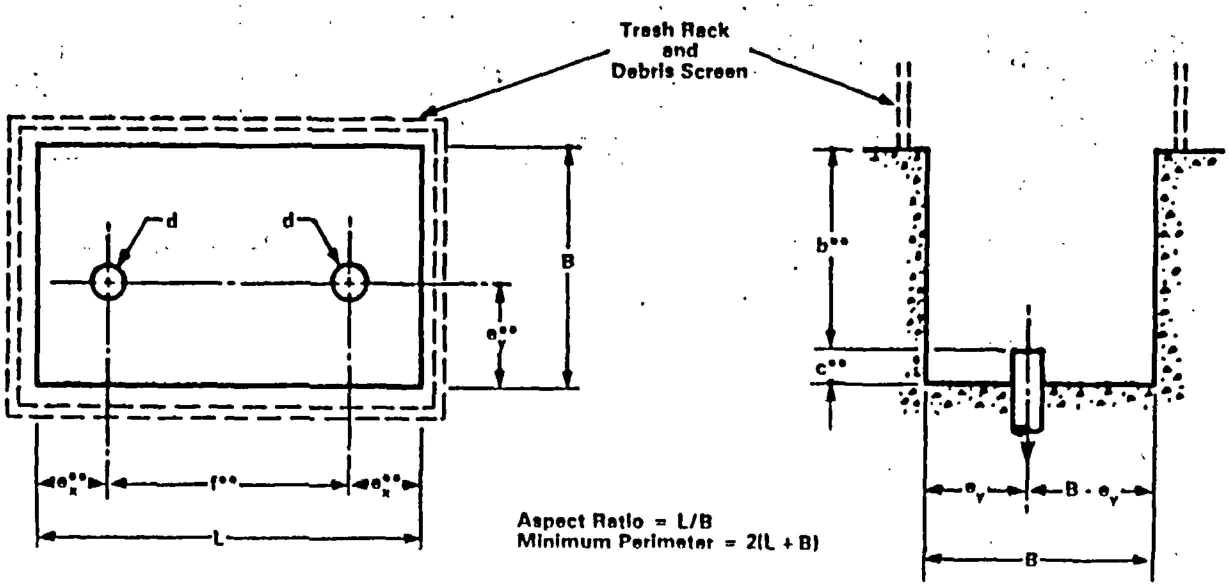
<table>
<tr><th rowspan="2">Sump Outlet</th><th colspan="3">Size</th><th colspan="6">Sump Outlet Position*</th><th colspan="2">Screen</th></tr>
<tr><th>Aspect Ratio</th><th>Min. Perimeter (ft)</th><th>(m)</th><th>e_y/d</th><th>$(B - e_y)/d$</th><th>c/d</th><th>b/d</th><th>f/d</th><th>e_x/d</th><th>Min. Area (ft^2)</th><th>(m^2)</th></tr>
<tr><td>Dual</td><td>1 to 5</td><td>36</td><td>11</td><td rowspan="2">≥ 1</td><td rowspan="2">≥ 3</td><td rowspan="2">≥ 1.5</td><td rowspan="2">≥ 1</td><td>≥ 4</td><td rowspan="2">≥ 1.5</td><td>75</td><td>7</td></tr>
<tr><td>Single</td><td>1 to 5</td><td>16</td><td>4.9</td><td>-</td><td>35</td><td>3.3</td></tr>
</table>



+Dimensions are always measured to pipe centerline.
*Preferred location.

Table 5.4 Geometric design envelope guidelines for vertical suction outlets +

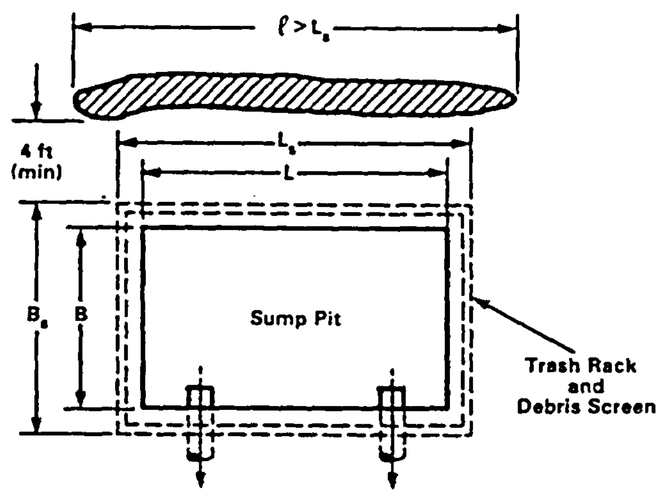
Sump Outlet	Size: Aspect Ratio	Size: Min. Perimeter (ft)	Size: Min. Perimeter (m)	Sump Outlet Position*: e_y/d	$(B - e_y)/d$	c/d	b/d	f/d	e_x/d	Screen: Min. Area (ft^2)	Screen: Min. Area (m^2)
Dual	1 to 5	36	11			≥ 0		≥ 4		75	7
				≥ 1	≥ 1		≥ 1		≥ 1.5		
Single	1 to 5	16	4.9			≤ 1.5		-		35	3.3



+Dimensions are always measured to pipe centerline.
*Preferred location.

Table 5.5 Additional considerations related to sump size and placement

1.	The clearance between the trash rack and any wall or obstruction of length ℓ equal to or greater than the length of the adjacent screen/grate (B_s or L_s) should be at least 4 feet (1.2 m).
2.	A solid wall or large obstruction may form the boundary of the sump on one side only (the sump must have three sides open to the approach flow).
3.	These additional considerations are provided to ensure that the experimental data boundaries (upon which Tables 5.1, 5.2, 5.3, and 5.4 are based) resulting from the experimental studies at ARL are noted.



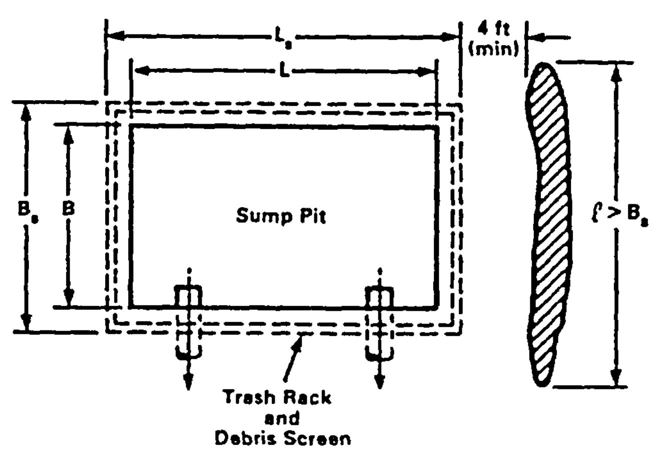
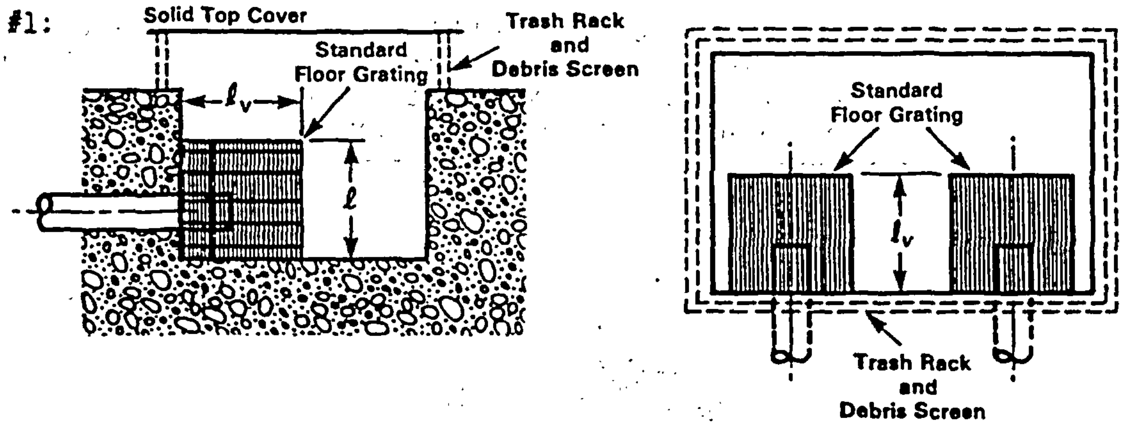


Table 5.6 Findings for selected vortex suppression devices*

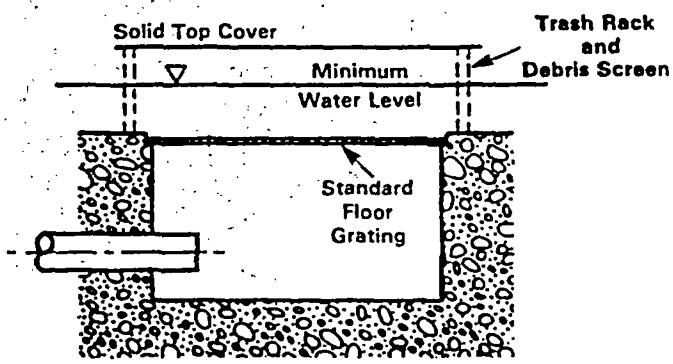
1. Cubic arrangement of standard 1-1/2-inch (38-mm) deep or deeper floor grating (or its equivalent) with a characteristic length, ℓ_v, that is $\geq$ 3 pipe diameters and with the top of the cube submerged at least 6 inches (15.2 cm) below the minimum water level. Noncubic designs with $\ell_v \geq$ 3 pipe diameters for the horizontal upper grate and satisfying the depth and distances to the minimum water level given for cubic designs are acceptable.

2. Standard 1-1/2-inch (38-mm) or deeper floor grating (or its equivalent) located horizontally over the entire sump and containment floor inside the screens and located between 3 inches (7.6 cm) and 12 inches (30 cm) below the minimum water level.

Design #1:



Design #2:



*Tests on these types of vortex suppressors at ARL have demonstrated their capability to reduce air ingestion to zero even under the most adverse conditions simulated.

Table 5.7 Screen, trash rack, and cover plate design findings*

1. Minimum plane face screen area should be obtained from Tables 5.3 and 5.4.

2. Minimum height of open screen (debris interceptors) should be 2 feet (0.61 m).

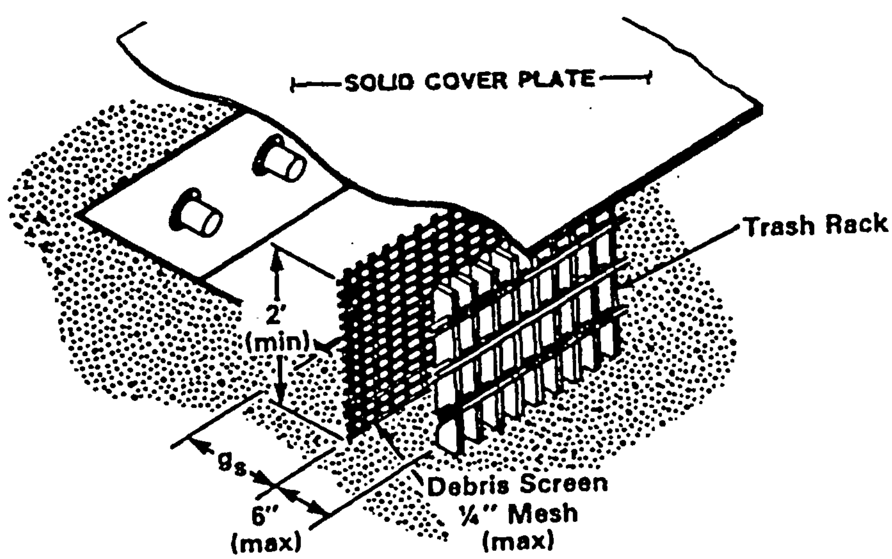
3. Distance from sump side to screens, g_s, may be any reasonable value.

4. Screen mesh should be 1/4 inch (6.4 mm) or finer.

5. Trash racks should be vertically oriented 1- to 1-1/2-inch (25- to 38-mm) standard floor grate or equivalent.

6. The distance between the screens and trash racks should be 6 inches (15.2 cm) or less.

7. A solid cover plate should be mounted above the sump and should fully cover the trash rack. The cover plate should be designed to ensure the release of air trapped below the plate (a cover plate located below the minimum water level is preferable).



*These design findings are based on full-scale tests conducted at the Alden Research Laboratory.

Table 5.8 Debris assessment considerations*

Consideration	Evaluate
(1) Debris generator (pipe breaks and location as identified in SRP Section 3.6.2)	Major pipe breaks and location Pipe whip and pipe impact Break jet expansion envelope (the (major debris generators)
(2) Expanding jets	Jet expansion envelope Piping and plant components targeted (e.g., steam generators) Jet forces on insulation Insulation that can be destroyed or dislodged by blowdown jets Sump and suction structures (i.e., screens), survivability under jet loading
(3) Short-term debris transport (by blowdown jet forces)	Jet/equipment interaction Jet/crane wall interaction Sump location relative to expanding break jet
(4) Long-term debris transport (transport to the sump during the recirculation phase)	Containment layout and sump (or suction) Locations Physical characteristics of debris Recirculation velocity Debris transport velocity
(5) Screen (or suction intake) blockage effects (impairment of flow and/or NPSH margin)	Screen (or suction strainer) area Water level under post-LOCA conditions Recirculation flow requirements Head loss across blocked screen or suction intakes
Key elements for assessment of debris effects	Estimated amount and type of debris that can reach sump Predicted screen (or suction) blockage ΔP across blocked screens or suction intakes NPSH required vs. NPSH available

*Per debris estimation methods described in Section 3.3

coolant pumps, pressurizer, tanks, etc.) that can become targets of expanding jet(s) identified under Item (1) are important in assessing debris generation. For BWRs, the feedwater and recirculation piping and the steamlines are important in assessing potential debris generation.

(3) Containment plan and elevation drawings showing high energy line piping runs, system components, and the piping that are sources of insulation debris should be reviewed. Structures and system equipment that become obstructions to debris transport as well as sump location(s), are important. Drawings showing PWR sump design and debris screen details are needed; for BWRs, drawings of downcomer inlet design (from drywell to wetwell), and RHR suction inlet and debris strainer design details are needed.

(4) Expected containment water levels and recirculation velocities during the post-LOCA recirculation period are needed to assess debris transport and NPSH effects (see Appendix D).

Generic findings regarding debris that might be generated, transported, and lodged against sump screens (and the plant-specific dependence of these phenomena) are discussed in Section 3.3. The following paragraphs summarize the findings.

Break locations, type and size of breaks, and break jet targets are major factors to consider in the estimation of potential quantities of debris generated. The break jet is a high energy, two phase expansion that is capable of shredding insulation and insulation coverings into small pieces or fibers by producing high impingement pressures and large jet loads.

If the PWR sump location can be directly targeted by an expanding break jet, a close examination should be made of possible jet load damage to such insulations at that location and their possible prompt transport to the sump; jet loads on sump screens, etc., also should be evaluated.

Low density insulations, such as calcium silicate and Unibestos™, that have closed cell structures can float. Thus, they are unlikely to impede flow through screens if water levels are above screen height. Partially submerged screens should, however, be evaluated for pulldown of floating debris (Uzuner, 1977). Low density hygroscopic insulations that, when wet, have submerged densities greater than water require plant-specific determinations of screen (or strainers) blockage effects.

Fibrous insulations (such as mineral wool and fiberglass materials) that are transported at low velocities have been shown to present the possibility for total screen blockages (NUREG/CR-2982). Even if these materials are deposited onto screens in layers of relatively small thickness (on the order of 1 inch or less), high pressure drops can result. The potential for screen blockage can be calculated using the methods provided in Sections 3.3.5, 3.3.6, and 3.3.7.

The methods for debris assessment noted above should also be reviewed in light of Appendices E and F. Appendix E provides information received from Diamond Power Company about HDR test results on MIRROR™ insulation performance during

LOCA conditions. Appendix F provides information received from Owens-Corning about HDR blowdown tests with NUKON™ insulation blankets. (The NRC staff response to this information is included in Appendix A.)

5.4 Pump Performance Under Adverse Conditions

The pump industry historically has determined NPSH requirements for pumps on the basis of a percentage of degradation in performance. The percentage is arbitrary, but generally is 1% or 3%. A 2% limit on allowed air ingestion was selected in this review because data show that air ingestion levels exceeding 2% have the potential to produce significant head degradation. Either the 2% limit in air ingestion or the NPSH requirement to limit cavitation may be used independently when the two effects act independently. However, air ingestion levels less than 2% will affect NPSH requirements. In determining these combined effects, the effects of air ingestion on NPSH required must be taken into account.

A calculational method for assessing pump inlet conditions is shown in Figure 5.2. For a given sump design, the following procedure can be followed:

(1) Determine the static water pressure at the sump suction pipe after debris blockage effects have been evaluated (see Section 5.3.). For PWRs, the water level in the sump should not be so low that a limiting critical water depth occurs at the sump edge in a way that flow is restricted into the sump.

(2) Assess the potential level of air ingestion (see Table 5.2) using the criteria in Section 5.2.

(3) Determine pressure losses between suction pipe inlet and pump inlet flange for the required RHR and CSS flows. If the pump inlet is located less than 14 pipe diameters from the suction pipe inlet, the effect of sump-induced swirl should be evaluated (see Section 3.4).

(4) Calculate the static pressure at the pump inlet flange. Static pressure is equal to containment atmospheric pressure plus the hydrostatic pressure due to pump elevation relative to sump or suppression pool surface level, less pressure losses and the dynamic pressure due to velocity. Note that no credit is allowed for containment overpressure, per SRP Section 6.2.2.

(5) Calculate the air density at the pump inlet, then calculate the air-volume flow rate at the pump inlet, incorporating the density difference from suction pipe to the pump.

(6) If the calculated air ingestion is found to be $\leq$ 2%, proceed to Step 7. If the calculated air ingestion is greater than 2%, reassess the sump design and operation per Section 5.1.

(7) Calculate the NPSH available.

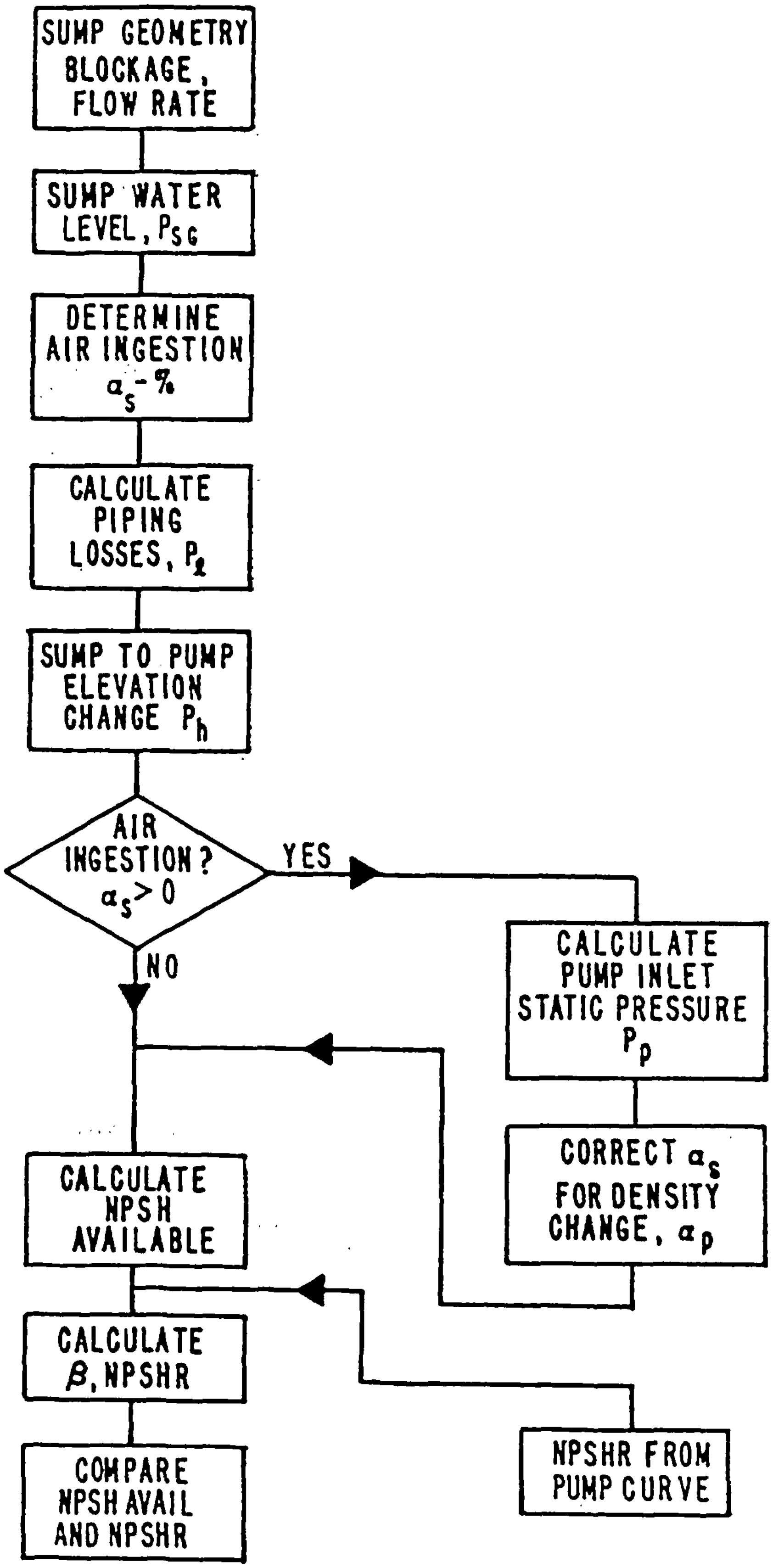


Figure 5.2 Flow chart for calculation of pump inlet conditions

(8) If air ingestion is indicated, correct the NPSH requirement from the manufacturer's pump curves by the following relationship:

$$NPSH_{required}(air/water) = NPSH_{required}(water) \times \beta$$

where

$$\beta = 1 + 0.5\ \alpha_p$$

and α_p is the air ingestion rate (in percent by volume) at the pump inlet flange.

(9) If the NPSH available from Step 7 is greater than the NPSH requirement from Step 8, inlet considerations will be satisfied.

If the above review procedure leads to the conclusion that an inadequate NPSH margin exists, further plant-specific discussions must be undertaken with the applicant/licensee for resolution of differences, uncertainties in calculations, plant layout details, etc. The lack of credit for containment overpressure should be recognized as a conservatism that should be assessed on a plant-specific basis.

In addition, small particulate (or debris) ingestion should be evaluated to assess pump bearing and seal design effects. Small particulates (which can pass through PWR screens or BWR suction strainers) should be assessed for adverse impacts on pump operation and pump bearings.

5.5 Combined Effects

The findings summarized in Sections 5.2, 5.3, and 5.4 can be combined as shown in Figure 5.3 for determination of adequate sump performance. This sequence is straight forward; it begins with assessing air ingestion potential, followed by assessing debris blockage effects on NPSH margins, and concluding with pump performance under post-LOCA conditions.

To facilitate first round, or scoping evaluations, the following guidance is provided:

(1) Air Ingestion Potential

(a) If submergence > 10 feet, intake velocity < 4 ft/sec, and Fr number < 0.25, $\alpha = 0$ (see Table 5.1).

(b) If $\alpha > 2\%$ (see Table 5.2), vortex suppressors should be considered to reduce α to 0 (see Table 5.6).

(2) Debris Blockage Potential

(a) If recirculation flow velocities are low ($\leq$ 0.15 ft/sec), transport of any debris is highly unlikely (see Table 5.9 for a scoping assessment).

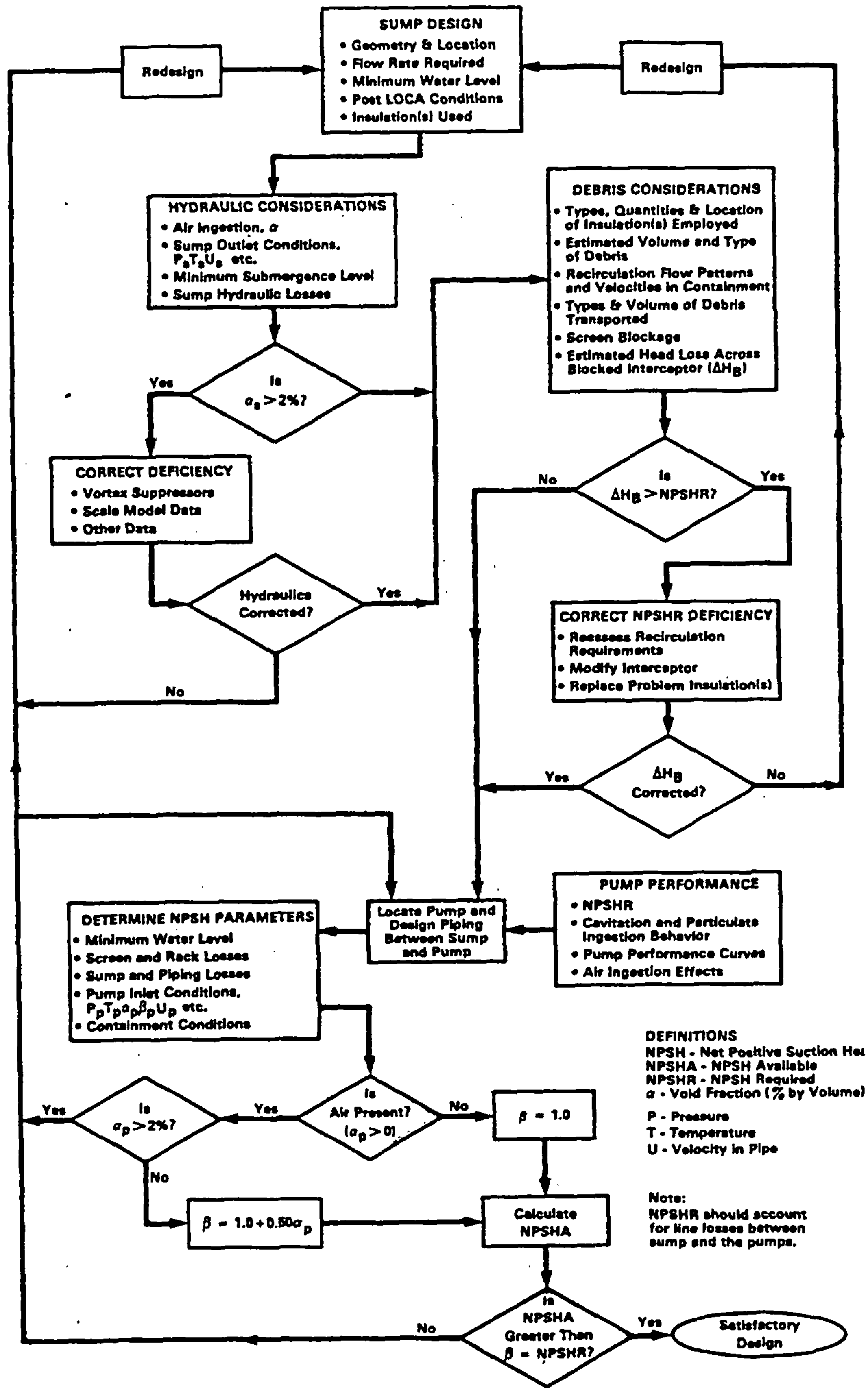


Figure 5.3 Combined technical considerations for sump performance

Table 5.9 First round assessment of screen blockage potential

Criteria for "Zero" Potential for Screen Blockage			
	Criteria 1	Criteria 2	Criteria 3
V_{fb}	0	0	> 0
V_{rm}	0	> 0	any value
V_{cc}	any value	any value	any value
V_{hg}	0	0	0
U_f	any value	$\leq$ 0.2 ft/sec	$\leq$ 0.15 ft/sec
H_w	$\geq H_s$	$\geq H_s$	$\geq H_s$

V_{fb} = volume of fibrous insulation employed
V_{rm} = volume of reflective metallic insulation employed
V_{cc} = volume of closed cell insulation with a specific gravity less than 1.0 (for $H_w \geq H_s$) this insulation will float on water surface above the sump.
V_{hg} = volume of hygroscopic insulation employed
H_w = water level at sump screen
H_s = sump screen height
U_f = flow velocity at the screen based upon the smaller of (1) the screen area that is shielded from prompt transport of insulation and below the minimum water level or (2) the smallest immediate, total approach-flow-area to the screens/grates below the minimum water level.

(b) When considerable quantities of fibrous (i.e., fiberglass) insulation are employed, the significance of potential blockage can be quickly scoped by assuming material within the 7 L/D cone envelope (see Figure 3.25) is totally destroyed and that debris volume is transported to the debris screen. Because fibrous debris blockage head losses (see Section 3.3.5) are a power function such as

$$\Delta H_B = a\ U^b t^c \qquad \text{Equation (1)}$$

which can be rewritten as

$$\Delta H_B = a\ (Q/A)^b (V/A)^c \qquad \text{Equation (2)}$$

where

Δ_B = head loss across blocked screen

Q = recirculation flow rate

A = effective (wetted) screen area

V = volume of fibrous debris transported to debris screen and distributed uniformly thereon

Therefore, a quick assessment of the head loss across a blocked screen area can be made and compared with the NPSHR. Figures 5.4 and 5.5 provide plots of transported debris volumes versus blockage head loss for high density and low density fiberglass debris and are based on experimentally derived head loss data for specific materials (see Section 3.3.6). Material density dependence is illustrated by these figures and necessitates obtaining similar correlation for other materials used. Thus, if a prior assumption is made that total transport occurs and the blocked screen calculated head loss is within NPSH margins, the most conservative calculation has been made.

If unacceptable screen blockage losses are calculated, more extensive evaluations, such as outlined in Figure 5.6, will be necessary.

(c) Reflective metallic insulation debris and associated blockage effects should be evaluated on a plant-specific basis utilizing the debris considerations and findings discussed in Sections 3.3.4 and 3.3.5.

(d) Combinations of insulations are more difficult to assess (see Section 3.3.7) and require estimating combined blockage effects.

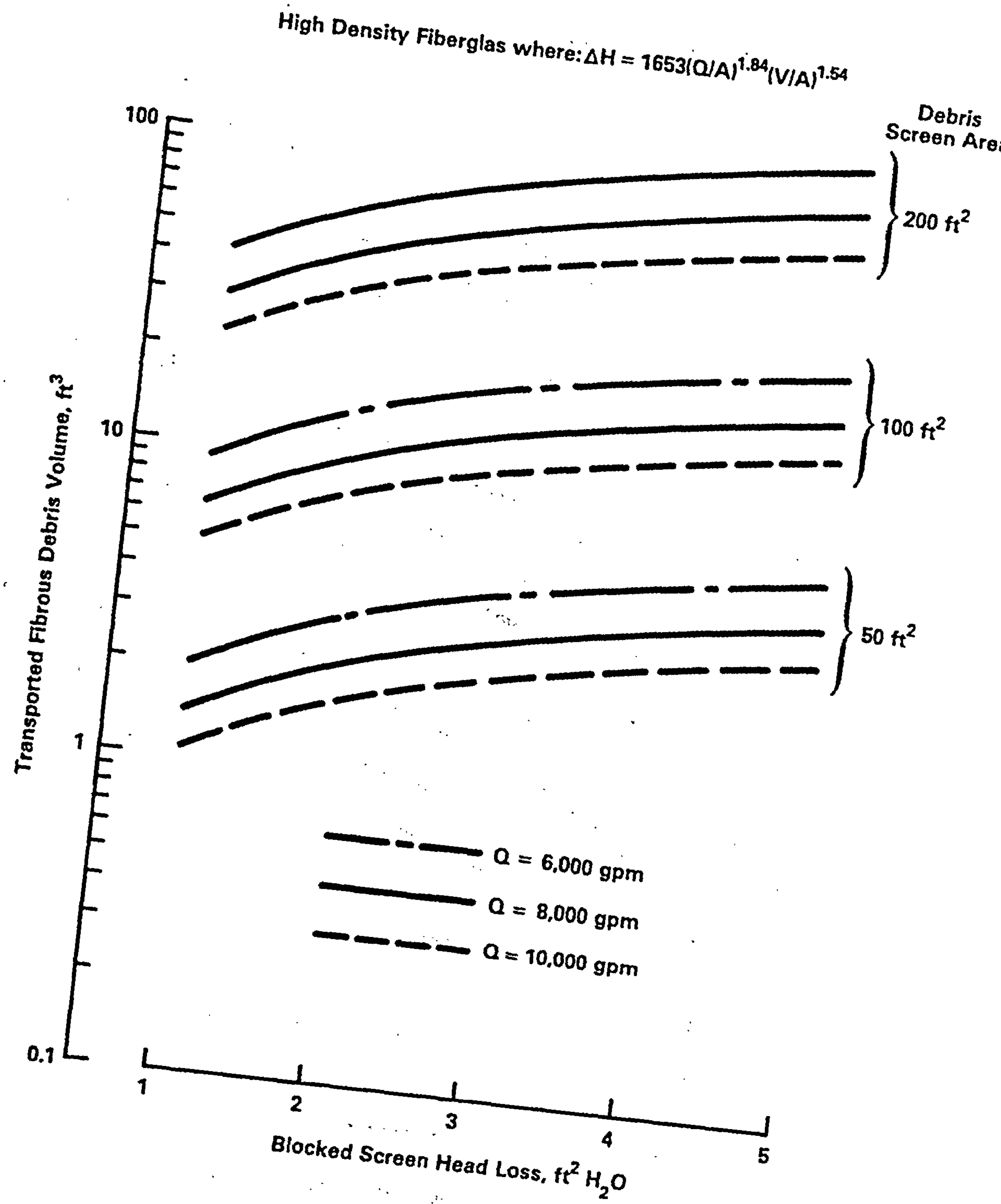


Figure 5.4 Debris volume versus debris screen area, recirculation flow rate, and blocked screen head loss, for high density fiberglass

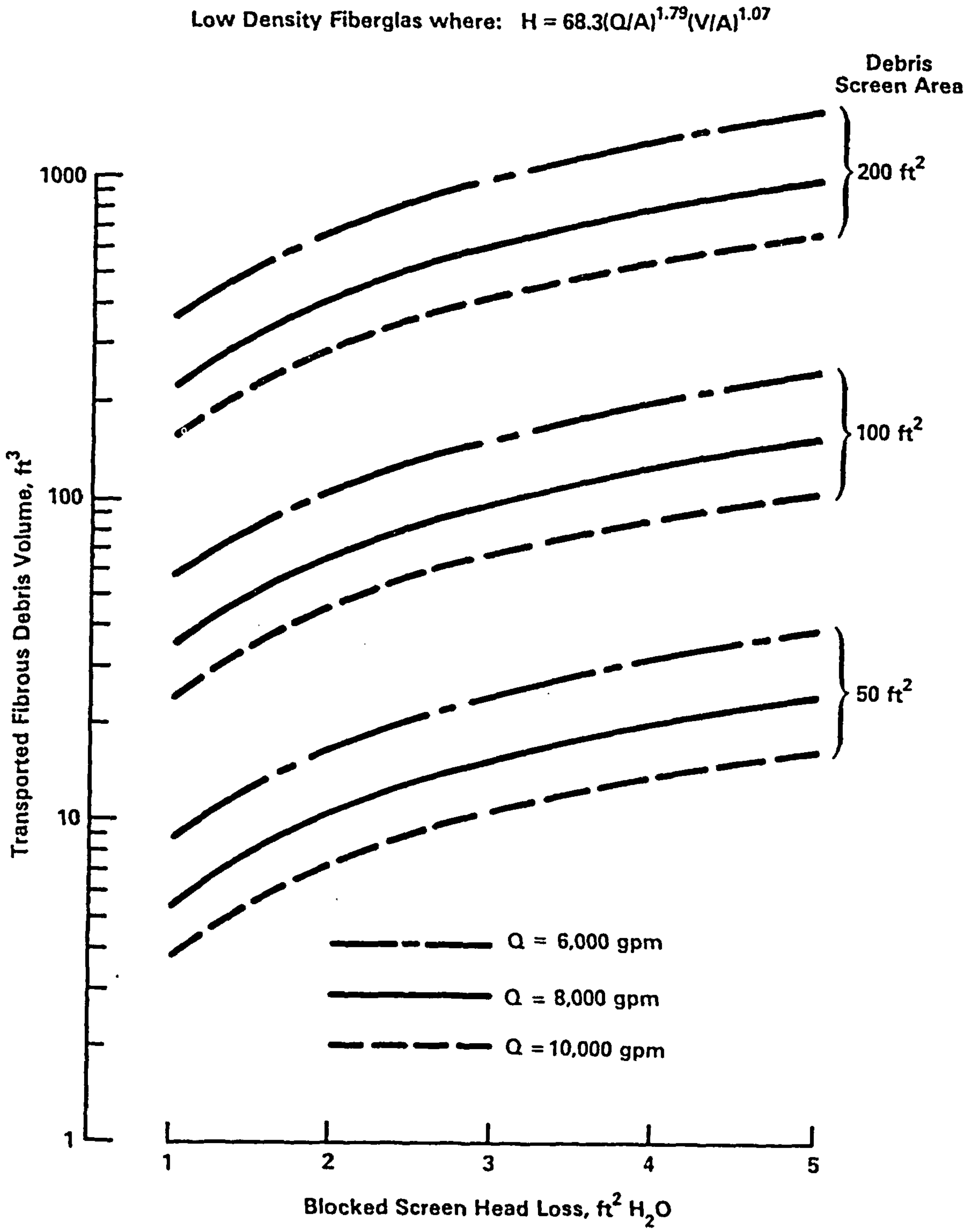
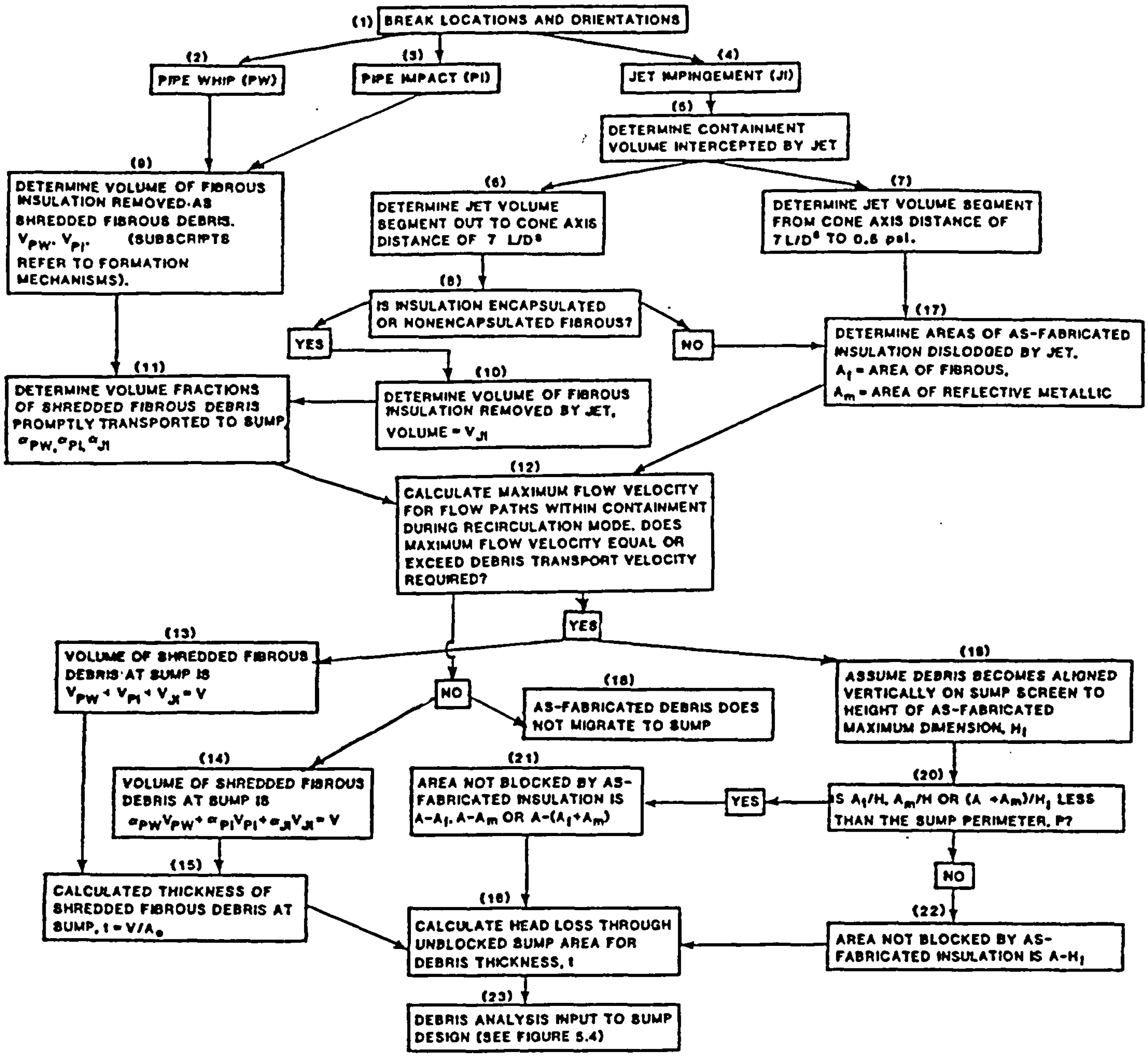


Figure 5.5 Debris volume versus debris screen area, recirculation flow rate, and blocked screen head loss, for low density fiberglass



V_{PW} - VOLUME OF SHREDDED FIBROUS INSULATION REMOVED BY PIPE WHIP. (FT^3)
V_{PI} - VOLUME OF SHREDDED FIBROUS INSULATION REMOVED BY PIPE IMPACT. (FT^3)
V_{JI} - VOLUME OF SHREDDED FIBROUS INSULATION REMOVED BY JET IMPINGEMENT. (FT^3)
α_{PW} - FRACTION OF VOLUME OF SHREDDED INSULATION CAUSED BY PIPE WHIP PROMPTLY TRANSPORTED TO SUMP.
α_{PI} - FRACTION OF VOLUME OF SHREDDED INSULATION CAUSED BY PIPE IMPACT PROMPTLY TRANSPORTED TO SUMP.
α_{JI} - FRACTION OF VOLUME OF SHREDDED INSULATION CAUSED BY JET IMPINGEMENT PROMPTLY TRANSPORTED TO SUMP.
L/D - RATIO OF JET LENGTH TO PIPE DIAMETER.
V - TOTAL VOLUME OF SHREDDED DEBRIS TRANSPORTED TO SUMP SCREEN. (FT^3)
A_f - AREA OF AS-FABRICATED FIBROUS INSULATION DISLODGED BY JET. (FT^3)
A_m - AREA OF AS-FABRICATED REFLECTIVE METALLIC INSULATION DISLODGED BY JET. (FT^2)
A - AREA OF SUMP SCREEN. (FT^2)
A_e - EFFECTIVE UNBLOCKED SUMP SCREEN AREA (AREA AVAILABLE FOR FLOW) (FT2)
H_I - MAXIMUM LINEAR DIMENSION OF AS-FABRICATED INSULATION. (FT)
P - PERIMETER OF EFFECTIVE SUMP SCREEN. (FT)
t - CALCULATED THICKNESS OF SHREDDED DEBRIS MAT ON SUMP SCREEN. (IN)

Figure 5.6 Flow chart for the determination of insulation debris effects

6 REFERENCES

U.S. Nuclear Regulatory Commission Documents

Information supplied during the public comment period on NUREG-0897, 1983 (available in the NRC Public Document Room, 1717 H Street, NW, Washington, DC 20555).

Information supplied by insulation companies during the public comment period on NUREG-0897, 1983 (available in the NRC Public Document Room, 1717 H Street, NW, Washington, DC 20555).

NUREG/CR-2403, "Survey of Insulation Used in Nuclear Power Plants and the Potential for Debris Generation," R. Reyer et al., Burns and Roe, Inc., October 1981.

NUREG/CR-2403, Supplement 1, "Survey of Insulation Used in Nuclear Power Plants and the Potential for Debris Generation," R. Kolbe and E. Gahan, Burns and Roe, Inc., May 1982.

NUREG/CR-2758, "A Parametric Study of Containment Emergency Sump Performance," G. G. Weigand et al., July 1982 (also Sandia National Laboratory, SAND-82-0624 and Alden Research Laboratory ARL-46-82).

NUREG/CR-2759, "Results of Vertical Outlet Sump Tests," Alden Research Laboratory/Sandia National Laboratory, joint report, September 1982 (also Alden Research Laboratory, ARL-47-92 and Sandia National Laboratory, SAND-82-1286).

NUREG/CR-2760, "Assessment of Scale Effects on Vortexing, Swirl, and Inlet Losses in Large Scale Sump Models," M. Padmanabhan and G. E. Hecker, Alden Research Laboratory, June 1982.

NUREG/CR-2761, "Results of Vortex Suppressor Tests, Single Outlet Sump Tests, and Miscellaneous Sensitivity Tests," M. Padmanabhan, Alden Research Laboratory, September 1982.

NUREG/CR-2772, "Hydraulic Performance of Pump Suction Inlet for Emergency Core Cooling Systems in Boiling Water Reactors," M. Padmanabhan, Alden Research Laboratory, June 1982.

NUREG/CR-2791, "Methodology for Evaluation of Insulation Debris Effects," J. J. Wysocki et al., Burns and Roe, Inc., September 1982.

NUREG/CR-2792, "An Assessment of Residual Heat Removal and Containment Spray System Pump Performance Under Air and Debris Ingesting Conditions," P. Kamath, T. Tantillo, and W. Swift, Creare, Inc., September 1982.

NUREG/CR-2913, "Two Phase Jet Loads," G. G. Weigand, S. L. Thompson, and D. Tomasko, Sandia National Laboratory, January 1983 (also Sandia National Laboratory, SAND-82-1935).

NUREG/CR-2982, Revision 1, "Buoyancy, Transport, and Head Loss of Fibrous Reactor Insulation," D. N. Brocard, Alden Research Laboratory, July 1983 (also Sandia National Laboratory, SAND-82-7205).

NUREG/CR-3170, "The Susceptibility of Fibrous Insulation Pillows to Debris Formation Under Exposure to Energetic Jet Flows," W. W. Durgin and J. Noreika, Alden Research Laboratory, January 1983 (also Sandia National Laboratory, SAND-83-7008).

NUREG/CR-3394, "Probabilistic Assessment of Recirculation Sump Blockage Due to Loss of Coolant Accident," J. J. Wysocki, Burns and Roe Inc., July 1983 (also Sandia National Laboratory, SAND-83-7116).

NUREG/CR-3616, "Transport and Screen Blockage Characteristics of Reflective Metallic Insulation Materials," D. N. Brocard, December 1983 (also Alden Research Laboratory, ARL-124-83 and Sandia National Laboratory, SAND-83-7471).

Other Documents

Brocard, D. N., "Transport and Head Loss Tests of Owens Corning Nukon™ Fiberglass Insulation," Alden Research Laboratory, ARL-110-83/M489F, Holden, MA, September 1983.

Durgin, W. W., and J. F. Noreika, "The Susceptibility of Nukon™ Insulation Pillows to Debris Formation Under Exposure to Energetic Jet Flows," Alden Research Laboratory, ARL-111-83/M489F, Holden, MA, September 1983.

Durgin, W. W., M. Padmanabhan, and C. R. Janik, "The Experimental Facility for Containment Sump Reliability Studies, NRC Generic Task A-43," Alden Research Laboratory, ALO-132, Holden, MA, 1980.

Florjancic, D., "Influence of Gas and Air Admission on the Behavior of Single- and Multi-Stage Pumps," Sulzer Research Number 1970, Sulzer Brothers, Ltd., Winterthur, Switzerland, 1970.

Glasstone, S., "Effects of Nuclear Weapons," Superintendent of Documents, U.S. Government Printing Office, Washington, D.C., 1981.

Hydraulic Institute, Hydraulic Institute Standards for Centrifugal, Rotary and Reciprocating Pumps, 13th edition, Cleveland, OH, 1975.

Imatran Voima Oy, "Model Tests of the Loviisa Emergency Cooling System," Report No. 275, August 1976, Civil Engineering Department, Construction Laboratory, Finland.

---, "Model Tests of Containment Sumps of the Emergency Core Cooling System," Report No. 291, April 1980, Civil Engineering Department, Construction Laboratory, Finland.

Merry, H., "Effects of Two-Phase Liquid/Gas Flow on the Performance of Centrifugal Pumps," Institution of Mechanical Engineers Conference on Pumps and Compressors, Paper No. C130/76, 1976, 1, Birdcage Walk, London, S.W. 1H 9JJ U.K.

Moody, F. J., "Fluid Reaction and Impingement Loads, Specialty Conference on Structural Design of Nuclear Plant Facilities," Vol 1, Chicago, IL, December

1973, "American Society of Civil Engineers, 345 East 47th Street, New York, N.Y. 10017.

Murakami, M. and K. Minemura, "Flow of Air Bubbles in Centrifugal Impellers and its Effect on Pump Performance," presented at sixth Australian Hydraulics and Fluid Mechanics National Conference, Publication No. 77/12, December 1977, Institution of Engineers, Adelaide, Australia.

Niyogi, K. K. and R. Lunt, "Corrosion of Aluminum and Zinc in Containment Following a LOCA and Potential for Precipitation of Corrosion Products in the Sump," United Engineers and Constructors, Inc., Philadelphia, PA, September 1981.

Uzuner, M. S., "Stability Analysis of Floating and Submerged Ice Floes," Proceedings of the American Society of Civil Engineers, 103, HY7, Journal of the Hydraulics Division, pp. 713-722, July 1977.

APPENDIX A

SUMMARY OF PUBLIC COMMENTS RECEIVED AND ACTIONS TAKEN

APPENDIX A

SUMMARY OF PUBLIC COMMENTS RECEIVED AND ACTIONS TAKEN

1 INTRODUCTION

The technical findings related to Unresolved Safety Issue (USI) A-43 were published for comment in May 1983. Notice of the publication was placed in the Federal Register on May 9, 1983. The official comment period lasted for 60 days and ended on July 11, 1983. However, comments were received into September 1983, with followup comments received into November 1983. A listing of those who responded during the period and afterwards is shown in Table 1. Copies of the comment letters are on file in the NRC Public Document Room, 1717 H Street, NW, Washington, DC.

A public meeting was held on June 1 and 2, 1983, at Bethesda, Maryland, to offer additional opportunity for public comments; however, attendance was very small. Followup discussions were held with respondees to clarify issues raised at this meeting and in the written comments.

An overview of the comments received is provided in Section 2 below. Section 3 contains summaries of significant comments and the actions planned to resolve them.

2 OVERVIEW OF COMMENTS RECEIVED

The major written comments received addressed seven specific subject areas. The comment categories and commentors are listed in Table 2 below. The commentors are identified in Table 2 as follows: Alden Research Laboratory (ARL); Atomic Industrial Forum (AIF); BWR Owners Group (BWR); Commonwealth Edison (CEd); Consumers Power Co. (CPC); Creare Research and Development (CRD); Diamond Power Co. (DPC); General Electric (GE); Gibbs and Hill, Inc. (GH); Northeast Utilities (NE); and Owens-Corning Fiberglass, Inc. (OCF). By category, the actions taken in response to these comments are as follows:

Categories 1 and 6: Tables have been added to NUREG-0897, Revision 1 and NUREG-0869, Revision 1 to include the additional plant insulation information provided during the public comment period. The text of the NUREGs has been revised to reflect recommended insulation definitions and the need to evaluate the specific insulation employed.

Categories 2 and 4: The cost estimates provided by different industry groups have varied over a wide range. With the exception of Diamond Power Company, respondees claimed that the cost estimates in value/impact analysis were too low. The revised value/impact analysis reflects an averaged value derived from costs provided.

Category 3: A detailed sump blockage probability analysis has been performed and is reported in NUREG/CR-3394. The results were used in the revised value/impact analysis. These results show a sump blockage probability range for pressurized water reactors (PWRs) of 10^{-6} to 5×10^{-5}/Rx-yr and a strong dependence on plant design.

Table 1 Persons who commented on the technical findings related to USI A-43*

Alden Research Laboratory (ARL), M. Padmanabhan, letter to A. Serkiz (NRC), "Comments on NUREG-0897 and 0869," June 13, 1983.

ARL, M. Padmanabhan, letter to A. Serkiz (NRC), "Revision to Table A-3 in NUREG-0869," June 22, 1983.

Atomic Industrial Forum, R. Szalay, letter to the Secretary of the Commission, "NRC's Proposed Resolution of Unresolved Safety Issue A-43, Containment Emergency Sump Performance, Contained in NUREG-0869," July 22, 1983.

Atomic Industrial Forum, J. Cook, letter to R. Purple (NRC) and enclosure "Examples of Staff Review Going Beyond Approved Regulatory Criteria," June 4, 1984.

BWR Owners Group, T. J. Dente, letter to T. P. Speis (NRC), "BWR Owners' Group Comments on Proposed Revision to Regulatory Guide 1.82, Rev. 1," October 18, 1983.

BWR Owners Group, D. R. Helwig, letter to V. Stello (NRC), BWR Owners' Group comments on Regulatory Guide 1.82, Revision 1, July 16, 1984.

Commonwealth Edison, D. L. Farrar, letter to the Secretary of the Commission, "NUREG-0897, Containment Emergency Sump Performance; Standard Review Plan Section 6.2.2, Rev. 4, Containment Emergency Heat Removal Systems; and NUREG-0869, USI A-43 Resolution Positions (48FR2089; May 9, 1983)," July 13, 1983.

Consumers Power, D. M. Budzik, letter to the Secretary of the Commission, "Comments Concerning Regulatory Guide 1.82, Proposed Revision 1 (File 0485.1, 0911.1.5, Serial: 23206)," July 15, 1983.

Creare, W. L. Swift, letter to P. Strom (SNL), "Comments on Figure 3-6 of NUREG-0897 and Table A-9 of NUREG-0869," June 13, 1983.

Diamond Power Company, R. E. Ziegler and B. D. Ziels, letter to K. Kniel (NRC), "Containment Emergency Sump Performance, USI A-43," July 11, 1983.

Diamond Power Specialty Company, B. D. Ziels, letter to A. Serkiz (NRC), "HDR Test Result Summary, MIRROR Insulation Performance During LOCA Conditions," December 6, 1984.

General Electric (GE), J. F. Quirk, letter to K. Kniel (NRC), "Comments on Emergency Sump Documents," July 11, 1983.

GE, J. F. Quirk, letter to T. P. Speis (NRC), "Comments on Proposed Regulatory Guide 1.82, Rev. 1," October 17, 1983.

---

*Including comments on NUREG-0869, NUREG-0897, proposed Revision 1 to Regulatory Guide 1.82, and proposed Revision 4 to Section 6.2.2 of the Standard Review Plan (SRP, NUREG-0800).

Table 1 (Continued)

Gibbs and Hill, Inc., M. A. Vivirito, letter to the Secretary of the Commission, "Comments on Proposed Revision No. 1 to RG 1.82," July 11, 1983.

Northeast Utilities, W. G. Counsil, letter to K. Kniel (NRC), "Haddam Neck, Millstone Nuclear Power Station, Unit Nos. 1, 2, and 3, Comments on NUREG-0897, SRP Section 6.2.2 and NUREG-0869," September 2, 1983.

Owens Corning Fiberglass (OCF), G. H. Hart, letter to A. Serkiz (NRC), "Comments on NUREG-0897 and NUREG-0869," June 23, 1983.

OCF, G. H. Hart, letter to A. Serkiz (NRC), "Updated Comments on NUREG-0897 and NUREG-0869," July 14, 1983.

OCF, G. P. Pinsky, letter to K. Kniel (NRC), "Comments on NUREG-0879 and -0896," July 14, 1983.

OCF, G. H. Hart, transmittal to A. Serkiz (NRC), "HDR Blowdown Tests with NUKON Insulation Blankets," February 18, 1985.

Power Component Systems, Inc., D. A. Leach, letter to A. Serkiz (NRC), "Nuclear Grade Blanket Insulation," November 8, 1984.

Table 2 Categories addressed in major written comments

Comment	Category	ARL	AIF	BWR	CED	CPC	CRD	DPC	GE	GH	NE	OCF
(1)	Survey of insulation used is not current or complete.							X				X
(2)	Cost estimates are low.		X								X	
(3)	Estimates of sump blockage probabilities are high.		X		X	X			X			
(4)	Value-impact analysis questioned.		X		X				X		X	
(5)	BWRs should be exempt; A-43 is a PWR issue.			X					X		X	
(6)	Insulation material definitions and descriptions need revision for clarity and completeness.							X			X	
(7)	Technical comments on and clarifications of subject matter in NUREG-0897 and NUREG-0869.	X			X	X	X	X		X	X	X

By category, the actions taken in response to these comments are as follows:

Categories 1 and 6: Tables have been added to NUREG-0897, Revision 1 and NUREG-0897, Revision 1 to include the additional plant insulation information provided during the public comment period. The text of the NUREGs has been revised to reflect recommended insulation definitions and the need to evaluate the specific insulation employed.

Categories 2 and 4: The cost estimates provided by different industry groups have varied over a wide range. With the exception of Diamond Power Company, respondees claimed that the cost estimates in value/impact analysis were too low. The revised value/impact analysis reflects an averaged value derived from costs provided.

Category 3: A detailed sump blockage probability analysis has been performed and is reported in NUREG/CR-3394. The results were used in the revised value/impact analysis. These results show a sump blockage probability range for pressurized water reactors (PWRs) of 10^{-6} to 5×10^{-5}/Rx-yr and a strong dependence on plant design.

Category 5: NUREG-0869 and Regulatory Guide 1.82 have been revised to specifically identify areas of concern for boiling water reactors (BWRs) and for PWRs.

Category 7: Technical corrections and clarifications have been made in the appropriate sections of NUREG-0897 and NUREG-0869.

The NRC staff greatly appreciates the review and comments provided by the respondees. The time and effort they have taken to review USI A-43 has resulted in an improved report that will reflect current findings and a balanced position with respect to this safety issue.

3 COMMENTS RECEIVED AND PROPOSED ACTION (OR RESPONSE) ACTIONS

The NRC staff has given complete and careful consideration to all comments received on USI A-43. Summaries of significant comments and the actions taken by the NRC staff in response are provided in Table 3. Comments are presented in alphabetical order, based on the name of the commenting institution.

Table 3 Comments received on USI A-43 and NRC staff response

Comment	NRC Staff Response
Alden Research Laboratory	
ARL noted typographical errors and proposed technical clarification to several tables	These corrections and clarifications have been incorporated into NUREG-0897 and NUREG-0869.
Atomic Industrial Forum	
The cost impact of $550,000/plant used in value/impact analysis is low by at least a factor of 2.	Costs impacts were re-evaluated based on cost estimate information received from AIF and other respondees
Economic considerations related to reduced probability of plant damage should be excluded from the cost-benefit balancing. Decisions should be based primarily on the value/impact ratio.	The essence of a value/impact analysis is that it attempts to identify, organize, relate, and make "visible" all the significant elements of value expected to be derived from a proposed regulatory action as well as all significant elements of impact. The net values are compared with the net impacts. Thus if a proposed safety improvement is accompanied by an adverse side effect, the impairment is is subtracted from the improvement to arrive at a net safety value for consideration in the value/impact assessment. Similarly, when the immediate and prospective cost impacts are summed, they should include all elements of economic impact on licensees, such as costs to design, plan, install, test, operate, maintain, etc. Plant downtime or decreased plant availability is included when applicable. The summed impacts, however, should be net impacts, for comparison with net values. Thus, any reductions in operating costs, improvements in plant availability, or reductions in the probability of plant damage are properly a factor in determining net adverse economic impact. Future economic costs and savings are appropriately discounted.

Table 3 (Continued)

Comment	NRC Staff Response
	Qualitative differences among impact elements are respected, and distinctive elements of impact (of which averted plant-damage probability, as a favorable rather than adverse impact, is a prominent example) are separately identified, for appropriate consideration in regulatory decision making. The ratio of avoided public dose to the gross cost of implementation is ordinarily a major decision factor. However, it is not by itself always a good guide to a sound regulatory decision. The issues involved are often too complex for a decision on this criterion alone. Other factors that enter, often in important ways, may include any economic benefits that reduce a net adverse economic impact, the safety importance of the issue, and values and impacts that cannot or cannot readily be quantified; for example, jeopardy to a defense layer in the defense-in-depth concept or expected reductions in plant availability that can be foreseen but not precisely estimated. A sound regulatory decision rests on adequate consideration of all significant factors. An overly simple approach can mislead if it simplifies away complexities that are the essence of the issue at hand.
The assumption that sump failure will occur in the case of 50% of the large LOCAs should be justified.	A detailed sump blockage probability analysis has been performed and is reported in NUREG/CR-3394. The results show a wide range of sump blockage failure probabilities (i.e., 3E-6 to 5E-5/reactor-year) and a high dependency on plant design and operational requirements. These results are reflected in a revised value impact analysis utilizing a range of sump failure probabilities.
The use of PWR release categories from WASH-1400 is too conservative. Containment failure probabilities used in WASH-1400	The containment failure probabilities and release categories used in the regulatory analysis for USI A-43 were based on information presented in WASH-1400, and also on

Table 3 (Continued)

Comment	NRC Staff Response
are inadequate to describe the nuclear industry's present knowledge in this field. Releases due to "vessel steam explosion" are unrealistic and should not be considered.	other considerations. The comments presented by an AIF subcommittee regarding the validity of continued use of WASH-1400 assumptions, etc. are being evaluated through other activities such as: reevaluation of source terms, SASA studies, etc. USI A-43 regulatory analyses were based on the following considerations and for the reasons noted: (1) WASH-1400 assumptions were utilized to provide a common baseline calculations for reference plants and were used to estimate increases in releases due to a postulated loss of recirculation flow capacity. Until revised failure mechanisms and new source terms are determined, this approach provides a consistent set of calculations. (2) Although using a small containment failure probability associated with steam explosion would be more appropriate, release category PWR-1 (which includes steam explosion) was not a dominant contributor to release. Release categories PWR-2, -4, and -6 were the dominant pathways contributing to increases releases due to a failed sump for the plants analyzed. (3) Basing release effects on the assumption of simultaneous failure of core cooling and loss of containment sprays is conservative. If containment were not lost (as would be the situation for PWRs that have dry containments with safety-grade fan cooler systems), the LOCA energy could be disspated without containment overpressurization and failure. Thus releases associated with PWR-2 and -4 categories could be discounted and PWR-6 releases only used. Such considerations have been incorporated into this revised regulatory analysis.

Table 3 (Continued)

Comment	NRC Staff Response
	(4) Other factors--such as containment structural design margins that argue against gross containment failures (as postulated in WASH-1400), realignment to alternate water sources, controlled venting for BWRs, etc.--have also been considered this revised regulatory analysis.
The use of the CRAC Code and a "no-evacuation," 50-mile-radius model to develop public doses is unrealistic.	The 50-mile radius reflects a substantial part (though not all) of the total population dose, and is thus a reasonable index of the radiological effect on the public. Standardization of calculations to that radius is helpful in comparing risks associated with different issues and average such risks for use with the 1000 person-rem/$M criterion. Evacuation of people is not considered because calculations suggest that, although it may sometimes be important for people directly affected, the effect of evacuation on the total population dose is likely to be small.
NRC should utilize information developed more recently (i.e., NUREG-0772) to reassess and reduce the source terms, rather than continue to use the PWR-2 and PWR-3 release categories from WASH-1400.	Possible changes in the source terms are being considered by the special task force established by the Commission to review the source-term issue. Changes would be premature before this group completes its evaluation and the new values are accepted by all parties involved.
NRC should utilize the "leak before break" concept in evaluating the safety significance of A-43.	Elastic-plastic fracture mechanics analysis techniques to analyze pipe break potential has been used in USI A-2, with a limited number of PWRs being analyzed. For USI A-2, the submittal of such analyses for specific break locations (on a plant-specific basis) will require obtaining an exemption from the requirements of GDC4. Submittal of such analyses to address the USI A-43 debris blockage issues would be reviewed by staff on a plant-specific basis, should a licensee or applicant elect to utilize this approach.

Table 3 (Continued)

Comment	NRC Staff Response
<u>BWR Owners Group</u>	
After quick review of the proposed revision to the regulatory guide, the BWR Owners Group and GE maintain that USI A-43 is not a generic issue for BWRs.	The requirement for long-term decay heat removal is applicable to light-water reactors, both BWRs and PWRs.
The revisions to RG 1.82, which now proposes specific criteria for BWRs, should apply only to light-water reactors that have any potential for harmful debris generation (i.e., light water reactors that extensively use fibrous insulation).	All types of insulation should be evaluated for the potential of debris generation, transport, and suction strainer blockage. The wide variation in plant designs and insulation employed does not support a generic statement.
These comments and any future comments by the BWR Owners Group should not substitute for the normal notice and comment procedure that allows potentially affected licensees to respond to proposed regulatory guide changes.	RG 1.82, Revision 1 (along with NUREG-0897, NUREG-0869 and SRP 6.2.2, Revision 4) was issued "for comment" in May 1983. Only 14 responses were received as of September 1983. Some of these comments (in particular GE's July 11, 1983 letter) cited a need to specifically address BWR-related concerns in the RG. This was done and copies were sent to GE and the BWR Owners Group. Given the previous extensive distribution of "for comment" reports and regulatory positions and the rather small number of responses, the staff does not plan to reissue RG 1.82, Revision 1 for comment. The NRC staff will incorporate additional valid technical points received from the BWR Owners Group and GE. The most recent input from the BWR Owners Group (July 16, 1984) does not provide new significant findings; rather this input re-expresses concerns previously voiced and stresses possible misinterpretations of wording in RG 1.82, Revision 1.

Table 3 (Continued)

Comment	NRC Staff Response
<u>Commonwealth Edison</u>	
The Commission has not sufficiently justified the need to impose retrofit requirements on either operating or near-term operating license units.	A-43 resolution does not mandate retrofits; rather, applicants are requested to assess long-term recirculation capability utilizing RG 1.82, Revision 1 and to then determine what corrective actions may be needed. The use of an information bulletin to the majority of the plants does not constitute imposition of a retrofit.
Cost estimates for surveys, design reviews, and retrofitting are questionable.	The A-43 value/impact evaluation has been revised based on detailed sump blockage probability studies (NUREG/CR-3394) and cost estimates received from industry responses.
The proposed RG 1.82 is overly conservative. However, given the need for assurance that the recirculation sump remains a reliable source of cooling water, the commentor agrees that an evaluation of sump designs, potential for debris, air ingestion, and adequate net positive suction head (NPSH) is fully justified.	The NRC staff acknowledges that conservatisms exist in RG 1.82, Revision 1. However, such conservatisms are prompted by the limited amount of available information regarding insulation destruction due to high pressure jets and attendant debris generation, and the wide variability of plant designs and types of insulation used.
The commentor questions the assumption that 50% of LOCAs lead to sump loss, the value/impact ratio given uncertainties in estimated costs, the basis for assuming 23 years remaining plant life, etc.	A detailed sump failure probability analysis was performed and is reported in NUREG/CR-3394. The "averaged" sump failure probability was 2E-5/reactor-year with a range of 3E-6 to 5E-5/reactor-year.
<u>Consumers Power</u>	
Regarding the proposed Revision 1 to RG 1.82, the commentor stated (1) that Appendix A should be clearly delineated as being an information and guidance source, not as presenting design requirements, and (2) that consistency is needed with respect to NPSH terminology.	Appendix A of proposed RG 1.82, Revision 1 was always intended to provide additional information and/or not design requirements. Appendix A has been clearly labeled as such.

Table 3 (Continued)

Comment	NRC Staff Response
Regarding the value/impact analysis, the commentor questioned the assumption that 50% of the loss-of-coolant accidents (LOCAs) lead to sump blockage and cites a sump failure frequency of 2×10^{-4} per demand from another probabilistic risk analysis.	That 50% of LOCAs lead to sump blockage has been reevaluated (see NUREG/CR-3394), and the results of that detailed study have been used in revising the A-43 release estimates.
The commentor questioned the direct application of core melt frequency reduction for computing avoided accident cost. The commentor disagrees with taking credit for loss of plant cost. Rather, the commentor states that loss-of-plant costs should be deducted from avoided accident costs.	The calculation of avoided accidents costs, loss-of-plant costs, etc., are consistent with current NRC staff evaluation practices. Recalculation of the parameters previously used will be carried out with the revised blockage frequencies.
<u>Creare</u>	
The beta factor used to predict a pump's required NPSH in an air/water mixture is based on data whose scatter was not reported. The NUREG should note this and caution the applicant and reviewer to carefully consider the adequacy of the NPSH margin if it is marginal.	Efforts were made to obtain the original data tapes and calculate the data's scatter; however, this information was not readily available. The suggested cautionary note has been added to NUREG-0897.
The use of an arbitrary minimum allowable NPSH margin, either as a fixed value (i.e., 1 foot) or as a percentage value (i.e., 0.5 x margin with no screen blockage), is not justifiable. It should be recognized that what constitutes a safe NPSH margin is a plant-specific judgment.	NUREG-0897 and RG 1.82, Revision 1 no longer recommend a minimum allowable NPSH margin. Instead, they note that whatever NPSH margin is available (after accounting for hydraulic and screen blockage effects) should be evaluated with respect to each plant's long-term recirculation requirements.
<u>Diamond Power Company</u>	
NUREG-0897 resolves a significant safety problem in a thorough and equitable manner.	The NRC staff concurs.

Table 3 (Continued)

Comment	NRC Staff Response
The commentor provides recommendations regarding the classification of various insulating materials, particularly on the need to distinguish between totally encapsulated insulation and jacketed insulation.	The proposed classifications have been combined with other similar proposals to revise and clarify the insulation classification and descriptions used in NUREG-0897.
The commentor provides listings of the types of insulations purchased since 1980 and the types of insulations used in recent retrofittings.	The information has been added to NUREG-0897 and NUREG-0869, along with data received from other manufacturers.
The commentor states that the costs in the value/impact analysis are in agreement with its costs and provides the following figures: Cost of MIRROR®™ reflective metallic insulation = $40/ft^2 for material alone. Installation cost, excluding material = $25/hour. Productivity = 1.24 hours/ft^2 of insulation.	This cost information has been reflected in the revised value/impact analysis (NUREG-0869), along with other industry cost figures.
Reflective metallic insulation is not the predominant type of insulation used in newer plants. Recently insulated plants mainly use fiberglass insulation.*	Information supplied by Owens-Corning Fiberglass Co. and the Diamond Power Co. regarding types of insulation used in existing and future reactors has been added to NUREG-0897 and NUREG-0869. These reports have been revised to reflect this new information. The trend appears to be toward a higher utilization of fibrous insulations.

* Letter of July 11, 1983.

Table 3 (Continued)

Comment	NRC Staff Response
A report on "HDR Test Results on MIRROR insulation performance during LOCA conditions was submitted to provide additional information to the existing data base used in resolution of USI A-43.*	This report has been included as Appendix E in NUREG-0897, Rev. 1. The results of this report do not support a hypothesis which postulates free and undamaged inner foils being available to transport at low velocities and to cause blockage. However, the limited data base precludes developing a detailed debris generation model.
<u>General Electric Company</u>	
SRP 6.2.2 and RG 1.82, Revision 1 make no distinction between BWRs and PWRs; regulatory criteria should differentiate between various plant designs.**	RG 1.82, Revision 1 and SRP 6.2.2 have been modified to identify PWR- and BWR-related concerns and renamed "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident."
Reference should be made to technical findings that imply that A-43 concerns do not pose a serious problem for BWRs.*	Based on the responses received, the A-43 technical findings will be revised to reflect (1) that there is a more extensive use of fibrous insulations (i.e., NUKON™) than previously identified and (2) that BWRs are reinsulating with NUKON™. NUREG-0897 will reflect current findings and identify both PWR- and BWR-related concerns.
The value impact analysis utilizes a PWR for the risk assessment and PWR-oriented industry impacts and, as such, is not directly applicable to BWRs.*	GE's point on utilizing a PWR probabilistic risk assessment for drawing conclusions for a BWR is acknowledged. Similar assessments have been made for BWRs and those results have been utilized in preparing this revised regulatory analysis.

*Letter of December 6, 1984.
**Letter of July 11, 1983.

Table 3 (Continued)

Comment	NRC Staff Response
General Electric has reviewed the proposed revisions and has concluded that the design requirements proposed in RG 1.82, Revision 1 are excessively prescriptive and not generically applicable to the BWR.*	The requirement for long-term decay heat removal is applicable to both BWRs and PWRs. RG 1.82, Revision 1 Appendix A contains a series of tables (or guidelines) that have been derived from extensive tests and analytical studies. This information is provided for of referral and can, or need not, be used--at the user's option. RG 1.82, Revision 1 is general, and not prescriptive. The applicant has the responsibility for design submittal and justification of the safety aspects thereof.
The proposed RG should be revised so that no further requirements are imposed on designs that have already taken design precautions that preclude air ingestion into, or blocking of, suction lines used for long-term decay heat removal.*	The technical findings in 1983 (versus earlier findings) are considerably different, particularly with respect insulation employed currently and the transport characteristics of insulation debris. The air ingestion potential has been experimentally quantified and found to be small. However the 50% blockage criterion in the current RG 1.82 permitted applicants to essentially bypass the debris blockage question. For those plants where design precautions can be clearly demonstrated, further actions (retrofits) are not necessary.
In addition, the proposed RG should be further revised to provide for alternative means of ensuring that long-term heat removal is not lost as a result of suction blocking or air ingestion.*	The licensee and/or applicant always has the option to propose alternate means to deal with a particular design or safety problem.
In the SER for GESSAR, the NRC indicated that USI A-43 posed no problem for the Mark III containment configuration.*	At the time the SER for GESSAR II was written, A-43 concerns relative to BWRs were still under evaluation. The staff's SER cited several elements of the

*Letter of October 17, 1983.

Table 3 (Continued)

Comment	NRC Staff Response
	GESSAR II design that tended to reduce the probability for blockage of the RHR suction inlets due to LOCA generated debris. The staff concluded that plants referencing the GESSAR II design could proceed pending resolution of USI A-43 without endangering the health and safety of the public while completing its evaluation of GESSAR. The unique aspects of each Mark III plant design should be evaluated during plant-specific reviews of A-43 concerns.
The tests performed by Alden Research Laboratory for Reference 3 may even be very conservative for BWRs since it appears the tests utilized sump screens directly on the sump floor.*	The comment is partially correct, because BWR RHR suction inlets are located at some elevated distance above the wetwell or suppression pool floor. However, the insulation debris transport characteristics (see NUREG/CR-2982, Rev. 1) showed that low velocities (i.e., 0.2 - 0.3 ft/sec) can transport fragmented debris and are applicable to both BWRs and PWRs.
The proposed regulatory guide should be revised to include criteria that will allow alternative measures for precluding loss of long-term decay heat removal due to air ingestion or blockage.*	RG 1.82, Revision 1 states: "This regulatory guide has been developed from an extension experimenal and analytical data base. The applicant is free to select alternate calculation methods which are founded in substantiating experiments and/or limiting analytical considerations." Thus, the applicant is free to select alternate methods or measures for precluding loss of long-term decay heat removal.
Earlier surveys on the use of insulation in light water reactors have concluded that most BWRs utilize metallic insulation, which minimizes the potential	As stated above, current findings do not support the earlier surveys or conclusions. NUREG-0897 is being revised to incorporate findings from public comments

*Letter of October 17, 1983.

Table 3 (Continued)

Comment	NRC Staff Response
for formation and subsequent transport of debris to the sump screens.*	received (particularly with respect to insulations currently used and the change to fibrous insulation from previously used reflective metallic insulations). Recent tests on the transport of thin stainless steel foils show that this material can be transported at low velocities (i.e., 0.2 to 0.3 ft/sec).
<u>Gibbs and Hill, Inc.</u>	
Section B does not discuss the fact that sump configurations that differ significantly from the criteria of Appendix A may be equally acceptable. Gibbs and Hill recommends adding the following concluding paragraph to Section B: "If the sump design differs significantly from the guidelines presented in Appendix A, similar data from full-scale or reduced-scale tests, or in-plant tests can be used to verify adequate sump hydraulic performance."	Appendix A (page 1-9) has wording very similar to the commentor's suggested wording.
Tables A-1 and A-3 are inconsistent and Table A-2 has inconsistencies in water level noted.	The inconsistencies have been corrected.
<u>Northeast Utilities</u>	
Tests show that gratings are as effective as solid cover plate in suppressing vortices.	Gratings were very effective in reducing air ingestion to essentially zero.
The procedure in Appendix B is too prescriptive. The NRC should allow licensees to define and develop their own evaluation methods.	Appendix B in NUREG-0897 presents the staff's technical findings for A-43. Appendix B was included to illustrate major considerations. RG 1.82, Revision 1 is the regulatory document.

*Letter of October 17, 1983.

Table 3 (Continued)

Comment	NRC Staff Response
Credit should be given for top screen area if it is deep enough to reduce the potential for clogging (RG 1.82, Revision 1, Section C, Item 7).	For those plant designs and calculated plant conditions where this point could be unconditionally substantiated, credit would be given.
The licensee should be free to determine methods of inspection and access requirements (RG 1.82, Revision 1, Section C, Item 14).	Section 4, Item 14 states: "The trash rack and screen structure should include access openings to facilitate inspection of the structure and pump suction intake."
RG 1.82, Revision 1 will be used to evaluate sumps in operating plants. This may require backfitting at substantial costs.	The need for backfitting will be based on plant-specific analyses that will reveal the need for, and the extent of backfitting that might be required. The cost of backfit should be weighed against core melt costs.
Appendix A to RG 1.82, Revision 1 requires obtaining performance data if sump design deviates significantly from the guidelines provided. For operating plants, this may result in costly sump testing.	Appendix A states: "If the sump design deviates significantly from the boundaries noted, similar performance data should be obtained for verification of adequate sump hydraulic performance."
NRC estimates for man-rem costs associated with insulation replacement are grossly underestimated.	The value impact analysis has been revised based on cost data received during "for comment" period.
The value impact analysis addresses only PWRs. If the NRC has concluded that this issue only applies to PWRs, then the document should reflect this.	The value impact analysis revision clearly addresses BWR and PWR concerns.
The commentor concurs with the comments submitted separately on this document by the AIF.	The AIF comments are addressed separately; see above.

Table 3 (Continued)

Comment	NRC Staff Response
<u>Owens-Corning</u>	
Detailed comments addressed the wide variation of insulations employed, descriptions, suggested terminology, etc.*	Detailed comments received on insulation types; descriptions, etc. have been used to revise NUREG-0897.
Comments recommended including transport and head loss data for NUKON™ fiberglass tests.*	Data from NUKON™ tests have been referenced and major findings summarized in the revised NUREG-0897.
The commentor questioned Table B-1, Criterion 2, that reflective metallic insulation foil debris would not be transported at velocities less than 2.0 ft/sec.*	Transport tests on reflective metallic foils were conducted and revealed that they can be transported at low velocities (0.2 - 0.5 ft/sec).
The commentor questioned the concept that if there is all reflective metallic insulation there is no problem.*	Inputs received have been used in revising NUREG-0869.
Comments on recommended changes to various tables as discussed at the June 1 and 2, 1983, public meeting.*	Inputs received have been used in revising NUREG-0869.
Numerous comments suggesting word changes that would minimize singling out fibrous type insulations as the screen blockage concern without considering blockages due to reflective metallic insulation materials.*	Inputs received have been used in revising NUREG-0869.
Comments on recommended revision to reflect current status of insulations employed in nuclear power plants.*	Inputs received have been used in revising NUREG-0869.

*Letter of June 23, 1983.

Table 3 (Continued)

Comment	NRC Staff Response
The potential for screen blockage by reflective metallic debris has not been adequately addressed. In particular, the water velocities required to to transport debris and hold it against the sump screen have not been studied.*	A set of experiments to determine transport velocities (similar to those performed on fibrous insulations) has been completed by Alden Research Laboratory. The results are summarized in NUREG-0897 and used in RG 1.82.
The assumption that all fibrous blankets and pillows within 7 L/D of a break are destroyed is overly conservative. Different designs of pillows have varying resistances to destruction by water jets.*	The 7 L/D criterion is based on experimental studies of representative samples of fibrous pillows exposed of high-pressure water jets. These small water jet studies showed that increasing pressure (40-60 psia) results in destruction of pillow covers and release of core material. Furthermore, blowdown experiments in the German HDR facility showed that fiberglass insulations (even when jacketed) were destroyed within 6 to 12 feet of the break, and distributed throughout containment as very fine particles. Unless conclusive experimental evidence is obtained that accurately replicates the variety of conditions that may exist in a LOCA, it is prudent to retain the conservative 7 L/D criterion. The 7 L/D envelope is a significant reduction from the previously proposed 0.5 psia stagnation pressure destruction criterion in NUREG/CR-2791 (September 1982) and (in general) limits the zone of maximum destruction to the primary system piping and lower portions of the steam generators.
The commentor stated that estimated costs for insulation installation and replacement are too low. OCF cost estimates that were provided are*	OCF cost data are utilized in revisions to the value/impact analysis.

*Letter of July 14, 1983.

Table 3 (Continued)

Comment	NRC Staff Response
Cost of NUKON™ = $\$90/ft^2$ for material (as fabricated) Cost of reflective metallic = $\$100/ft^2$ for material (as fabricated) Installation cost = $\$112/ft^2$ for labor and related support	
The commentor provided recommendations for classification of various insulating materials Stressing differences between NUKON™ (an OCF product) and other fiberglass and mineral wool materials. The commentor noted the differences between NUKON™ and high density fiberglass.*	Descriptive classifications provided for insulation types have been combined with similar classifications obtained from Diamond Power Company for inclusion in NUREG-0897, Revision 1 and NUREG-0869, Revision 1.
The commentor identified 14 reactor plants that have been reinsulated with NUKON™, are in the process of installing NUKON™, or may install NUKON™.*	OCF plant information have been utilized, along with information from Diamond Power Company, to develop a current picture of insulation utilization in nuclear power plants. The major finding is that the number of plants using or are planning to use fibrous insulation is larger than previously estimated. For example, the Diamond Power list reveals that 25 of 130 operating and projected plants are utilizing fibrous insulation on primary system components.
The commentor recommended inspection surveys of plants to identify actual insulations employed and recommended the modification of a draft generic letter to include this requirement.*	The recommendation for physical plant surveys (or inspection to identify types and quantities insulations employed) is a good one. However, the use of a generic letter is to reconfirm adequate NPSH

*Letter of July 14, 1983.

Table 3 (Continued)

Comment	NRC Staff Response
	margins, and will be based on the actual types and quantities of insulation employed within a given plant without imposing a need to report in detail.
A report on "HDR Blowdown Tests with NUKON Insulation Blankets" was submitted as a supportive document for the capability on NUKON™ insulation to withstand the impact of a high pressure steam-water blast.**	This report has been included as Appendix F in NUREG-0897 Rev. 1. The tests demonstrated that jacketed and unjacketed NUKON™ blankets within 7 L/D will be nearly totally destroyed. However NUKON™ blankets enclosed in standard NUKON™ stainless steel jackets withstood the blast better. But these were insufficient number of tests to draw conclusions for similar insulations.
Power Component Systems, Inc.	
A report on "Buoyancy, Transport and Head Loss Characteristics of Nuclear Grade Insulation Blankets"was submitted as a supportive document for relative efforts in the area of fibrous insulations.***	The formula provided for fibrous debris blockage head loss is included in Section 5 of NUREG-0897, Rev. 1.

*Letter of July 14, 1983.
**Letter of February 18, 1985
***Letter of November 8, 1984.

APPENDIX B

PLANT SUMP DESIGNS AND CONTAINMENT LAYOUTS

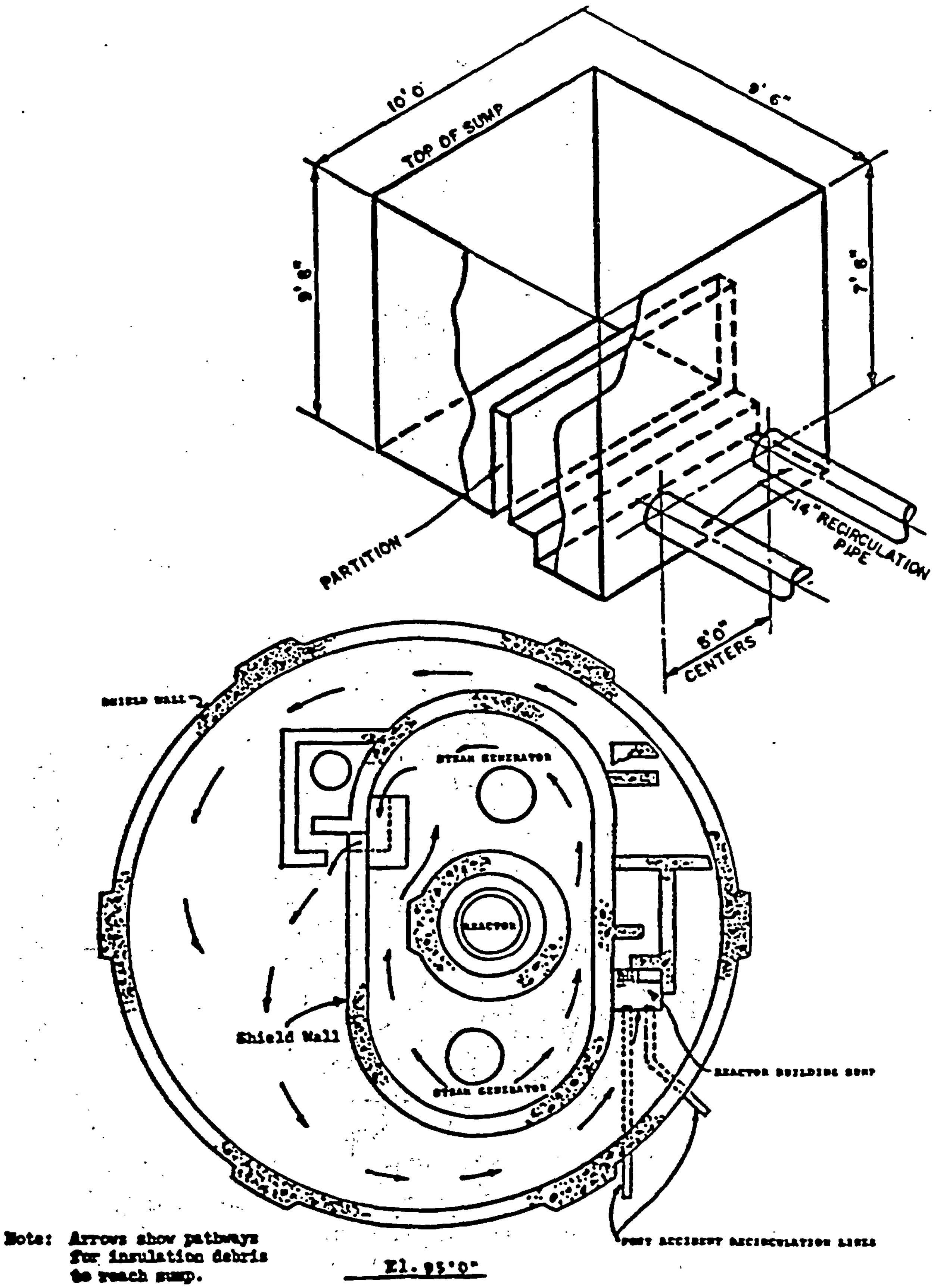


Figure B.1 ECCS sump and containment building layout, Crystal River Unit 3

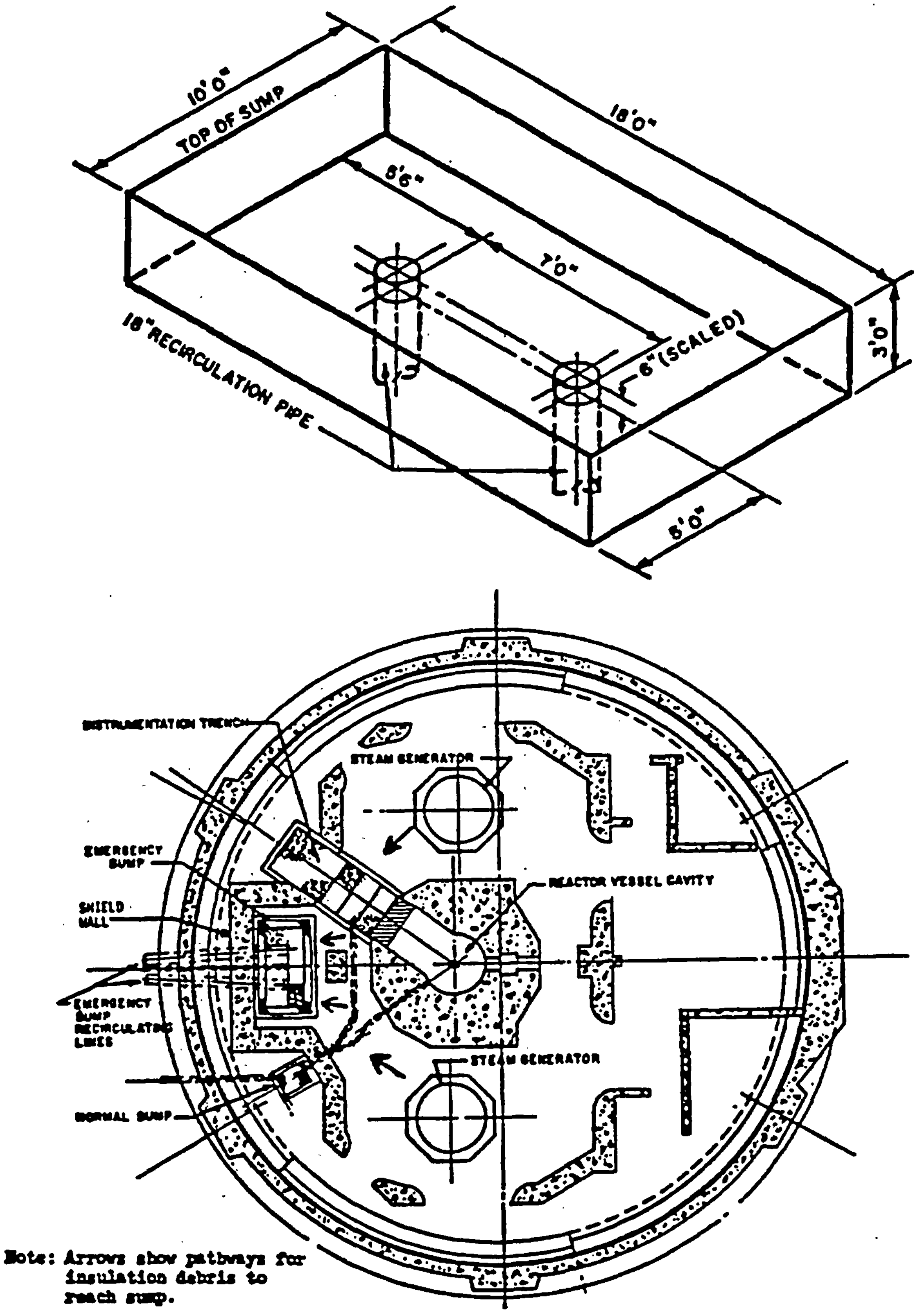


Figure B.2 ECCS sump and containment building layout, Oconee Unit 3

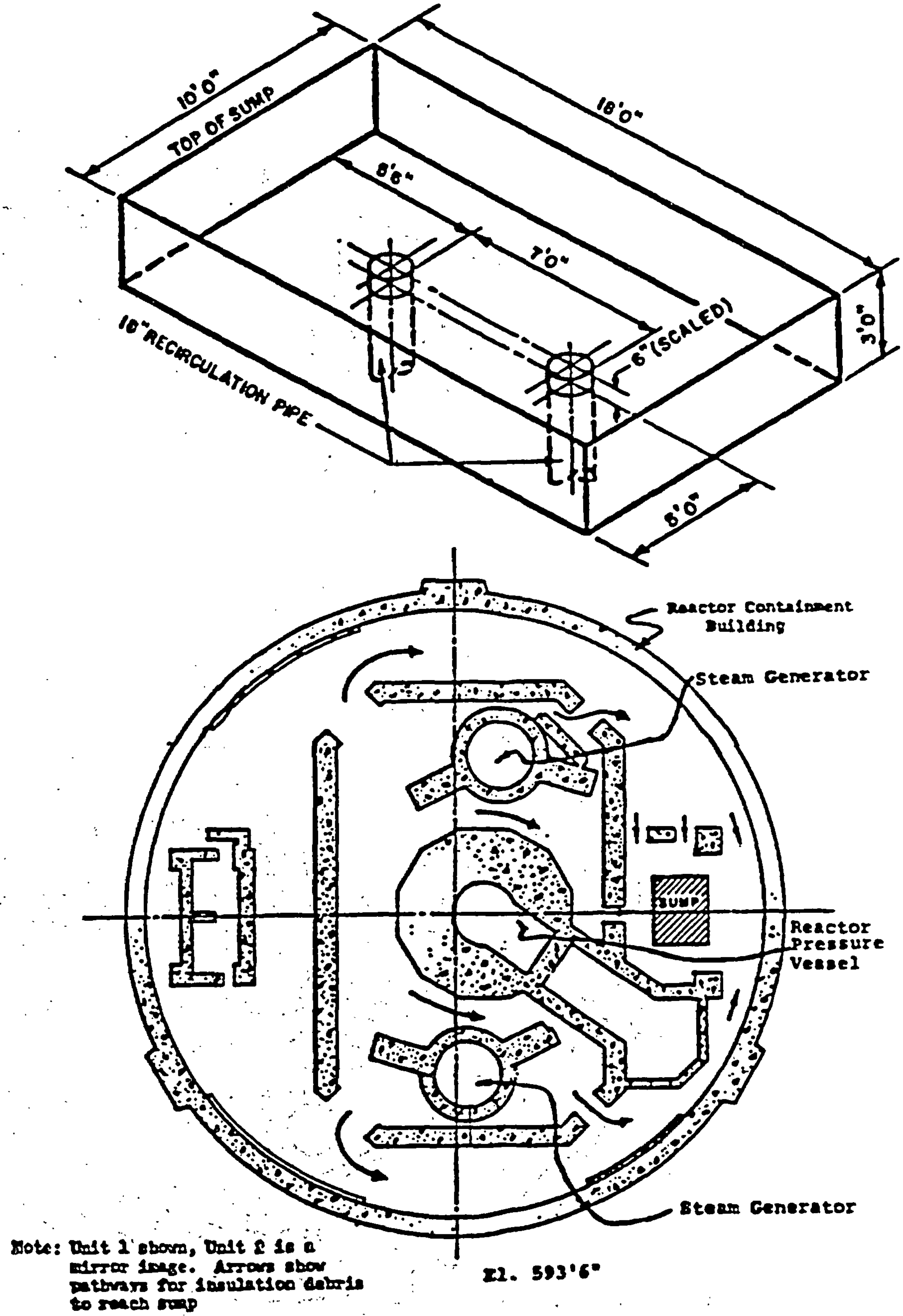
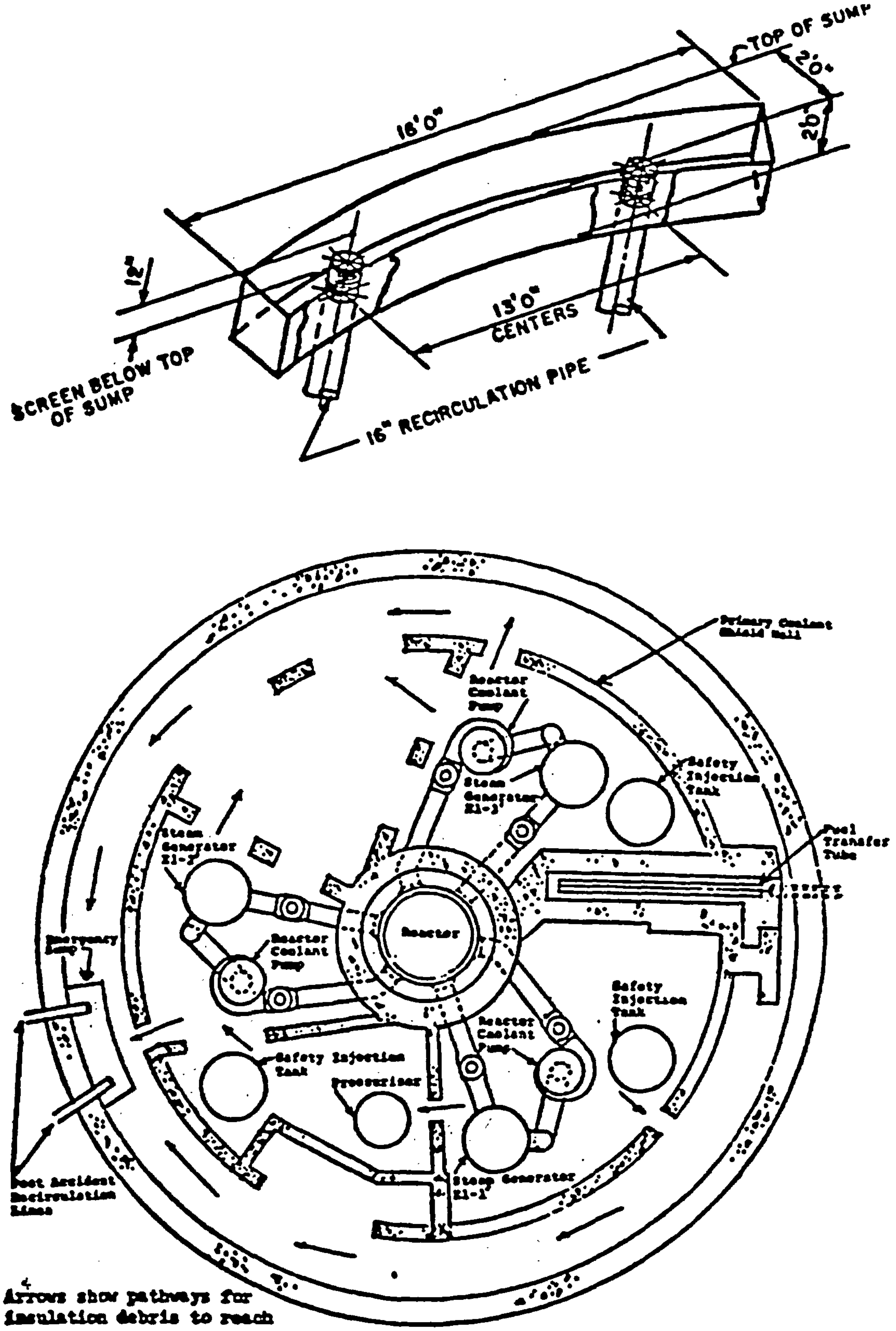


Figure B.3 ECCS sump and containment building layout, Midland Unit 2



Note: Arrows show pathways for insulation debris to reach sump.

Figure B.4 ECCS sump and containment building layout, Maine Yankee

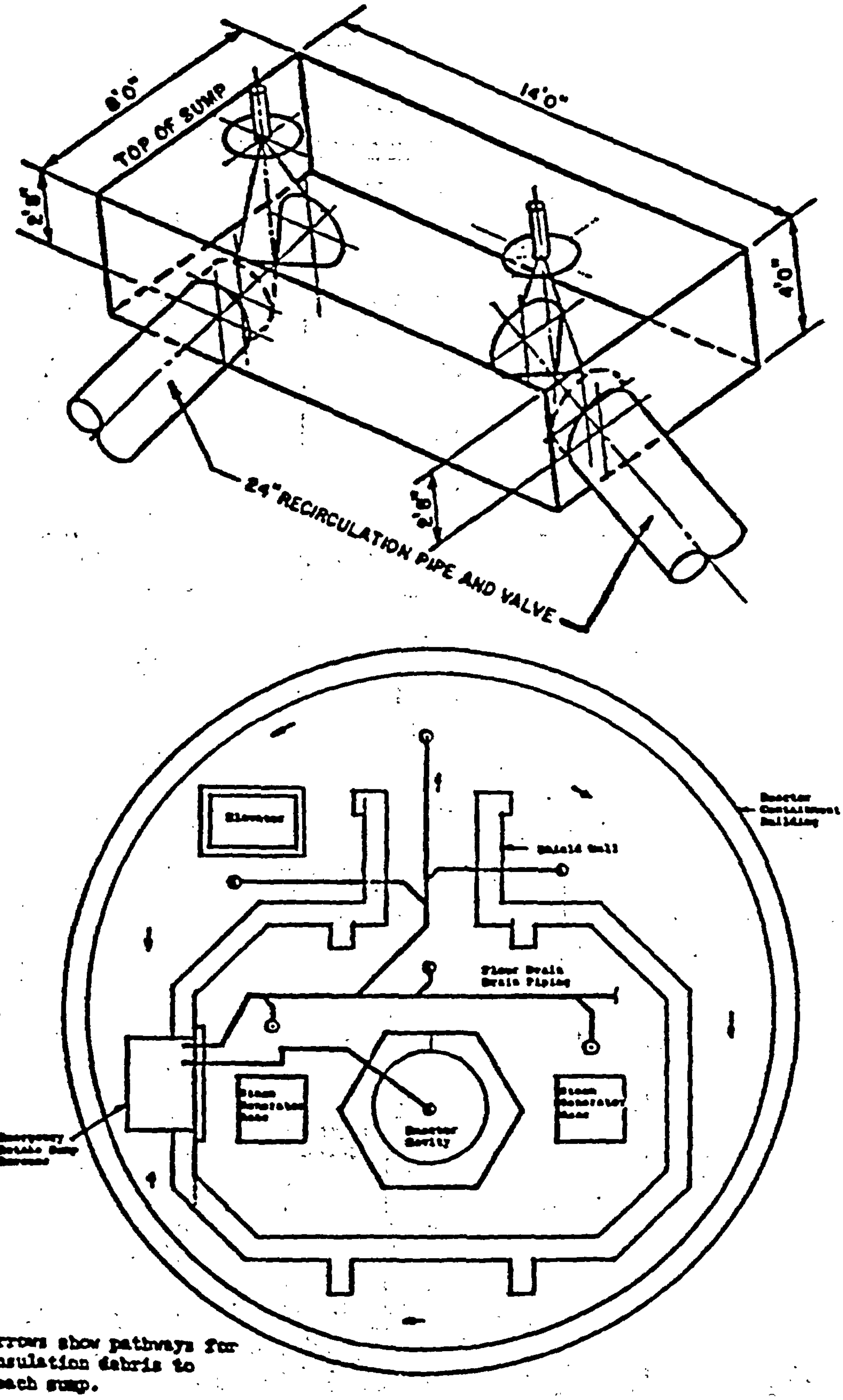


Figure B.5 ECCS sump and containment building layout, Arkansas Unit 2

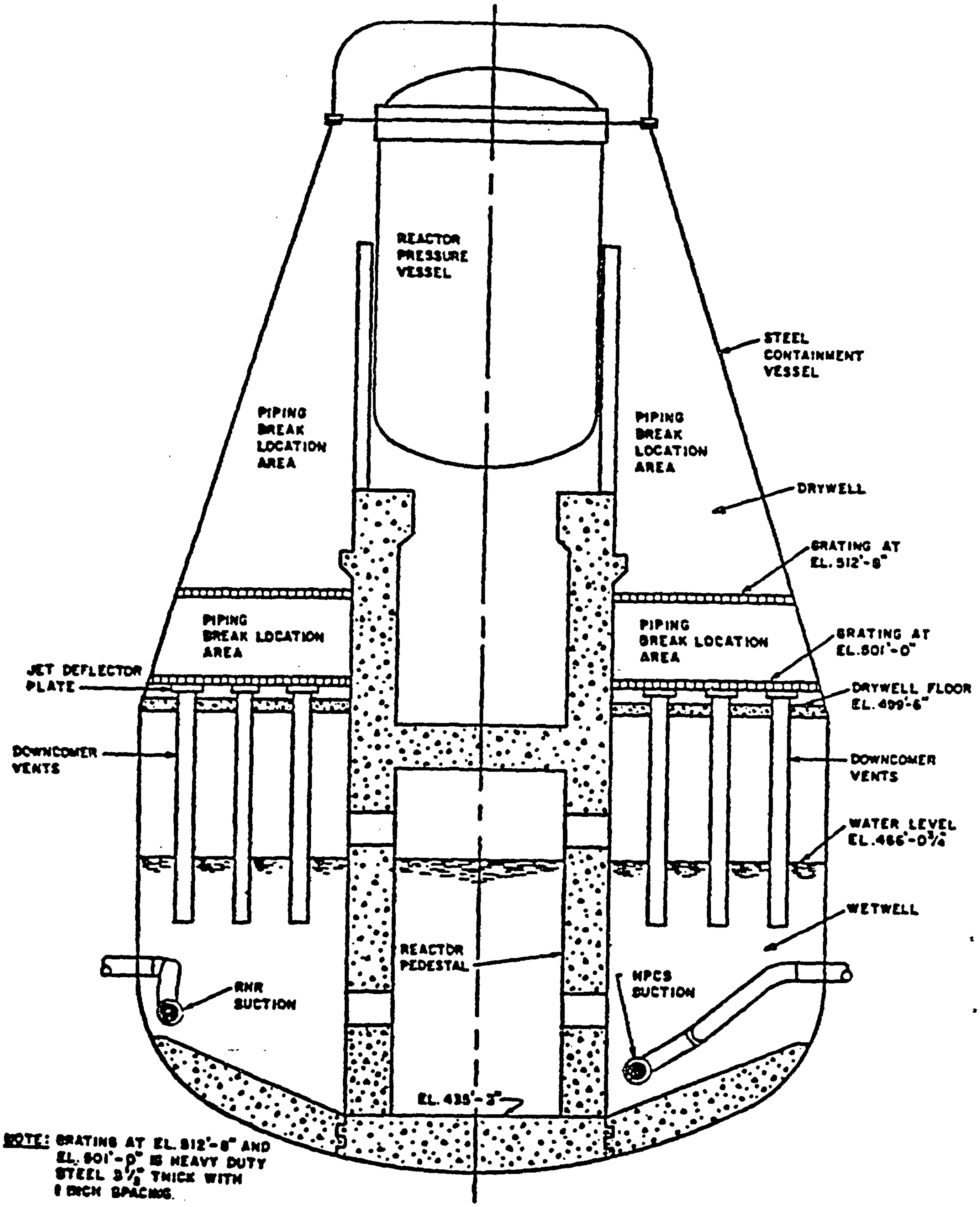


Figure B.6 Primary containment vessel, WPPSS Unit 2

APPENDIX C

INSULATION DAMAGE EXPERIENCED IN THE HDR PROGRAM

APPENDIX C

INSULATION DAMAGE EXPERIENCED IN THE HDR PROGRAM

<u>HDR Program and Facility Description*</u>

The Heissdampfreaktor(HDR) safety program (PHDR) represents a major research effort in the Federal Republic of Germany addressing the safety of nuclear power plants. Funded by the Federal Ministry for Research and Technology (Bundesministerium für Forschung and Technologie, or BMFT) and directed by the Kernforschungszentrum Karlsruhe (KfK), HDR experiments at a decommissioned nuclear power plant cover a broad range of topics relevant to nuclear safety. The program was conceived with two basic objectives

(1) to improve understanding of reactor system behavior under upset conditions and define margins of safety

(2) to evaluate and improve design and testing techniques for nuclear systems and components

HDR research is concentrated in the following five areas:

(1) reactor pressure vessel blowdown from typical LWR operating conditions

(2) response of structures and components to such extreme external loads as earthquakes and aircraft impact

(3) structural response and fracture behavior of pressure vessels and piping under both thermal and mechanical loading

(4) nondestructive examination of materials

(5) measurement of containment leak rates, both under normal operating conditions and following simulated accidents

The HDR (Heissdampfreaktor or superheated steam reactor) achieved initial criticality in October 1969 as a prototype 100-MWt boiling water reactor (BWR). Although the facility was originally intended to demonstrate the commercial feasibility of direct nuclear superheat, numerous operating problems forced its final shutdown after less than 2000 hours of operation. Rather than restart the HDR as a nuclear facility, the BMFT decided in late 1973 to refit the HDR for light water reactor (LWR) safety research. The reactor internals were removed and the facility decontaminated; new equipment was installed specifically for test purposes. The first blowdown tests at the recommissioned HDR test facility took place in 1977.

The HDR is a real nuclear power plant. That is to say, although it was originally designed nearly 20 years ago, the HDR still offers the following test capabilities relevant to more modern commercial plants:

---

*Source: Scholl and Holman, 1983.

(1) Actual reactor systems and components can be tested up to 1:1 scale.

(2) HDR systems and components are generically similar in construction and materials to those in use today.

(3) The HDR containment provides a representative basis for investigating pressurization and flow effects in multicompartmented structures following a loss-of-coolant accident.

(4) The HDR can be placed under thermal-hydraulic conditions that subject systems and components to pressure, temperature, and mass flow loads typical of postulated accident scenarios.

The initial thermal-hydraulic state required for HDR blowdown tests (typically 110 bar, 310°C) reflects nominal PWR operating conditions and is produced by a specially designed test loop. The experimental test loop (Versuchskreislauf, or VKL) includes a 4-MW electric boiler for heating circulating water, a cooler with 8 MW of heat rejection capacity, and an appropriate volume and pressure control system. Warm water is fed in at the top of the reactor pressure vessel (RPV) and cold water at the bottom, and water at a mix temperature is withdrawn through a feedwater inlet nozzle. The system is designed to produce either pressure vessel temperature gradients typical of normal PWR operating conditions or uniform (standby) temperatures. Initial tests on the VKL proved it capable of maintaining pressures stable within 1 bar, and temperatures stable within 3°C.

Damage Incurred During Blowdown Tests*

Blowdown tests conducted in the HDR facility showed there were high dynamic loads in the vicinity of the immediate break area. Inspections following these blowdown tests revealed: spalled concrete (attributed to thermal shock), blown open and damaged hatchways (in some compartments doors were torn from their frames), bent metal railings, damaged protective (or painted) coatings, peeled and heavily damaged thermal insulation on the piping and vessels, and scattered insulation debris throughout the containment building. The damage to, and the scattering of, glass wool insulation was particularly severe.

Figure C-1 shows the HDR containment and break compartment. The large number of compartments at various elevations should be noted and utilized when making use of findings for application to U.S. nuclear plants, which are generally much more open, without many intervening compartments. Figures C-2 to C-6 are photographs illustrating damage that occurred. Figure C-7 shows a typical pressure and temperature plot for containment following a blowdown.

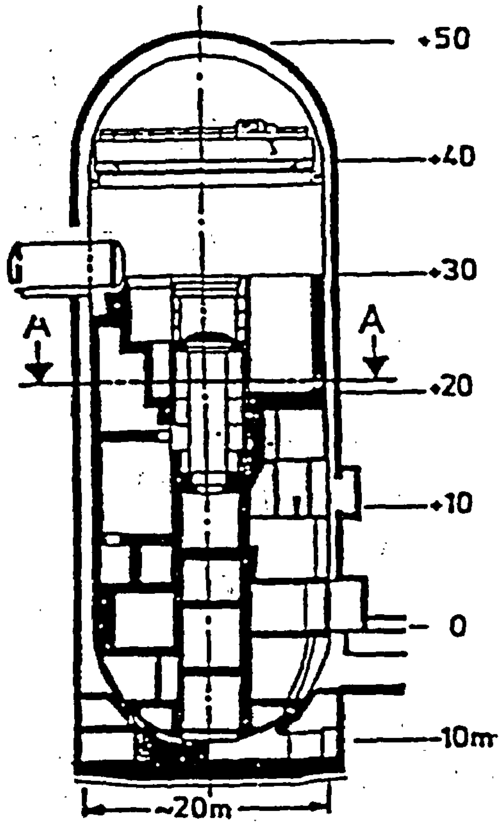
Insulation Damage Experienced During Blowdown Tests

NUREG/CP-0033 reports insulation damage as described below.

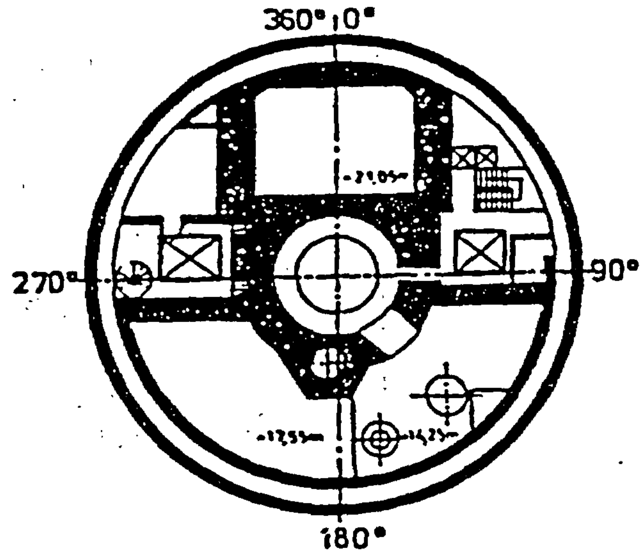
(1) Insulation (Vessel and Piping)

Standard glass wool insulation with sheet metal covering was torn away within a radius of 3 to 5 meters and distributed throughout containment. A significant

*See Holman, Müeller-Dietsche, and Mīller, 1983.



HDR containment.



Cross Section A-A

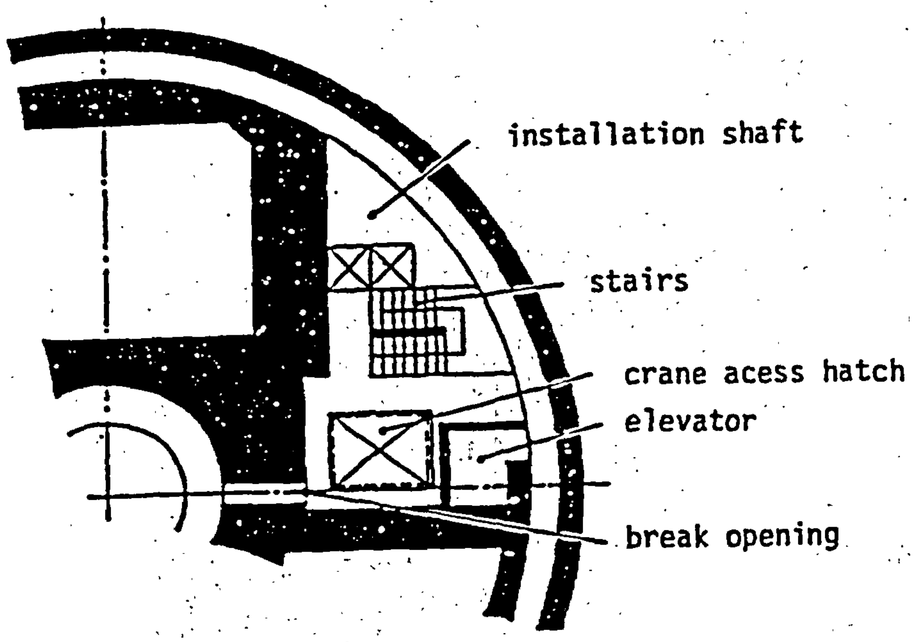


Figure C-1 HDR containment and break compartment

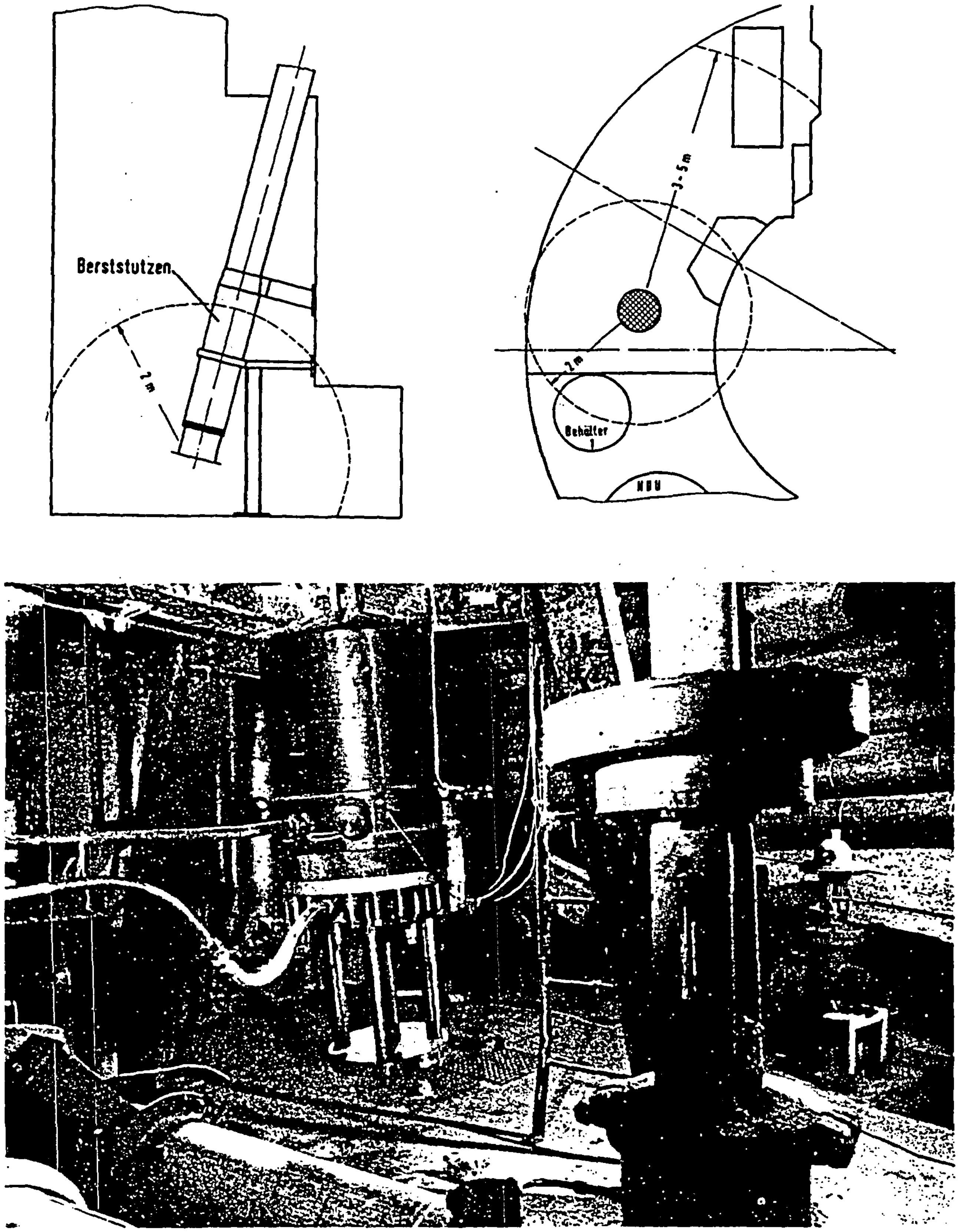


Figure C-2 HDR blowdown compartment and photo of local damage near the break nozzle

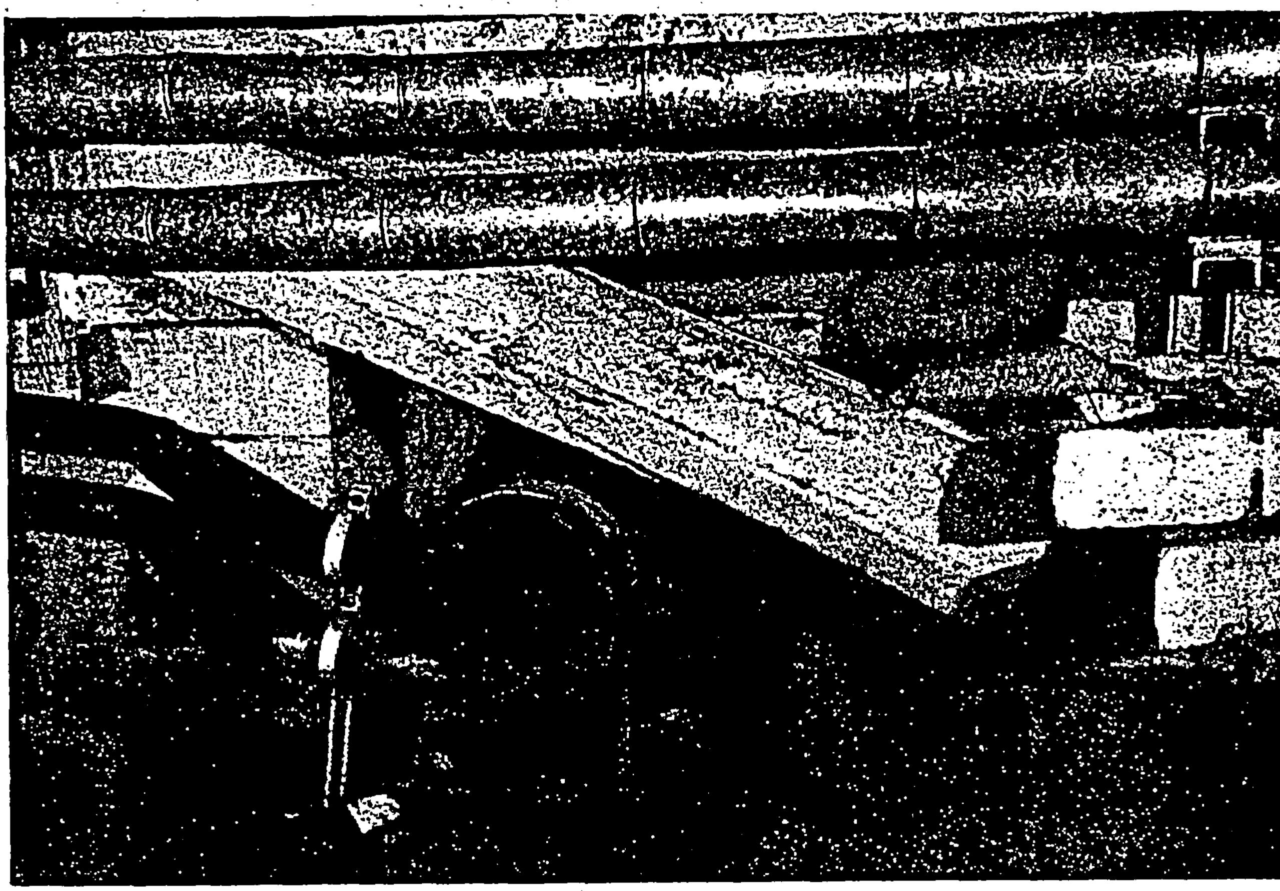


Figure C-3 HDR post-blowdown damage to concrete structures within blowdown compartment

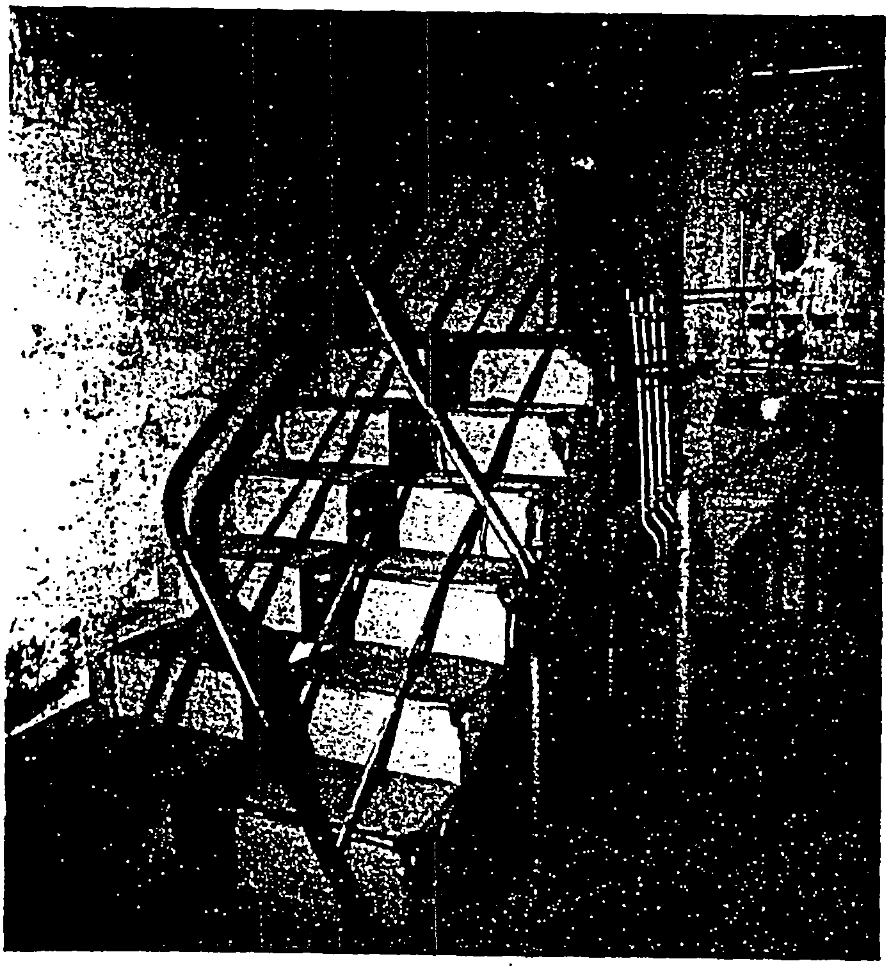


Figure C-4 HDR post-blowdown damage to railing structures in compartment near blowdown chamber

Figure C-5 HDR post-blowdown damage to compartment doors due to pressure wave. Debris shown is spalled concrete from stairwell located near blowdown compartment.

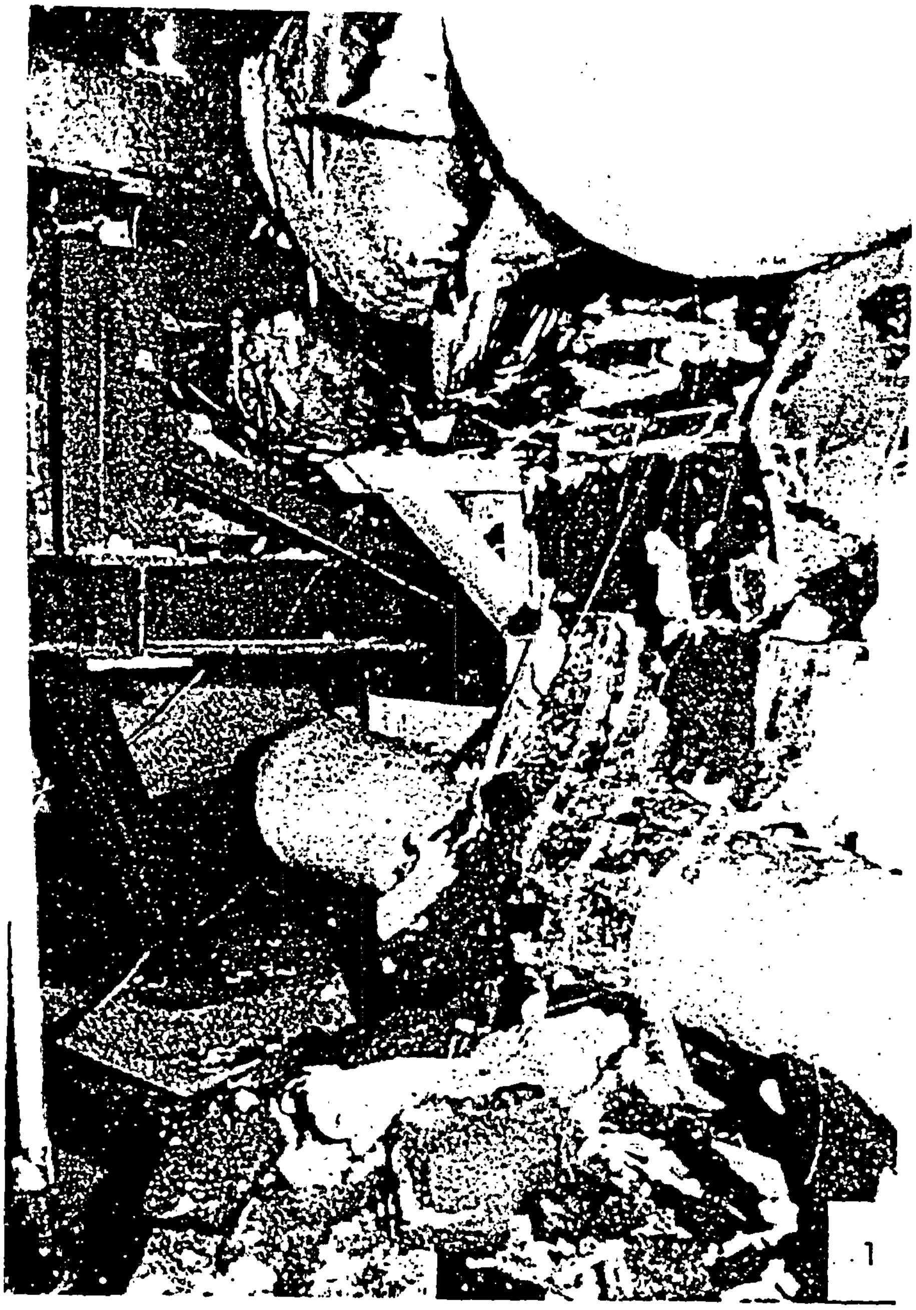


Figure C-6 HDR post-blowdown damage to insulated piping within the blowdown compartment

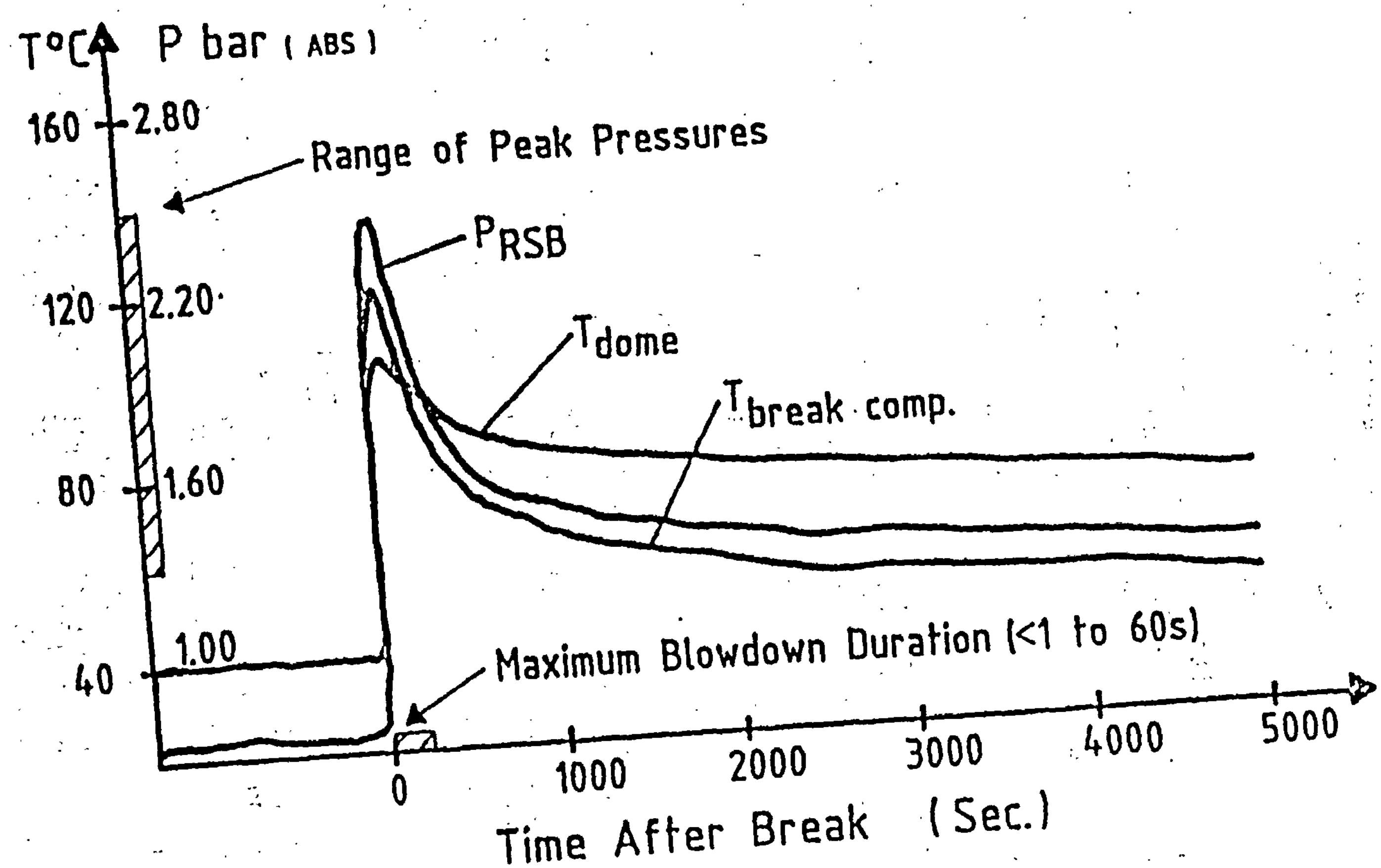


Figure C-7 Typical pressure and temperature in containment after blowdown improvement was achieved through replacing the glass wool insulation with foam glass insulation on the pipes.

Insulation on larger vessels in the pressure wave path could be protected by steel bands as long as the pressure loading was from outside to the inside. However, at times the wave pressure loading penetrated beneath the surface and lifted off the protective sheathing.

(2) Insulation (RPV)

The RPV, with its nozzle openings and complex flow patterns, is an exception because the pressure wave propagates to a certain extent from inside to outside. Several types of insulation were tested here with the following results:

° Glass wool with sheet metal sheathing was peeled off and destroyed.

° Foam glass was destroyed by larger inner overpressure because of its brittleness.

Foam glass insulation sheathed in stainless steel proved more resistant to pressure waves and jet impingement loads because its connecting joints yield to inner overpressure and suppress it.

Insulation mats with glass wool inserts and pure textile or wire-weave-strengthened covers resisted pressure waves and jet forces equally well.

Figures C-8 and C-9 illustrate the insulation damage incurred.

Two letters from the HDR staff that provide further information regarding insulation are included in their entirety in this appendix. Two other documents that are pertinent to this subject are "Investigations of the Transport Behavior of Particles During a Blowdown Test at HDR," GKSS Report 83/E/9, and "Considerations Related to Accident Induced Debris Distribution in a Pressurized Water Reactor Containment," GKSS Report 83/E/8, December 1982. Both documents were written by M. Kreubig and translated by G. Holman of Lawrence Livermore National Laboratory.

REFERENCES

Holman, G., W. Müeller-Dietsche, and K. Muller, "Behavior of Components Under Blowdown and Simulated Seismic Loading," paper presented at the ASME Pressure Vessel and Piping Conference, Orlando, Florida, July 2, 1982.

Müller, K., G. Holman, and G. Katzenmier, "Behavior of Containment Structures During Blowdown and Static Pressure Tests at the HDR Plant," in Proceedings of the Workshop on Containment Integrity, Vol II of II, U.S. Nuclear Regulatory Commission Report NUREG/CP-0033, October 1982 (see also Sandia National Laboratory, SAND-82-1659).

Scholl, K. H. and G. S. Holman, "Research at Full Scale: the HDR Program" in Nuclear Engineering International, January 1983.

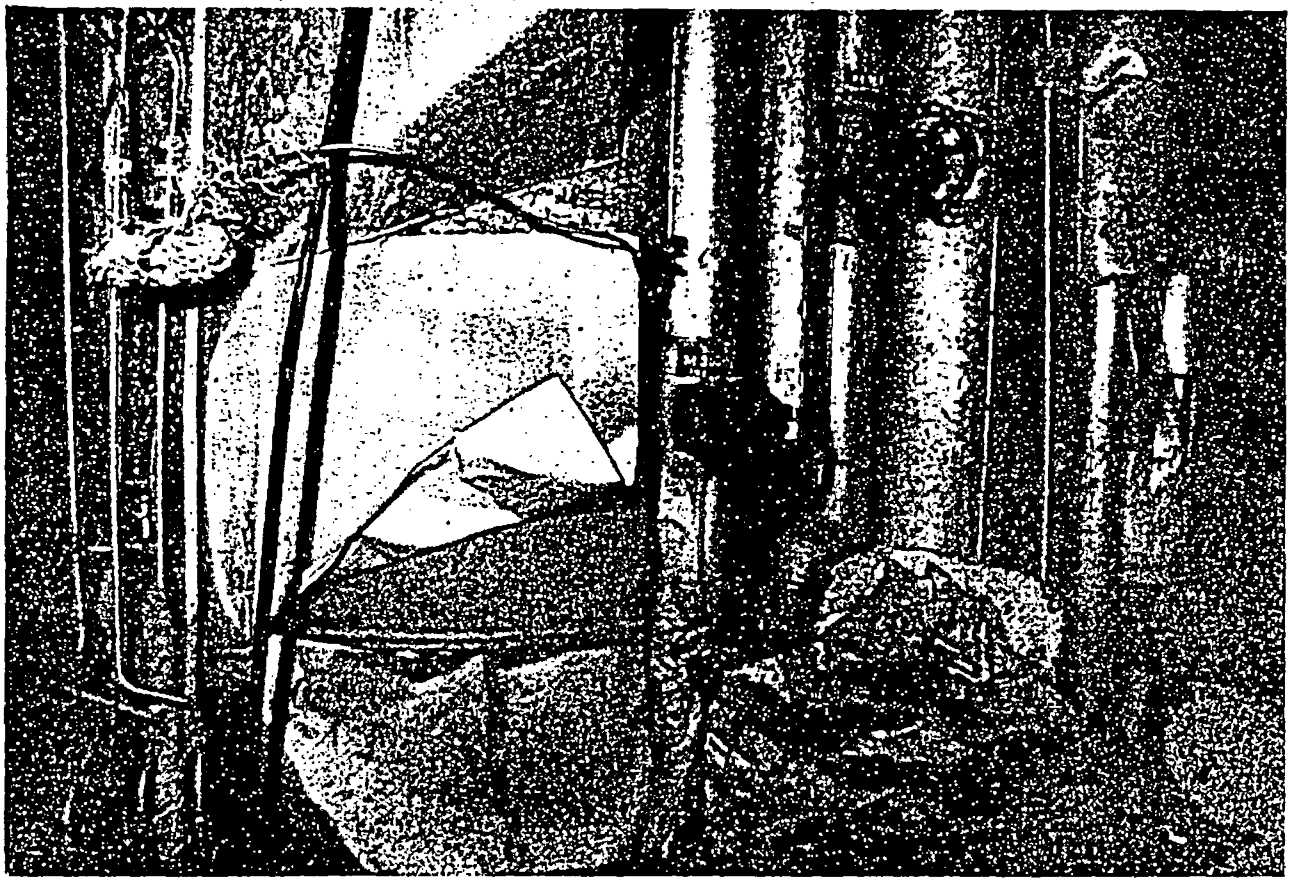


Figure C-8 Damage to jacketed fiberglass insulation located on the HDR blowdown test. Source: letter from G. Holman to A. Serkiz, NRC, "Photographs of HDR blowdown damage," April 18, 1983.

Figure C-9 Foam glass insulation damage following a blowdown in the HDR. Foam glass insulation withstood blowdown tests better than fiberglass. Source: letter from G. Holman to A. Serkiz, NRC, "Photographs of HDR Blowdown Damage," April 18, 1983.

Kernforschungszentrum Karlsruhe
Gesellschaft mit beschränkter Haftung

Kernforschungszentrum Karlsruhe GmbH Postfach 3640 D-7500 Karlsruhe 1

Mr. A.W. Serkiz
Task-Manager
Generic Issues Branch
Division of Safety
Technology
U.S. NRC
Mail Stop NL-5650

Washington, D.C. 20555
U.S.A.

Projekt HDR-Sicherheitsprogramm
Leiter: Dipl.-Ing. W. Müller-Dietsche

Datum: Aug. 02, 1983 - bo
Bearbeiter: Kl. Müller
Telefon: 07247/ 82 4343
Ihre Mitteilung:

Dear Mr. Serkiz:

I will send copies of our papers concerning equipment qualification next week to G.S. Holman (LLNL) for translation.

These papers conclude

- behavior of components during blowdown and in post blowdwon atmosphere
- distribution of isolation materials
- distribution of debries during blowdown in direction to the sump-area
- behavior of containment structures during blowdown.
- proposal of using HDR as a equipment qualification testbed.

Kernforschungszentrum Karlsruhe, Wissenschaftlich-Technische Einrichtungen und Verwaltung 7514 Eggenstein-Leopoldshafen,
Tel. (07247) 821, Telex: 7826484, Drahtwort: Reaktor Karlsruhe; Stadtbüro u Postanschrift: D-7500 Karlsruhe 1, Weberstraße 5, Postfach 3640
Vorsitzender des Aufsichtsrats: Staatssekretar Hans Hilger Haunschild,
Vorstand: Prof. Dr. Rudolf Harde, Vorsitzender; Dr. Hellmut Wagner, Stellv. Vorsitzender; Prof. Dr. Horst Bohm, Dr. Hans Henning Hennies, Prof. Dr. Wolfgang Klose
Handelsregister: Amtsgericht Karlsruhe HRB 302; Baden-Württembergische Bank AG., Karlsruhe, Kto Nr. 400 24713 00 (BLZ 660 200 20); Commerzbank AG., Karlsruhe, Kto Nr. 2 221 000 (BLZ 660 400 18); Deutsche Bank AG., Karlsruhe, Kto Nr 0236 521 (BLZ 660 700 04); Dresdner Bank AG., Karlsruhe Kto Nr. 5 634 398 (BLZ 660 800 52)

Mr. Wind of HDR-project will join the 11th WRS-Meeting end October 1983. Perhaps you can contact him together with G. Holman. He will answer additional questions and if needed from your side, he can illustrate component behavior and damage by slides. In this case please contact me during September 1983 by phone or telex.

With best regards
Kernforschungszentrum Karlsruhe GmbH
Project HDR Safety Program

Project HDR Safety Program
Kl. Müller

Kernforschungszentrum Karlsruhe

Aug. 1, 1983 - bo

NOTA: INSULATION DAMAGE IN THE HDR BLOWDOWN EXPERIMENTS

1. Glas Fibre Insulation

HDR was equipped with this typ of insulation at the begin of the experiments. In the break compartment all glas fibre insulation was destroyed at 2 m around the break nozzle and distributed through the whole reactor in very fine particles on the walls and floor. The iron wrappers were thrown away from vessels within 4 m around the break nozzle, the glas fibre being untouched. With enforced shieldings (steel bandages) around the vessels nothing happened.

2. Glas Foam Insulation

Glas foam insulation around pipes up to 200 mm Ø withstood the blowdown impact even in a distance of about 2 m around the nozzle, except a small area where the mass flow touched the pipe. At these placed the insulation was cut out. The insulation of the pressure vessel was destroyed at great areas around the break nozzle caused by the first pressure wave cracking the material (short break nozzle RDB-E experiment).

Kernforschungszentrum Karlsruhe: Wissenschaftlich-Technische Einrichtungen und Verwaltung: 7514 Eggenstein-Leopoldshafen
Tel. (07247) 821, bei Durchwahl 82 Telex: 7826484, Drahtwort. Reaktor Karlsruhe; Stadtbüro u. Postanschrift: D-7500 Karlsruhe 1, Weberstraße 5; Postfach 3640

The glas foam then was cracked into great pieces not leaving the break compartment and a great amount of fine particels following the blowdown pathes up to the sump inlet.

3. Glas Foam with Stainless Steel Shielding

This material withstood all impacts and retained intact even installed about 1 m around the break nozzle.

4. Insolating Matrazes

They consist of an special cloth outside eventually reinforced by steel wires filled with glass fibre or stone wool. This material withstood all impacts even good as material point 3. Nevertheless there were some corrosion effects on the cloth caused by demineralized water at high temperatures.

More detailed information you will get on request from Mr. Wind of K. Mueller.

Kernforschungszentrum Karlsruhe

Gesellschaft mit beschränkter Haftung

Kernforschungszentrum Karlsruhe GmbH Postfach 3640 D-7500 Karlsruhe 1

Mr. A.W. Serkiz
Task-Manager
Generic Issues Branch
Division of Safety Technology
U.S. NRC
Mail Stop NL 5650

Washington D.C. 20555
U.S.A.

Projekt HDR-Sicherheitsprogramm
Leiter: Dipl.-Ing. W. Müller-Dietsche

Datum: Sept. 12, 1983
Bearbeiter: K. Müller
Telefon: 07247/ 82 4343
Ihre Mitteilung:

Dear Mr. Serkiz,

I read the reports you send to me with great interest. There are some additional remarks concerning Nureg/CR-2982 coming from our experiments.

We found out; the jet forces are main cause for debris generation and distribution; pipe whip etc. are negligable.

Jet forces act only in a diameter of 2 - 5 m around the nozzle, depending on break diameter and break geometry.

We did these experiments with pure steam and pure water jet with nozzle diameters of 200 - 450 mm Ø.

First the pressure wave mainly destroys covers around fibre-glass and mineral wool and brittle insulation materials as glas foam. Than the impact of the fluid peels off the unprotected "wool layer" or cuts out the foam glas around pipes.

The jet and the following turbulences transport even heavy weight fragments to the next compartments. Here heavy parts are normaly fixed by drag force and only light wight particles will be transported further especially into the dome.

Kernforschungszentrum Karlsruhe, Wissenschaftlich-Technische Einrichtungen und Verwaltung. 7514 Eggenstein-Leopoldshafen.
Tel. (07247) 821, Telex: 7826484, Drahtwort: Reaktor Karlsruhe; Stadtbüro u. Postanschrift. D-7500 Karlsruhe 1, Weberstraße 5; Postfach 3640
Vorsitzender des Aufsichtsrats: Staatssekretär Hans Hilger Haunschild.
Vorstand. Prof. Dr. Rudolf Harde, Vorsitzender; Dr. Hellmut Wagner, Stellv. Vorsitzender; Prof. Dr. Horst Böhm, Dr. Hans Henning Hennies, Prof. Dr. Wolfgang Klose
Handelsregister: Amtsgericht Karlsruhe HRB 302; Baden-Württembergische Bank AG., Karlsruhe, Kto Nr. 400 24713 00 (BLZ 660 200 20), Commerzbank AG., Karlsruhe, Kto Nr. 2 221 000 (BLZ 660 400 18); Deutsche Bank AG., Karlsruhe, Kto Nr. 0236 521 (BLZ 660 700 04), Dresdner Bank AG., Karlsruhe, Kto Nr. 5 634 398 (BLZ 660 800 52)

All grids and components within the building act as a screen for fixing these light wight particles, so in containments with a complicate interior most of the generated debris are fixed before reaching the sump area.

So only a break location with direct access to the sump area will block the screens in the way described in your papers.

In the post blowdown phase when the emergency cooling system is fed by the sump water there are only some "main water ways" left leading from the nozzle to the sump. These "main water ways" will not cause pump failure.

From my opinion you will get more debris collected and settled within the core barrel and other core internals than reactivated by the back flow of the water to the sump.

Even if activating the Containment spraysystem you will get more problems with the blockage of the injection nozzles of a water spray system by the debris than blocking the pump or sump inlet.

Yours sincerely,
Kernforschungszentrum Karlsruhe GmbH
Projekt HDR-Sicherheitsprogramm

Ø: G. Holman, LLNL
F. Wind, PHDR

APPENDIX D

DETERMINATION OF RECIRCULATION VELOCITIES

APPENDIX D

DETERMINATION OF RECIRCULATION VELOCITIES

1.0 General

During the recirculation mode of operating the ECCS, water on the reactor floor will drain to the sump, the source of water for pumps which provide long-term cooling of the reactor. This flow of water on the reactor floor may be at sufficient velocity such that insulation debris is transported with the flow, resulting in blockage of the sump screens and a pressure drop across the screens. Of major concern is the impact of this potential pressure drop on the pump flow and on the available pump NPSH compared to the required NPSH.

Various types of insulation materials have been tested to determine what flow velocities will initiate movement and transport of this debris. Of equal importance is the determination of what flow velocities will exist in a given plant during the recirculation mode, as it is the relative magnitude of the actual recirculation velocities to the experimentally determined transport velocities which determines the probability of insulation debris blocking the sump screens.

Due to the arrangement of plant walls, structures, and equipment, there will be only certain flow paths available from each postulated break location to the sump(s). Some plant layouts will result in a few obvious flow paths; in other plants, the flow paths may be numerous and not so easily defined. Those paths having the shortest length and offering the least resistance (losses) will produce the greatest velocities (i.e., have the most water surface slope). For a given velocity, the flow path with the largest cross-sectional area will carry the largest discharge. Local velocities will be considerably different from average velocities due to local flow contractions. Losses may be produced by surface friction, drag due to the flow past appurtenant structures, equipment, or pipes, expansion losses downstream from constricted openings, bends of the flow path, and any other phenomena causing turbulent energy dissipation.

This appendix will review various means for determining the recirculation velocities, such that an assessment of debris transport can be made. If a preliminary analysis using simplified methods indicates recirculation velocities are within a factor of about two (2) compared to the experimentally derived transport velocity for the insulation type(s) under study, then more refined analyses are warranted. For example, if recirculation velocities are up to about 50% less than the predetermined debris transport velocities, transport may still actually occur since many approximations are inherent in the preliminary analyses. On the other hand, if the recirculation velocities are up to about twice the transport velocities, transport may be less severe than indicated for similar reasons. To be conservative, it should be assumed that all flow is returned by the safety injection system since this maximizes recirculation velocities on the containment floor.

2.0 Review of Network Resistance Method

A preliminary method of estimating recirculation velocities is to define a system of possible flow paths with varying resistance. This flow/resistance network is simplified by finding equivalent resistances to series and parallel paths, until one equivalent flow path remains. Since the total flow is known and the equivalent resistance may be estimated from coefficients of friction and losses available in handbooks, the total head drop from the break to the sump may be calculated. As all parallel flow paths are subject to this same total drop, the individual flows in all other paths may then be determined. Knowledge of flows per path allows local velocities to be determined from the known local cross-sectional areas. This preliminary analytic method is presented in NUREG/CR-2791, and is summarized below (using conventional hydraulic terms).

As an illustration, assume the simplified situation shown in Figure D-1 (taken from NUREG/CR-2791). Flow from the break may reach the sump in all combination of the paths illustrated, and this combination may be reduced to the flow/resistance diagram shown, where resistances R1 through R8 correspond to the similarly numbered flow paths. The resistance may be determined from the following set of equations, starting with the well known Darcy-Weisbach resistance formula (4, 8).

$$h_L = f \frac{L}{4R_H} \frac{V^2}{2g} \tag{D-1}$$

where
- h_L = the drop in water level (piezometric head) (ft)
- f = friction factor (dimensionless)
- L = flow path length (ft)
- R_H = hydraulic radius = flow area/wetted perimeter (ft)
- V = average cross-section flow velocity (ft/sec)
- g = acceleration due to gravity (ft/sec^2)

Values of f, which depend on the relative roughness of the flow path and the flow Reynolds number, are available from standard text and handbooks, such as (4, 8). Since

$$V = Q/A \tag{D-2}$$

where

- Q = flow rate (ft^3/sec)
- A = cross sectional flow area (ft^2)

letting

$$C = 1/2g$$

$$K = \frac{fL}{4R_H}$$

then

$$h_L = \frac{KC}{A^2} Q^2 \tag{D-3}$$

By setting

$$R = KC/A^2$$

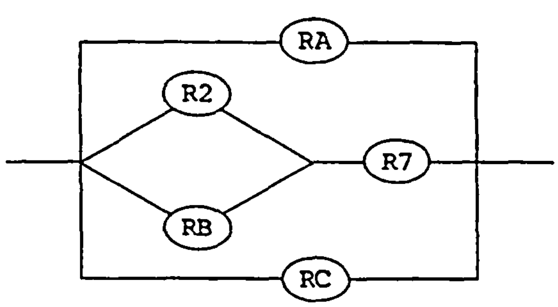
we obtain the usual system loss equation

$$h_L = RQ^2 \tag{D-4}$$

indicating greater resistance (higher values for R) for paths having greater friction, longer lengths, and smaller cross-section areas, and vice versa.

Equivalent resistances may be found for combined flow paths by use of the above equations and continuity, noting that the loss for each parallel flow path equals the total loss. The result is that resistances in series add, and resistances in parallel follow a reciprocal law.

Therefore, the network in Figure D-1 may be simplified to

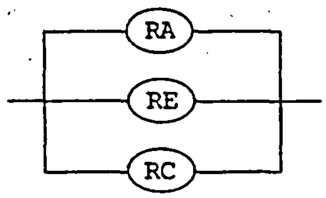


where RA = R1 + R8
RB = R3 + R6
RC = R4 + R5

Parallel resistances such as R2 and RB may be combined by finding an equivalent resistance

$$RD = \left[\frac{1}{\sqrt{\frac{1}{R2}} + \sqrt{\frac{1}{RB}}} \right]^2$$

such that the network is now simplified to



where RE = RD + R7

which in turn is reduced to one equivalent resistance RF by application of the reciprocal law for parallel resistances. Therefore

$$h_L = (RF)Q^2$$

and h_L may now be calculated, because RF is estimated from the individual branch resistances, and the total flow of the ECCS is known. Given the calculated h_L, which is the same for all parallel branches, flow in each branch may be calculated using the individual resistances for that branch. For example,

$$Q_1 = Q_8 = \sqrt{\frac{h_L}{R1 + R8}}$$

and the velocity at any section along flow path 1-8 may be determined by dividing the above determined flow rate by the cross-sectional area at the section of interest. It is important to consider local flow contractions to less than actual structural openings. A typical flow contraction can be as low as about 0.65 of the actual available opening, depending on the geometry involved.

The above summarized method appearing in NUREG/CR-2791, although sound in principle, includes many approximations. A basic problem is that values for f are available only for straight, prismatic channels, and that average values of f and R_H are used for the entire flow path. This may be overcome by using much shorter flow paths, each having the proper value of f and R_H, but this makes the calculation more laborious. It should also be recognized that most of the flow resistance is due to drag of various objects in the flow path, to bends, and due to flow expansions from contracted areas. Drag losses may be expressed as (4)

$$h_D = C_D \frac{V^2}{2g}$$

where C_D = a dimensionless drag coefficient.

A similar expression is used for losses due to bends

$$h_B = C_B \frac{V^2}{2g}$$

where C_B will vary with the bend radius.

Values for C_D and C_B are available for a variety of shapes in standard text and handbooks (4, 8). Head losses due to flow expansions are given by (1, 4 & 8)

$$h_E = \left[1 - \frac{a}{A} \right]^2 \frac{V_a^2}{2g} = C_E \frac{V_a^2}{2g}$$

where a = contracted flow area (ft^2)

A = downstream cross-sectional area (ft^2)

V_a = contracted area velocity (ft/sec)

The contracted velocity may be related by continuity to the average flow velocity of the branch, and C_E expressed in terms of V instead of V_a. The total head loss for a given flow path may thus be calculated from

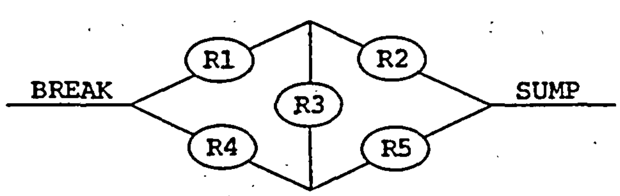
$$h_L' = [f \frac{L}{4R_H} + C_D + C_B + C_E] \frac{V^2}{2g} = R'Q^2 \tag{D-5}$$

where

$$R' = (\frac{fL}{4R_H} + C_D + C_B + C_E) \frac{1}{2gA^2} \tag{D-6}$$

The above illustrated calculations will be improved by the addition of these terms, but numerous flow paths must be defined such that the available values of R_H, C_D, C_B, and C_E really apply to that section, as average or effective values of these coefficients for varying path characteristics cannot be determined.

Despite the possible refinements to this method, not all flow/resistance networks can be simplified to one equivalent resistance. Consider, for example, the following simple case.



This problem may be overcome by using a different type of analyses, as illustrated below for a more complex flow network postulated for a given plant.

3.0 Complex Network Analysis

In the example illustrated in Figure D-2 there are 28 flow paths and 18 junctions, A to R. For each flow path Eq. (D-5) is applicable. For example, for the flow path 5,

$$H_C - H_B = R_5' Q_5^2 \qquad \text{(D-7)}$$

where H = piezometric head at the junction identified by the sub-script (ft)

Q = flowrate along the flow path identified by the subscript (ft^3/sec)

R' = an overall resistance factor as defined in equation (D-6) for the flow path identified by the subscript (sec^2/ft^5)

Similar to Eq. (D-7), 28 equations corresponding to the 28 flow paths are available. Also, for each junction the continuity equation can be applied.

For example, in Figure D-2, for junction J, assuming inflow from flow paths 16 and 21

$$Q_{17} = Q_{16} + Q_{21} \qquad \text{(D-8)}$$

Combining Eq. (D-8) with head loss relationships similar to Eq. (D-7) gives

$$\left(\frac{H_J - H_A}{R_{17}'}\right)^{1/2} = \left(\frac{H_K - H_J}{R_{16}'}\right)^{1/2} + \left(\frac{H_I - H_J}{R_{21}'}\right)^{1/2} \qquad \text{(D-9)}$$

For each of the junctions, one could write an equation similar to Eq. (D-9). Hence, if flow directions are first assumed, 18 junction equations are obtained to form a system of nonlinear equations with the 18 unknown piezometric heads at junctions. One of the most widely used method for solving such a system numerically is the Newton-Ralphson method (5) which iteratively solves the system of equations. Computer programs using the Newton-Ralphson method are readily available in many books on pipe network analysis (5, 10) or on numerical analysis of nonlinear equations (3). To use the Newton-Ralphson method, one assumes the flow directions and provides an initial estimate of the piezometric heads conforming to the assumed flow directions. Since the method is iterative, the acceptable error in final solution should also be indicated. The method usually converges very fast, although convergences may not be obtained if initial values are unreasonable and too far from actual values. The flow directions, if wrong, will be automatically corrected by the calculation procedure to conform to the values of the piezometric heads obtained after each iteration.

For the example considered, the piezometric head at the sump and the total flow into the sump Q_T would be known. Referring to Figure D-2, H_A, the piezometric head at junction A is known. If the piezometric heads at each of the junctions B to R are determined, one could calculate the flows using the flow path equations similar to Equation (D-7). There are 17 unknowns, namely H_B to H_R, and the required 17 equations can be obtained by writing the continuity equations at each of the junctions A to Q. For example at junction A,

$$\left(\frac{H_J - H_A}{R'_{17}}\right)^{1/2} + \left(\frac{H_Q - H_A}{R'_2}\right)^{1/2} + \left(\frac{H_B - H_A}{R'_6}\right)^{1/2}$$

$$+ \left(\frac{H_B - H_A}{R'_7}\right)^{1/2} + \left(\frac{H_B - H_A}{R'_8}\right)^{1/2} - Q_T = 0 \qquad \text{(D-10)}$$

The Newton-Ralphson method can be used to solve the 17 equations similar to (D-10) for the 17 unknowns H_B to H_R. The method is iterative and solves a linear matrix as explained below:

Let the 17 non-linear equations be,

$$
\begin{aligned}
F_1 &= 0 \\
F_2 &= 0 \\
&----\\
&----\\
&----\\
F_{17} &= 0
\end{aligned}
$$

A linear matrix is written as,

$$
\begin{bmatrix}
\frac{\partial F_1}{\partial H_B}, & \frac{\partial F_1}{\partial H_C}, & ----, & \frac{\partial F_1}{\partial H_R} \\
\frac{\partial F_2}{\partial H_B}, & \frac{\partial F_2}{\partial H_C}, & ----, & \frac{\partial F_2}{\partial H_R} \\
---- & ---- & ---- & ---- \\
---- & ---- & ---- & ---- \\
\frac{\partial F_{17}}{\partial H_B}, & \frac{\partial F_{17}}{\partial H_C}, & ----, & \frac{\partial F_{17}}{\partial H_R}
\end{bmatrix}
\begin{bmatrix} Z_B \\ Z_C \\ - \\ - \\ Z_R \end{bmatrix}
=
\begin{bmatrix} F_1 \\ F_2 \\ - \\ - \\ F_{17} \end{bmatrix}
\tag{D-11}
$$

Using the initial guesses of H_B to H_R, the values of $\partial F_1/\partial H_B$, $\partial F_1/\partial H_C$ etc. and F_1, F_2 etc. are calculated first and the linear matrix (D-11) is solved to obtain, Z_B to Z_R, the corrections to initial guesses of H_B to H_R. Note that the values of F_1, F_2 etc might be non-zero, since the initial guesses are not actual values.

The corrected values of H_B to H_R are used for the next iteration, and the calculations are repeated until Z_B to Z_R are within stated acceptable error margins. After several iterations, the final corrected values of H_B to H_R will be considered as the actual values, and these are then used in the flow path equations similar to (D-7) to obtain the flows in each flow path.

Alternatively, a network analysis based on corrections to the flows in each loop could be performed. The flow system given in Figure D-2 could be transformed to an eight loop network as given in Figure D-3 by replacing parallel pipes with equivalent pipes. In this case, initial guesses of flows along each flow path should be made such that the continuity equation is satisfied at each junction. Referring to Figure D-3, there are 8 loops. For each loop the algebraic sum of the head losses around the loop would be zero. The positive direction of flow must be defined, such as clockwise around each loop. For example, referring to Figure D-3, for loop 6 with assumed flow directions let the initial guesses of flows be Q_{13}, Q_4, Q_{11}, and Q_{12} along the flow paths 13, 4, 11, and 12 respectively. Since the algebraic sum of the head losses around the loop would be zero, we get

$$F_6 = R'_{13}\ (Q_{13} + \Delta Q_6)^2 - R'_4\ (Q_4 - \Delta Q_6 + \Delta Q_1)^2$$

$$- R'_{11}\ (Q_{11} - \Delta Q_6 + \Delta Q_5)^2 + R'_{12}\ (Q_{12} + \Delta Q_6 - \Delta Q_7)^2$$

$$= 0 \qquad \text{(D-12)}$$

where ΔQ_i is the correction to flows in the loop i required to convert initial estimates to actual values of flows. When a flow path is common to more than one loop, corrections from each of the loops have to be included to get the actual flow for that flow path.

Writing similar equations for each loop, we get

$F_1 = 0$
$F_2 = 0$
- - -
- - -
$F_8 = 0$

As a first iteration, the unknowns ΔQ_1 to ΔQ_8 are solved by Newton-Ralphson method for the assumed initial guess of the flows around each loop. Then the flows are corrected with the obtained values of ΔQ_1 to ΔQ_8 and the next iteration is carried out. The procedure is repeated until ΔQ_1 to ΔQ_8 become acceptably small.

This method is quicker in that a lesser number of unknowns (equal to number of flow loops) is involved. However, it is difficult to give initial estimates of flows satisfying continuity equation at each junction.

Instead of the Newton-Ralphson method, other iterative methods can be used, such as Hardy-Cross or linear methods, to solve the nonlinear system of equations (5).

Irrespective of the method of analysis for large networks, the time consuming part is providing the initial data of R' values for each flow path and the initial estimates of piezometric heads or flows. It must also be realized that many break locations must be considered, with each location requiring re-evaluation (perhaps redefinition) of the flow network. Therefore two other methods to predict flow patterns and local velocities are addressed below.

4.0 Two-Dimensional Analyses

Rather than pre-defining flow paths, another approach is to use a two-dimensional numerical model which, by its nature, accounts for the shape and size of the various flow paths and obstructions in the containment building. The flow to the sump being basically horizontal, the complete three-dimensional flow equations are integrated vertically over the water depth (depth averaged) and solved numerically using one of several techniques. The two basic classes of techniques are the finite differences and finite elements methods. In the former, a grid is defined covering the flow field, and the derivatives appearing in the differential equations are approximated based on the values of the variables at the nodes of the grid. The most common type of finite differences grid is rectangular, with possibility of variable resolution, but other grids are possible, particularly circular grids for problems with obvious circular characteristics.

In finite elements methods, the variations of the variables of interest are approximated continuously over elements through pre-defined "basis functions" (or interpolating functions) and nodal values. The most common type of element is triangular with nodes at the vertices, but there is no limitation on the shape of the elements that can be used (rectangular and curvilinear are common), and the number of nodes per element depends on the choice of basis function. One of the advantages of the finite elements method over the finite difference method is that the flow domain can be approximated more closely and that variations of resolution are more convenient with finite elements. As an example, a grid of triangular finite elements is shown in Figure D-4 for the previously discussed application. Finite elements solutions, however, tend to require larger computation times than finite differences solutions.

There are many other differences between available two-dimensional models. These other differences concern the details of the numerical technique used, such as the way in which the nonlinear terms are treated or handling of the advective terms (which tend to create numerical instabilities), or the way time integration is performed. Another important difference between available models is the way in which turbulence and the corresponding Reynolds stresses are simulated. A common approach is to use an eddy viscosity concept but flow separation is then difficult to reproduce, and the values of the eddy viscosity has a large effect on numerical stability, making the selection of this parameter all the more critical. The so-called κ-ε method of turbulence simulation has recently been shown to be very powerful, at the expense of an increased number of differential equations to be solved.

For such two-dimensional analyses, the break flow is simulated by a flow source term(s) at one or more nodes at the break location. The sump may be simulated either by sink terms for nodes around the sump, or specifying values of normal velocity components at these locations. Various assumptions regarding the distribution of velocity or flow around the sump may be made. Losses due to friction and distributed drag from small pipes or structures are estimated and appropriate values of f selected. Losses due to flow eddies and large-scale turbulence may be simulated depending on the grid detail and on the analytic model. For practical grid sizes such as on Figure D-4, proper modeling of flow separation is doubtful. Initial values must be prescribed for velocities and water depths at all nodes, and zero velocities and a horizontal water surface are convenient initial conditions. At solid boundaries, zero normal (perpendicular) velocities must exist, although the tangential velocity component may be either zero or unprescribed.

Several two-dimensional models which are applicable to this problem are available, including those by Wang & Connor (9), Leendertse (7), Benque et al (2) and Launder and Spaulding (6). Application of any of these models to the calculation of recirculating flow patterns in containment buildings should, however, be subject to careful evaluation as a number of features exist in the proposed application for which the analytic models have not been fully tested. A notable feature to be checked is the flow separation that can be expected behind obstructions.

Results of two-dimensional models are flow velocities and water surface elevation at the node points versus time. For this application, transient effects would probably be negligible, but the computation time would remain large because of the fine grids required to account for the geometrical details of the domain. In spite of the relatively dense grid shown in Figure D-4, it is not possible to closely follow the actual bounding geometry in regions of small clearances and local contractions.

None of the analytic techniques described above includes consideration of the initial break flow momentum, nor do they closely simulate the complex geometry of the containment and appurtenant equipment, as either one- or two-dimensional approximations are made. Also, losses must be independently estimated. If complex flow patterns have significant effects on the problem under consideration, it is accepted practice that a physical (hydraulic) model study may be necessary.

5.0 Hydraulic Model Studies

Depending on how close any analytically predicted recirculation velocities are to the experimentally determined debris transport velocities and the need to further refine the evaluation of potential debris transport to the sump, it may be advantageous to use a physical (hydraulic) model. Such a model would include all geometric features of the containment floor area which could affect flow patterns. A portion of the type of model which would be suitable is illustrated in Figure D-5. Although a full-scale simulation of the reactor floor and sump geometry may be considered, it is more efficient to use a reduced scale model (and there is no technical reason to the contrary). NUREG/CR-2760 reports on studies specifically designed to evaluate potential scale effects on sump hydraulics. These studies show no scale effects as long as model flow Reynolds numbers exceed certain limiting values, such as typically achieved at geometric scale ratios of about 1:4.

The advantages of using a hydraulic model are

(1) There is no need to make assumptions regarding loss and contraction coefficients as these are implicitly included.

(2) Flow paths are reproduced to their actual geometry rather than simulated by one- or two-dimensional techniques, allowing accurate spatial definition of velocity variations.

(3) The break flow momentum can be scaled, and numerous break locations can be evaluated without model reconstruction.

Basic debris transport phenomena, such as relative volumes moved and downstream settling in lower velocity areas, can be demonstrated using simulated (scaled) debris.

5.1 Similitude Requirements

The main similitude requirement is based upon scaling the two dominant forces in free surface flow, gravity and inertia. These primary forces are embodied in the Froude number, F,

$$F = \frac{V}{\sqrt{gH}}$$

(where V, g, and H are as previously defined) and equality of Froude number between model and prototype leads to proper scaling of flow patterns from the break to the sump. The selected geometric scale ratio must be large enough, however, such that viscous forces involved with friction and drag are properly scaled. This will be true if the model Reynolds number is large enough such that loss coefficients are equal to those of the prototype. Alternately, adjustments in the size of components causing losses may be made to compensate for the lower model Reynolds number. The use of standard laboratory velocity meters may also influence the choice of the model scale ratio.

It should be noted that the actual reactor pressures and water temperature do not have to be scaled in the hydraulic model. The gas pressure over the water is constant in space and will have no effect on flow patterns. Water temperature affects the water viscosity and surface tension, but neither parameter influences flow patterns for sufficiently large geometric scale ratios and model Reynolds number.

Simulation of the insulation debris transport, if desired, is more complex. Since it may not be possible to directly scale all relevant parameters as is the case for other analogous hydraulic models simulating material transport, test results are more qualitative than quantitative. One approach is to find a model material which is transported at the model velocity scaling the known actual transport velocity for that insulation material. Alternately, the actual insulation material may be used (at scaled size and volume) if the model flow and velocity is increased to actual (prototype) values, while maintaining the water depth. For a scale ratio of 1:4, this involves doubling the model flow and velocities from that given by normal Froude scaling. It should be demonstrated that such flow increases do not change the flow patterns as determined from running the model at Froude scaled flows.

REFERENCES

1. Baumeister, Theodore, editor, Marks' Standard Handbook for Mechanical Engineers, Chapter 3, Eighth Edition; McGraw-Hill Book Company, Inc., 1978.

2. Benque, J.P., J.A. Cunge, J. Feuillet, A. Hauguel, and F.M. Holly, "New Method for Tidal Current Computation," in Journal of the Waterways, Port, Coastal and Ocean Division, ASCE, Vol. 108, No. WW3, August 1982.

3. Conte, S.D., and C. DeBoor, Elementary Numerical Analysis, McGraw Hill, 1965.

4. Daily, J.W., and D.R.F. Harleman, Fluid Dynamics, Addison-Wesley Publishing Company, Inc., 1966.

5. Jeppson, R.W., Analysis of Flow in Pipe Networks, Ann Arbor Science, 1979.

6. Launder, B.E., and D.B. Spaulding, Lectures in Mathematical Models of Turbulence, Academic Press, New York, 1972.

7. Leendertse, J.J., "Aspect of a Computational Model for Long Period Water Wave Propagation," Rand Corporation, Memorandum RM-5294-PR, May 1967.

8. Streeter, V.L., Handbook of Fluid Dynamics, Section 3, First Edition; McGraw-Hill Book Company, Inc., 1961.

9. Wang, J.D., "Real Time Flow in Unstratified Shallow Water," in Journal of the Waterways, Port, Coastal and Ocean Division, ASCE, Vol. 104, No. WW1, February 1978.

10. Watters, G.Z., Modern Analysis and Control of Unsteady Flow in Pipelines, Ann Arbor Science, 1979.

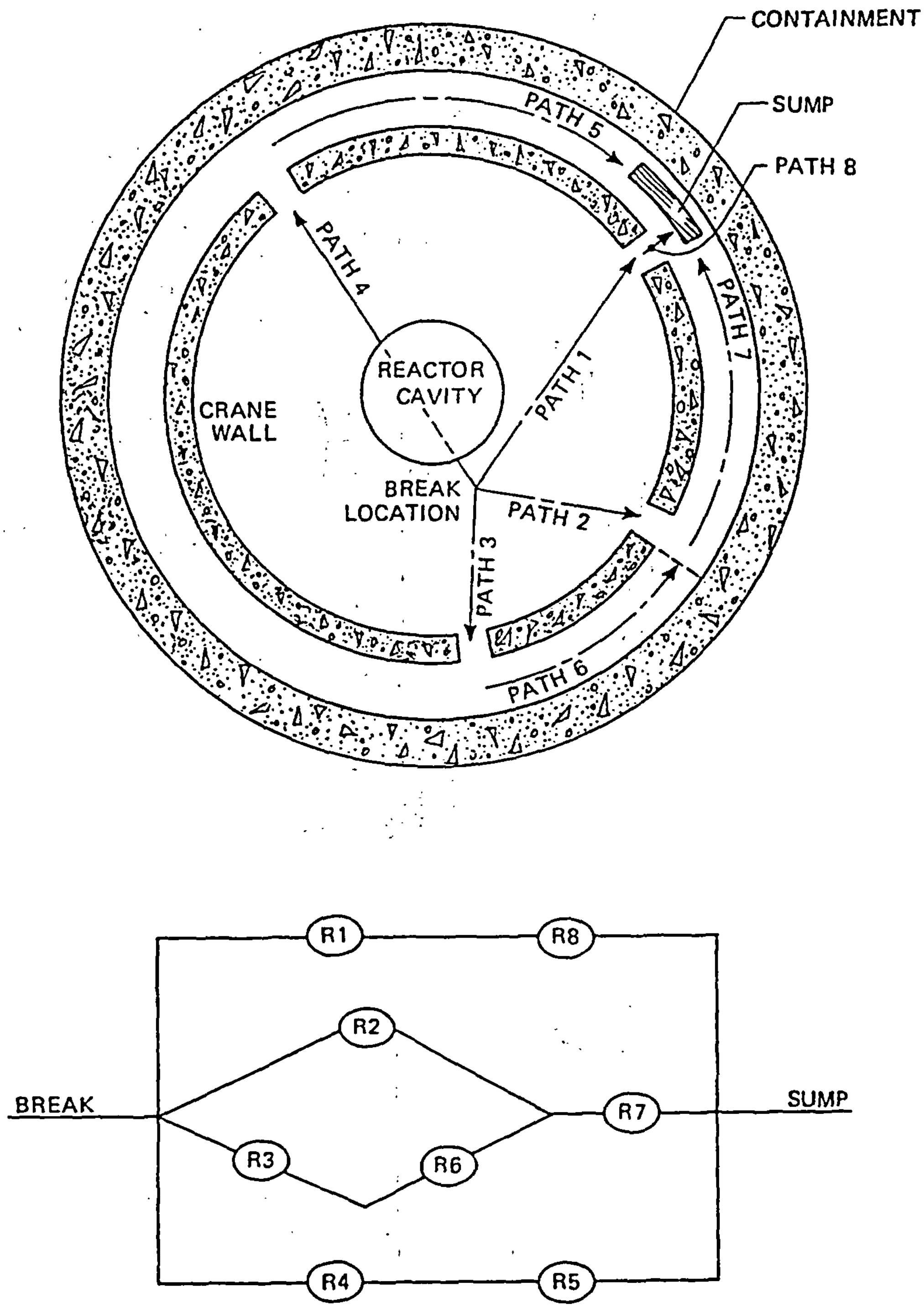
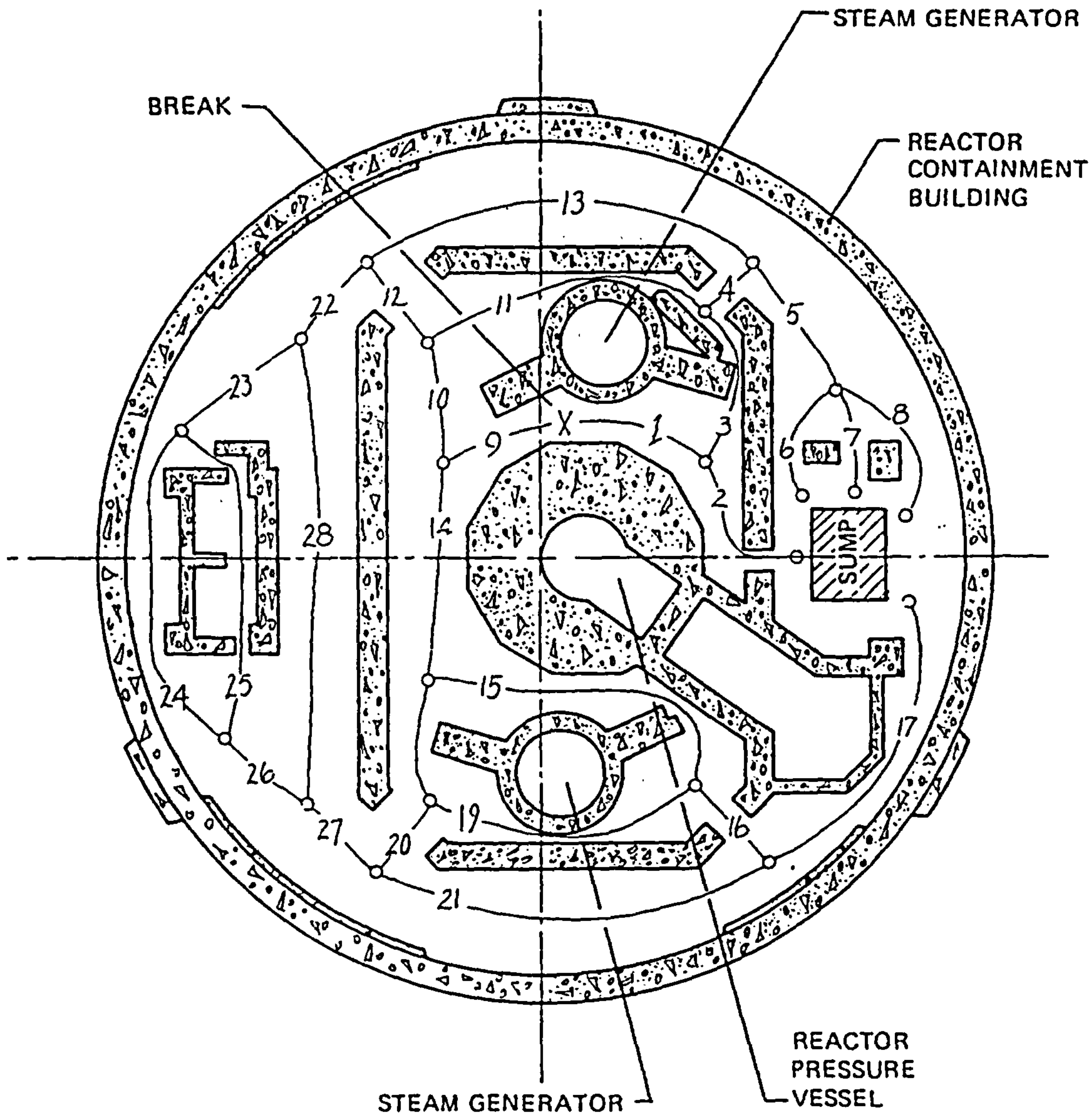


FIGURE D-1 SIMPLIFIED FLOW PATTERN WITHIN CONTAINMENT



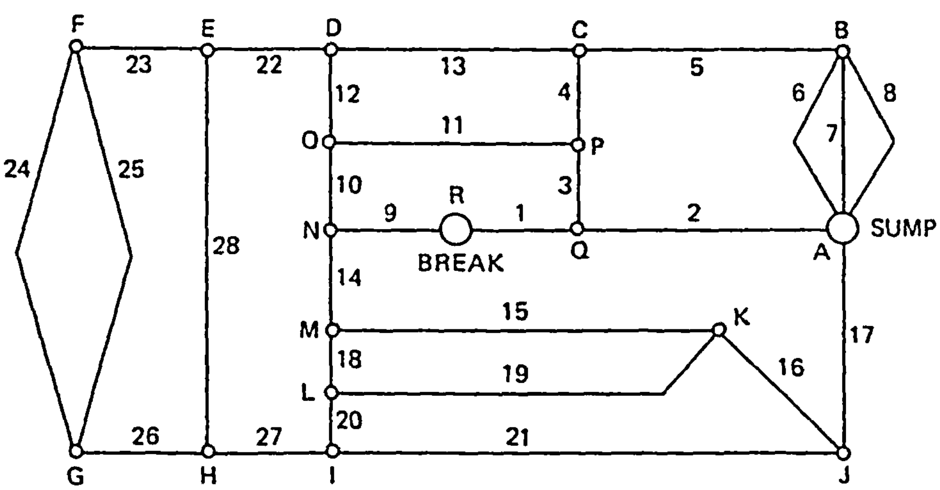


FIGURE D-2 COMPLEX FLOW NETWORK FOR CONTAINMENT

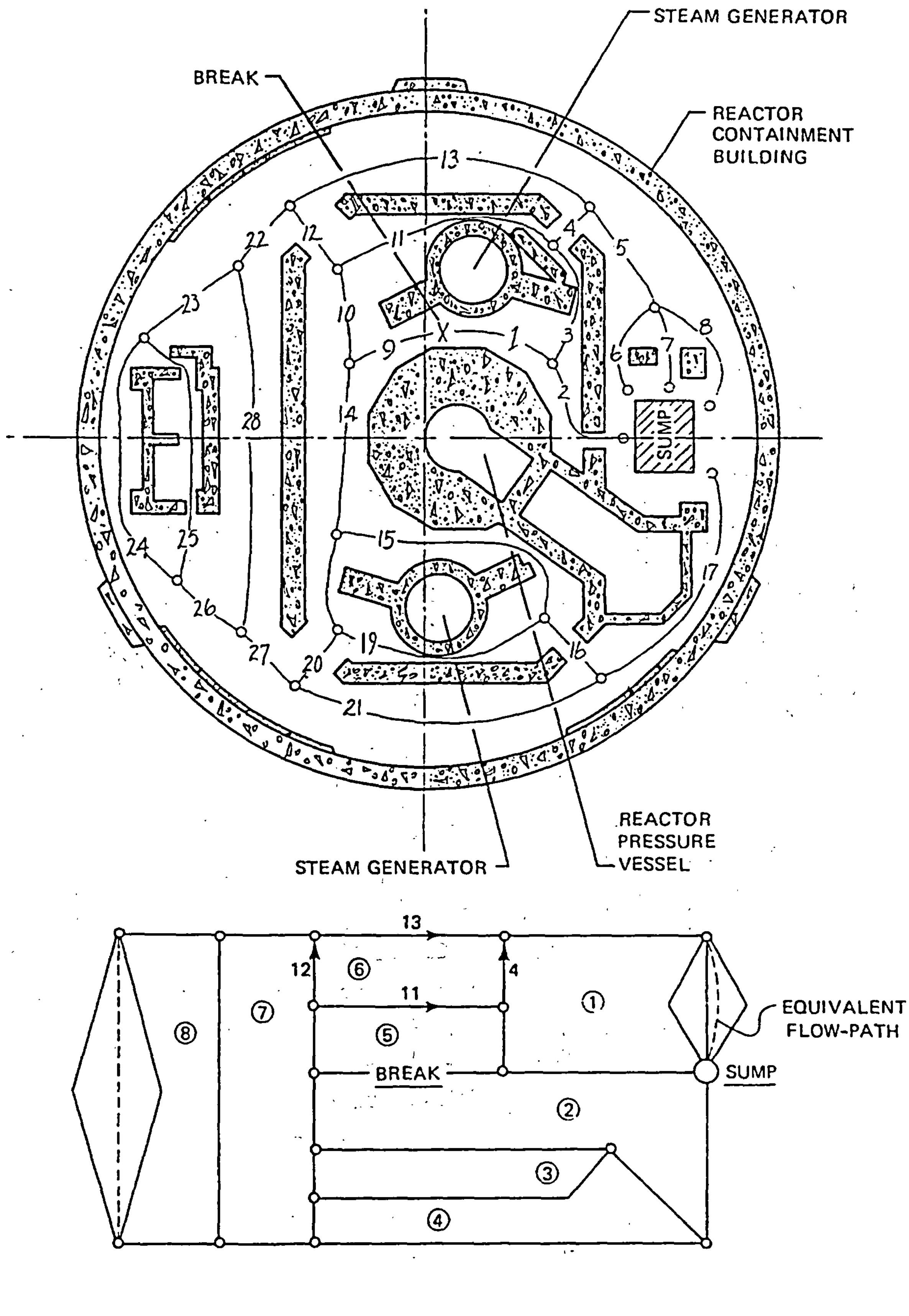


FIGURE D-3 COMPLEX NETWORK WITH FLOW LOOPS IDENTIFIED

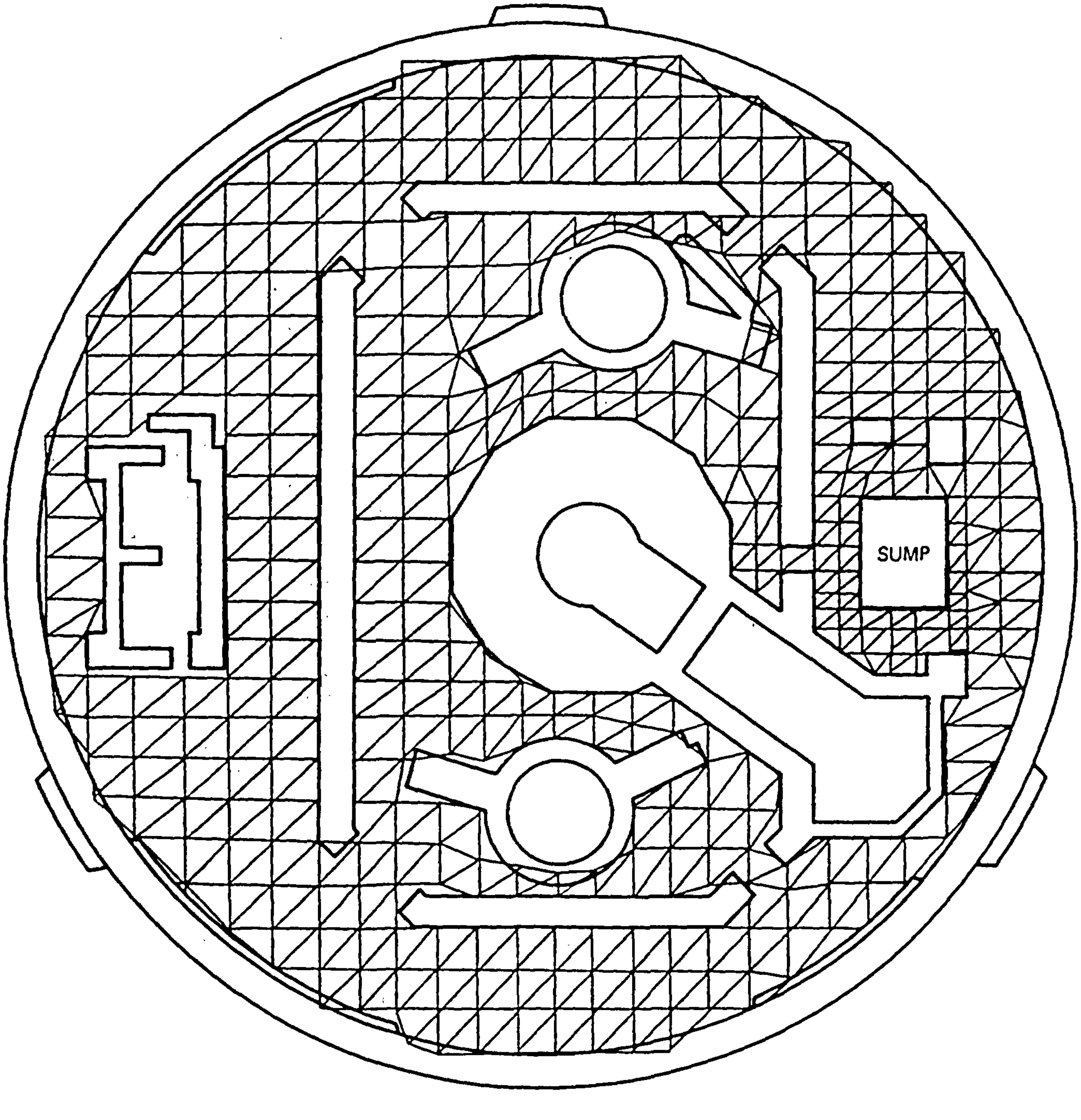
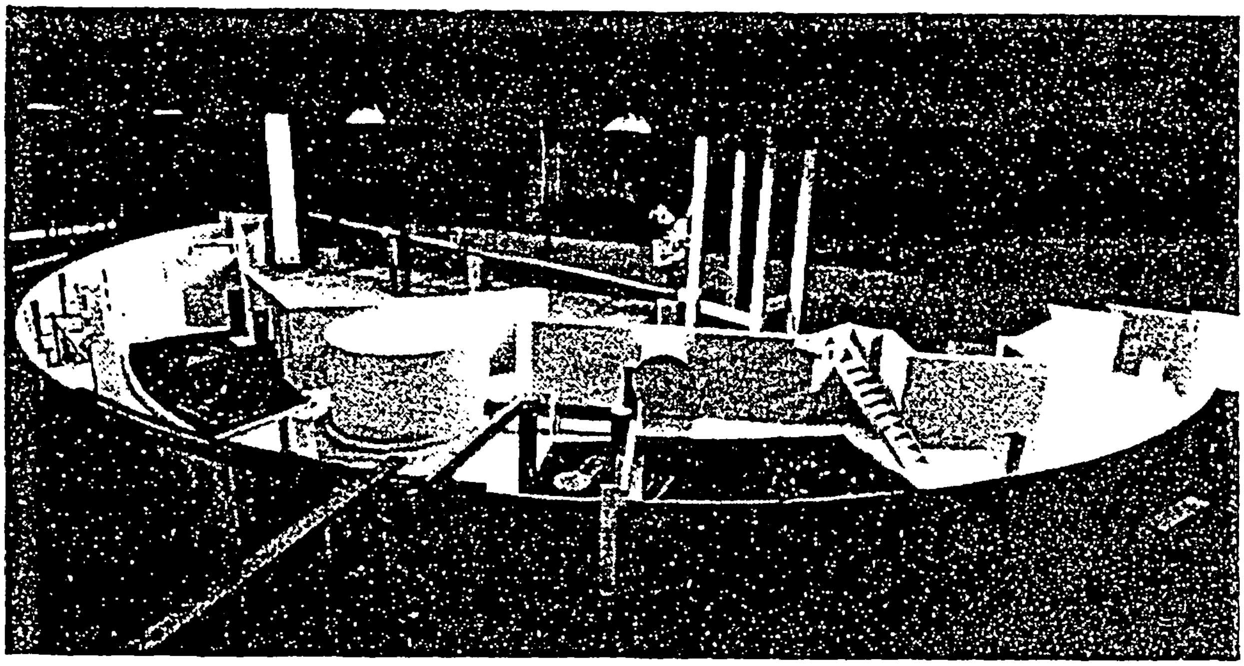


FIGURE D-4 TWO DIMENSIONAL REPRESENTATION OF CONTAINMENT RECIRCULATION FLOW



NOTE: PHOTO COURTESY OF ALDEN RESEARCH LABORATORY,
WORCESTER POLYTECHNIC INSTITUTE

FIGURE D-5 HYDRAULIC MODEL OF CONTAINMENT FLOOR AREA

APPENDIX E

MIRROR® INSULATION PERFORMANCE DURING LOCA CONDITIONS

PROVIDED BY

DIAMOND COMPANY

HDR TEST RESULT SUMMARY
MIRROR® INSULATION PERFORMANCE DURING LOCA CONDITIONS
DCN AE6609-111984-02

Prepared By:
Kim T. Gilbert
K. T. Gilbert
Senior Engineer
Technical Support
Mirror Insulation

Reviewed By:
R. L. Patel
R. L. Patel
Senior Group Supervisor
Technical Support
Mirror Insulation

Approved By:
Burton Ziels
B. D. Ziels
Manager
Mirror Engineering

Introduction

Insulation reaction to LOCA jet forces ultimately relates to the Emergency Core Cooling System's ability to perform its intended function, since insulation debris has the potential to block sump screens and reduce the pump's ability to recirculate the cooling medium. Based on postulated damage thresholds and modes, testing has been performed to determine the transport potential and resulting screen blockage patterns for components of metallic reflective insulation (NUREG/CR-3616). The subject testing was designed to answer questions related to how reflective insulation reacts when exposed to LOCA magnitude jet forces:

- How is metallic reflective insulation damaged by jet forces?
- Will it be removed from the pipe or remain in place as installed?
- If the insulation is torn apart by the jet forces, what sizes and shapes of debris will be generated?

The test results summarized in the following pages provide valuable insight for understanding the fundamental questions of damage potential and mode, so that the most realistic assessment of screen blockage potential can be made. The test results summarized here are the only test results available that provide information on reflective insulation reaction to high pressure jet forces. For more details of the test program, please refer to Diamond Power Specialty Co. report #DCN AE6609-111984-01, "HDR TEST RESULTS, MIRROR® INSULATION PERFORMANCE DURING LOCA CONDITIONS."

Test Facility Description

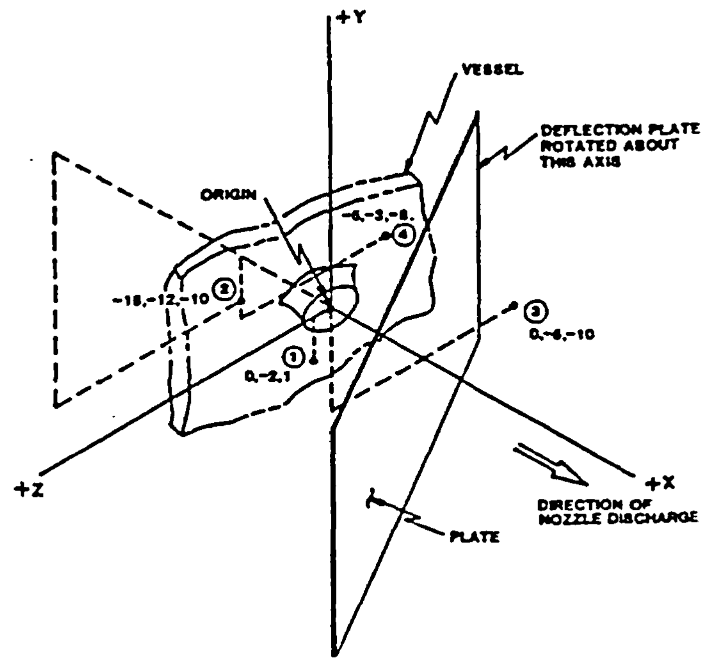
A decommissioned nuclear reactor (100MW prototype BWR) has been refitted to allow full scale testing with conditions representative of commercial LWR operation. The "HDR" (paraphrased meaning "superheated steam reactor") is a real nuclear power plant which offers test capabilities relevant to modern commercial plants.

The initial thermal hydraulic state for this specific HDR blowdown test was 110 bar, 318°C (saturated steam), which reflects nominal PWR operating conditions. The reactor vessel design parameters include 10m height, 2.96m I.D., 75m³ volume, 110 bar design pressure and 360°C design temperature. The design basis break was simulated by means of a set of rupture disks mounted in a 450mm I.D. nozzle. Approximately four feet from the nozzle end, a large deflection plate is located to guide the jet away from direct impact with the compartment walls.

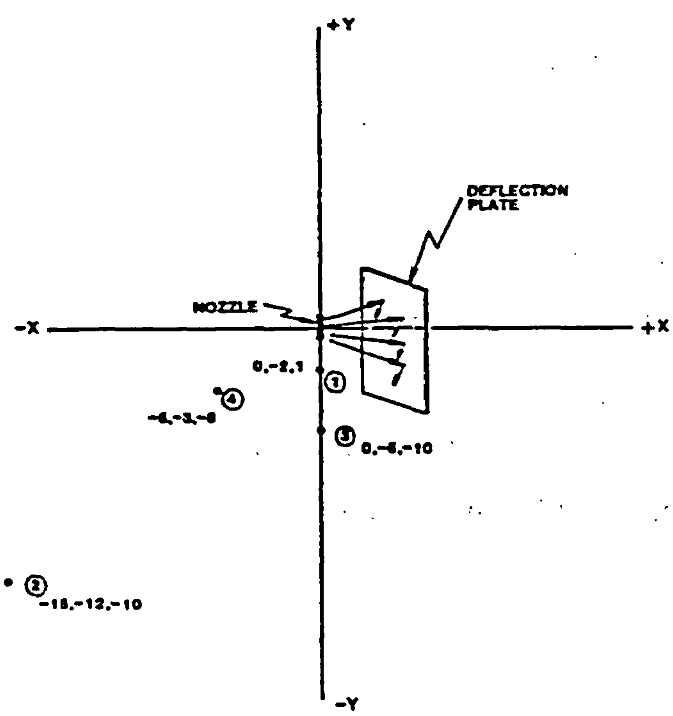
Four specimens of MIRROR® metallic reflective insulation were provided for simultaneous testing. Each of the specimens was designed and manufactured using standard practice and materials. All materials used were 304 stainless steel and all fasteners (screws, buckles, pop rivets) were standard production components. The location of the test specimens relative to the test nozzle and deflection plate are shown in Figure 1.

- Origin = Center of nozzle discharge
- X Axis along axis of nozzle
- Dimensions to approximate center of installed specimen
- Coordinates are (X,Y,Z)

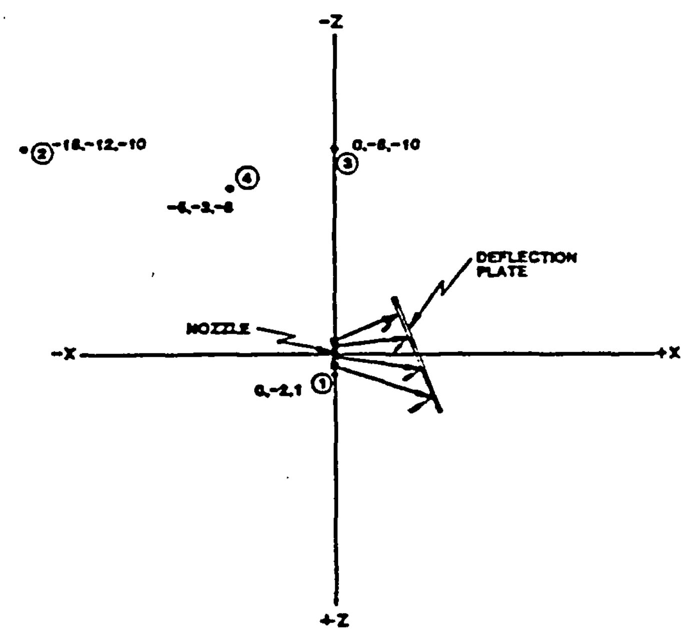
SPECIMEN NUMBER	DISTANCE FROM NOZZLE DISCHARGE (APPROXIMATE)
1	2.5 FT
2	22 FT
3	11 FT
4	10 FT



Isometric



View From X-Y Plane or Elevation View



View From X-Z Plane or Plan View

Figure 1 Isometric location drawing

Test Results

"Before" and "after" photographs support the following general observations:

- No large, flat pieces of sheet metal were released from insulation units.

- Forces required to "tear apart" insulation units tear and deform thin gage liner material into many irregular shaped and/or small pieces.

- Insulation installed farther than approximately 10 feet from the break location remained in its installed location and essentially undamaged.

- Metallic components/debris did not affect test (measurement) instrumentation or plant instrumentation.

- No insulation debris was transported outside the test compartment by either the blowdown jet forces or subsequent flow velocities.

Representative photographs are included on the following pages. The arrows superimposed on the "before test" photographs represent the best estimate of the steam jet directions relative to the insulation specimens.

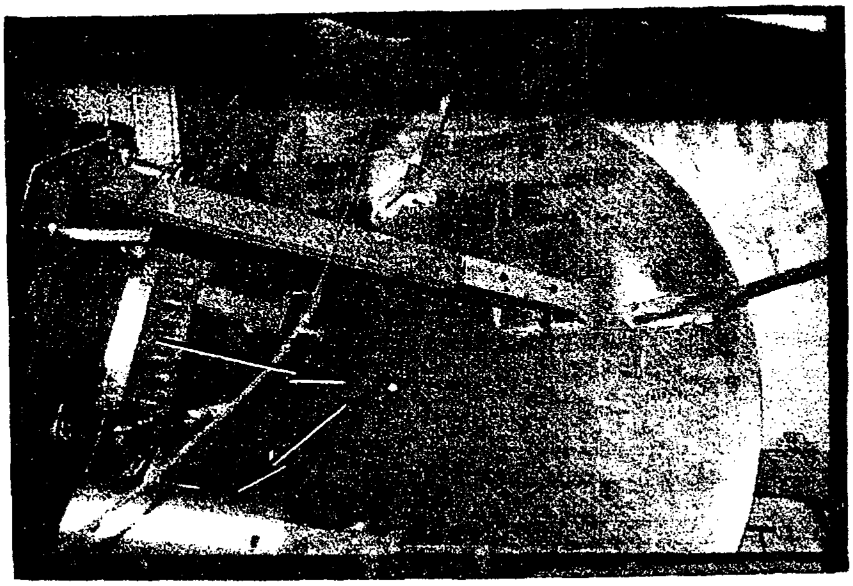


Figure 2 Specimen 1 before test

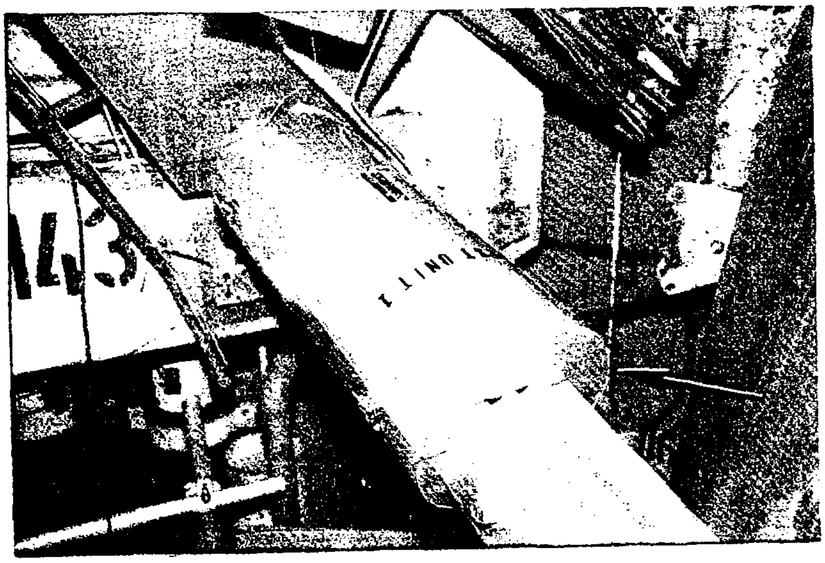


Figure 3 Specimen 1 before test

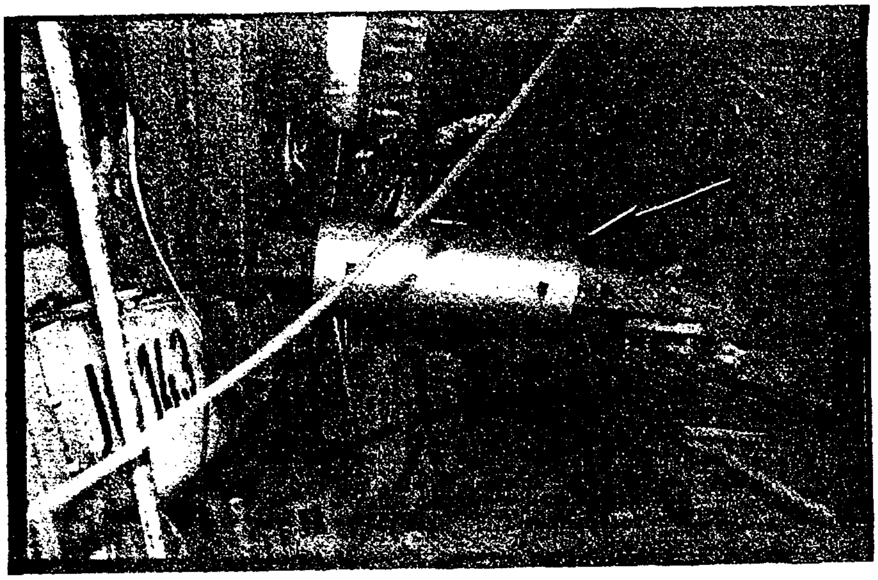


Figure 4 Specimen 1 before test

SPECIMEN 1, BEFORE TEST

Figure 2 shows the insulation (in the lower left corner) relative to the nozzle and the deflector plate. Figure 3 shows a closeup of the test unit with the strut connection to the deflector plate in the lower right hand corner. Figure 4 clearly shows the test specimen installed on the strut.

Details:

O.D. of insulation=12"
Length of Unit=30"
Thickness of insul=2.0"
Liner material=.0025"
Material=All 304 S.S.
Distance from
nozzle ℄ ≅ 2.5 ft.

Figure 5 Specimen 1 after test

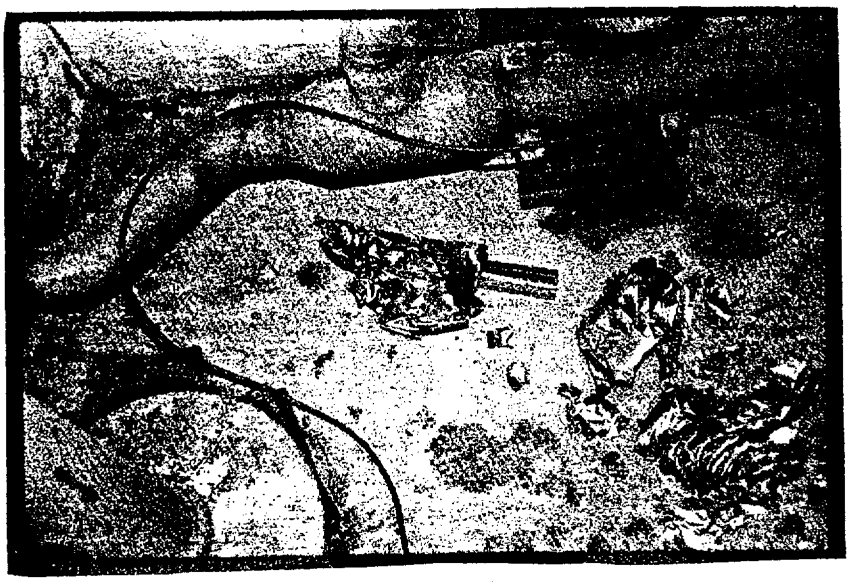


Figure 6 Specimen 1 after test

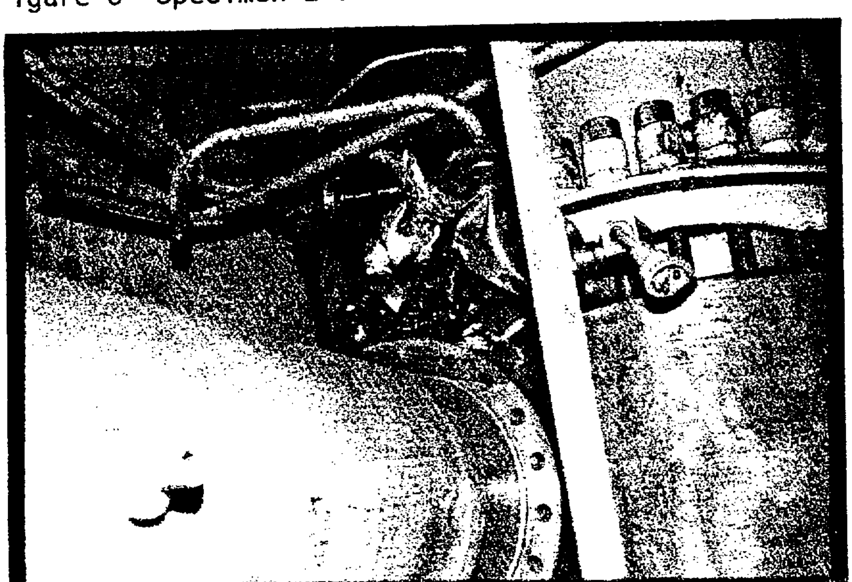


Figure 7 Specimen 1 after test

SPECIMEN 1
AFTER TEST

Figures 5 and 6 show debris from test specimen 1. These photos indicate that thin gauge liner material debris is torn and crumpled by forces required to tear apart the insulation section. Note that no evidence of large, flat sheets of liner material resulted from the test.

Figure 7 demonstrates the tendency of insulation sections to remain intact, even under severe destructive conditions. No components from this section were set loose, even though it is severely crushed and deformed.

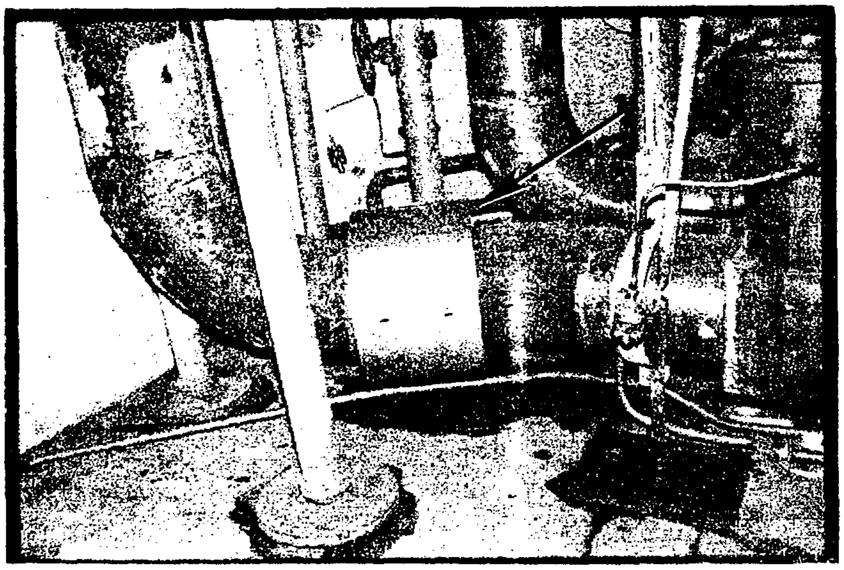


Figure 8 Specimen 2 after test

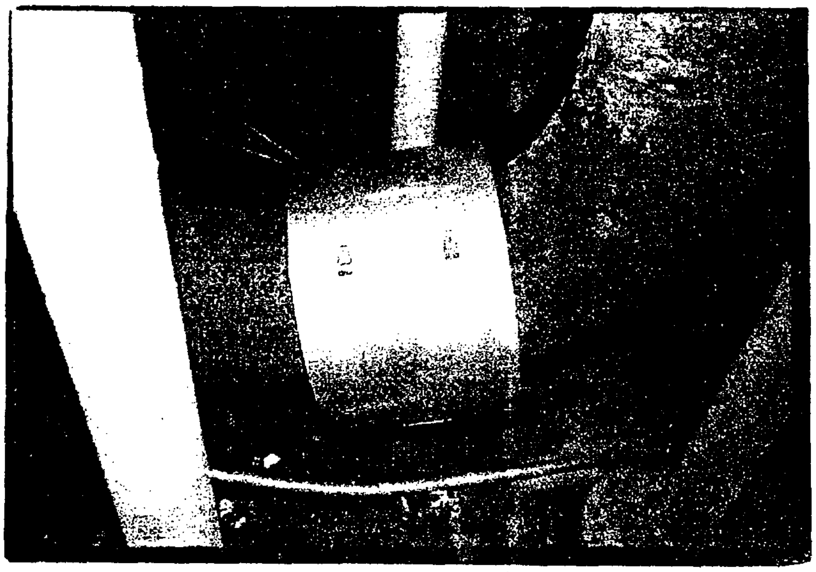


Figure 9 Specimen 2 after test

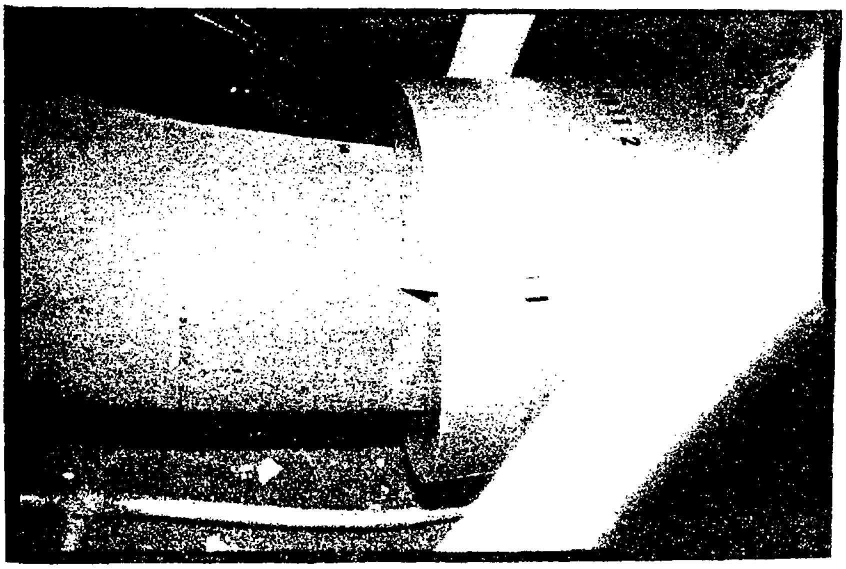


Figure 10 Specimen 2 after test

SPECIMEN 2

Figure 8 shows the insulation specimen installed on the pipe prior to the test.

Details:

O.D. of insulation=24"
Length of unit=15.75"
Thickness of insul=3.0"
Liner material = .0025"
Material = All 304 S.S.
Distance from
nozzle ℄ ≅ 22 ft.

Figures 9 and 10 show the test unit on the pipe following the test (photos from opposite side of pipe). Note that the test unit has been moved along the pipe and has sustained a small amount of deformation on one end disc due to the motion relative to very rough pipe surface.

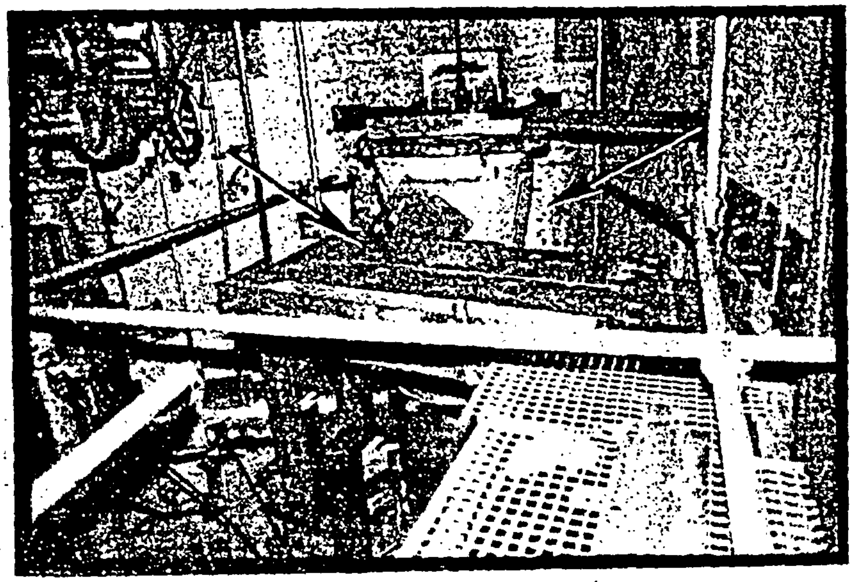


Figure 11 Specimen 3 after test

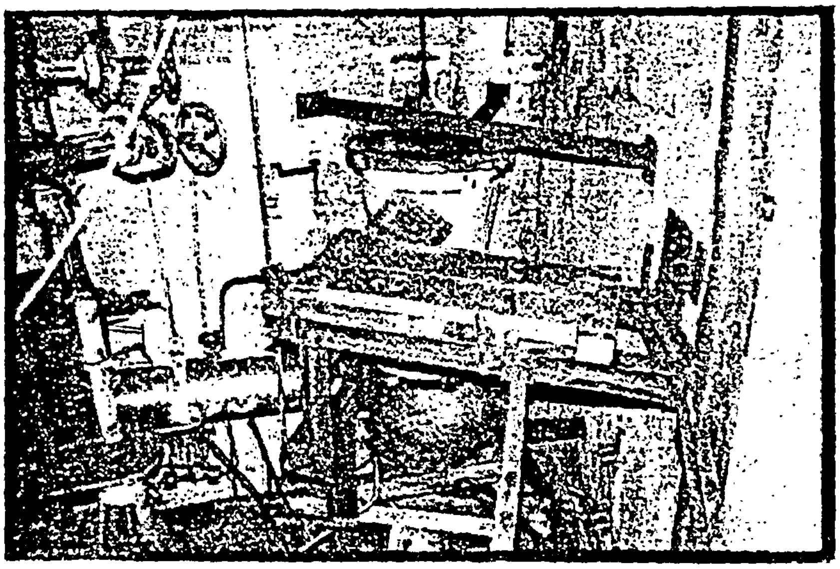


Figure 12 Specimen 3 after test

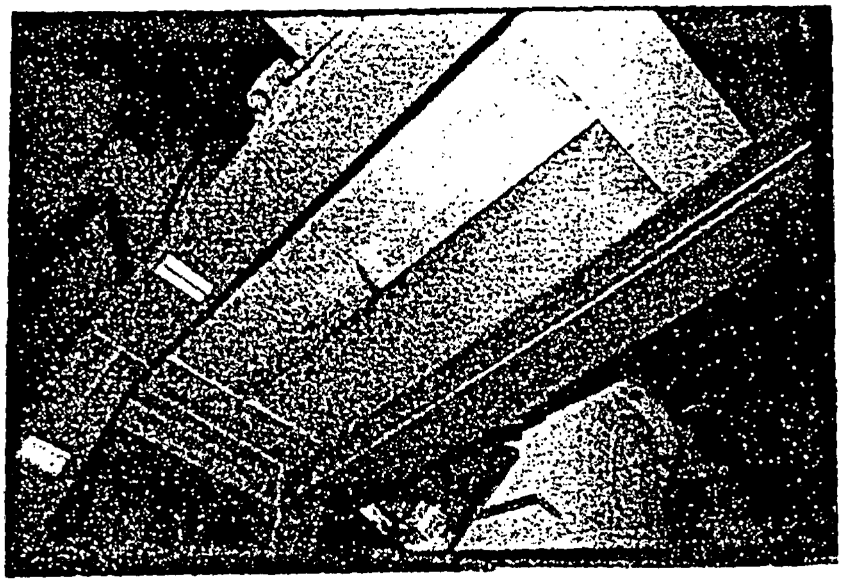


Figure 13 Specimen 3 after test

SPECIMEN 3

Figure 11 shows the insulation installed. The insulation is fastened together with buckles and screws and is supported on the edges on I-Beams.

Details:
Panel size: 11.7x46"
Thickness of insul=4.0"
Liner material=.0025"
Material=All 304 S.S.
Distance from nozzle ℄ ≈ 11 ft

Figures 12 and 13 show the insulation sections after the test was performed. No damage was apparent from above the insulation. The slight damage observed from below the specimen (Figure 13) must have been caused by impact from a foreign object, since no damage was observed from above.

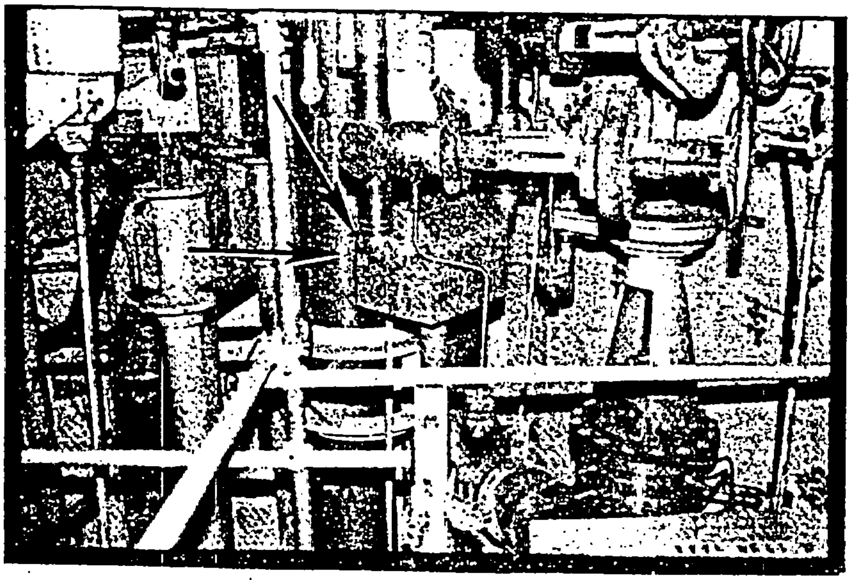


Figure 14 Specimen 4 after test

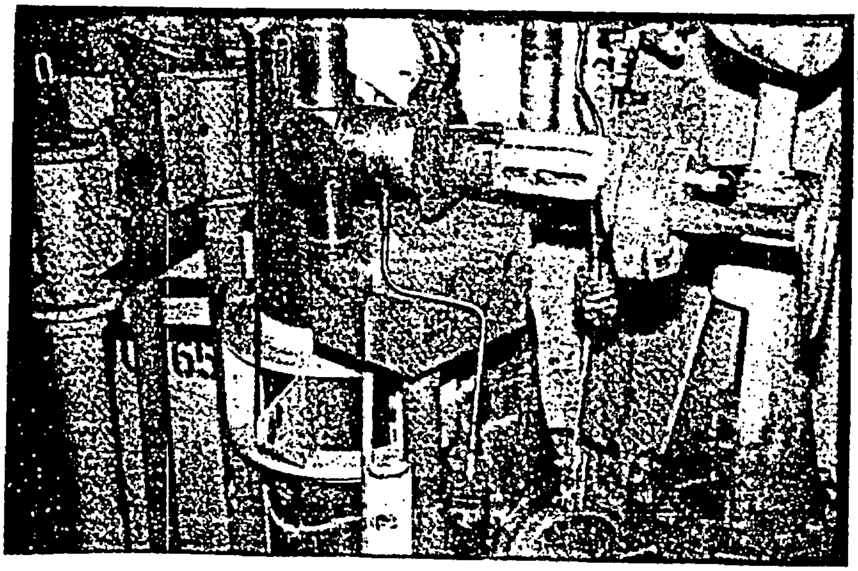


Figure 15 Specimen 4 after test

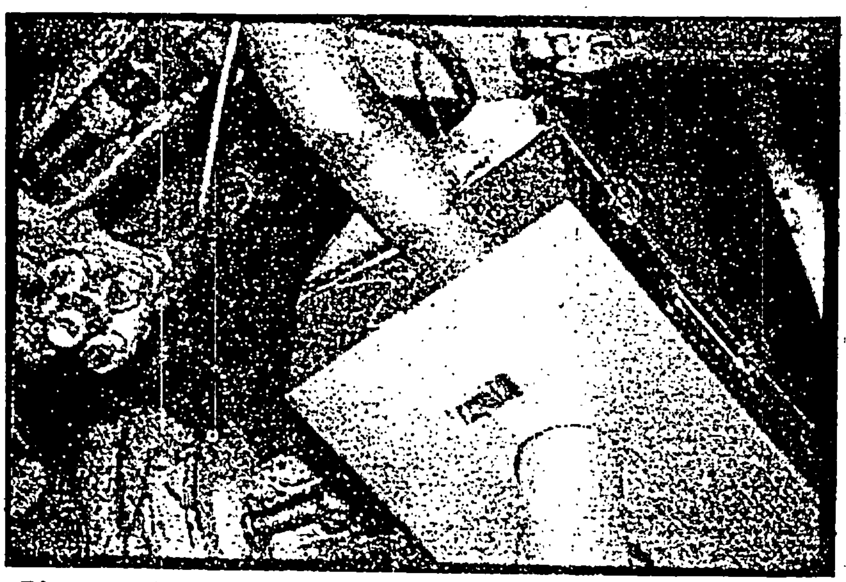


Figure 16 Specimen 4 after test

SPECIMEN 4

Figure 14 shows the U-shaped box insulation installed on a tee.

Details

Length of unit=16.0"
Thickness of insul=2.0"
Diameter of circular section = 12.0"
Liner material = All 304 S.S.
Distance from nozzle ℄ ≅ 10 ft.

Figures 15 and 16 show the test unit after the test was performed. Damage was confined to local areas around the end disc (vertical surfaces). Damage shown in the lower photograph is believed due to impact from a foreign object.

APPENDIX F

HDR BLOWDOWN TESTS WITH NUKON™ INSULATION BLANKETS

PROVIDED BY

OWENS-CORNING FIBERGLAS CORPORATION

Test Report:

HDR Blowdown Tests

With NUKON Insulation Blankets

Gordon H. Hart

Insulation Technology Laboratory

Owens-Corning Fiberglas Corporation

Research and Development Division

Granville, Ohio

Revised: March 27, 1985

SUMMARY:

This report summarizes the results of the two tests conducted this past summer at the HDR facility in West Germany. For Owens-Corning, the objective of these tests was to determine the capability of the NUKON nuclear containment area insulation to withstand the impact of a high pressure steam-water blast and to determine the size distribution of the fibrous insulation debris resulting from the impacted blankets. The report summarizes the test procedure and the results; it contains, in addition, "before" and "after" photographs and weight tables of the various components.

In short, the tests demonstrated that unjacketed NUKON blankets, or NUKON blankets covered in a metal mesh, that are located within nine pipe diameters of the simulated pipe break, can be totally destroyed but may not be, depending on the orientation (i.e., over 90 percent of the wool insulation was reduced to fine fibers). However, NUKON blankets enclosed in the standard NUKON 22 gage stainless steel jackets withstood the blast to such an extent that less than 50 percent of the metal jacketed wool insulation was reduced to fine fibers (for pipe insulation within seven pipe diameters from the simulated pipe break). These test results are unique to NUKON insulation systems since they likely depend on type and thickness of stainless steel jackets, strength of jacket latches, type of insulation wool, type of fabric covers, strength of fabric to fabric seams, strength of the Velcro joints, and strength of the Velcro to fabric seams. Further, it should be emphasized that the success of the metal jacketed NUKON pipe insulation in resisting the blast constitutes only two data points. These should not be used as points of extrapolation to cover different materials or conditions. While it is reasonable to assume that a flat NUKON blanket, covered with 22 gage stainless steel jacketing, would also resist damage by a water-steam jet blast, no actual data for this configuration was obtained in these tests.

ANALYSIS OF THESE RESULTS:

A. Overview:

Attachments 1 and 7 show the layouts of the nozzle, the impingement plate, and the support strut for both Tests No. 1 and 2. The center of the impingement plate was positioned 1.5 m from the burst plate of the nozzle. The impingement plate was a 2.6 m diameter disk with its center positioned 2.0 m from the ceiling, or its upper edge 0.85 m from the ceiling (note that the plate and the ceiling are not perpendicular).

Set-up for Test No. 1 (conducted on June 15):

A single blanket of NUKON pipe insulation (measuring 870 mm long, 50 mm thick, and of adequate stretch-out to cover a 100 mm by 120 mm rectangular bar that was used to simulate a pipe) was placed on one of three rectangular steel struts. See Attachment 2 for positioning of the pipe blanket relative to the 450 mm inner diameter nozzle. This blanket was left unjacketed. The center of this blanket was located within a 350mm cone of the nozzle, representing 0.8 pipe diameters (0.8 D). However, as it was likely for this blanket to be hit by water reflected from the impingement plate, the reflected distance was 1850mm, or 4.1 D (to the center of the blanket in both cases). Two flat blankets, each measuring 500 mm by 750 mm by 50 mm (thick), were attached to the ceiling, directly above the axis of the jet nozzle; see Attachments 1A and 1B. This plate was oriented perpendicular to the axis of the nozzle. See Attachment 3 for photographs of the flat blankets before the test. Attachments 1A and 1B show the position of the nozzle, the insulated bar, the impingement plate, and other support elements. The impingement plate was positioned 1.5 m from the burst plate of the nozzle with the insulated strut extending between the two and slightly below the center axis of the nozzle. The flat blankets were not located within a "90 degree cone" extending out from the center of the nozzle. Therefore, for the purposes of impact, their distance from the nozzle, was calculated to be 3320mm or 7.4 D pipe diameters, assuming that they would be impacted by water from the impingement plate. See the NRC report, "Methodology for Evaluation of Insulation Debris Effects", pp. 22-26, for an explanation of the 90-degree cone extending out from the nozzle. The blankets were attached to the ceiling with Velcro strips and pins with speed washers (with the pins imbedded into the concrete ceiling).

Results for Test No. 1:

Both types of unjacketed NUKON blankets were badly destroyed by the jet blast which originated from water-steam heated to 310 degrees C and 110 bar. The flat insulation was totally destroyed, with only pieces of cloth, which were caught on pipe supports, able to be retrieved. The pipe insulation was largely destroyed although 15 percent of the wool was left intact, enclosed in the fiberglass fabric. All pieces were located around the test room but none in original positions. See Attachments 4 and 5 for photographs of the resulting fabric material. These results are inconsistent since material located 7.4D from the nozzle was totally destroyed while other material, located 4.1D from the nozzle, was left 15 percent intact.

Set-up for Test No. 2: (Conducted on July 4)

For the second test, the impingement plate was angled about 30 degrees from the center line axis. With this orientation, a greater strut length was available for insulating. Therefore, two pieces of pipe insulation blankets, each of the same size as for Test No. 1, were able to be placed on the strut that was on the side impacted by water reflecting off the angled impingement plate. Each pipe blanket was covered with standard NUKON 22 gage stainless steel jacketing. These shall be referred to as Pipe Blankets A and B. See Attachments 7A, and 7B, and 8, lower photograph.

The flat blankets were positioned exactly as for Test No. 1. This time, however, they were covered with metal scrim jacketing. See Attachment 8, upper photograph. The test conditions for Test No. 2 were about the same as for Test No. 1: 310 degrees C, 110 bar.

Results for Test No. 2:

The flat blankets, located 7.4 D from the nozzle, were totally destroyed by the blast, as in Test No. 1. The fabric and the metal scrim were again strewn about and caught on components in the test chamber. Of the two pipe blankets, the one closest to the nozzle (Pipe Blanket A) was just slightly damaged, retaining 93 percent of its original weight of wool (the end of the blanket was slightly torn up). Its center was located 125mm, or 0.3D, from the nozzle itself, although it is likely that reflected water-steam had the greatest impact. For the reflected case, the distance was about 2830mm, or 6.3D. Blanket A remained in its original position. The one closest to the impingement plate (Pipe Blanket B) was partially damaged, retaining 25 percent of its original weight of wool. Its metal jacketing, badly deformed, remained on the bar as did the piece of blanket that contained the wool. This latter blanket had its center located 1350mm, or 3.0D from the nozzle, although for reflected water-steam, its distance was 1930mm, or 4.3D. See Attachments 9 and 10 for photographs of this pipe insulation after the blast.

Attachment 11 shows photographs of the metal jacketing for Pipe Blanket B. A more careful examination of this jacket leads to several conclusions. The majority of the jacket damage can be attributed to the water-steam pressure. The rivets for the latches all appear to be blown radially straight out by an internal pressure. Most cracks in the steel had been initiated from an internal pressure pushing out. The fracture shown in Attachment 11 occurred along the initial bend of the rectangular jacket; apparently, internal water pressure ripped the metal jacket where the added stress of the bend caused a weak spot. There was some question as to whether or not the burst plate damaged the steel jacketing. Two cracks in the jacket showed very clean edges and evidence of abrasion. It is quite possible that they were caused by the flying burst plate. Dents and cracks in Attachment 12 strengthen this conclusion.

B. Summary of the Tests:

Table 1 gives a summary of the weights of the blankets both before and after each test. The flat blankets are, of course, separated from the pipe blankets on this weight table. The Velcro, used to attach the pipe blankets and sewn onto the fabric, is treated as part of the fabric. The weights of the NUKON base wool are separated since the wool, not the fabric, is considered to pose the greater sump blockage potential and hence its fate was of most interest in this testing. In Table 1, what is of greatest significance is the difference between the results of Test No. 1 and Test No. 2. By metal jacketing the pipe insulation, the percentage of pipe wool reduced to debris was dropped from 85 percent to 41 percent. This is significant because it demonstrates the effectiveness of metal jacketing in protecting the blankets. It also demonstrates that a substantial portion of the wool insulation, that started within seven pipe diameters (7D) of the break, was not reduced to fine fibers. This contradicts the assumption made by the NRC that all fibrous insulation located within 7D of a break would be reduced to fine fibers by a blast.

On the other hand, the flat blankets, placed on the ceiling directly above the impingement plate and at 7.4D of the break, were totally reduced to fibrous debris. The addition of the metal mesh jacketing apparently had no effect whatever in protecting the blanket. On this basis, if calculations show there is a need to reduce the sump blockage potential, it is recommended that flat, or nearly flat, blankets placed on steam generators be covered by stainless steel jacketing, not by mesh. However, it would be advisable to obtain actual test data on flat, metal jacketed blankets subjected to a blast.

C. Previous Jet Impingement Tests:

The NRC assumption, that all blankets within 7D of a break will be reduced to loose fiber debris, is a rational one. It is based on "jet impingement" tests conducted in 1982 and 1983 at the Alden Research Laboratories (ARL). These tests demonstrated that, in the worst case, blankets made of fibrous insulation will be torn apart by dynamic water pressures of 20 psig or greater when located within a "90 degree cone". Using this pressure, the NRC backed out the "7D" assumption. On this basis, the assumption is rational.

However, the ARL tests were performed using cold water emerging from a two-inch diameter nozzle. In an actual two-phase blast, such as would occur in a pressurized water reactor containment area accident, the water-steam jet would have less momentum at 20 psig than a cold water jet, hence it would have less destructive potential. Also, because it could not constitute a defined jet, it would likely have less destructive potential. However, the NRC was justified in using the ARL data because it was the only data available at that time.

The two HDR tests, showing that metal jacketing can be used to protect fibrous insulation, really only constitutes a single data "point". That data point should not be extrapolated in other directions to predict the behavior of other types of wool, fabric, stitching, metal jacketing, latches, or insulation system designs. A variation in any of these variables could have had a profound effect on the results presented in the two OCF tests conducted at HDR.

D. Issue of Size Distribution of Fibrous Debris:

One of the original objectives of this testing was to obtain a size distribution of the fibrous debris. This distribution, it was reasoned, could then be used with ARL water transport data to calculate the quantity of debris that could reach a sump screen in a specific plant sump analysis. However, such a size distribution could not be obtained. The wool that was torn from the blankets was not able to be found and, hence, was assumed to be entirely reduced to loose fibers. All the wool retrieved was still enclosed in fiberglass fabric; hence, its size distribution was not an issue (i.e., enclosed in fabric, it would not transport to the sump screen as loose fibers). Therefore, the results of the test were binary: wool that remained enclosed in the fabric was not transportable, while wool that was torn from the fabric enclosure was reduced to loose fibers.

E. Conclusions and Recommendations:

From the HDR Blowdown Tests No. 1 and 2 on NUKON insulation blankets, several conclusions can be drawn:

1. Unjacketed blankets, and those jacketed in metal mesh, located about 7.4 pipe diameters from the jet nozzle, were reduced to shredded fabric and unretrievable loose insulation fibers. Most of the fabric generated by the tests was caught on protrusions in the area. The wool not retrieved was assumed to be reduced to loose fibers. On the other hand, unjacketed pipe insulation, located within 0.8D of the nozzle, was only 85 percent destroyed.

2. Some of the 22 gage metal pipe jacketing in Test No. 2 was badly bent by the blast; however, it was not torn away from its position around the strut it had covered. The suggestion is that the reflected jet, rather than the primary jet, inflicted the greatest damage.

3. The use of metal jacketing over pipe blankets was effective in reducing the level of wool destruction from 85 percent (Test No. 1) to 41 percent (Test No. 2), or 75 percent for Pipe Blanket B and 7 percent for Pipe Blanket A.

It is recommended that for sump analyses involving pipes insulated with metal jacketed NUKON blankets, Attachment 13 replace Figure 3.26 in the NRC report, NUREG-0897. The curves on these graphs have been redrawn by using data collected in these tests. It is also recommended that, if possible, further testing be conducted. This would include metal jacketed flat NUKON blankets and insulation samples placed at various other positions and orientations. Ideally, the impingement plate should be removed and insulation samples should be impacted by the primary jet, not only by reflected water.

Finally, it is recommeded that these results not be extended to insulation materials fabricated with different gage metal jacketing, matal latches, compressibility of insulation, etc. Variations could have a profound effect on the results. Also, caution should be urged on extrapolating these results to so-called "encapsulated" insulation since that is not a clearly defined type of insulation and since its behavior could be significantly different.

TEST #1

Original Weights (kg)*

Piece of NUKON	Cloth	Scrim	Velcro	Wool	Total
1. Pipe	1.03	0.09	0.025	1.50	2.66
2. Flat A	0.54	0.05	----	0.70	1.27
3. Flat B	0.54	0.05	----	0.70	1.29

Comparative Weights from Test #1 (kg)

Piece	Cloth & Velcro			Wool		
	Before*	After	% Lost	Before*	After	% Lost
1. Pipe	1.05	0.83	21%	1.50	0.22	85%
2. Flat A	0.54	0.51	} 12%	0.70	0	100%
3. Flat B	0.54	0.44		0.70	0	100%

TEST #2

Original Weights (kg)*

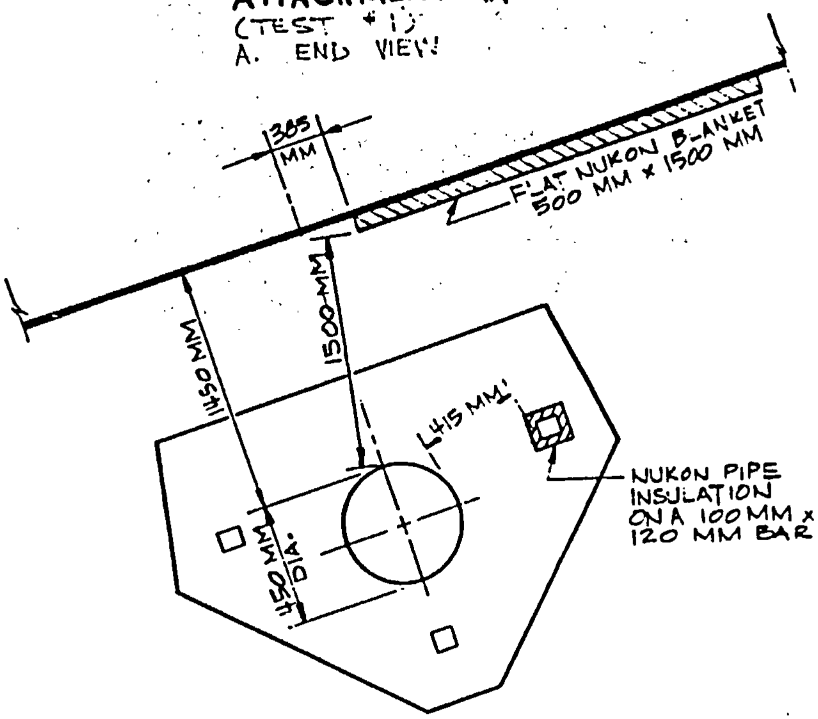
NUKON Blanket	Cloth	Scrim	Velcro	Wool	Total
1. Pipe A	1.03	0.09	0.025	1.50	2.66
2. Pipe B	1.03	0.09	0.025	1.50	2.66
3. Flat A	0.54	0.05	----	0.70	1.30
4. Flat B	0.54	0.05	----	0.70	1.29

Comparative Weights from Test #2 (kg)

NUKON Blanket	Cloth & Velcro			Wool			
	Before*	After	% Lost	Before*	After	% Lost	
1. Pipe A	1.05	1.03	} 80%	1.50	1.39	7%	} 41%
2. Pipe B	1.05	0.64		1.50	0.38	75%	
3. Flat A	0.54	} 0.27		0.70	0	100%	
4. Flat B	0.54		----	0.70	0	100%	
5. Unidentifiable Fabric Scraps	----	0.49	----	----	----	----	
6. All Blankets	3.18	2.43	24%	----	----	----	

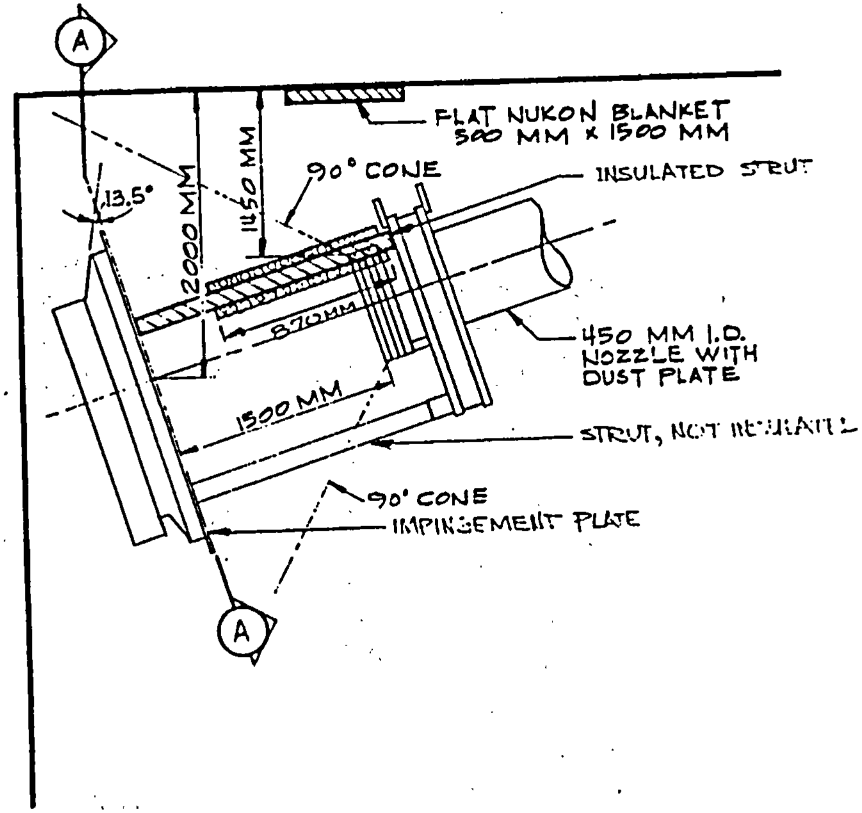
* Based on average values of the weights of the materials for six different blankets constructed for the tests.

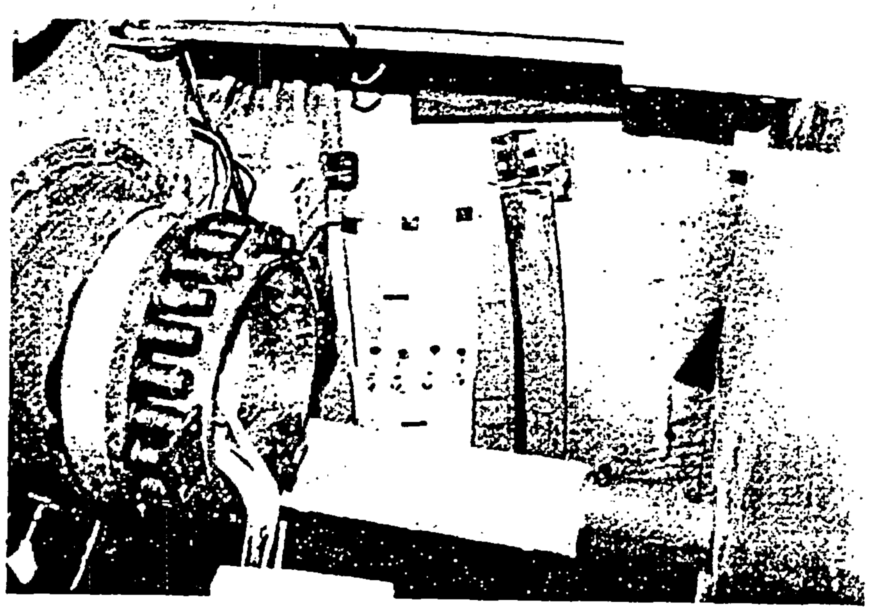
ATTACHMENT 1A
(TEST #1)
A. END VIEW



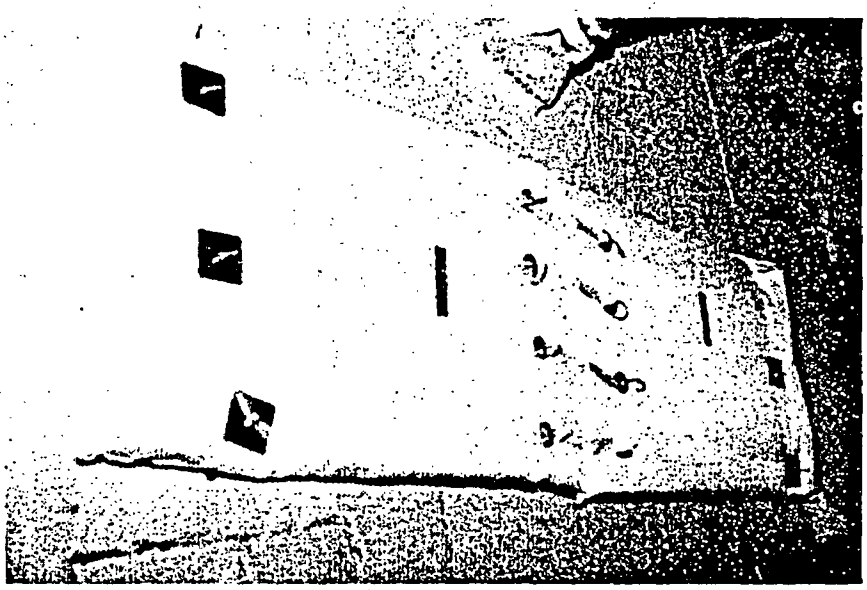
SECTION A-A

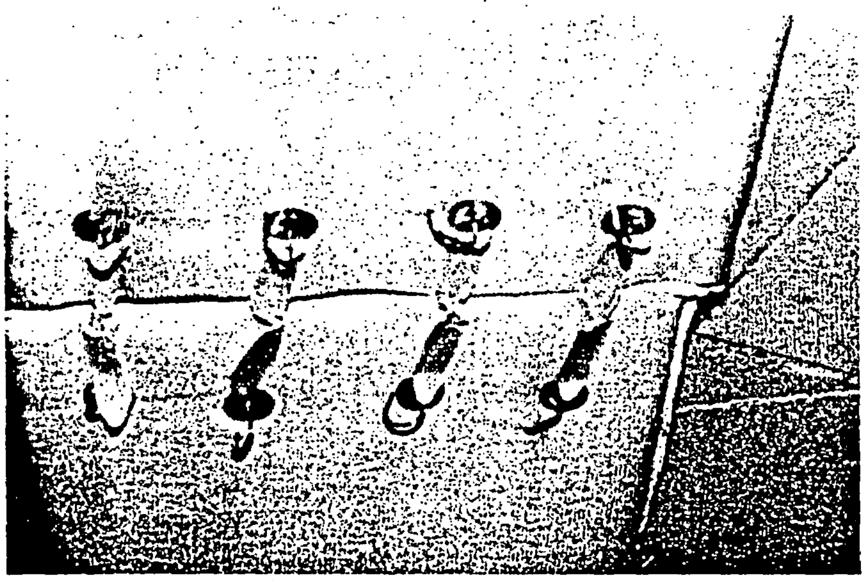
ATTACHMENT 1B
B. SIDE VIEW



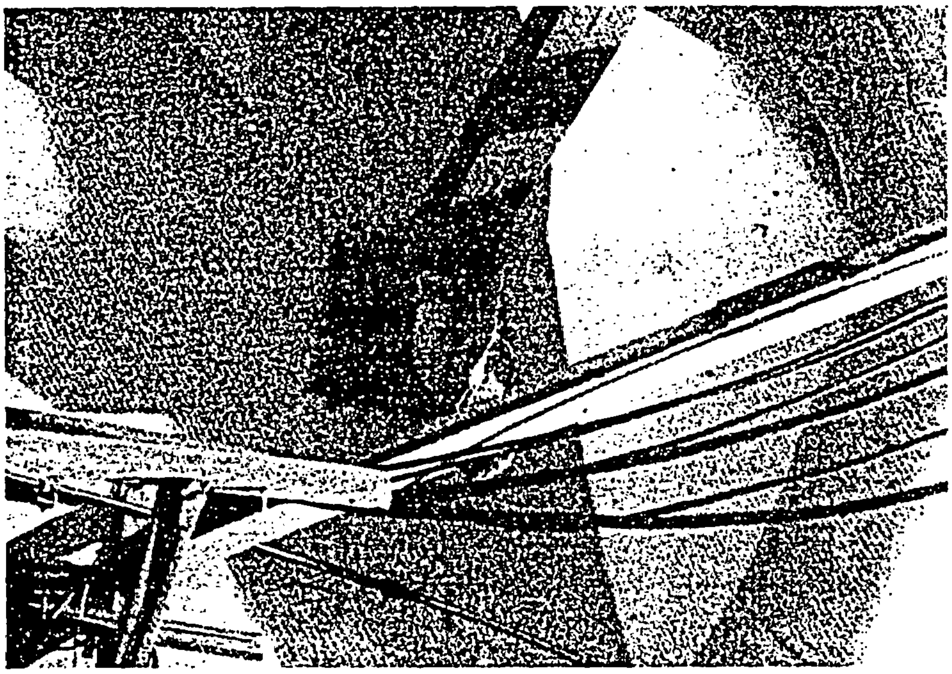


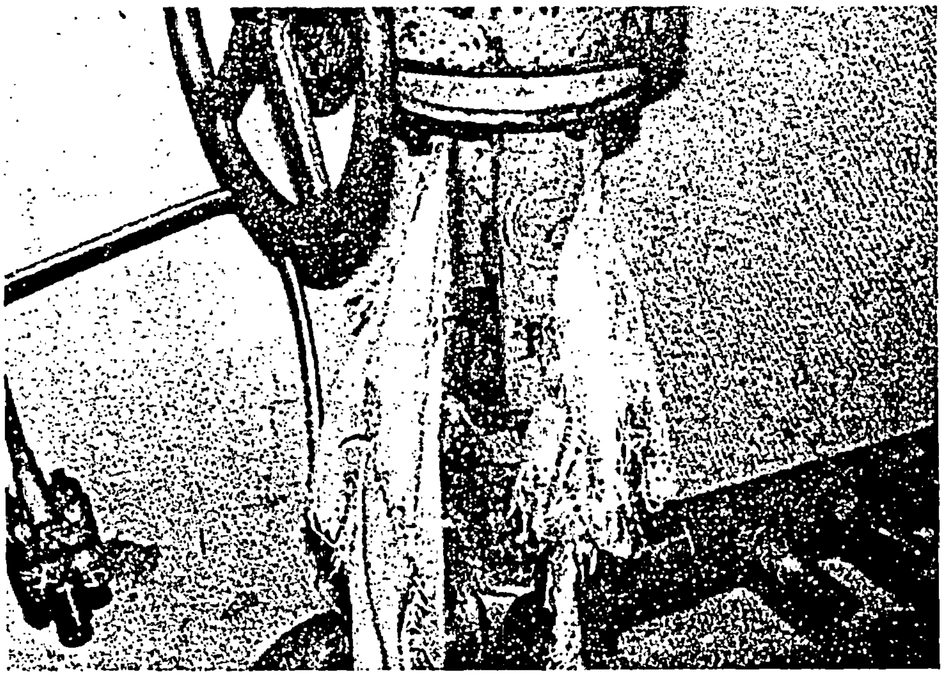
INSTALLED INSULATION BEFORE TEST 1





INSTALLATION OF FLAT BLANKETS BEFORE TEST 1



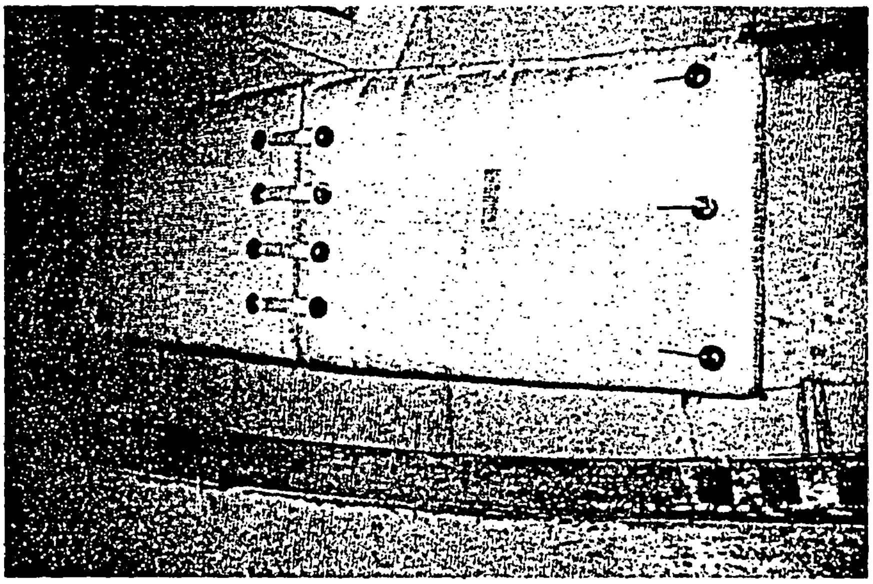


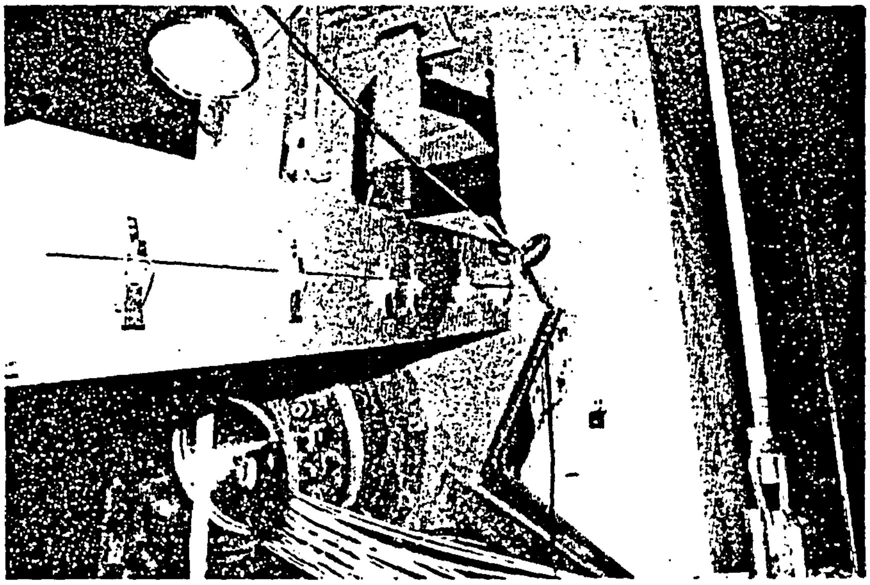
RESULTS OF TEST 1

RESULTS OF TEST 1

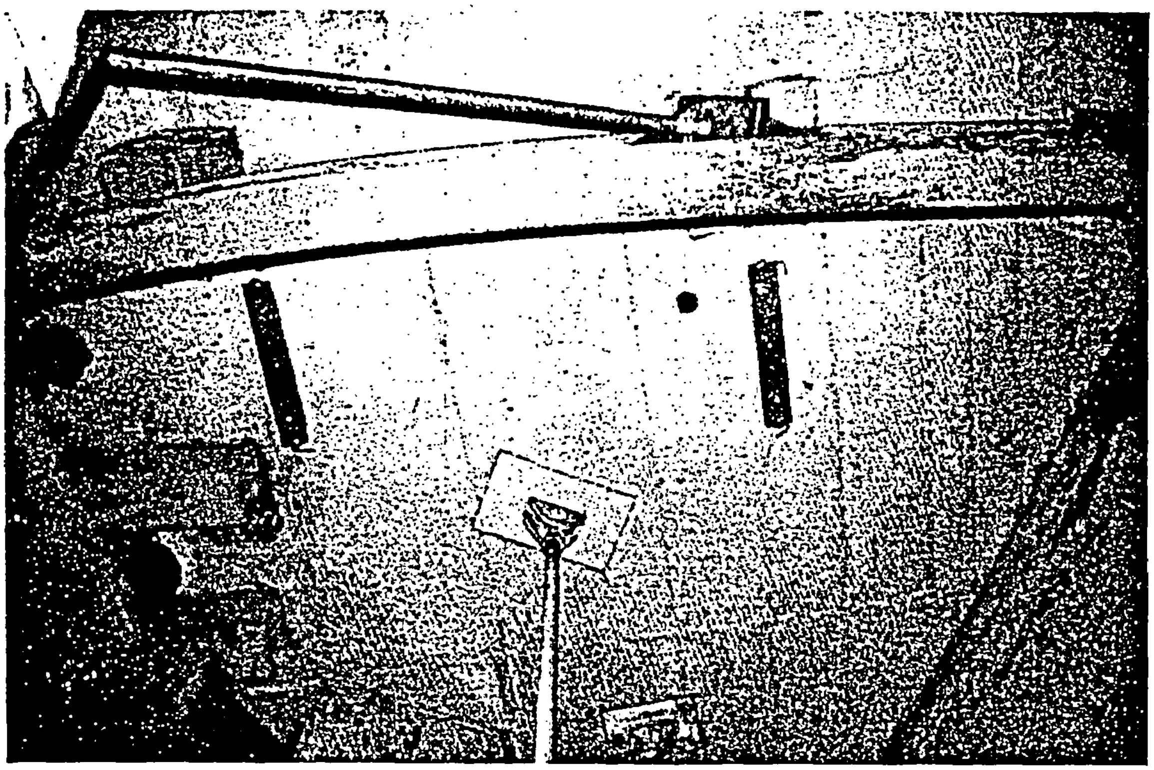
Attachment 8

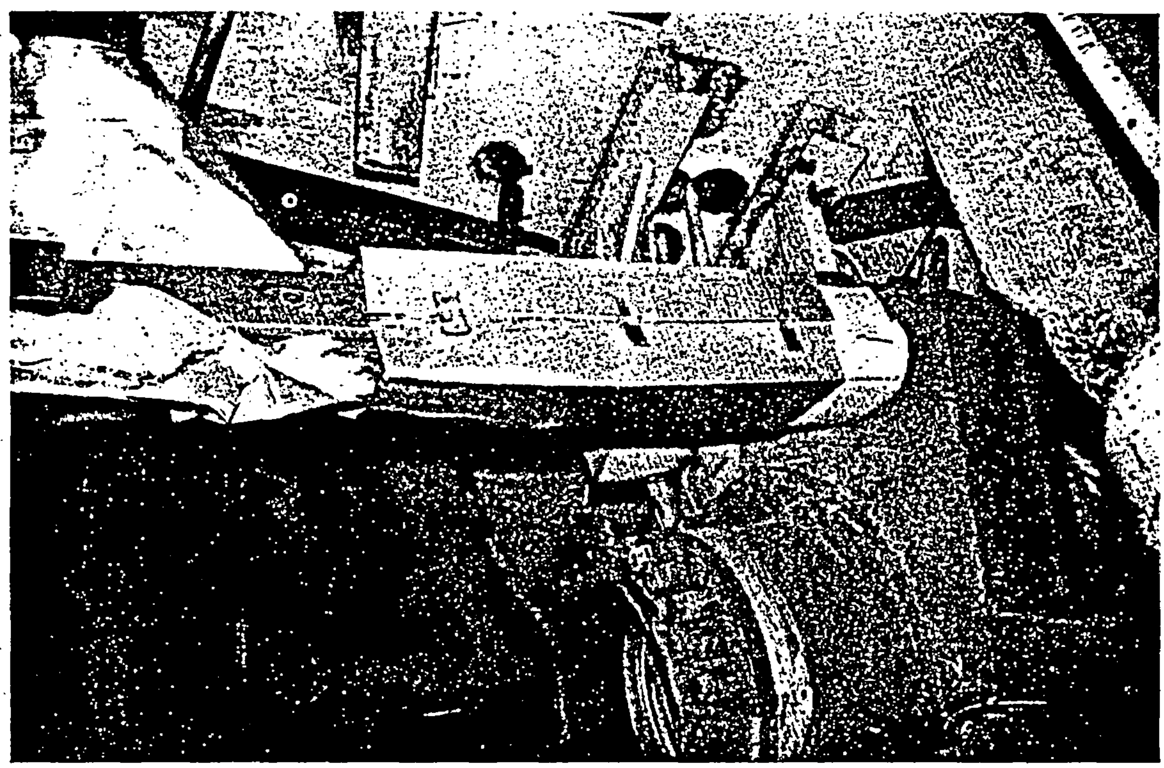
TR 35947



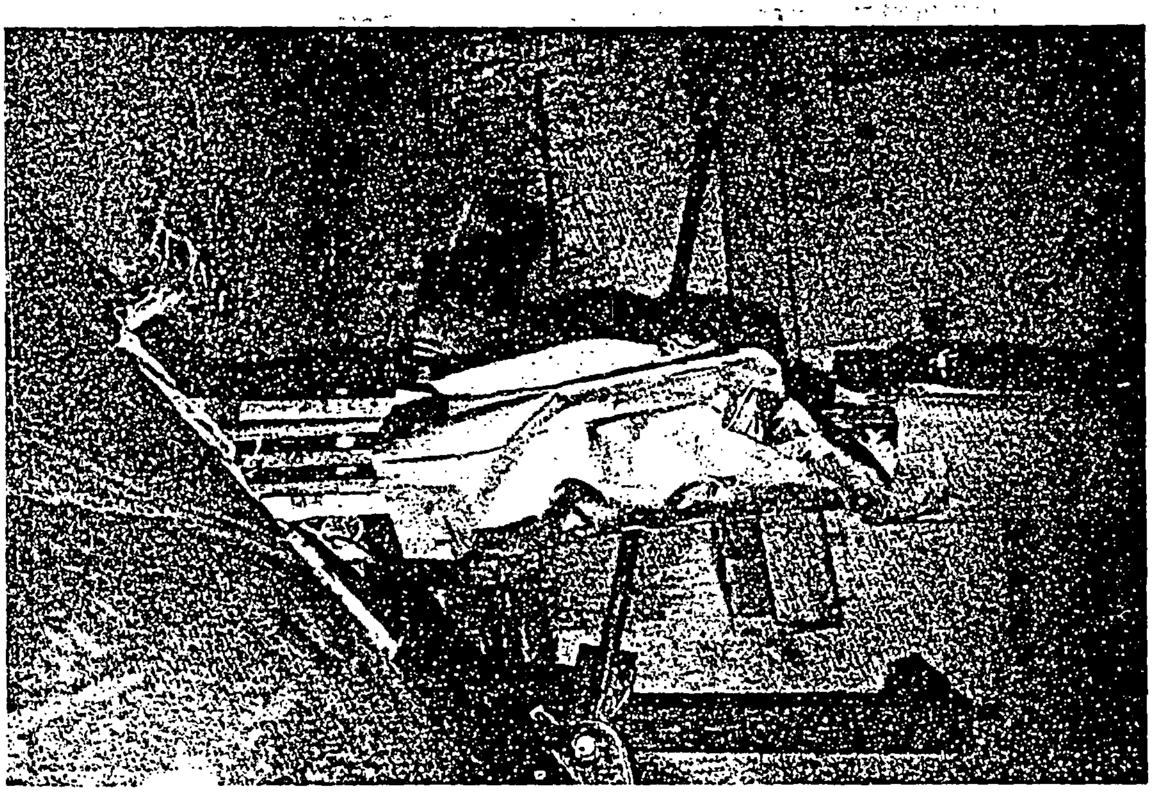


INSTALLED INSULATION BEFORE TEST 2

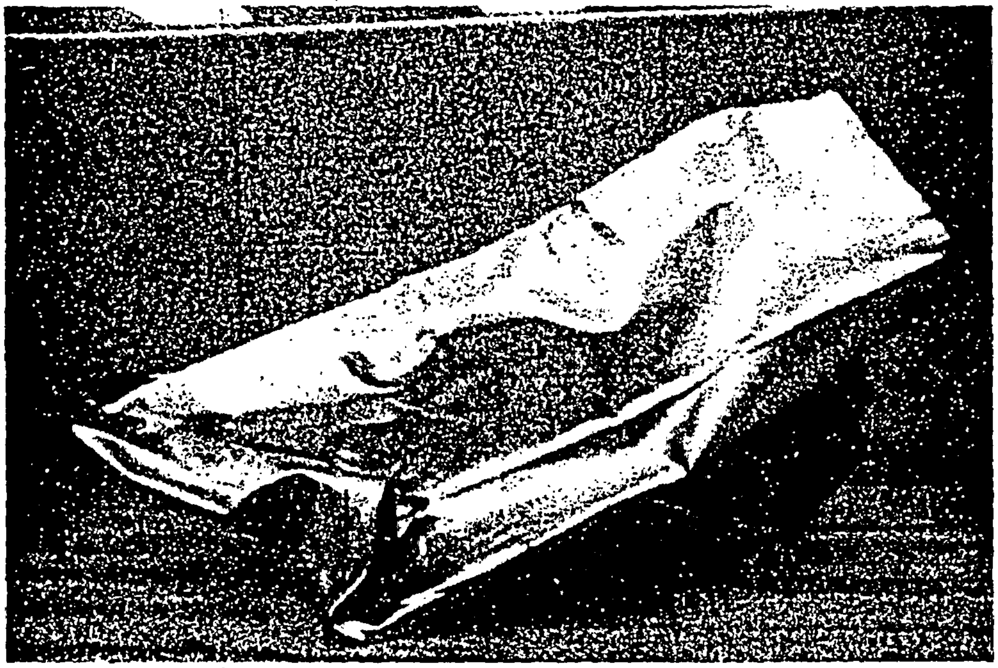




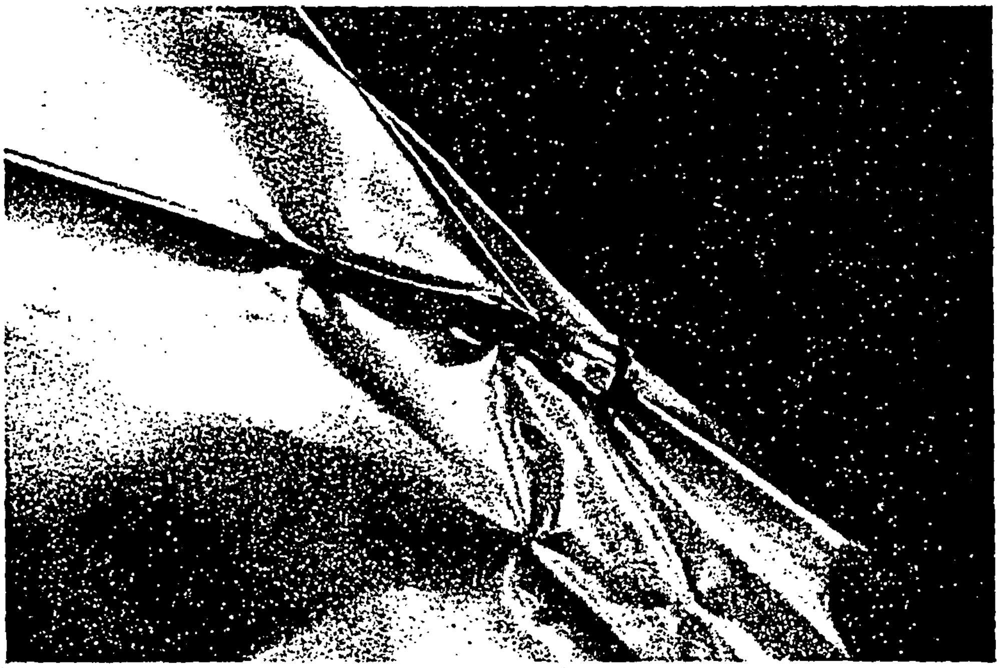
RESULTS OF TEST 2



RESULTS OF TEST 2



JACKET NEAREST THE IMPINGEMENT PLATE

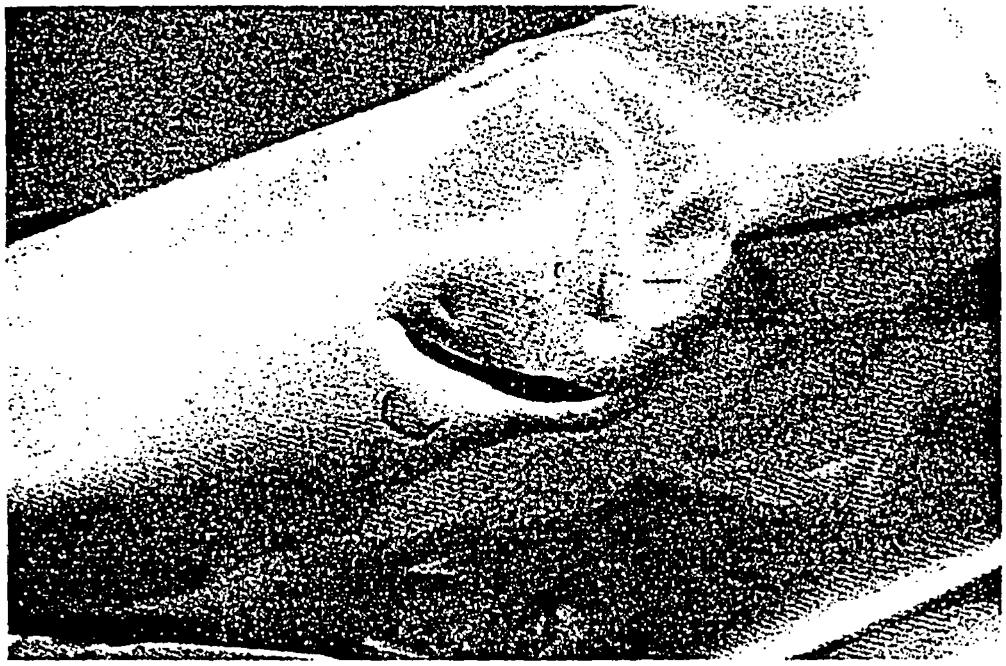


FRACTURE CAUSED BY WATER PRESSURE

STUDY OF STAINLESS STEEL JACKET FAILURE

Attachment 12

TR 35947



SUSPECTED BURST PLATE DAMAGE

INITIAL BURST PLATE DAMAGE PROPAGATED BY
INTERNAL WATER PRESSURE

STUDY OF STAINLESS STEEL JACKET FAILURE

NRC FORM 335 (2-84) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE.

1. REPORT NUMBER (Assigned by TIDC, add Vol No., if any)
NUREG-0897
Rev. 1

2. TITLE AND SUBTITLE
Containment Emergency Sump Performance
(Technical Findings Related to Unresolved Safety Issues)

3. LEAVE BLANK

4. DATE REPORT COMPLETED

MONTH	YEAR
March	1985

5. AUTHOR(S)
A. W. Serkiz

6. DATE REPORT ISSUED

MONTH	YEAR
October	1985

7. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Division of Safety Technology
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

8. PROJECT/TASK/WORK UNIT NUMBER

9. FIN OR GRANT NUMBER

10. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Same as 7. above.

11a. TYPE OF REPORT
Technical

b. PERIOD COVERED (Inclusive dates)

12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

This report summarizes key technical findings related to Unresolved Safety Issue (USI) A-43, Containment Emergency Sump Performance. Although this issue was formulated considering pressurized water reactor (PWR) sumps, the generic safety questions apply to both boiling water reactors (BWRs) and PWRs. Hence, both BWRs and PWRs are considered in this report.

The technical findings in this report provide information on post-LOCA recirculation performance. These findings have been derived from extensive experimental studies, generic plant studies, and assessments of sumps and pumps used for long-term cooling. The results of hydraulic tests have shown that the potential for air ingestion is less severe than previously hypothesized. The effects of debris blockage on NPSH margin must be dealt with on a plant-specific basis. These findings have been used to develop revisions to Regulatory Guide 1.82 and Standard Review Plan Section 6.2.2 (NUREG-0800).

14. DOCUMENT ANALYSIS - a. KEYWORDS/DESCRIPTORS
USI A-43
Technical Findings USI A-43
Containment Emergency Sump Performance

b. IDENTIFIERS/OPEN-ENDED TERMS

15. AVAILABILITY STATEMENT
Unlimited

16. SECURITY CLASSIFICATION
(This page) Unclassified
(This report) Unclassified

17. NUMBER OF PAGES

18. PRICE

NUREG-0869
Rev. 1

USI A-43 Regulatory Analysis

- Regulatory Analysis for USI A-43, "Containment Emergency Sump Performance"
- Proposed Resolution
- Summary of Public Comments Received and Action Taken
- CRGR Minutes (Ref. USI A-43)
- Draft Generic Letter

Office of Nuclear Reactor Regulation

A. W. Serkiz, Task Manager

UNITED STATES NUCLEAR REGULATORY COMMISSION

NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 1717 H Street, N.W.
   Washington, DC 20555
2. The Superintendent of Documents, U.S. Government Printing Office, Post Office Box 37082, Washington, DC 20013-7082
3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC Office of Inspection and Enforcement bulletins, circulars, information notices, inspection and investigation notices; Licensee Event Reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, and NRC booklets and brochures. Also available are Regulatory Guides, NRC regulations in the *Code of Federal Regulations*, and *Nuclear Regulatory Commission Issuances*.

Documents available from the National Technical Information Service include NUREG series reports and technical reports prepared by other federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal and periodical articles, and transactions. *Federal Register* notices, federal and state legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, and are available there for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

NUREG-0869
Rev. 1

USI A-43 Regulatory Analysis

- Regulatory Analysis for USI A-43, "Containment Emergency Sump Performance"
- Proposed Resolution
- Summary of Public Comments Received and Action Taken
- CRGR Minutes (Ref. USI A-43)
- Draft Generic Letter

Manuscript Completed: June 1985
Date Published: October 1985

A. W. Serkiz, Task Manager

Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ABSTRACT

This report consists of: (1) the regulatory analysis for Unresolved Safety Issue (USI) A-43, "Containment Emergency Sump Performance"; (2) the proposed resolution; (3) a summary of public comments received and action taken; (4) the Committee to Review Generic Requirements (CRGR) minutes related to this USI; and (5) appendices that summarize assumptions, calculational methods, consequence analyses, and cost estimates used in this regulatory analysis.

CONTENTS

CONTENTS (continued)

LIST OF TABLES

FOREWORD

This report has been prepared to provide a self-contained reference that includes the regulatory analysis for Unresolved Safety Issue A-43, "Containment Emergency Sump Performance"; the proposed resolution; public comments received on the for comment issuance of NUREG-0869 and actions taken in response; minutes of meetings of the Committee to Review Generic Requirements regarding this issue; and summaries of assumptions, calculational methods, consequence analyses, and cost data utilized.

This report was originally issued as NUREG-0869 for public comment in May 1983. Comments received were reviewed, and those of substantive technical or informational content were incorporated into this revision. These are summarized in Appendix A.

For historical purposes, it should be noted that two draft versions of this report--NUREG-0869, Revision 1A, May 1984, and NUREG-0869, Revision 1B, July 1985--were prepared for use in the predecisional review process. This formal issuance--NUREG-0869, Revision 1, October 1985--is the concluding regulatory analysis applicable to the resolution of Unresolved Safety Issue A-43.

It should also be clearly noted that this report is not a substitute for requirements set forth in General Design Criteria 16, 35, 36, 38, 40, and 50 in Appendix A of Title 10 of the Code of Federal Regulations Part 50, nor is it a substitute for guidelines set forth in NRC's Standard Review Plan (SRP, NUREG-0800), regulatory guides, or other regulatory directives. The draft generic letter in Appendix H is for completeness of record and will be implemented through a normal regulatory issuance.

REGULATORY ANALYSIS FOR USI A-43,
"CONTAINMENT EMERGENCY SUMP PERFORMANCE"

1 STATEMENT OF PROBLEM

1.1. Summmary of Safety Issue

Unresolved Safety Issue (USI) A-43 deals with the concerns about the availability of adequate recirculation cooling water following a loss-of-coolant accident (LOCA) when long-term recirculation cooling from the containment sump in a pressurized water reactor (PWR) or the residual heat removal (RHR) system suction intake in a boiling water reaction (BWR) must be initiated and maintained to prevent core melt. These safety concerns can be summarized as follows:

> In the recirculation cooling mode, will the sump design (PWRs) or the RHR suction intakes (BWRs) provide sufficient water to the RHR and containment spray system (CSS) pumps, and will this water be sufficiently free of LOCA-generated debris and ingested air so that pump performance is not impaired to the point of seriously degrading long-term recirculation flow capability?

The USI A-43 safety concerns can be separated into three parts:

(1) the effects of potential air ingestion and elevated temperatures and break flow on sump (or suction intake) hydraulic performance under post-LOCA adverse conditions

(2) the effects of LOCA-generated insulation debris (resulting from a pipe break jet that destroys large quantities of insulation) that is transported to the sump debris screen(s) and blocks the sump screen (or suction strainer), reducing NPSH margin below that required for the recirculation pumps to maintain long-term cooling

(3) the effects of air or debris ingestion or other problems (such as the effects of particulate ingestion on pump seal and bearing systems) on the capability of RHR and CSS pumps to continue to function

Although USI A-43 was derived principally from concerns about PWR containment emergency sump performance, the concern about debris blockage applies to BWRs as well. The RHR suction strainers in a BWR are analogous to the PWR sump debris screen, and both BWRs and PWRs must have adequate recirculation cooling capacity to prevent core melt.

1.2 Technical Findings

The staff investigated these safety concerns on a generic basis, and reported its technical findings in NUREG-0897, Revision 1, October 1985. These findings can be summarized as follows:

(1) Extensive, full-scale sump hydraulic tests generally show that low levels of air ingestion (less than 1% to 2%) will occur. These tests also demonstrate that vortex observations alone cannot be used to quantify levels of air ingestion (as has been done in the past). The test results have been used to develop PWR sump and BWR suction intake hydraulic design guidelines for minimizing, or eliminating, air ingestion and have eliminated the need for plant-specific sump tests or model tests.

(2) Plant insulation surveys, development of methods for estimating debris generation and transport, debris transport experiments, and information received in public comments show that the effects of debris blockage depend on the types and quantities of insulation employed, the layout of the primary system within containment, post-LOCA recirculation patterns and velocities, and post-LOCA recirculation flow rates. Thus, the staff concluded that a single generic solution is not possible, but rather that debris blockage effects are governed by plant-specific design features and post-LOCA recirculation flow requirements.

The test results also show that the 50% screen blockage criterion in Regulatory Guide (RG) 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems," should be replaced with a requirement that debris blockage effects be assessed on a plant-specific basis. The 50% screen blockage criterion does not require a plant-specific evaluation of the debris-blockage potential and may result in a nonconservative analysis.

(3) Reviews of available data on the effects of air ingestion on pumps and discussions with the U.S. manufacturers of RHR and CSS pumps show (a) that low levels of air ingestion (2% or less) will not significantly degrade pump performance and (b) that the types of pumps employed in nuclear plants will tolerate ingestion of the insulation debris and other types of post-LOCA particulates that can pass through PWR sump screens or BWR suction strainers.

In summary, these findings show (1) that the potential safety problems that could result from vortex formation and air ingestion are less than previously hypothesized, and (2) that the potential for a loss of recirculation cooling capability as a result of LOCA-generated debris blocking a screen is potentially a more significant concern.

These technical findings were in NUREG-0897, which was issued for comment in May 1983. Information received during the public comment period included technical data, plant-specific data, cost information, and cost impact data. Apendix A summarizes the comments received and how the staff addressed them. Those comments that were principally technical in nature were incorporated into NUREG-0897, Revision 1, October 1985. Plant-specific design information and backfit cost data received during the comment period were used in preparing this report. Appendix B contains minutes of meetings of the Committee to Review Generic Requirements on USI A-43.

2 OBJECTIVES

The general objective of the proposed regulatory actions discussed below is to provide assurance that the safety concerns associated with USI A-43 will be adequately addressed in the licensing process. The goal is to meet the requirements of General Design Criterion (GDC) 35, "Emergency Core Cooling," and GDC 38, "Containment Heat Removal" (in Appendix A to Title 10 of the Code of Federal Regulations Part 50).

The technical findings developed as the result of the work performed to resolve USI A-43 have established a need to revise current licensing guidance on these matters. Therefore, the staff's technical findings (NUREG-0897, Revision 1, October 1985) have been used to revise RG 1.82, Revision 0, and Standard Review Plan (SRP, NUREG-0800) Section 6.2.2, "Containment Heat Removal Systems." The issuance of these revised regulatory documents would change staff review practices in light of current technical findings. The issuance and need for implementation of the revised regulatory guide and SRP section are discussed in Section 3 below.

3 ALTERNATIVES

The following approaches were considered as alternatives to resolve USI A-43:

(1) Issue RG 1.82, Revision 1 and SRP Section 6.2.2, Revision 4, and require <u>all</u> licensees and applicants to evaluate the potential effects of debris blockage (per RG 1.82, Revision 1) for confirmation of adequate net positive suction head (NPSH) margin.

(2) Issue RG 1.82, Revision 1, and SRP Section 6.2.2, Revision 4, and require an evaluation of the potential effects of debris blockage (per RG 1.82, Revision 1) for confirmation of adequate NPSH margin from only those licensees of and applicants for plants where loss of recirculation could lead to core melt <u>and</u> loss of containment integrity. The risk from offsite radioactive exposure depends both on the probability of core melt from loss of recirculation and on the effectiveness of the containment in mitigating offsite exposure. If a licensee/applicant can demonstrate that the containment integrity of a plant will be maintained even if the core should melt because of a loss of recirculation, that licensee/applicant could be exempt from a requirement to evaluate debris blockage effects because public risk would be low.

This alternative involves the staff's determinating--on the basis of value-impact analyses for different types of containments (e.g., PWR dry containments, ice condenser plants, Mark Is and IIs and Mark IIIs)--whether the benefits outweigh the cost impacts. This approach is used to assess both Options (1) and (2) and is discussed in Section 4.

(3) Issue a generic letter <u>for information only</u> to all licensees and applicants describing the potential safety concerns associated with insulation debris blockage of the sump screen, which could lead to the loss of adequate NPSH margin during recirculation. Make clear that the 50% blockage criterion provided in RG 1.82, Revision 0, is not a necessarily conservative way to perform an assessment of debris blockage, particularly if fibrous insulation is used on the primary system piping and components. Provide (with this generic letter) NUREG-0897, Revision 1, October 1985, which contains the staff's technical findings, for information. State clearly that there is no requirement for analysis or modification for any operating plant or

plant now under construction. Issue RG 1.82, Revision 1, and SRP Section 6.2.2 Revision 4, for implementation on Standard Plant designs and new construction permit (CP) applications only, to be effective 6 months from date of issuance of those documents.

In summary, under this alternative the significant safety information derived under the A-43 program would be provided to all licensees and applicants but there would be no requirement for any action from licensees of operating plants or applicants for plants now under construction. Standard Plant or new CP applicants would be required to address RG 1.82, Revision 1, and SRP Section 6.2.2, Revision 4, within 6 months after the date they are issued.

(4) Make no revision to RG 1.82 or SRP Section 6.2.2; publish NUREG-0897, Revision 1 (the staff's technical findings for USI A-43) as an information only document.

The need for the proposed actions can be summarized as follows:

(1) Issuance of NUREG-0897, Revision 1 (the staff's technical findings) will provide a comprehensive description of the technical issues, along with an extensive data base for designing and assessing PWR sump designs and BWR RHR suction inlets. These findings (which show that vortex observations do not quantify air ingestion) will replace the assumptions that lead to previously required inplant tests (or model tests); they also provide a common technical data base for licensees, applicants, and the staff to use, thereby reducing the regulatory burden in future assessments.

(2) Revising RG 1.82 will bring that guide into conformance with more than 3 years of experiments and generic studies related to sump design and performance, and will remove the 50% blockage criterion, which is an arbitrary assumption and not necessarily conservative from the viewpoint of sump debris blockage effects.

(3) Revising SRP Section 6.2.2 will make the review considerations consistent with the USI A-43 technical findings and with RG 1.82, Revision 1.

4 CONSEQUENCES

This section assesses the consequences (values versus impacts) of each of the alternatives given in Section 3 above. Alternatives 1 and 2--which would require licensees of or applicants for all (Alternative 1) or selected plants (Alternative 2) to perform analyses of potential debris blockage to determine if recirculation capability might be lost, and to undertake necessary plant modifications to reduce such potential risks--are the subject of this section. Alternative 3--issuance of a generic information letter and implementation of revised licensing criteria in future reviews--would not impact current licensees or applicants. The impact (cost) of including the revised licensing criteria in new designs is considered to be very small, and the resulting value/impact would be favorable; therefore no detailed quantitative analysis is deemed to be necessary. Alternative 4--do nothing--does not involve any impact.

Because of the difficulty of treating all reactors as a homogeneous group (both because of their design differences with respect to the sump and type of insulation used, as well as because of the differences in the capability of the containment to survive), the staff has developed averted risk and value/impact analyses for each of the five major types of plant containments. These are PWRs with large, dry containments; PWRs with subatmospheric containments; PWRs with ice condensers; BWRs with Mark I and II containments; and BWRs with Mark III containments. For many plants, the staff expects that evaluations (per RG 1.82, Revision 1) would show that adequate NPSH would exist, despite the potential for debris blockage. For others, a plant-specific value/impact analysis would not support imposing backfit requirements. Without performing individual plant assessments, the staff cannot determine the number of plants in these two categories. Therefore, rather than attempting to develop a value/impact (V/I) assessment for the total plant population, the staff developed estimates of releases and value/impacts for each plant and associated containment type, assuming that a significant probability of sump blockage exists. For each such class, the staff considered: (1) the potential reduction in core melt frequency, (2) the potential reduction in public risk if backfit is required (estimated averted releases in person-rem for the remaining plant life time), (3) the costs of such backfits, and (4) the resulting V/I ratio of such modifications. The results of these analyses are given below.

Blockage of the PWR containment emergency sumps or of the BWR RHR suction intakes during the recirculation phase following a LOCA can lead to core melt and containment overpressurization unless alternate recirculation cooling water sources can be made available. Core melt accompanied by containment failure can lead to a release of radioactivity and public radiation exposure.

The probability of these effects stems from

(1) pipe failure (LOCA) probability

(2) the probability of a sump blockage that would lead to loss of NPSH margin and a loss of recirculation capability

(3) the probability that the containment structural can survive overpressurization

The A-43 pipe break probabilities were calculated from a 1977 data base that included piping failures of all types known at that time, including materials not used in nuclear plant piping (see Appendix C). The estimated probabilities were 3×10^{-6} Rx/yr for large pipes ($\geq$ 28 inches) to 3×10^{-4} Rx/yr for small pipes (2 to 6 inches). The more recent experimental and analytical work, which is based on mechanistic fracture mechanics, results in probabilities of the rupture of large-size ductile piping (unaffected by IGSCC) significantly lower (better by several decades in magnitude) than those employed in the A-43 analyses. Therefore, if pipe failure probabilities are extremely low--because of such considerations as leak-before-break, etc.--these calculations would result in very low estimated releases, and backfits would not be supportable on the basis of value/impact criteria.

The probability that the sump would be blocked depends on the amount and type of debris that could be generated by a LOCA. The amount and type of LOCA-generated debris, in turn, depends on the size and location of the break, the layout of the containment, sump location and design (size of the debris screens), recirculation flow requirements, available NPSH margins, and type of insulation employed. Because these vary greatly from plant to plant, estimating sump blockage probabilities becomes highly plant-specific and arriving at a single, or generic, value is not possible. Table 4.1 shows the factors needed to estimate sump blockage probabilities. Appendix D provides a more detailed discussion and gives examples of the calculations used to develop the sump blockage probabilities.

Table 4.1 Assessment of sump blockage probability

Event	Technical consideration(s)	Safety implication
Pipe break	Break probability Break size Break type	If break probability is extremely low, debris blockage potential is negligble.
Debris generated	Break size and location Target(s) location Type(s) of insulation Extent of jet damage (L/D)	Small pipes (< 10-inch diameter) generate small amounts of debris; therefore debris blockage effects produced by small pipes are not significant.
Transport of debris	Break location Plant layout and sump location Type of debris Recirculation velocities (U)	If $U < 0.2$ ft/sec, transport is not likely to occur; therefore blockage would not occur.
Debris screen blockage	Amount transported Available screen area Type of blockage	Fibrous insulation debris transports and coats total screen area. If debris screen areas are large, pressure drop is minimized.
NPSH impact	Type of blockage Type of debris Recirculation flow required Blockage head loss (ΔH_B)	If $\Delta H_B >$ NPSHA, loss of recirculation can occur.

The survival of the containment structure and the maintenance of containment integrity also are key factors in determining the potential consequences of sump blockage (see Appendices E and F). Even though core melt can be postulated to result from loss of recirculation cooling flow, the survival of the containment structure and the maintenance of containment integrity (or the ability to withstand an overpressure transient) would significantly limit the levels of radioactive release. Recent studies dealing with containment structural capabilities have shown that overpressure design margins do exist; therefore containment integrity can be maintained through structural overpressure design margins, alternate containment cooling means, or controlled venting (as is done in BWRs). However, the variability in the containment design (e.g., PWR dry containment versus PWR ice condenser design, BWR Mark I versus Mark III design) precludes a single conclusion.

Table 4.2 summarizes the number and types of containments in use.

Table 4.2 Types of nuclear plant containments and their license review status

Type of containment	Number of plants	OLs issued[1]
PWR dry w/SGFC[2]	57	51
PWR dry w/o SGFC	14	
PWR subatmospheric	7	5
PWR ice condenser	10	7
PWR TOTAL	88	63
BWR Mark I	23	21
BWR Mark II	10	7
BWR Mark III	8	1
BWR TOTAL	41	29
INDUSTRY TOTAL	129	92

[1]As of December 1984.
[2]SGFC = safety-grade fan cooler.

4.1 Estimated Consequences Associated with Different Containment Types

4.1.1 PWR Dry Containments

The PWR dry containment design concept is used extensively in U. S. nuclear power plants. Currently 71 PWR dry containment plants are licensed or in final license review (of approximately 125 plants docketed). Fifty-seven of these 71 plants use safety-grade fan coolers (SGFCs) to help control LOCA containment pressures and temperatures.

As noted above, pipe break probability is the first key factor to consider in evaluating sump blockage probability because the LOCA is the debris generator. A range of estimated pipe failure probabilities was developed for USI A-43; these are discussed in Appendix C. The estimated probabilities were 3×10^{-6}/Rx-yr for large pipes ($\geq$ 28 inches) to 3×10^{-4}/Rx-yr for small pipes (2 to 6 inches). These estimated pipe break frequencies do not include more recent leak-before-break considerations.

Salem Unit 1 was used as a reference PWR to model piping layouts, weld locations, break locations, insulation distribution, and major insulated plant components. Because of the variability of plant and sump designs and operational requirements, the following design specifications were analyzed parametrically:

Recirculation Flow = 6,000 to 10,000 gpm
Available Debris Screen Area = 50 to 200 ft^2
Available NPSH Margin (w/o blockage) = 1 to 5 ft H_2O

These design ranges are representative of 19 PWR designs for which the staff has detailed information. The calculational methods and results are reported in NUREG/CR-3394.

The principal purpose of calculating sump failure probability is to estimate core melt frequency. For the range of parameters given above, the estimated PWR sump failure probabilities (see Appendix D) ranged from 3×10^{-6} to 5×10^{-5}/Rx-yr. Assuming that sump failure leads to core melt (which is the result if other actions cannot be or are not taken), core melt frequencies from a blocked sump would be 3×10^{-6} to 5×10^{-5}/Rx-yr. Two significant points to be noted are: (1) these sump failure probabilities were based on the assumption that all fibrous debris was transported to the sump and led to blockage (this would not be the case for PWRs that have recirculation velocities less than 0.2 ft/sec, which many large dry containments are known to have), and (2) no credit was given for detection of blockage buildup and operator corrective actions in these sump blockage estimates.

To estimate the frequency of core melt resulting from loss of NPSH due to sump blockage, it was assumed that in 50% of the cases this loss would lead to core melt. For the remaining cases it was assumed that the operator would detect the onset of blockage and take action to maintain recirculation flow. Thus the estimated core melt frequency is 1.5×10^{-6} to 2.5×10^{-5}/Rx-yr for this type of plant.

As noted above, the second approach to assessing the safety significance of this USI is based on estimating the effects on the public of potential releases of radioactivity associated with sump failure (see Appendices E and F).

The estimated conditional consequences from core melt for PWRs with large dry containments without safety-grade fan coolers are estimated to be 5×10^5 person-rem. Using the estimated core melt frequency of 1.5 to 25×10^{-6} and an outstanding reactor life span of 25 years results in an estimated averted risk range of 20 to 300 person-rem/Rx. These are low levels of averted risk and indicate the safety issue to be of moderate-to-low significance.

The value/impact (V/I) ratio range that can be calculated for PWRs with large dry containments (based on an estimated cost of $1.5M/Rx for replacing insulation) is 10 to 200 person-rem/$million. If less severe backfit actions are required (at an estimated cost of $0.4M/Rx), the V/I range is 50 to 800 person-rem/$ million. Plant backfit cost estimates are discussed in Appendix G, and are based on a composite average of industry estimates received during the public comment period. The V/I is in all cases less than the criterion of 1000 person-rem/$ million.

PWR dry containments with SGFCs have an additional safety system capable of rejecting post-LOCA containment heat loads. SGFCs are designed to operate independently in the post-LOCA environment and would, therefore, not be directly affected by loss of the sump or containment sprays. This independent heat rejection capability will ensure that the containment would not fail because of overpressure. Thus, although core melt could still be postulated, maintenance of containment integrity (by the SGFCs) would ensure that the radionuclides were contained and the public risk would be very low (see Appendix E).

4.1.2 PWR Subatmospheric Containments

Only seven PWRs have subatmospheric containments, and five of these have received an OL. Most of the discussion above on PWR large dry containments applies to this plant class as well, except that subatmospheric containments do not have safety-grade fan coolers.

The estimated range of core melt frequency resulting from sump blockage discussed in Appendix D applies to these plants, and is in the range from 1.5×10^{-6} to 2.5×10^{-5}/Rx-yr. As noted above, these numbers were derived assuming fibrous debris would transport to the sump and that the operator would take corrective action to maintain recirculation in 50% of the cases.

The estimated averted risk (for those plants where sump design problems may exist) is 20 to 310 person-rem/Rx. The value-impact ratio is the same as for PWR large dry containments without safety-grade fan coolers (10 to 200 person-rem/$M (at $1.5M/Rx) or 50 to 800 person-rem/$M (at $0.4M/Rx).

4.1.3 PWR Ice Condenser Plants

Ice condenser plants are the most prone to eventual overpressure failure if loss of recirculation occurs (see Appendix E). Although hydrogen igniters would protect against deflagration effects, the containment could fail as a result of steam production when the reactor vessel fails or within a few hours thereafter. Therefore, the consequences of sump blockage in an ice condenser plant are higher than the consequences of sump blockage in PWR dry and subatmospheric containment plants.

Three different utilities have built (or are building) 10 ice condenser plants as five twin units. Consequently, these units have many similar design features, including the details of insulation, sump design, and interior layout. Information on these design features was specifically obtained by the staff and is discussed further in Appendix E; however, the major points of similarity are summarized as follows:

(1) All ice condenser plants use reflective metallic insulation (RMI) on primary system piping and major components. Sump blockage effects associated with RMI are less severe than those associated with fibrous insulation debris.

(2) The majority of these plants have approach velocities in the vicinity of the sump of less than 0.2 ft/sec. Therefore debris transport and blockage are not likely.

(3) NPSH margins for the majority of these plants are in excess of 5 feet of water; however, the PWR sump failure probabilities were derived on the basis of a 1- to 5-foot blockage loss criterion.

The net effect is that the sump failure probabilities employed for the PWR dry and subatmospheric containments should be reduced when applied to the ice condenser plants. It is estimated that sump blockage probability leading to loss of NPSH could be reduced to 1 to 9 x 10^{-6}/Rx-yr, or lower. The estimated core melt frequency is 0.5 to 4.5 x 10^{-6}/Rx-yr.

The averted release would then be based on an estimated consequence value of 5 x 10^{6} person-rem, and for a 25-year plant life would be 60 to 560 person-rem/Rx. This estimate is in the same range as for other PWRs discussed previously.

The estimated V/I ratio for ice condenser plants is 160 to 1400 person-rem/$M (at an estimated cost of $0.4M/Rx) and is 25 to 380 person-rem/$M (at estimated cost of $1.5M/Rx).

4.1.4 BWRs with Mark I and Mark II Containments

The potential blockage of BWR RHR suction intakes is similar to that estimated for PWRs, particularly because BWRs are being reinsulated with fiberglass and newer BWRs are being built with fiberglass insulation on the primary pressure boundary piping. BWR intakes have suction strainer areas of typically 50 to 150 ft^2 (which is on the lower side of the sump debris screen area in PWRs) and somewhat higher suction flows (8,000 to 12,000 gpm/train).

On the other hand, suppression pool velocities are generally low (<0.2 ft/sec for bulk pool velocity) and the drywell-versus-wetwell design and separation tend to inhibit insulation debris transport. Although no detailed BWR RHR intake blockage probability analysis has been done, the staff has estimated that the probability of BWR intake blockages will be somewhat lower; therefore, the staff estimated a core melt frequency (equivalent to half the intake blockage probability) of 2 to 10 x 10^{-6}/Rx-yr in the calculations that follow. Such a reduction is supported by analyses done for Limerick 1 (Philadelphia Electric, 1984) (using the proposed RG 1.82, Revision 1 analysis guidelines), which showed that plant had adequate NPSH margins (principally because the plant layout resulted in high NPSH availability).

The estimated conditional consequence associated with core melt for Mark I and Mark II containments (see Appendix E) is 5 x 10^{6} person-rem. The use of a remaining reactor life of 25 years and the estimated RHR intake blockage probabilities noted above results in averted releases of 250 to 1250 person-rem/Rx; these values apply only to those plants where blockage leading to loss of NPSH has been determined.

The estimated value/impact ratios for Mark I and Mark II containments (assuming containment failure) are therefore approximately 630 to 3100 person-rem/$M (at an estimated cost of $0.4M/Rx) and 170 to 830 person-rem/$M (at an estimated cost of $1.5 M/Rx).

If credit is given for containment spray recovery and containment venting (which would ensure containment integrity despite loss of recirculation as a result of blockage), the consequences are reduced to 5 x 10^5 person-rem (see Appendix E). The estimated averted releases would then be reduced to 25 to 125 person-rem/Rx.

The estimated V/I ratios would then reduce to 63 to 300 person-rem/$M (at an estimated cost of $0.4 M/Rx) and 17 to 80 person-rem/$M (at an estimated cost pf $1.5M/Rx).

4.1.5 BWRs with Mark III Containments

Blockage considerations for Mark III containment are similar to those discussed in Section 4.1.4, although it could be argued that they are somewhat lower because of the wetwell-versus-drywell structural design. For these calculations, an estimated blockage probability of 4 to 20 x 10^{-6}/Rx-yr was used. The estimated core melt frequency is 2 to 10 x 10^{-6}/Rx-yr, and the estimated release consequence is 5 x 10^5 person-rem (see Appendix E). This release is less than that for Mark I and II containments because fission products would bubble through a subcooled suppression pool, and a significantly reduced source term would result.

Therefore, the estimated averted releases for Mark III containments are 25 to 125 person-rem/Rx (assuming again an outstanding reactor life of 25 years). The calculated V/I ratios are 60 to 310 person-rem/$M (at an estimated cost of $0.4 M/Rx) and 17 to 80 person-rem/$m (at an estimated cost of $1.5 M/Rx).

4.2 Estimated Occupational Exposure

Estimates of inplant radiological exposures associated with insulation replacement can be derived from actual experience during the steam generator repair and replacement at the Surry and Turkey Point plants. Table 4.3 shows the work categories applicable to insulation replacement (as reported in NUREG/CR-3540) and the attendant exposures.

Table 4.3 Radiation exposure to workers during insulation replacement (in person-rem)

Work	Surry 2	Surry 1	Turkey Pt 3	Turkey Pt 4
Installing scaffolding	46.5	40.9	9.95	34.19
Removing insulation	15.16	19.35	70.80	63.64
Reinstalling insulation	57.80	6.30	85.72	4.17
Total	119.46	65.55	166.47	102.00

Table 4.3 shows both the benefit of preplanning (or learning from experience at the first plant), as well as the variation between plants. It should also be noted that the scaffolding installed at these plants was designed to remove

steam generators, and that entire steam generators were stripped of insulation and reinsulated. Thus, the average of these exposures, 115 person-rem, is considered to be higher than that expected as a result of the insulation replacement needed to resolve A-43.

Discussions with Surry site personnel during the 1983 for comment period for NUREG-0869 indicated that a 50-person-rem exposure level for insulation replacement is realistic if the job is thoroughly preplanned. On this basis, the occupational exposure for major insulation replacement is estimated to be 50 person-rem per plant. Exposures associated with alternate actions (such as increasing debris screen size) would be less.

Using the estimated averted risks developed in Appendix E and the replacement exposures given in Table 4.3, the net radiological effects of backfitting shown in Table 4.4 can be developed.

Table 4.4 Radiological effects of backfitting

Plant type	Backfit exposure (person-rem/Rx)	Estimated averted risk (person-rem)/Rx	Net averted exposure (person-rem/Rx)
PWR ice condenser	50	40 to 560	-10 to 510
PWR dry w/o SGFCs and subatmospheric	50	20 to 300	-30 to 250
Mark I and II	50	250 to 1250	200 to 1200
Mark III	50	25 to 125	-25 to 75

The staff estimates that fewer than 10 plants would have the higher values of net averted exposure.

4.3 Impact on NRC Operations

With respect to NRC staff review time, the impact of the proposed actions will be minimal. The guidelines in Appendix A of the revised RG 1.82 and in NUREG-0897, Revision 1, October 1985 (and its supporting references) provide the technical information and specific guidance the staff reviewer needs to perform an evaluation in a reasonable time. It is estimated that about 2 weeks of staff and related licensing review time will be needed per plant (estimated cost $5K per plant) to review analyses submitted. Assuming 5 to 20 such detailed responses are received, the estimated staff cost would be $25K to $100K.

The experimental data and generic plant information and calculations reported in NUREG-0897, Revision 1, October 1985 (and supporting references) represent an investment of nearly $3.0M by NRC and the Department of Energy. This information is of value to both the NRC and industry, because similar tests need not be duplicated. In addition, this extensive hydraulic performance data base provides the basis for eliminating unnecessary inplant or sump model testing to examine vortex formation, which has been required for many plants at the OL stage.

4.4 Impact on Other Government Agencies

Because nuclear plant design review and acceptance are done solely by the NRC staff, no impact on other government agencies is projected.

4.5 Public Impact

If the recommendations in this report are adopted, there would be no impact the public. Rather, there would be a value to the public: added reassurance that adequate sump designs exist for ensuring operability in the recirculation mode following a postulated LOCA. As discussed above, only a small number of plants are expected to be susceptible to LOCA-generated debris. Issuance of a generic letter and the staff's technical findings to licensees and OL applicants outlining potential safety concerns associated with insulation debris would add to the "defense-in-depth" concept, which ensures that the health and safety of the public is being maintained. Because insulation is periodically replaced in operating plants, these findings could be used in the licensee's process for insulation selection and replacement.

4.6 Other Constraints

The Commission has proposed to amend its requirements governing the backfitting of commercial power reactors and certain licensed nuclear facilities (see NRC Notice 84-137, dated November 30, 1984). The regulatory analysis herein is consistent with those proposed requirements.

5 DECISION RATIONALE

The regulatory alternatives pertinent to the resolution of USI A-43 were identified in Section 3, and the consequences (values versus impacts) were discussed in Section 4. This section compares regulatory alternatives and considers the value/impact analysis results.

5.1 Comparison of Regulatory Alternatives

The regulatory alternatives can be compared as follows:

(1) Requiring <u>all</u> licensees and applicants to evaluate adequate NPSH margin would give the NRC a determination of which plants are most susceptible to debris blockage problems, and would identify plants that would benefit from corrective action. This option would, however, result in the greatest overall industry cost impact: it is expected that only 5 to 10 plants would be identified as having a potential debris blockage problem that required backfit actions, while all licensees and applicants would incur analysis costs.

(2) Requesting evaluations from only the licensees for those plants that are judged to have a high probability of containment failure would focus this

safety issue on minimizing (or averting) the public risk of radiation exposure as the result of loss-of-recirculation capability. This option would reduce the industry impact and would concentrate the analysis effort on those plants (a) that are subject to containment failure and (b) that would benefit from corrective action.

Identification of only those plants that have a high probability of debris blockage would require a plant-specific analysis because of the plant-to-plant variabilities discussed in Section 4; no generic conclusion (or identification) is possible. Therefore, this option is similar to Option (1) but would involve fewer plants. Both options are discussed as a single option in the decisionale rationale presented below.

(3) The forward fit of RG 1.82, Revision 1, and SRP 6.2.2, Revision 4, to Standard Plants and new construction permits (CPs) would address sump design and debris blockage effects in new plants at a very low incremental cost. Continued use of RG 1.82, Revision 0 (which has the 50% blockage criterion) does not adequately address this issue, and is technically inconsistent with the technical findings developed for the resolution of USI A-43. This option would include the issuance of NUREG-0897, Revision 1, October 1985, and of a generic letter for information, providing the industry with technical findings that would be useful for industry safety assessments regarding routine change of insulation materials.

(4) Issuing NUREG-0897 only, without using current technical findings to revise RG 1.82 and SRP Section 6.2.2, would be contrary to (a) addressing safety concerns using current and the most reliable technical findings and (b) the need to remove the current 50% blockage criterion in RG 1.82, Revision 0.

5.2 Rationale for Selecting the Recommended Resolution

The rationale for selecting a resolution position is based in part on the values and impacts discussed in Section 4, which presents the consequences of sump blockage problems for different containment design categories. Conclusions are based on core melt frequency, potentially averted releases (ΔR), and value/impact ratio, utilizing the 1000 person-rem/$M criterion for backfit consideration. Table 5.1 summarizes the values and impacts developed in Section 4.

5.2.1 Options 1 and 2

Options 1 and 2 are discussed together because Option 2 is a modification of Option 1.

PWR Dry Containments

PWR dry containment plants fall into two categories, those with safety-grade fan coolers (SGFCs) and those without. PWR dry containments with SGFCs have an additional capability to reject post-LOCA decay heat and prevent containment overpressurization, thereby ensuring containment integrity. Furthermore, large dry containments are least susceptible to abrupt failures as a result of hydrogen burns, steam spikes, etc. Therefore, it is the staff's opinion that even if loss of sump should occur, current designs would contain postulated core

Table 5.1 Summary of calculated values and impacts associated with various containment designs for resolution of USI A-43

Type containment	Estimated core melt probability[1] (1/Rx-yr)	Calculated risk averted, ΔR (person-rem/Rx)	Calculated value/ impact ratio[3] (person-rem/$M)
PWR dry w/o SGFCs and subatmospheric	1.5 to 25 x 10^{-6}	19 to 310	48 to 780 12 to 210
PWR dry w/SGFCs	1.5 to 25 x 10^{-6}	--	--
PWR ice condenser[3]	0.5 to 4.5 x 10^{-6}	60 to 560	160 to 1400 25 to 380
PWR dry w/o SGFCs and subatmospheric w/spray recovery	1.5 to 25 x 10^{-6}	2 to 31	5 to 33 1 to 21
Mark I and II	2 to 10 x 10^{-6}	250 to 1250	630 to 3100 170 to 830
Mark III	2 to 10 x 10^{-6}	25 to 125	60 to 310 17 to 80
Mark I and II w/venting and spray recovery	2 to 10 x 10^{-6}	25 to 125	60 to 310 17 to 83

[1]The estimated core melt frequency is based on the conditional consequences discussed in Appendix E, the sump blockage frequency estimates discussed in Appendix D, and the assumption that 50% of the time blockage occurs leads to loss of NPSH and core melt follows. This assignment of a conditional core melt probability of 0.5 is felt to be realistic from the viewpoint of potential detection of flow degradation, potential operator followup action to correct this situation, and sump design variability.

[2]The value/impact ratios have been calculated for an estimated cost of $0.4M/Rx (this assumes backfit costs would be minimal) and for an estimated cost of $1.5M/Rx (this cost assumes replacement of troublesome insulation(s)); see Appendix G for a discussion of estimated costs.

[3]A separate estimate of sump blockage probability made for the ice condenser plants (see Appendix E) takes into account their specific design features.

melt effects and maintain public releases at appropriate levels. Thus, backfit action (including analyses) for PWR dry containments with SGFCs is not justifiable, and Option (1), which includes such a requirement for PWR dry containments, is not attractive.

PWR dry containments (without SGFCs) have been evaluated (see Section 4.1.1) and the results are as follows:

(1) Probabalistic risk analyses (PRAs) discussed in Section 4 have concluded that the core melt probability from this class of plants for the sequence involving a blocked sump is in the range 1.5 x 10^{-6} to 2.5 x 10^{-5}. For a number of reasons as discussed above, the staff believes believes the best estimate of core melt frequency for this sequence is at the lower end of this range, about 3 x 10^{-6}. This level of core melt frequency does not support a backfit requirement.

(2) The calculated range of averted risk (assuming containment failure, as per WASH-1400 (NUREG-75/014)) is approximately 20 to 300 person-rem/Rx. This is a low-to-moderate level. If corrective operator action to restore containment sprays (should debris blockage be encountered) is begun before containment failure, then the estimated averted risk is 2 to 30 person-rem/Rx.

(3) The calculated V/I ratios were

50 to 800 person-rem/$M (low retrofit cost)
10 to 200 person-rem/$M (high retrofit cost)

Utilization of a 1000 person-rem/$M criterion does not support a backfit requirement.

(4) If operator detection of the onset of blockage and taking corrective actions are included in these consequence calculations, the estimated releases noted above would be reduced by 10% and the V/I ratios would also be reduced by 10%

Therefore, based on the above assessment, it is the staff's opinion that all PWR dry containments can be excluded from any backfit requirements.

PWR Subatmospheric Containments

The release consequences resulting from an assumed failed containment associated with PWR subatmospheric containments are estimated to be the same as those for PWR dry containments without SGFCs (see Appendix E). Sump blockage probabilities are judged to be in the same range also. Therefore, the averted releases and V/I estimates are the same as noted in the preceding section. Restoration of containment spray capability by operator action reduces estimated releases by a factor of 10.

Therefore, backfit requirements for subatmospheric containments are not supported for the same reasons cited for not requiring such backfits for PWR dry containments without SGFCs.

PWR Ice Condenser Plants

As noted in Section 4.1.3 and Appendix E, PWR ice condenser plants are most prone to overpressure failure. However, fewer than 10 plants have this type of containment design, layout, and recirculation flow features (see Section 4.1.3). For these plants, the staff has determined a separate value of sump failure probability based on known plant design parameters. The result is that the probability for sump blockage is lower, thus offseting the higher estimated consequences from core melt.

The calculated averted risk is 60 to 560 person-rem/Rx (similar in value to that for the PWR dry containments) and the calculated V/I ratios are 160 to 1400 person-rem/$M (low cost retrofit estimates) and 40 to 380 person-rem/$M (high retrofit cost estimate).

Thus, based on the calculated averted risk and V/I noted above, a backfit requirement for PWR ice condenser plants is not indicated.

BWR Mark I and Mark II Plants

NRC and industry PRA studies report an estimated BWR core melt frequency (attributable to all causes) of 2×10^{-5} to 3×10^{-4}/Rx-yr. The LOCA contribution to this total frequency is 1% to 10%, with large LOCAs having the lower value. Thus, core melt frequency related to this safety issue is judged to be on the order of 10^{-7} to 10^{-6}/Rx-yr. If backfit actions are viewed from the perspective of a reduction in core melt frequency, the gain would minimal (this would apply only to BWRs where debris blockage was identified to be significant). Therefore, from this viewpoint, a backfit requirement is not indicated.

The calculated averted risk for this class of BWRs is 250 to 1250 person-rem/Rx (see Section 4.4), based on blockage probabilities derived from PWR studies. For reasons discussed in Section 4, these values are conservative. The corresponding V/I ratios are 630 to 3100 person-rem/$M (low retrofit cost estimate) and 170 to 830 person-rem/$M (high retrofit cost estimate).

On the basis of the averted risk and V/I assessment, an argument could be made for proceeding with some type of plant assessment. However, this view must be balanced with the knowledge that only a few BWRs may fall into the higher V/I category. In addition (as for PWRs), it is the staff's opinion that the majority of BWRs will be at the lower end of the blockage frequency spectrum. An example is the analysis submitted for Limerick 1 (Philadelphia Electric, 1984). When this analysis was evaluated according to the guidelines in the proposed RG 1.82, Revision 1, it showed the plant had adequate NPSH margin.

Another factor to be considered is suction realignment capability if blockage should occur. Mark I plants can be realigned to alternate water sources (e.g., the condensate storage tank). Twenty-three of 41 BWRs have Mark I containments. Mark II containments do not have the option of being aligned to the condensate storage tank, although realignment to the fuel storage pool would be a possibility if such procedures were provided to the operators.

An additional significant factor is the capability for controlled venting. Controlled venting (an option that can be submitted for staff approval on a plant-specific basis) provides a means to maintain containment integrity by avoiding overpressurization. With wetwell venting, the calculated averted risk is reduced by 10% and the V/I cited above decreases accordingly. It is the staff's understanding that the BWR Owners Group is supporting implementation of controlled venting for all plants.

Therefore, given the above considerations, it is the staff's view that backfit requirements for Mark I and Mark II plants are of marginal significance and therefore are not proposed.

Mark III Plants

The consequences (releases) associated with Mark III containments are estimated to be 10% lower than those for Mark I and II plants because, should the containment fail, the Mark III containments channel fission products through the pool before they are released to the environment with or without wetwell venting (see Appendix E).

Therefore the calculated averted risk for Mark III containments is 25 to 125 person-rem/Rx. The V/I ratios are 60 to 310 person-rem/$M (low retrofit cost estimate) or 17 to 80 person-rem/$M (high retrofit cost estimate).

On the basis of these values, a backfit requirement is not supportable for the Mark III containments.

5.2.2 Option 3

Option 3, which is a forward fit application of RG 1.82, Revision 1, to Standard Plant and new CP applications, would have no impact on current licensees and applicants. The incremental impact of requiring a a sump blockage analysis for a new design would be very small; therefore the V/I is favorable.

An important aspect of Option 3 is that all licensees and applicants would be informed of the technical findings of A-43 and the recommended actions in RG 1.82, Revision 1, for performing a sump blockage analysis. This information would give them a basis for considering analysis and corrective action as they deem necessary. This information also would give each licensee or applicant a better basis for considering changes of insulation. (The periodic changing of insulation is a common practice at operating plants.) Therefore, because of its obvious high value and benefit amd very low impact, the staff recommends adopting Option 3.

5.2.3 Option 4

Option 4 would use no overt means to inform the industry regarding the new information and understanding developed, and the existing incorrect NRC guidance with respect to sump blockage would remain in place. Although this option has no impact, it also has no value. The prospect of having incorrect guidance standing is unacceptable to the staff; thus this option was rejected.

5.3 Recommended Regulatory Action

On the basis of the discussion in Section 5.2, the staff recommends adoption of Option 3 and the following specific actions:

(1) Issue the staff's technical findings (NUREG-0897, Revision 1, October 1985) for use as a technical information source.

(2) Issue SRP Section 6.2.2, Revision 4, and RG 1.82, Revision 1. These revisions reflect the staff's technical findings reported in NUREG-0897, Revision 1, October 1985. This revised regulatory guidance would apply only to future CP applications, Preliminary Design Approvals (PDAs), Final Design Approvals (FDAs) that have not received prior approval, and future applications for license to manufacture. It would be effective 6 months following issuance.

(3) Issue a generic letter for information only to all holders of an Operating License or Construction Permit outlining the safety concerns regarding potential debris blockage and recirculation failure due to inadequate NPSH. It is suggested (but not required) that licensees utilize RG 1.82, Revision 1, as guidance for conduct of the 10 CFR 50.59 review for future plant modifications involving replacement of insulation on primary system piping and/or equipment. If, as a result of NRC staff review of licensee actions associated with replacement or modification to insulation, the staff decides that SRP 6.2.2, Revision 4, and/or RG 1.82, Revision 1, criteria should be (or should have been) applied by the licensee, and the staff seeks to impose these criteria, then the NRC will treat such an action as a plant-specific backfit pursuant to 10 CFR 50.109.

6 PLAN FOR IMPLEMENTATION

The proposed resolution of USI A-43 would be accomplished through the following actions:

(1) Issue the staff's technical findings (NUREG-0897, Revision 1, October 1985) for use as an information source by applicants, licensees, and the staff.

(2) Issue Revision 1 of RG 1.82 to reflect the technical findings reported in NUREG-0897, Revision 1, October 1985. In particular, the 50% screen blockage criterion would be replaced by a plant-specific debris blockage assessment. RG 1.82, Revision 1, provides specific guidance acceptable to the staff for assessing sump performance and RHR suction intakes including debris blockage effects. RG 1.82, Revision 1, would apply to:

   (a) construction permit applications and Preliminary Design Approvals (PDAs) that are docketed more than 6 months following issuance of RG 1.82, Revision 1;

   (b) applications for Final Design Approvals (FDAs), for standardized designs that are intended for referencing in future construction permit applications, that have not received approval 6 months following issuance of RG 1.82, Revision 1; and

   (c) applications for licenses to manufacture that are docketed more than 6 months following issuance of RG 1.82, Revision 1.

(3) Issue NRC SRP Section 6.2.2, Revision 4, "Containment Heat Removal Systems," to reflect the guidance in RG 1.82, Revision 1, and the technical findings

in NUREG-0897, Revision 1, October 1985. The revised SRP section would apply to new CP applications and Standard Plant designs only, effective 6 months after the revisions are issued.

(4) Issue a Generic Letter to all holders of an operating license or construction permit outlining the potential safety concerns related to potential post-LOCA debris blockage and the inadequacy of the current 50% blockage criterion contained in Revision 0 of RG 1.82. This action is significant because licensees should be made aware of the potential for recirculation blockage from insulation following a LOCA, and replacing insulation in operating plants is common practice. (A draft generic letter is provided in Appendix H.) The proposed generic letter contains no requirements and no request for a response.

7 STATUTORY CONSIDERATIONS

7.1 NRC Authority

Because the proposed changes are revisions to RG 1.82 and SRP Section 6.2.2, these actions fall within the statutory authority of the NRC. Further, the recommendation to require applicants/licensees to demonstrate adequate sump performance falls within the statutory authority of the NRC to regulate and ensure the safe operation of nuclear power plants.

7.2 Need for National Environmental Policy Act (NEPA) Statement

The proposed changes and potential plant backfits do not warrant a NEPA statement.

8 BIBLIOGRAPHY

The following U.S. Nuclear Regulatory Commission documents were used in the preparation of this report:

NUREG-75/014, "Reactor Safety Study," 1975 (formerly WASH-1400).

NUREG-0800, "Standard Review Plan," July 1981. Revised Standard Review Plan Section 6.2.2, Revision 4, "Containment Heat Removal Systems," available from the NRC Division of Technical Information and Document Control, 1717 H Street, NW, Washington, DC 20555.

NUREG-0897, Revision 1, "Containment Emergency Sump Performance, Technical Findings Related to USI A-43," November 1984.

NUREG/CP-0933, SAND82-1659, "Proceedings of the Workshop on Containment Integrity," Volume II of II, October 1982.

NUREG/CR-2403: see Reyer.

NUREG/CR-2403, Supplement No. 1: see Kolbe.

NUREG/CR-2759: see Argonne.

NUREG/CR-2760: see Padmanabhan and Hecker, June 1982.

NUREG/CR-2761: see Padmanabhan, September 1981.

NUREG/CR-2772: see Padmanabhan, June 1982.

NUREG/CR-2982: see Brocard, July 1983.

NUREG/CR-2791: see Wysocki.

NUREG/CR-2792: see Kamath.

NUREG/CR-3394: see Wysocki, July 1983.

NUREG/CR-3540: see Parkhurst, December 1983

NUREG/CR-3616: see Brocard, December 1983.

Regulatory Guide 1.82, Revision 1, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," July 1985.

Other documents used in the production of this report were:

Argonne National Laboratory, "Results of Vertical Outlet Sump Tests," joint ARL/Sandia National Laboratory report, ARL47-82/SAND82-1286/NUREG/CR-2759, September 1982.

Brocard D., "Buoyancy, Transport and Head Loss of Fibrous Reactor Insulation," U.S. Nuclear Regulatory Commission report, NUREG/CR-2982, Revision 1, July 1983.

---, "Transport and Screen Blockage Characteristics of Reflective Metallic Insulation Materials," U.S. Nuclear Regulatory Commission report, NUREG/CR-3616, December 1983.

Ferrell, W. L. et al., "Probabilistic Assessment of USI A-43," Science Applications, Inc. report, September 1982.

Kamath, P., T. Tantillo, and W. Swift, "An Assessment of Residual Heat Removal and Containment Spray Pump Performance Under Air and Debris Ingesting Conditions," Creare, Inc., Hanover, NH, U.S. Nuclear Regulatory Commission report, NUREG/CR-2792, September 1982.

Kolbe, R. and E. Gahan, "Survey of Insulation Used in Nuclear Power Plants and the Potential for Debris Generation," Burns and Roe, Inc., Oradell, NJ, U.S. Nuclear Regulatory Commission report, NUREG/CR-2403, Supplement 1, May 1982.

Padmanabhan, M., "Results of Vortex Suppressor Tests, Single Outlet Sump Test and Miscellaneous Sensitivity Tests," Alden Research Laboratory, Worcester Polytechnic Institute, Holden, MA, U.S. Nuclear Regulatory Commission report, NUREG/CR-2761, September 1982.

---, "Hydraulic Performance of Pump Suction Inlet for Emergency Core Cooling Systems in Boiling Water Reactors," Alden Research Laboratory, Worcester Polytechnic Institute, Holden, MA, U.S. Nuclear Regulatory Commission report, NUREG/CR-2772, June 1982.

Padmanabhan, M., and G. E. Hecker, "Assessment of Scale Effects on Vortexing, Swirl, and Inlet Losses in Large Scale Sump Models," Alden Research Laboratory, Worcester Polytechnic Institute, Holden, MA, U.S. Nuclear Regulatory Commission report, NUREG/CR-2760, June 1982.

Parkhurst, M. A., L. Rathbun, and D. Murphy, "Radiological Assessment of Steam Generator Repair and Replacement," Pacific Northwest Laboratory, Richland, WA, U.S. Nuclear Regulatory Commission report, NUREG/CR-3540, December 1983.

Philadelphia Electric Company, "Limerick Generating Station, Units 1 and 2 Containment Emergency Sump Performance," April 2, 1984.

Reyer, R., et al., "Survey of Insulation Used in Nuclear Power Plants and the Potential for Debris Generation," Burns and Roe, Inc., Oradell, NJ, U.S. Nuclear Regulatory Commission report, NUREG/CR-2403, Supplement 1, October 1981.

Wysocki, J. J., "Methodology for Evaluation of Insulation Debris," Burns and Roe, Inc., Oradell, NJ, U.S. Nuclear Regulatory Commission report, NUREG/CR-2791, September 1982.

---, "Probabilistic Assessment of Recirculation Sump Blockage Due to Loss-of-Coolant Accidents, U.S. Nuclear Regulatory Commission report, NUREG/CR-3394, Volumes 1 and 2, July 1983.

APPENDIX A

SUMMARY OF PUBLIC COMMENTS RECEIVED AND ACTIONS TAKEN

APPENDIX A

SUMMARY OF PUBLIC COMMENTS RECEIVED AND ACTIONS TAKEN

1 INTRODUCTION

The technical findings related to Unresolved Safety Issue (USI) A-43 were published for comment in May 1983. Notice of the publication was placed in the Federal Register on May 9, 1983. The official comment period lasted for 60 days and ended on July 11, 1983. However, comments were received into September 1983, with followup comments received into November 1983. A listing of those who responded during the period and afterwards is shown in Table 1. Copies of the comment letters are on file in the NRC Public Document Room, 1717 H Street, NW, Washington, DC.

A public meeting was held on June 1 and 2, 1983, at Bethesda, Maryland, to offer additional opportunity for public comments; however, attendance was very small. Followup discussions were held with respondees to clarify issues raised at this meeting and in the written comments.

An overview of the comments received is provided in Section 2 below. Section 3 contains summaries of significant comments and the actions planned to resolve them.

2 OVERVIEW OF COMMENTS RECEIVED

The major written comments received addressed seven specific subject areas. The comment categories and commentors are listed in Table 2 below. The commentors are identified in Table 2 as follows: Alden Research Laboratory (ARL); Atomic Industrial Forum (AIF); BWR Owners Group (BWR); Commonwealth Edison (CEd); Consumers Power Co. (CPC); Creare Research and Development (CRD); Diamond Power Co. (DPC); General Electric (GE); Gibbs and Hill, Inc. (GH); Northeast Utilities (NE); and Owens-Corning Fiberglass, Inc. (OCF). By category, the actions taken in response to these comments are as follows:

Categories 1 and 6: Tables have been added to NUREG-0897, Revision 1 and NUREG-0869, Revision 1 to include the additional plant insulation information provided during the public comment period. The text of the NUREGs has been revised to reflect recommended insulation definitions and the need to evaluate the specific insulation employed.

Categories 2 and 4: The cost estimates provided by different industry groups have varied over a wide range. With the exception of Diamond Power Company, respondees claimed that the cost estimates in value/impact analysis were too low. The revised value/impact analysis reflects an averaged value derived from costs provided.

Category 3: A detailed sump blockage probability analysis has been performed and is reported in NUREG/CR-3394. The results were used in the revised value/impact analysis. These results show a sump blockage probability range for pressurized water reactors (PWRs) of 10^{-6} to 5×10^{-5}/Rx-yr and a strong dependence on plant design.

Table 1 Persons who commented on the technical findings related to USI A-43*

Alden Research Laboratory (ARL), M. Padmanabhan, letter to A. Serkiz (NRC), "Comments on NUREG-0897 and 0869," June 13, 1983.

ARL, M. Padmanabhan, letter to A. Serkiz (NRC), "Revision to Table A-3 in NUREG-0869," June 22, 1983.

Atomic Industrial Forum, R. Szalay, letter to the Secretary of the Commission, "NRC's Proposed Resolution of Unresolved Safety Issue A-43, Containment Emergency Sump Performance, Contained in NUREG-0869," July 22, 1983.

Atomic Industrial Forum, J. Cook, letter to R. Purple (NRC) and enclosure "Examples of Staff Review Going Beyond Approved Regulatory Criteria," June 4, 1984.

BWR Owners Group, T. J. Dente, letter to T. P. Speis (NRC), "BWR Owners' Group Comments on Proposed Revision to Regulatory Guide 1.82, Rev. 1," October 18, 1983.

BWR Owners Group, D. R. Helwig, letter to V. Stello (NRC), BWR Owners' Group comments on Regulatory Guide 1.82, Revision 1, July 16, 1984.

Commonwealth Edison, D. L. Farrar, letter to the Secretary of the Commission, "NUREG-0897, Containment Emergency Sump Performance; Standard Review Plan Section 6.2.2, Rev. 4, Containment Emergency Heat Removal Systems; and NUREG-0869, USI A-43 Resolution Positions (48FR2089; May 9, 1983)," July 13, 1983.

Consumers Power, D. M. Budzik, letter to the Secretary of the Commission, "Comments Concerning Regulatory Guide 1.82, Proposed Revision 1 (File 0485.1, 0911.1.5, Serial: 23206)," July 15, 1983.

Creare, W. L. Swift, letter to P. Strom (SNL), "Comments on Figure 3-6 of NUREG-0897 and Table A-9 of NUREG-0869," June 13, 1983.

Diamond Power Company, R. E. Ziegler and B. D. Ziels, letter to K. Kniel (NRC), "Containment Emergency Sump Performance, USI A-43," July 11, 1983.

Diamond Power Specialty Company, B. D. Ziels, letter to A. Serkiz (NRC), "HDR Test Result Summary, MIRROR Insulation Performance During LOCA Conditions," December 6, 1984.

General Electric (GE), J. F. Quirk, letter to K. Kniel (NRC), "Comments on Emergency Sump Documents," July 11, 1983.

GE, J. F. Quirk, letter to T. P. Speis (NRC), "Comments on Proposed Regulatory Guide 1.82, Rev. 1," October 17, 1983.

*Including comments on NUREG-0869, NUREG-0897, proposed Revision 1 to Regulatory Guide 1.82, and proposed Revision 4 to Section 6.2.2 of the Standard Review Plan (SRP, NUREG-0800).

Table 1 (Continued)

Gibbs and Hill, Inc., M. A. Vivirito, letter to the Secretary of the Commission, "Comments on Proposed Revision No. 1 to RG 1.82," July 11, 1983.

Northeast Utilities, W. G. Counsil, letter to K. Kniel (NRC), "Haddam Neck, Millstone Nuclear Power Station, Unit Nos. 1, 2, and 3, Comments on NUREG-0897, SRP Section 6.2.2 and NUREG-0869," September 2, 1983.

Owens Corning Fiberglass (OCF), G. H. Hart, letter to A. Serkiz (NRC), "Comments on NUREG-0897 and NUREG-0869," June 23, 1983.

OCF, G. H. Hart, letter to A. Serkiz (NRC), "Updated Comments on NUREG-0897 and NUREG-0869," July 14, 1983.

OCF, G. P. Pinsky, letter to K. Kniel (NRC), "Comments on NUREG-0879 and -0896," July 14, 1983.

OCF, G. H. Hart, transmittal to A. Serkiz (NRC), "HDR Blowdown Tests with NUKON Insulation Blankets," February 18, 1985.

Power Component Systems, Inc., D. A. Leach, letter to A. Serkiz (NRC), "Nuclear Grade Blanket Insulation," November 8, 1984.

Table 2 Categories addressed in major written comments

Comment	Category	ARL	AIF	BWR	CED	CPC	CRD	DPC	GE	GH	NE	OCF
(1)	Survey of insulation used is not current or complete.							X				X
(2)	Cost estimates are low.		X								X	
(3)	Estimates of sump blockage probabilities are high.		X		X	X			X			
(4)	Value-impact analysis questioned.		X		X				X		X	
(5)	BWRs should be exempt; A-43 is a PWR issue.			X					X		X	
(6)	Insulation material definitions and descriptions need revision for clarity and completeness.							X			X	
(7)	Technical comments on and clarifications of subject matter in NUREG-0897 and NUREG-0869.	X			X	X	X	X		X	X	X

By category, the actions taken in response to these comments are as follows:

Categories 1 and 6: Tables have been added to NUREG-0897, Revision 1 and NUREG-0897, Revision 1 to include the additional plant insulation information provided during the public comment period. The text of the NUREGs has been revised to reflect recommended insulation definitions and the need to evaluate the specific insulation employed.

Categories 2 and 4: The cost estimates provided by different industry groups have varied over a wide range. With the exception of Diamond Power Company, respondees claimed that the cost estimates in value/impact analysis were too low. The revised value/impact analysis reflects an averaged value derived from costs provided.

Category 3: A detailed sump blockage probability analysis has been performed and is reported in NUREG/CR-3394. The results were used in the revised value/impact analysis. These results show a sump blockage probability range for pressurized water reactors (PWRs) of 10^{-6} to 5×10^{-5}/Rx-yr and a strong dependence on plant design.

Category 5: NUREG-0869 and Regulatory Guide 1.82 have been revised to specifically identify areas of concern for boiling water reactors (BWRs) and for PWRs.

Category 7: Technical corrections and clarifications have been made in the appropriate sections of NUREG-0897 and NUREG-0869.

The NRC staff greatly appreciates the review and comments provided by the respondees. The time and effort they have taken to review USI A-43 has resulted in an improved report that will reflect current findings and a balanced position with respect to this safety issue.

3 COMMENTS RECEIVED AND PROPOSED ACTION (OR RESPONSE) ACTIONS

The NRC staff has given complete and careful consideration to all comments received on USI A-43. Summaries of significant comments and the actions taken by the NRC staff in response are provided in Table 3. Comments are presented in alphabetical order, based on the name of the commenting institution.

Table 3 Comments received on USI A-43 and NRC staff response

Comment	NRC Staff Response
<u>Alden Research Laboratory</u>	
ARL noted typographical errors and proposed technical clarification to several tables	These corrections and clarifications have been incorporated into NUREG-0897 and NUREG-0869.
<u>Atomic Industrial Forum</u>	
The cost impact of $550,000/plant used in value/impact analysis is low by at least a factor of 2.	Costs impacts were re-evaluated based on cost estimate information received from AIF and other respondees
Economic considerations related to reduced probability of plant damage should be excluded from the cost-benefit balancing. Decisions should be based primarily on the value/impact ratio.	The essence of a value/impact analysis is that it attempts to identify, organize, relate, and make "visible" all the significant elements of value expected to be derived from a proposed regulatory action as well as all significant elements of impact. The net values are compared with the net impacts. Thus if a proposed safety improvement is accompanied by an adverse side effect, the impairment is is subtracted from the improvement to arrive at a net safety value for consideration in the value/impact assessment. Similarly, when the immediate and prospective cost impacts are summed, they should include all elements of economic impact on licensees, such as costs to design, plan, install, test, operate, maintain, etc. Plant downtime or decreased plant availability is included when applicable. The summed impacts, however, should be <u>net</u> impacts, for comparison with <u>net</u> values. Thus, any reductions in operating costs, improvements in plant availability, or reductions in the probability of plant damage are properly a factor in determining net adverse economic impact. Future economic costs and savings are appropriately discounted.

Table 3 (Continued)

Comment	NRC Staff Response
	Qualitative differences among impact elements are respected, and distinctive elements of impact (of which averted plant-damage probability, as a favorable rather than adverse impact, is a prominent example) are separately identified, for appropriate consideration in regulatory decision making. The ratio of avoided public dose to the gross cost of implementation is ordinarily a major decision factor. However, it is not by itself always a good guide to a sound regulatory decision. The issues involved are often too complex for a decision on this criterion alone. Other factors that enter, often in important ways, may include any economic benefits that reduce a net adverse economic impact, the safety importance of the issue, and values and impacts that cannot or cannot readily be quantified; for example, jeopardy to a defense layer in the defense-in-depth concept or expected reductions in plant availability that can be foreseen but not precisely estimated. A sound regulatory decision rests on adequate consideration of all significant factors. An overly simple approach can mislead if it simplifies away complexities that are the essence of the issue at hand.
The assumption that sump failure will occur in the case of 50% of the large LOCAs should be justified.	A detailed sump blockage probability analysis has been performed and is reported in NUREG/CR-3394. The results show a wide range of sump blockage failure probabilities (i.e., 3E-6 to 5E-5/reactor-year) and a high dependency on plant design and operational requirements. These results are reflected in a revised value impact analysis utilizing a range of sump failure probabilities.
The use of PWR release categories from WASH-1400 is too conservative. Containment failure probabilities used in WASH-1400	The containment failure probabilities and release categories used in the regulatory analysis for USI A-43 were based on information presented in WASH-1400, and also on

Table 3 (Continued)

Comment	NRC Staff Response
are inadequate to describe the nuclear industry's present knowledge in this field. Releases due to "vessel steam explosion" are unrealistic and should not be considered.	other considerations. The comments presented by an AIF subcommittee regarding the validity of continued use of WASH-1400 assumptions, etc. are being evaluated through other activities such as: reevaluation of source terms, SASA studies, etc. USI A-43 regulatory analyses were based on the following considerations and for the reasons noted: (1) WASH-1400 assumptions were utilized to provide a common baseline calculations for reference plants and were used to estimate increases in releases due to a postulated loss of recirculation flow capacity. Until revised failure mechanisms and new source terms are determined, this approach provides a consistent set of calculations. (2) Although using a small containment failure probability associated with steam explosion would be more appropriate, release category PWR-1 (which includes steam explosion) was not a dominant contributor to release. Release categories PWR-2, -4, and -6 were the dominant pathways contributing to increases releases due to a failed sump for the plants analyzed. (3) Basing release effects on the assumption of simultaneous failure of core cooling and loss of containment sprays is conservative. If containment were not lost (as would be the situation for PWRs that have dry containments with safety-grade fan cooler systems), the LOCA energy could be disspated without containment overpressurization and failure. Thus releases associated with PWR-2 and -4 categories could be discounted and PWR-6 releases only used. Such considerations have been incorporated into this revised regulatory analysis.

Table 3 (Continued)

Comment	NRC Staff Response
	(4) Other factors--such as containment structural design margins that argue against gross containment failures (as postulated in WASH-1400), realignment to alternate water sources, controlled venting for BWRs, etc.--have also been considered this revised regulatory analysis.
The use of the CRAC Code and a "no-evacuation," 50-mile-radius model to develop public doses is unrealistic.	The 50-mile radius reflects a substantial part (though not all) of the total population dose, and is thus a reasonable index of the radiological effect on the public. Standardization of calculations to that radius is helpful in comparing risks associated with different issues and average such risks for use with the 1000 person-rem/$M criterion. Evacuation of people is not considered because calculations suggest that, although it may sometimes be important for people directly affected, the effect of evacuation on the total population dose is likely to be small.
NRC should utilize information developed more recently (i.e., NUREG-0772) to reassess and reduce the source terms, rather than continue to use the PWR-2 and PWR-3 release categories from WASH-1400.	Possible changes in the source terms are being considered by the special task force established by the Commission to review the source-term issue. Changes would be premature before this group completes its evaluation and the new values are accepted by all parties involved.
NRC should utilize the "leak before break" concept in evaluating the safety significance of A-43.	Elastic-plastic fracture mechanics analysis techniques to analyze pipe break potential has been used in USI A-2, with a limited number of PWRs being analyzed. For USI A-2, the submittal of such analyses for specific break locations (on a plant-specific basis) will require obtaining an exemption from the requirements of GDC4. Submittal of such analyses to address the USI A-43 debris blockage issues would be reviewed by staff on a plant-specific basis, should a licensee or applicant elect to utilize this approach.

Table 3 (Continued)

Comment	NRC Staff Response
BWR Owners Group	
After quick review of the proposed revision to the regulatory guide, the BWR Owners Group and GE maintain that USI A-43 is not a generic issue for BWRs.	The requirement for long-term decay heat removal is applicable to light-water reactors, both BWRs and PWRs.
The revisions to RG 1.82, which now proposes specific criteria for BWRs, should apply only to light-water reactors that have any potential for harmful debris generation (i.e., light water reactors that extensively use fibrous insulation).	All types of insulation should be evaluated for the potential of debris generation, transport, and suction strainer blockage. The wide variation in plant designs and insulation employed does not support a generic statement.
These comments and any future comments by the BWR Owners Group should not substitute for the normal notice and comment procedure that allows potentially affected licensees to respond to proposed regulatory guide changes.	RG 1.82, Revision 1 (along with NUREG-0897, NUREG-0869 and SRP 6.2.2, Revision 4) was issued "for comment" in May 1983. Only 14 responses were received as of September 1983. Some of these comments (in particular GE's July 11, 1983 letter) cited a need to specifically address BWR-related concerns in the RG. This was done and copies were sent to GE and the BWR Owners Group. Given the previous extensive distribution of "for comment" reports and regulatory positions and the rather small number of responses, the staff does not plan to reissue RG 1.82, Revision 1 for comment. The NRC staff will incorporate additional valid technical points received from the BWR Owners Group and GE. The most recent input from the BWR Owners Group (July 16, 1984) does not provide new significant findings; rather this input re-expresses concerns previously voiced and stresses possible misinterpretations of wording in RG 1.82, Revision 1.

Table 3 (Continued)

Comment	NRC Staff Response
Commonwealth Edison	
The Commission has not sufficiently justified the need to impose retrofit requirements on either operating or near-term operating license units.	A-43 resolution does not mandate retrofits; rather, applicants are requested to assess long-term recirculation capability utilizing RG 1.82, Revision 1 and to then determine what corrective actions may be needed. The use of an information bulletin to the majority of the plants does not constitute imposition of a retrofit.
Cost estimates for surveys, design reviews, and retrofitting are questionable.	The A-43 value/impact evaluation has been revised based on detailed sump blockage probability studies (NUREG/CR-3394) and cost estimates received from industry responses.
The proposed RG 1.82 is overly conservative. However, given the need for assurance that the recirculation sump remains a reliable source of cooling water, the commentor agrees that an evaluation of sump designs, potential for debris, air ingestion, and adequate net positive suction head (NPSH) is fully justified.	The NRC staff acknowledges that conservatisms exist in RG 1.82, Revision 1. However, such conservatisms are prompted by the limited amount of available information regarding insulation destruction due to high pressure jets and attendant debris generation, and the wide variability of plant designs and types of insulation used.
The commentor questions the assumption that 50% of LOCAs lead to sump loss, the value/impact ratio given uncertainties in estimated costs, the basis for assuming 23 years remaining plant life, etc.	A detailed sump failure probability analysis was performed and is reported in NUREG/CR-3394. The "averaged" sump failure probability was 2E-5/reactor-year with a range of 3E-6 to 5E-5/reactor-year.
Consumers Power	
Regarding the proposed Revision 1 to RG 1.82, the commentor stated (1) that Appendix A should be clearly delineated as being an information and guidance source, not as presenting design requirements, and (2) that consistency is needed with respect to NPSH terminology.	Appendix A of proposed RG 1.82, Revision 1 was always intended to provide additional information and/or not design requirements. Appendix A has been clearly labeled as such.

Table 3 (Continued)

Comment	NRC Staff Response
Regarding the value/impact analysis, the commentor questioned the assumption that 50% of the loss-of-coolant accidents (LOCAs) lead to sump blockage and cites a sump failure frequency of 2×10^{-4} per demand from another probabilistic risk analysis.	That 50% of LOCAs lead to sump blockage has been reevaluated (see NUREG/CR-3394), and the results of that detailed study have been used in revising the A-43 release estimates.
The commentor questioned the direct application of core melt frequency reduction for computing avoided accident cost. The commentor disagrees with taking credit for loss of plant cost. Rather, the commentor states that loss-of-plant costs should be deducted from avoided accident costs.	The calculation of avoided accidents costs, loss-of-plant costs, etc., are consistent with current NRC staff evaluation practices. Recalculation of the parameters previously used will be carried out with the revised blockage frequencies.
<u>Creare</u>	
The beta factor used to predict a pump's required NPSH in an air/water mixture is based on data whose scatter was not reported. The NUREG should note this and caution the applicant and reviewer to carefully consider the adequacy of the NPSH margin if it is marginal.	Efforts were made to obtain the original data tapes and calculate the data's scatter; however, this information was not readily available. The suggested cautionary note has been added to NUREG-0897.
The use of an arbitrary minimum allowable NPSH margin, either as a fixed value (i.e., 1 foot) or as a percentage value (i.e., 0.5 x margin with no screen blockage), is not justifiable. It should be recognized that what constitutes a safe NPSH margin is a plant-specific judgment.	NUREG-0897 and RG 1.82, Revision 1 no longer recommend a minimum allowable NPSH margin. Instead, they note that whatever NPSH margin is available (after accounting for hydraulic and screen blockage effects) should be evaluated with respect to each plant's long-term recirculation requirements.
<u>Diamond Power Company</u>	
NUREG-0897 resolves a significant safety problem in a thorough and equitable manner.	The NRC staff concurs.

Table 3 (Continued)

Comment	NRC Staff Response
The commentor provides recommendations regarding the classification of various insulating materials, particularly on the need to distinguish between totally encapsulated insulation and jacketed insulation.	The proposed classifications have been combined with other similar proposals to revise and clarify the insulation classification and descriptions used in NUREG-0897.
The commentor provides listings of the types of insulations purchased since 1980 and the types of insulations used in recent retrofittings.	The information has been added to NUREG-0897 and NUREG-0869, along with data received from other manufacturers.
The commentor states that the costs in the value/impact analysis are in agreement with its costs and provides the following figures: Cost of MIRROR°™ reflective metallic insulation = $40/ft^2 for material alone. Installation cost, excluding material = $25/hour. Productivity = 1.24 hours/ft^2 of insulation.	This cost information has been reflected in the revised value/impact analysis (NUREG-0869), along with other industry cost figures.
Reflective metallic insulation is not the predominant type of insulation used in newer plants. Recently insulated plants mainly use fiberglass insulation.*	Information supplied by Owens-Corning Fiberglass Co. and the Diamond Power Co. regarding types of insulation used in existing and future reactors has been added to NUREG-0897 and NUREG-0869. These reports have been revised to reflect this new information. The trend appears to be toward a higher utilization of fibrous insulations.

* Letter of July 11, 1983.

Table 3 (Continued)

Comment	NRC Staff Response
A report on "HDR Test Results on MIRROR insulation performance during LOCA conditions was submitted to provide additional information to the existing data base used in resolution of USI A-43.*	This report has been included as Appendix E in NUREG-0897, Rev. 1. The results of this report do not support a hypothesis which postulates free and undamaged inner foils being available to transport at low velocities and to cause blockage. However, the limited data base precludes developing a detailed debris generation model.
<u>General Electric Company</u>	
SRP 6.2.2 and RG 1.82, Revision 1 make no distinction between BWRs and PWRs; regulatory criteria should differentiate between various plant designs.**	RG 1.82, Revision 1 and SRP 6.2.2 have been modified to identify PWR- and BWR-related concerns and renamed "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident."
Reference should be made to technical findings that imply that A-43 concerns do not pose a serious problem for BWRs.*	Based on the responses received, the A-43 technical findings will be revised to reflect (1) that there is a more extensive use of fibrous insulations (i.e., NUKON™) than previously identified and (2) that BWRs are reinsulating with NUKON™. NUREG-0897 will reflect current findings and identify both PWR- and BWR-related concerns.
The value impact analysis utilizes a PWR for the risk assessment and PWR-oriented industry impacts and, as such, is not directly applicable to BWRs.*	GE's point on utilizing a PWR probabilistic risk assessment for drawing conclusions for a BWR is acknowledged. Similar assessments have been made for BWRs and those results have been utilized in preparing this revised regulatory analysis.

*Letter of December 6, 1984.
**Letter of July 11, 1983.

Table 3 (Continued)

Comment	NRC Staff Response
General Electric has reviewed the proposed revisions and has concluded that the design requirements proposed in RG 1.82, Revision 1 are excessively prescriptive and not generically applicable to the BWR.*	The requirement for long-term decay heat removal is applicable to both BWRs and PWRs. RG 1.82, Revision 1 Appendix A contains a series of tables (or guidelines) that have been derived from extensive tests and analytical studies. This information is provided for of referral and can, or need not, be used--at the user's option. RG 1.82, Revision 1 is general, and not prescriptive. The applicant has the responsibility for design submittal and justification of the safety aspects thereof.
The proposed RG should be revised so that no further requirements are imposed on designs that have already taken design precautions that preclude air ingestion into, or blocking of, suction lines used for long-term decay heat removal.*	The technical findings in 1983 (versus earlier findings) are considerably different, particularly with respect insulation employed currently and the transport characteristics of insulation debris. The air ingestion potential has been experimentally quantified and found to be small. However the 50% blockage criterion in the current RG 1.82 permitted applicants to essentially bypass the debris blockage question. For those plants where design precautions can be clearly demonstrated, further actions (retrofits) are not necessary.
In addition, the proposed RG should be further revised to provide for alternative means of ensuring that long-term heat removal is not lost as a result of suction blocking or air ingestion.*	The licensee and/or applicant always has the option to propose alternate means to deal with a particular design or safety problem.
In the SER for GESSAR, the NRC indicated that USI A-43 posed no problem for the Mark III containment configuration.*	At the time the SER for GESSAR II was written, A-43 concerns relative to BWRs were still under evaluation. The staff's SER cited several elements of the

*Letter of October 17, 1983.

Table 3 (Continued)

Comment	NRC Staff Response
	GESSAR II design that tended to reduce the probability for blockage of the RHR suction inlets due to LOCA generated debris. The staff concluded that plants referencing the GESSAR II design could proceed pending resolution of USI A-43 without endangering the health and safety of the public while completing its evaluation of GESSAR. The unique aspects of each Mark III plant design should be evaluated during plant-specific reviews of A-43 concerns.
The tests performed by Alden Research Laboratory for Reference 3 may even be very conservative for BWRs since it appears the tests utilized sump screens directly on the sump floor.*	The comment is partially correct, because BWR RHR suction inlets are located at some elevated distance above the wetwell or suppression pool floor. However, the insulation debris transport characteristics (see NUREG/CR-2982, Rev. 1) showed that low velocities (i.e., 0.2 - 0.3 ft/sec) can transport fragmented debris and are applicable to both BWRs and PWRs.
The proposed regulatory guide should be revised to include criteria that will allow alternative measures for precluding loss of long-term decay heat removal due to air ingestion or blockage.*	RG 1.82, Revision 1 states: "This regulatory guide has been developed from an extension experimenal and analytical data base. The applicant is free to select alternate calculation methods which are founded in substantiating experiments and/or limiting analytical considerations." Thus, the applicant is free to select alternate methods or measures for precluding loss of long-term decay heat removal.
Earlier surveys on the use of insulation in light water reactors have concluded that most BWRs utilize metallic insulation, which minimizes the potential	As stated above, current findings do not support the earlier surveys or conclusions. NUREG-0897 is being revised to incorporate findings from public comments

*Letter of October 17, 1983.

Table 3 (Continued)

Comment	NRC Staff Response
for formation and subsequent transport of debris to the sump screens.*	received (particularly with respect to insulations currently used and the change to fibrous insulation from previously used reflective metallic insulations). Recent tests on the transport of thin stainless steel foils show that this material can be transported at low velocities (i.e., 0.2 to 0.3 ft/sec).
<u>Gibbs and Hill, Inc.</u>	
Section B does not discuss the fact that sump configurations that differ significantly from the criteria of Appendix A may be equally acceptable. Gibbs and Hill recommends adding the following concluding paragraph to Section B: "If the sump design differs significantly from the guidelines presented in Appendix A, similar data from full-scale or reduced-scale tests, or in-plant tests can be used to verify adequate sump hydraulic performance."	Appendix A (page 1-9) has wording very similar to the commentor's suggested wording.
Tables A-1 and A-3 are inconsistent and Table A-2 has inconsistencies in water level noted.	The inconsistencies have been corrected.
<u>Northeast Utilities</u>	
Tests show that gratings are as effective as solid cover plate in suppressing vortices.	Gratings were very effective in reducing air ingestion to essentially zero.
The procedure in Appendix B is too prescriptive. The NRC should allow licensees to define and develop their own evaluation methods.	Appendix B in NUREG-0897 presents the staff's technical findings for A-43. Appendix B was included to illustrate major considerations. RG 1.82, Revision 1 is the regulatory document.

*Letter of October 17, 1983.

Table 3 (Continued)

Comment	NRC Staff Response
Credit should be given for top screen area if it is deep enough to reduce the potential for clogging (RG 1.82, Revision 1, Section C, Item 7).	For those plant designs and calculated plant conditions where this point could be unconditionally substantiated, credit would be given.
The licensee should be free to determine methods of inspection and access requirements (RG 1.82, Revision 1, Section C, Item 14).	Section 4, Item 14 states: "The trash rack and screen structure should include access openings to facilitate inspection of the structure and pump suction intake."
RG 1.82, Revision 1 will be used to evaluate sumps in operating plants. This may require backfitting at substantial costs.	The need for backfitting will be based on plant-specific analyses that will reveal the need for, and the extent of backfitting that might be required. The cost of backfit should be weighed against core melt costs.
Appendix A to RG 1.82, Revision 1 requires obtaining performance data if sump design deviates significantly from the guidelines provided. For operating plants, this may result in costly sump testing.	Appendix A states: "If the sump design deviates significantly from the boundaries noted, similar performance data should be obtained for verification of adequate sump hydraulic performance."
NRC estimates for man-rem costs associated with insulation replacement are grossly underestimated.	The value impact analysis has been revised based on cost data received during "for comment" period.
The value impact analysis addresses only PWRs. If the NRC has concluded that this issue only applies to PWRs, then the document should reflect this.	The value impact analysis revision clearly addresses BWR and PWR concerns.
The commentor concurs with the comments submitted separately on this document by the AIF.	The AIF comments are addressed separately; see above.

Table 3 (Continued)

Comment	NRC Staff Response
<u>Owens-Corning</u>	
Detailed comments addressed the wide variation of insulations employed, descriptions, suggested terminology, etc.*	Detailed comments received on insulation types; descriptions, etc. have been used to revise NUREG-0897.
Comments recommended including transport and head loss data for NUKON™ fiberglass tests.*	Data from NUKON™ tests have been referenced and major findings summarized in the revised NUREG-0897.
The commentor questioned Table B-1, Criterion 2, that reflective metallic insulation foil debris would not be transported at velocities less than 2.0 ft/sec.*	Transport tests on reflective metallic foils were conducted and revealed that they can be transported at low velocities (0.2 - 0.5 ft/sec).
The commentor questioned the concept that if there is all reflective metallic insulation there is no problem.*	Inputs received have been used in revising NUREG-0869.
Comments on recommended changes to various tables as discussed at the June 1 and 2, 1983, public meeting.*	Inputs received have been used in revising NUREG-0869.
Numerous comments suggesting word changes that would minimize singling out fibrous type insulations as the screen blockage concern without considering blockages due to reflective metallic insulation materials.*	Inputs received have been used in revising NUREG-0869.
Comments on recommended revision to reflect current status of insulations employed in nuclear power plants.*	Inputs received have been used in revising NUREG-0869.

*Letter of June 23, 1983.

Table 3 (Continued)

Comment	NRC Staff Response
The potential for screen blockage by reflective metallic debris has not been adequately addressed. In particular, the water velocities required to to transport debris and hold it against the sump screen have not been studied.*	A set of experiments to determine transport velocities (similar to those performed on fibrous insulations) has been completed by Alden Research Laboratory. The results are summarized in NUREG-0897 and used in RG 1.82.
The assumption that all fibrous blankets and pillows within 7 L/D of a break are destroyed is overly conservative. Different designs of pillows have varying resistances to destruction by water jets.*	The 7 L/D criterion is based on experimental studies of representative samples of fibrous pillows exposed of high-pressure water jets. These small water jet studies showed that increasing pressure (40-60 psia) results in destruction of pillow covers and release of core material. Furthermore, blowdown experiments in the German HDR facility showed that fiberglass insulations (even when jacketed) were destroyed within 6 to 12 feet of the break, and distributed throughout containment as very fine particles. Unless conclusive experimental evidence is obtained that accurately replicates the variety of conditions that may exist in a LOCA, it is prudent to retain the conservative 7 L/D criterion. The 7 L/D envelope is a significant reduction from the previously proposed 0.5 psia stagnation pressure destruction criterion in NUREG/CR-2791 (September 1982) and (in general) limits the zone of maximum destruction to the primary system piping and lower portions of the steam generators.
The commentor stated that estimated costs for insulation installation and replacement are too low. OCF cost estimates that were provided are*	OCF cost data are utilized in revisions to the value/impact analysis.

*Letter of July 14, 1983.

Table 3 (Continued)

Comment	NRC Staff Response
Cost of NUKON™ = $90/ft^2$ for material (as fabricated) Cost of reflective metallic = $100/ft^2$ for material (as fabricated) Installation cost = $112/ft^2$ for labor and related support	
The commentor provided recommendations for classification of various insulating materials stressing differences between NUKON™ (an OCF product) and other fiberglass and mineral wool materials. The commentor noted the differences between NUKON™ and high density fiberglass.*	Descriptive classifications provided for insulation types have been combined with similar classifications obtained from Diamond Power Company for inclusion in NUREG-0897, Revision 1 and NUREG-0869, Revision 1.
The commentor identified 14 reactor plants that have been reinsulated with NUKON™, are in the process of installing NUKON™, or may install NUKON™.*	OCF plant information have been utilized, along with information from Diamond Power Company, to develop a current picture of insulation utilization in nuclear power plants. The major finding is that the number of plants using or are planning to use fibrous insulation is larger than previously estimated. For example, the Diamond Power list reveals that 25 of 130 operating and projected plants are utilizing fibrous insulation on primary system components.
The commentor recommended inspection surveys of plants to identify actual insulations employed and recommended the modification of a draft generic letter to include this requirement.*	The recommendation for physical plant surveys (or inspection to identify types and quantities insulations employed) is a good one. However, the use of a generic letter is to reconfirm adequate NPSH

*Letter of July 14, 1983.

Table 3 (Continued)

Comment	NRC Staff Response
	margins, and will be based on the actual types and quantities of insulation employed within a given plant without imposing a need to report in detail.
A report on "HDR Blowdown Tests with NUKON Insulation Blankets" was submitted as a supportive document for the capability on NUKON™ insulation to withstand the impact of a high pressure steam-water blast.**	This report has been included as Appendix F in NUREG-0897 Rev. 1. The tests demonstrated that jacketed and unjacketed NUKON™ blankets within 7 L/D will be nearly totally destroyed. However NUKON™ blankets enclosed in standard NUKON™ stainless steel jackets withstood the blast better. But these were insufficient number of tests to draw conclusions for similar insulations.
<u>Power Component Systems, Inc.</u>	
A report on "Buoyancy, Transport and Head Loss Characteristics of Nuclear Grade Insulation Blankets"was submitted as a supportive document for relative efforts in the area of fibrous insulations.***	The formula provided for fibrous debris blockage head loss is included in Section 5 of NUREG-0897, Rev. 1.

*Letter of July 14, 1983.
**Letter of February 18, 1985
***Letter of November 8, 1984

APPENDIX B

BACKGROUND AND SUMMARY OF MINUTES OF MEETINGS OF
COMMITTEE TO REVIEW GENERIC REQUIREMENTS (CRGR)
REGARDING UNRESOLVED SAFETY ISSUE (USI) A-43 RESOLUTION
(CRGR MEETINGS NOS. 26, 28, AND 66)

BACKGROUND AND SUMMARY OF MINUTES OF MEETINGS OF
COMMITTEE TO REVIEW GENERIC REQUIREMENTS
REGARDING UNRESOLVED SAFETY ISSUE A-43 RESOLUTION
CRGR MEETING NOS. 26, 28, AND 66)

BACKGROUND

The staff's proposed resolution of Unresolved Safety Issue (USI) A-43, "Containment Emergency Sump Performance," was sent to the Committee to Review Generic Requirements (CRGR) on October 27, 1982 and was discussed in meetings with CRGR on November 24, 1982 and December 21, 1982. The December 21, 1982 CRGR minutes state that CRGR agrees with the staff's findings and proposed changes to Standard Review Plan Section 6.2.2, "Containment Heat Removal Systems," and Regulatory Guide 1.82, "Sump for Emergency Core Cooling and Containment Spray Systems." However, CRGR agrees only with "forward fit" implementation. The CRGR minutes cite the Deputy Executive Director for Regional Operations and Generic Requirements (DEDROGR) staff analyses. The CRGR questioned four key assumptions in the Office of Nuclear Reactor Regulation (NRR) calculations of averted public dose and stated that the DEDROGR staff felt that the dose was high by a factor of 100. In conclusion, the CRGR recommended that the NRR staff review these risk-reduction calculations, re-affirm or revise the proposed backfit actions, and then meet again with CRGR.

In response to the CRGR recommendations, the staff made additional calculations to estimate the frequency of large loss-of-coolant accidents. These calculations were based on a detailed piping-and-break-probability analysis and estimates of the percentage of these breaks that could lead to sump screen blockage. The results of these calculations are in NUREG/CR-3394, which was published in July 1983. These findings, along with public comments received during the for comment period for USI A-43 (May-June 1983), were used in revising NUREG-0897 and NUREG-0869.

A third meeting was held with the CRGR on July 11, 1984. The summary minutes of Meeting No. 66 pertaining to USI A-43 are those noted as Enclosure 3 to the minutes for CRGR Meeting No. 66, which are included in this appendix. CRGR's views are as noted in this enclosure. After the July 11, 1984 meeting, the staff again revised the proposed resolution of USI A-43.

SUMMARY OF CRGR MEETING NO. 26 (November 24, 1982)*

The CRGR met on Wednesday, November 24, 1982, from 1:00 - 6:00 p.m. S. Hanauer, NRR, presented for CRGR review the NRR recommendations to resolve USI A-43, Containment Emergency Sump Performance. The overall safety concern embodied in USI A-43 is related to the capability of the containment emergency sump to provide an adequate water source to sustain long-term recirculation cooling following a large LOCA.

The problem can be subdivided into (a) sump hydraulic performance, (b) LOCA-generated debris effects, and (c) recirculation pump performance under post-LOCA conditions. Each has been studied by NRR and the technical findings are reported in NUREG-0897 and associated references. With this view, NRR proposed the following actions:

(1) Revise the NRC Standard Review Plan (SRP) Section 6.2.2, "Containment Heat Removal Systems," and Section 6.3, "Emergency Core Cooling Systems." Issuance of the proposed revisions to the SRP is needed to correct previous sump review criteria that are not supported by current findings from full-scale sump tests and generic plant studies (i.e., judgment of sump hydraulic acceptability principally on vortex formation).

(2) Revise Regulatory Guide (RG) 1.82 to reflect the findings in NUREG-0897, "Containment Emergency Sump Performance," to incorporate the results of 2 years of sump testing and generic plant studies and to correct deficiencies such as the 50% screen blockage criterion. Generic plant calculations addressing LOCA-generated debris effects have shown that the 50% blockage criterion can be excessive in some plants and nonconservative in other plants. Continued use, without revision, of this regulatory guidance would permit the sump designer to bypass the need to assess debris blockage effects and to continue to show that a 50% blocked screen does not result in excessive head loss. Appendix A has been included in the proposed revision to RG 1.82 to provide guidance and criteria for assessing sump hydraulic performance, LOCA-induced debris effects, and pump performance under adverse conditions. A combined consideration of these three aspects is necessary to determine overall sump performance and acceptability with respect to assurance that adequate pump NPSH margin will exist.

(3) Operating plants should assess the extent of debris blockage potential and, based on the outcome of plant assessments, action should be taken to modify the sump screens or to replace all fibrous insulation with encapsulated insulation.

*Verbatim copy.

The Committee commended the staff for the thorough technical analysis described in NUREG-0897 and agreed with recommendations (1) and (2) above, which reduce requirements on future OL applicants. In support of recommendation (3), NRR presented cost-benefit analyses which showed the benefits (using $1000 per person-rem averted) outweighed the costs of the proposed requirements in (3) for operating plants. The Committee suggested that the benefits (reduction in core melt probability) appeared to be overstated by at least a factor of 10, and perhaps 100, and that the costs appeared to be understated. Thus, it was not clear to the Committee that recommendation (3) could be justified on a cost-benefit basis, even though it was acknowledged to be good engineering practice to replace unencapsulated fibrous insulation with encapsulated insulation.

In response to a question whether the staff has considered the effects of paint debris on sump performance, NRR said they had not considered it in the context of USI A-43, but they agreed to review what consideration had been given to paint debris in previous staff reviews. The Committee decided to discuss USI A-43 in a subsequent meeting after information on the potential effects of paint debris has been received from NRR.

SUMMARY OF CRGR MEETING NO. 28 (December 21, 1982)*

The CRGR met with respresentatives of NRR to further pursue questions regarding USI A-43 Containment Emergency Sump Performance. The CRGR, during Meeting No. 26, had questioned the potential for sump blockage due to paint removed from containment surfaces during a LOCA. The question of the potential for sump blockage due to paint removal and transport to the sumps was addressed in a memorandum from H. Denton to V. Stello dated December 16, 1982. The NRR position on the paint blockage issue was that:

(1) Analyses indicate that there is not a basis for concern as a generic safety issue;

(2) The issue will be further evaluated within established NRR procedures for treating proposed new generic issues, to determine the priority for further evaluation;

(3) The possible issue of paint removal therefore should not delay obtaining industry and public comment on the defined A-43 issue.

The CRGR accepted the NRR position on the paint blockage issue.

THe CRGR addressed the level of risk reduction, or benefit, to be obtained from the analyses and potential modifications proposed to be required of the several licensees that might be found to have combined insulation/sump designs that could lead to failure of long-term recirculation cooling.

The Committee (as reflected in the minutes of CRGR Meeting No. 26 November 24, 1982) has agreed with the forward-fit aspects of the NRR proposed requirements. A revised Standard Review Plan Section 6.2.2 and a revised Regulatory Guide 1.82 would incorporate changes in design criteria that would provide greater assurance of sump performance, but would be imposed only on Operating License and Construction Permit applicants filing Final or Preliminary Safety Analysis Reports at some time after the effective dates of the revised Standard Review Plan Section and the revised Regulatory Guide.

To support the proposed backfit requirements, NRR provided a generic value/impact assessment comprised of a probabilistic risk analysis of the effects of loss of sump function and estimated costs of the backfit requirements proposed for licensees to reduce the risks of such loss. The probabilistic risk analyses resulted in an expected value of offsite public dose (person-rems) that could be averted from the estimated six to ten plants that are expected to need modifications. Key assumptions in this NRR analysis are:

*Verbatim copy.

(1) The expected value of large LOCA (greater than 6" diameter pipe) incidence is 10^{-4} per reactor-year.

(2) For those plants having fibrous insulation that could potentially result in sump blockage, it is assumed that 50% of all LOCAs in piping greater than 6" diameter will result in complete failure to pump any water from any containment sump.

(3) The assumed failure of recirculation flow (from sump) is assumed to conditionally fail both reactor building spray and emergency core cooling, thereby leading to a core melt with containment failure by overpressure. No credit was given for potential beneficial operator action to prevent sump blockage by throttling the emergency core cooling system pump or to utilize alternate water sources and systems to prevent either core melt or loss of containment function. Thus, for the class of plants above, the NRR analysis assumed the core melt frequency for this LOCA sequence is 5×10^{-5}/Rx-Yr.

(4) The offsite consequence model used to predict expected values of population dose assumed an average site, a 50-mile radius, and no evacuation of population during the accident.

An analysis by the DEDROGR staff indicated that each of the assumptions above was probably too conservative and that the NRR predicted value of averted public dose of about 65 person-rems per plant per year was too high by a factor of at least 100. If this were indeed the case, the proposed implementation plan actions would not appear to be justified. The CRGR recommended that NRR review its risk reduction analysis in light of the analysis performed by the DEDROGR staff with the objective of developing the most realistic assessment of averted radiological dose. NRR should then reaffirm or revise the proposed backfit actions, and discuss with CRGR again if they believe the cost benefit analysis justifies the proposed backfit actions.

SUMMARY OF CRGR MEETING NO.66 (July 11, 1984)

Enclosure 3 to the Minutes for CRGR Meeting No. 66

CRGR REVIEW OF THE PROPOSED RESOLUTION TO UNRESOLVED SAFETY ISSUE A-43 "CONTAINMENT EMERGENCY SUMP PERFORMANCE"

The NRR Division of Safety Technology representatives T. Speis, F. Schroeder, K. Kniel and A. Serkiz presented the proposed resolution for CRGR review. The package submitted for review was transmitted by a memorandum dated June 14, 1984 from H. Denton to V. Stello, Jr; it included the following documents:

1. Summaries of USI A-43 References.

2. NUREG 0897, Revision 1, March 1984, describing the technical findings of the effort.

3. Regulatory Guide 1.82, Rev. 1, May 1984, "Sump Design and Water Sources for Emergency Core Cooling."

4. Standard Review Plan Section No. 6.2.2, Revision 4, "Containment Heat Removal Systems."

5. NUREG 0869, Rev. 1, USI A-43 Regulatory Analysis, containing a value/impact analyses, summary of public comments received and action taken, and a proposed generic letter for implementation of R.G. 1.82, Rev. 1.

6. Draft Generic Letter, subject: "Assessment of Available NPSH Margin for Long Term Cooling."

7. Note to A. Serkiz from B. Shields, April 3, 1984, Subject: Generic Letter on Containment Emergency Sump Performance.

The NRR presenters at the CRGR meeting also provided a handout titled "USI A-43, Containment Emergency Sump Performance," which is attached.

CRGR was requested to recommend to the EDO approval of the following final actions:

1. Issue NUREG-0897, Revision 1 as the technical findings for resolving USI A-43;

2. Issue Regulatory Guide (RG) 1.82, Revision 1 and SRP Section 6.2.2, Revision 4 for guidance and use in the OL and CP review cycle as part of the normal review process.

3. Issue a generic letter to all LWR licensees and applicants pursuant to 10 CFR 50.54(f). This letter would request a plant-specific assessment of

post-LOCA debris blockage effects (using guidance provided in Appendix A of RG 1.82, Revision 1) on net positive suction head (NPSH) margin, and would request the responders to submit the calculated NPSH margin available, and a description of any plant modifications shown to be necessary by this assessment.

The staff did not propose a generic solution to the variety of potential deficiencies that were postulated to exist among licensees. The proposed generic backfit requirement was for licensees to complete an assessment and report to NRC using the staff specified criteria. Further staff/licensee interaction to define and approve acceptable solutions to design/operational deficiencies would be pursued on a plant-specific basis. Five to 20 plants were expected to be in this category.

The safety rationale for the proposed requirements was that the containment recirculation mode of long term decay heat removal following a LOCA must be assured. The proposed backfit would not provide substantial additional protection over that thought to exist, but would assure a level of safety previously thought to exist. The backfit could in most cases be limited to analysis to verify the sump/pump performance. Staff review and followup actions would be limited to those plants with identified problems. CP and OL applicants would be required to demonstrate adequate design margin for NPSH during the licensing review, in accordance with the SRP and Regulatory Guide revisions presented in the CRGR package.

The staff proposal originally presented to the CRGR at two meetings in November and December 1982 was issued for public comment in early 1983. The current proposal includes revisions made as a result of public comment. In addition, a more sophisticated analysis was completed to assess the likelihood of sump blockage than was presented in support of the initial proposal in late 1982.

The current proposal was presented as based on

(1) A review of expected LOCA probability as a function of pipe size, weld type, and joint configuration.

(2) An examination of the Salem-1 plant containment layout and selection of 238 locations in pipe as those expected to represent all LOCAs significant with respect to then current staff guidelines for selection of postulated breaks in high energy pipe within containment (SRP Sections 6.2).

(3) A mathematical model describing the volume of insulation removed by LOCA as a function of break size and location relative to adjacent insulated pipes and vessels.

(4) Investigations of the likelihood of transport of stripped insulation to the sump. Five operating plant layouts were modeled analytically to evaluate the water velocity through expected pathways for transport of insulation.

(5) Results of experiments with various insulation materials to determine minimum water velocities necessary to transport typical insulation debris.

(6) Calculations of pressure loss across typical sump screens due to insulation deposition on the screens.

A value-impact analysis was presented based on the calculations of sump failure probability and cost data, much of which was supplied by industry during the public comment period. Core melt probability was assumed to be identical with sump failure probability, based on an SAI report [1/] completed in September 1982. To calculate net safety benefit from reducing this risk, the assumption was also made that given a core melt, the containment fails with a probability of 1. This containment failure assumption led to offsite radioactivity releases commensurate with the PWR 2 release category of WASH 1400. Calculation of offsite dose commitment in person-rem was done using the CRAC code and a site population and meteorology selected to represent an average of all plants.

The value-impact results produced by the staff are summarized on page 9 of the attachment to this enclosure.* The staff pointed out that the value/impact ratio was only moderate and that for most plants no risk reduction was anticipated. However, the staff's position is that the deterministic licensing bases of 10 CFR 50.46 (b)(5) mandated the confirmatory analysis steps proposed in the draft generic letter. An additional concern voiced by the staff was the observed recent tendency of utilities to change insulation materials to types that are more likely to block a sump.

The following major points were made in the discussion of the staff proposal at this meeting:

1. The application of the initiating event probability data in Table 1, Appendix B, page B-4, NUREG 0869, was considered unclear by the CRGR. The application of the data was called into question because the specification of 238 postulated break locations seemed to be a low number of locations relative to the total length of piping in containment subject to a high pressure LOCA. The tabulated probabilities as a function of pipe size were intended to represent the population of all pipe in containment subject to LOCA failures.

   Further, the Committee was concerned that the selected break locations were such that the probability of pipe displacement to give the postulated double ended or even single ended flow model used to predict insulation removal was much lower than the probability of pipe rupture given in the

1/ Ferrell, W.L., et al, "Probabilistic Assessment of USI A-43", Science Applications Inc., September 1982

* Table on Page 9 of attachment is duplicated on Page B-11

tabulated data. The tabulated data is representative of a data base that includes all kinds of disruptive failures in pipe, not just the sudden explosive rupture with deflection at the break necessary to provide a full flow area blowdown.

2. The CRGR believed that there was considerable but undescribed conservatism in the model for determining, given a pipe break of full flow area, how much insulation is torn off a designated target area or volume. Two such conservatisms would be:

   a. The reduction in (initiating event) probability due to the selection of pipe break direction such that the maximum amount of insulation is targeted.

   b. The use of a hemispherical volume of radius 7 x the pipe diameter oriented to contain the most target insulation, and the assumption that all insulation within that volume is stripped from the component even if, as in the case of a steam generator, the "back" side of the component is not impinged directly by the break jet flow.

3. The CRGR noted that there was considerable conservatism in the conclusion drawn from the analyses of five plants that all insulation debris torn off components would be transported to the sump. NUREG/CR 2791, where the analyses are reported, reports that in four plants insulation debris arriving at the sump would not approach amounts necessary to significantly degrade sump operation. In one plant, Maine Yankee, material might adversely affect the sump based on the assumption that all material completes the trip to the sump, a scenario which seemed unlikely to the report's authors.

4. The CRGR noted further conservatisms in considering the effects of material at the sump, even given that material arrives at the sump in quantities necessary to cause sufficient screen blockage to unacceptably reduce the net positive suction head (NPSH) at the recirculation pump.

   Two such conservatisms were noted:

   (a) The assumption that insulation debris distributes uniformly over the sump screens,

   (b) The assumption that recirculation flow in the range 6 to 10 thousand gallons per minute is the appropriate design flow requirement for the sump screens. It was stated in the meeting that most plants have flow instruments in ECCS recirculation lines, so operators could be expected to reduce pump flow to the minimum required to cool the reactor core after vessel refill immediately following blowdown after a large LOCA. This flow requirement is likely to be in the range of 300 to 1000 gallons per minute, considerably less than the flows postulated in the NRR proposal. Since sump blockage according to the NRR presentation is not expected in less than 1-2 hours, the lesser cooling flow requirements are highly likely.

5. The CRGR noted probable conservatism in the postulated WASH 1400 release categories used in the analysis. It was suggested that the release source terms associated with an early, energetic, above ground containment failure such as the WASH 1400 PWR-2 category would be excessively conservative. Evidence pointed to at most a delayed core meltdown with eventual core melt through the containment base mat, resulting in releases no higher than that associated with a PWR 5 or 6 release.

6. The CRGR overall consensus was that given the questions raised about the assumptions and/or levels of conservatism of the analyses, NRR's position that the proposals based on the analysis are of only moderate benefit versus cost importance, the probability that further clarification or resolution of CRGR concerns may suggest a lower safety benefit than does the current proposal, and the estimated applicability of the proposed solution to only a few plants, all combine to argue against approval of a backfit requiring all licensees to expend resources to demonstrate that their designs are acceptable.

Conclusions

The Committee concluded the following as a result of its review of the material transmitted and the meeting discussion:

1. The proposed requirements package was rated of medium importance by NRR. CRGR review of the information presented on the safety benefit to be achieved resulted in uncertainty about the validity of the analyses, with the possibility that the proposed potential risk reductions may be far overestimated and would apply to relatively few plants.

2. The proposed backfit requirement on operating reactors to analyze containment sump performance and report to the NRC should not be promulgated at this time due to the uncertainties raised in item 1 above.

3. The CRGR would be willing to reconsider this proposal or a modified proposal by NRR at a regularly scheduled meeting after receipt of responses to the issues raised at the meeting and discussed in the meeting summary.

VALUE/IMPACT OVERVIEW

PLANTS W/O (CATEGORY A):

ANALYSIS COST = $45K/PLANT

AVERTED RELEASE = 0 MAN-REM/Rx

PLANTS W/SOME BACKFITS (CATEGORY B):

ANALYSIS COST = $ 65K/PLANT

BACKFIT COST = $300K/PLANT

$365K/PLANT

PLANTS W/INSULATION REPLACEMENT (CATEGORY C):

ANALYSIS COST = $ 85K/PLANT

BACKFIT COST = $820K/PLANT

$905K/PLANT

AVERTED RELEASE = 650 MAN-REM/RX

INDUSTRY DISTRIBUTION:

CATEGORY	ASSUMED NO. OF PLANTS	COST ($M)	AVERTED RELEASE (MAN-REM)	V-I RATIO (MAN-REM/$M)
A	90	4.1	0	0
B	15	5.5	9,750	1,770
C	5	4.5	3,250	720
TOTALS:	110	14.1	13,000	920

SUMMARY OF CRGR MEETING NO. 80 (SEPTEMBER 9, 1985)

Enclosure 3 to the Minutes of CRGR Meeting No. 80
USI A-43 Containment Emergency Sump Performance

Dr. T. Speis and A. Serkiz of NRR presented for CRGR review a proposed final resolution to USI A-43. The CRGR was requested to recommend several actions pursuant to resolving USI A-43:

1. Issue the staff's technical findings in NUREG-0897, Revision 1 for use as an information source by applicants, licensees, and the staff in addressing the design and operation of containment emergency sumps and BWR RHR suction intakes.

2. Issue Regulatory Guide 1.82, Revision 1, to include the technical findings reported in the new NUREG-0897 Revision 1. This would provide improved regulatory staff positions as guidance for assessment of sump performance and BWR RHR suction intakes, including debris blockage effects.

3. Issue NRC Standard Review Plan (SRP) Section 6.2.2, Revision 4, to incorporate the guidance provided by the revised Regulatory Guide 1.82 and the technical findings in NUREG-0897, Revision 1.

4. Issue a generic letter to all applicants and licensees outlining the potential for safety concerns related to post-LOCA sump blockage and the fact that the original Regulatory Guide 1.82 (Revision 0) is inadequate in light of the more current information resulting in Revision 1 to Regulatory Guide 1.82.

   A copy of the viewgraphs used for the A-43 proposal is attached. The proposed A-43 resolution discussed was a revision to a previously presented proposal which was discussed at CRGR meeting number 66 on July 11, 1984. The new proposal was to take actions to assure that the technical findings and new guidance are available to the nuclear industry, to advise the industry of the inherent safety benefit to be gained by using the new guidelines, to state the staff's position that this new guidance will be used by the NRC only in new CP reviews and certain standardized design reviews, and to recommend that the staff's technical findings on A-43 should be considered when insulation is replaced at operating plants.

Revisions in the prior regulatory analyses (discussed at meeting number 66) were discussed at this meeting in support of the proposed resolution. These are highlighted briefly:

1. Credit for operator action to recover ECCS flow that may have been lost is now given as a 50 percent likelihood that the operator will detect and mitigate a loss of flow given a sump blockage event.

2. Plants of varying containment types and having different accident mitigating systems were evaluated to better define the risks inherent in each design type. Expected values of offsite consequences and

value/impact ratios were calculated for all the plant types considered and were generally such that much more than $1000/person-rem would be required to effect safety-beneficial changes in the plants. In addition, the conservatism inherent in pipe break probabilities used in the regulatory analysis was recognized in the regulatory analysis and discussed in the CRGR meeting. Pipe break data used in the analysis has since been superseded by advanced fracture mechanics analysis and experiments. The more recent work shows that breaks in ductile pipes larger than eight inches in diameter, the size range necessary to provide a significant sump blockage probability, may occur at frequencies that are several orders of magnitude less than the frequencies used in the A-43 analyses.

The CRGR decided to recommend approval of the proposed A-43 resolution presented, with several specific comments on changes that should be incorporated in the various documents:

1. The more recent work on fracture-mechanics resulting in lower estimates of pipe-break frequency should be explicitly recognized and referenced in the summary section of the regulatory analysis.

2. Implementation wording in the Regulatory Guide and SRP section should be modified to clearly show that the NRC staff use of the new review material will be forward-fit only.

3. The generic letter should clearly state that NRC application of the new guidance to an operating plant, particularly with respect to the NRC staff reviews of licensee 10 CFR 50.59 reviews, will be treated by the NRC staff as a plant-specific backfit action pursuant to 10 CFR 50.109.

REFERENCES

H. R. Denton to V. Stello, Jr., memorandum dated October 27, 1982, "CRGR Review of Proposed Revisions to SRP Section 6.2.2 and RG 1.82 and the Supporting Technical Information Document NUREG-0897, as related to USI A-43, 'Containment Emergency Sump Performance.'

V. Stello, Jr. to W. J. Dircks, memorandum dated December 10, 1982, "Minutes of CRGR Meeting Number 26."

H. R. Denton to V. Stello, Jr., memorandum dated December 16, 1982, "Potential Sump Screen Blockage Due to 'Paint Sheets' (Ref. CRGR Meeting of 11/24/82 on USI A-43)."

V. Stello, Jr. to W. J. Dircks, memorandum dated January 11, 1983, "Minutes of CRGR Meeting Number 28."

H. R. Denton to V. Stello, Jr., memorandum dated February 28, 1983, "Response to CRGR Comments on USI A-43."

V. Stello, Jr. to W. J. Dircks, memorandum dated July 24, 1984, "Minutes of CRGR Meeting Number 66."

H. R. Denton to W. J. Dircks, memorandum dated August 20, 1984, "Feedback and Closure: CRGR Meeting Number 66 (RE: Proposed Resolution of USI A-43)."

V. Stello, Jr. to W. J. Dircks, memorandum dated September 13, 1985, "Minutes of CRGR Meeting Number 80."

APPENDIX C

ESTIMATION OF PIPING FAILURE PROBABILITY

ESTIMATION OF PIPING FAILURE PROBABILITY

For USI A-43 evaluations, it was necessary to consider a large number of potential break locations over a wide range of piping sizes used for an actual reactor coolant system design. This, in turn, necessitated assigning piping failure probabilities to each of the potential break locations and pipe sizes considered. The assigned pipe failure probabilities shown in Table 1 were used as basic inputs for subsequent computerized calculations that weighted these values by the number of welds and the number of piping sections. The calculations were done for particular diameter size categories and for the plant piping designs utilized (see NUREG/CR-3394).

The pipe failure (rupture) probabilities shown in Table 1 were derived from the assessments presented by Dr. S. H. Bush in his October 1977 paper entitled, "Reliability of Piping in Light Water Reactors," IAEA-SM-218/11. This paper included assessment of the validity of piping failure probabilities cited from various world-wide literature sources and their applicability to nuclear systems. Bush also evaluated the safety significance of reported failures in nuclear piping systems and presented information that would allow an estimate to be made of the relative probability of severance because of (1) general faults (e.g., inadequate piping flexibility) and (2) various components (e.g., straight piping runs, joints, tees, and elbows), depending on their size and potential crack orientation. Using those world-wide sources and failure experiences deemed relevant to nuclear piping systems, Bush reached the following conclusions:

(1) The failure probabilities for larger sizes of nuclear piping are considered to be in the range of 10^{-4} to 10^{-6} per reactor year, exclusive of intergranular stress corrosion cracking (IGSCC).

(2) Small pipe sizes of lesser safety significance have much higher failure rates.

(3) In boiling water reactors (BWRs), IGSCC can cause failure rates much higher than 10^{-4} per reactor year (10^{-2} per reactor year) in piping 4 to 10 inches (102 to 205 mm) in diameter.

(4) Suggested failure mechanisms apply in most instances exclusive of IGSCC.

(5) Catastrophic failure would appear more likely from operator error or design and construction errors (water hammer, improper handling of dynamic loads, and undetected fabrication defects) rather than conventional flaw initiation and growth or fatigue.

Utilizing the failure information that Bush deemed relevant to nuclear plants, an overall piping failure probability of 3×10^{-4}/reactor-year can be derived from his paper for pipes greater than or equal to 3 inch diameter.

TABLE 1

Piping failure probability estimates

Pipe size (inches)	Diameter class	Failure Probability (1/Rx-Yr)	Weld failure probability distribution		
			Weld type 1	Weld type 2	Weld type 3
D	J	P_j	W_n	W_a	W_e
2 to <6	1	$3x10^{-5}$	0.7	0.15	0.15
6 to <10	2	$4x10^{-5}$	0.5	0.30	0.20
10 to <16	3	$3x10^{-5}$	0.5	0.30	0.20
16 to 28	4	$3x10^{-6}$	0.5	0.30	0.20
$\geq$28	5	$3x10^{-6}$	0.5	0.30	0.20

$$\Sigma P_j = 3.76x10^{-4}/Rx\text{-}yr$$

Weld Type 1 = fabricated and non-standard joints
Weld Type 2 = high restraint joints and tees with joints
Weld Type 3 = elbows, reducers, and straight piping runs with joints

This value is consistent with pipe failure estimates assigned in WASH-1400 for piping in the size range of more than 2 inches to less than 6 inches in diameter. The Bush paper can also be used to develop a failure probability distribution as a function of pipe diameter of about 4 x 10^{-4}/reactor-year frequency, with the relative distributions (as an approximate percentage value) being

6 inches to < 10 inches diameter 13%
10 inches to < 16 inches diameter 10%
16 inches to 28 inches diameter 1%

Furthermore, the Bush paper indicates that circumferential cracks (if they exist) would be expected to be of greater significance (by about a factor of 2) than axial cracks relative to the rupture probability. High restraint, fabricated joints would also be expected to make a higher contribution to the overall rupture probability than would straight runs of pipe. Therefore, the above percentages can be further redefined, this time as a function of piping joints, etc. The results of these types of considerations are reflected in the piping and joint failure multipliers shown in Table 1, which were utilized in the USI A-43 sump blockage assessments (see Appendix D).

In his paper Bush also observed that a well-planned program of periodic inspection should dramatically reduce the probability of catastrophic failure of piping, and he cites various studies that have suggested that inspection benefits could result in reducing estimated failure frequencies by as much as 1 to 3 orders of magnitudes.

Experimental and analytical work based on mechanistic fracture mechanics that has been done after the 1977 assessment by Bush (see, for example, NUREG-1061, Vol 1, August 1984) also indicates that the rupture probability of large-size ductile piping (unaffected by IGSCC) could also be significantly less than the assigned values of Table 1, perhaps by about several orders of magnitude. The term "leak before break" has been coined from this later work. The sump blockage assessment performed toward resolution of USI A-43 did not, however, give any explicit probabilistic credit for this "leak before break" concept.

References

Bush, S. H., "Reliability of Piping in Light-Water Reactors," International Atomic Energy Agency, IAEA-SM-218/11, October 1977.

U. S. Atomic Energy Commission, WASH-1400, "Reactor Safety Study," October 1975 (also issued by NRC as NUREG-75/018).

U. S. Nuclear Regulatory Commission, NUREG-1061, "Report of the USNRC Piping Review Committee, Investigation and Evaluation of Stress Corrosion Cracking in Piping of Boiling Water Reactor Plants," Vol 1, August 1984.

APPENDIX D

ESTIMATION OF PWR SUMP FAILURE PROBABILITY

ESTIMATION OF PWR SUMP FAILURE PROBABILITY

SUMP FAILURE

Sump failure is defined as a loss of pressurized water reactor (PWR) sump (or boiling water reactor (BWR) suction intake) capability to provide an adequate water source and net positive suction head (NPSH) margin to the residual heat removal (RHR) and containment spray system (CSS) pumps during the period after a loss-of-coolant accident (LOCA) because of the effects of debris blockage. Stated another way: does the head loss across a debris-blocked screen or suction strainer exceed the NPSH margin available under zero blockage conditions?

SUMP FAILURE PROBABILITY

The sump failure probability (because of debris effects) is a function of:

(1) The probability of a pipe break or weld failure, because the LOCA is the initiating event that can destroy insulation.

(2) The potential break sizes and locations within containment with respect to other piping and insulated primary system components (e.g., steam generators, pressurizer, pumps, safety injection tanks) because the expanding jet will destroy insulation that falls within the jet expansion envelope.

(3) The types and quantities of insulation employed, because blockage effects will vary with insulation type (i.e., fibrous debris versus metallic insulation debris) and the location of such insulation.

(4) The containment layout and sump (or suction inlet) location, which can control debris transport. (Can the sump be directly targeted by the break jet, resulting in prompt transport, or does the debris transport occur later because of recirculation flow drag?).

(5) The size of debris screens (or suction inlet strainers), because larger screens can accommodate larger quantities of debris without incurring large head losses.

(6) Post-LOCA recirculation flow and pump NPSH requirements, which determine whether a blocked screen situation will result in loss of NPSH margin and pumping capacity.

Thus, arriving at an estimated sump failure probability becomes a complex and plant-specific evaluation based on: (1) probabilistic estimates (i.e., pipe failure probabilities), (2) plant design features, and (3) a deterministic analysis of debris generated, potential transport to the sump, and potential attendant blockage which could lead to loss of NPSH. Such an evaluation begins with an estimation of pipe failure probabilities (which are a function of pipe size and weld type), followed by an estimate of the volume of debris that can be generated by any break postulated (which is a function of break size, break-to-target locations and possible combinations, and break-jet model); debris transport potential; and blocked screen head loss (which is a function of the quantity of debris transported, the available debris screen area, and the post-LOCA recirculation flow rate requirements). The examples that follow are provided to illustrate such evaluations.

ESTIMATING PIPE WELD FAILURE PROBABILITIES

The first step is to estimate the probability of pipe (or weld) failure to calculate the initiating event probability (i.e., LOCA probability). The probabilities shown in Table 1 were estimated by M. Taylor (DEDROGR staff based on his review of "Reliability of Piping in Light Water Reactors" by S. Bush (IAEA-SM-218/11, October 1977; see also Appendix C of this report.) They represent the estimated failure probabilities for all piping in a typical nuclear plant for the diameter classes shown. For example, the estimated pipe failure probability of any pipe in the 10- to 16-inch diameter range is 3E-5 per Rx-yr, and the failure probability of a fabricated or non-standard weld in this diameter range is 1.5E-5 per Rx-yr.

To estimate pipe failure probabilities as a function of pipe diameter size and the type of weld (the assumption being that failure would occur at the weld joint), the data shown in Table 1 can be used to calculate a weld failure probability (P_{wk}), as follows:

$$P_{wk} = \frac{(P_j)(N_n W_{jn} + N_a W_{ja} + N_e W_{je})}{(X_n W_{jn} + X_a W_{ja} + X_e W_{je})} \qquad \text{Eq (1)}$$

Where:

- P_{wk} = probability of a weld failure in diameter class "k" pipe, weld type weighted
- P_j = probability of pipe break in any pipe in diameter class "j"
- N_n = number of welds of type "n" and diameter "k"
- X_n = number of welds of type "n" in diameter class "j"
- W_{jn} = probability weighting factor for type "n" welds
- N_a = number of welds of type "a" and diameter "k"
- X_a = number of welds of type "a" in diameter class "j"
- W_{ja} = probability weighting factor for type "a" welds
- N_e = number of welds of type "e" and diameter "k"
- X_e = number of welds of type "e" in diameter class "j"
- W_{je} = probability weighting factor for type "e" welds

TABLE 1
Piping failure probability estimates

Pipe size (inches)	Diameter class	Failure probability (1/Rx-Yr)	Weld failure probability distribution		
			Weld type 1	Weld type 2	Weld type 3
D	J	P_j	W_n	W_a	W_e
2 to <6	1	$3x10^{-4}$	0.7	0.15	0.15
6 to <10	2	$4x10^{-5}$	0.5	0.30	0.20
10 to <16	3	$3x10^{-5}$	0.5	0.30	0.20
16 to 28	4	$3x10^{-6}$	0.5	0.30	0.20
$\geq$28	5	$3x10^{-6}$	0.5	0.30	0.20

$$\Sigma P_j = 3.76x10^{-4}/\text{Rx-yr}$$

Weld Type 1 = fabricated and non-standard joints
Weld Type 2 = high restraint joints and tees with joints
Weld Type 3 = elbows, reducers, and straight piping runs with joints

The weld failure probabilities that were derived from the failure assumptions shown above and treated algebraically as described in Equation 1 were used to estimate the LOCA probability, using the Salem 1 plant and piping layout. NUREG/CR-3394 details those analyses.

The weld sizes and distributions derived from a typical Salem 1 plant primary cooling piping loop are shown in Table 2. The breaks were assumed to occur at weld locations [following the criteria in Section 3.6.2 of the NRC Standard Review Plan (NUREG-0800)]. The loop analyzed contained 238 welds associated with piping that can be classified as LOCA-sensitive piping.

Because the estimated pipe failure probabilities in Table 1 must be distributed on a per weld basis, the first step is to apportion (or redistribute) the total probability for a diametric size class to all existing pipe sizes. Table 3 has been constructed to illustrate how Equation 1 was used to develop such a distribution. If weld type is ignored in the first step, the per weld failure probabilities distribute as a function of the fraction of the number of welds (of a given pipe size) to the total number of welds in a particular diametric class (that is, j = 1, 2, 3, 4, and 5, as shown in Table 3). The summed totals (for a particular j class) must always sum to the total probability for that diameter class extracted from Table 1. Table 3 illustrates this process. In addition, Table 3 compares this type of fractional distribution with the weld type weighted values from NUREG/CR-3394. The variability within a particular diameter class (j = 1, 2, 3, 4, and 5) results from the distribution of weld types in the Salem 1 plant piping layout analyzed (see again Table 2). NUREG/CR-3394 analyses were based on both weld type and pipe segment probability distributions.

The initiating event probability (P_0) attributable to all 237 welds was calculated to be 3.7E-4/Rx-yr and is in agreement with the ΣP_j value shown in Table 1. Thus this type of single loop analysis is applicable to all loops. Had the distribution methodology discussed above been applied to four loops, the same initiating break probability would have been obtained, because the overall probability cannot exceed the summation noted in Table 1.

Because satisfactory operation of the sump is essential when breaks occur in the primary coolant system LOCA-sensitive pressure boundary, the overall probability (P_0) discussed above should be reduced (for purposes of estimating sump blockage probabilities) to account only for such breaks in the primary coolant system. In many PWRs (such as Salem 1) such piping is located within the crane wall region and that break probability is designated P_i. Therefore, elimination of secondary system piping weld locations and piping outside the crane wall resulted in a calculated P_i of 1.84E-4/Rx-yr for Salem 1. These break locations (associated with P_i) were further analyzed for debris generation as discussed below.

TABLE 2
Break size distribution for Salem plant analysis

System weld distribution

System designation	System	No. of welds	No. of welds @ pipe diameter (inches)
1	Hot leg	8	8 @ 34
2	Cold leg	6	6 @ 36.3
3	Cross over	6	6 @ 32.3
4	Safety injection/cold leg	41	15 @ 10, 4 @ 11 11 @ 6, 11 @ 2
5	Safety injection/hot leg	33	6 @ 6, 27 @ 2
6	Chemical volume and control	13	5 @ 16, 8 @ 14
7	Feedwater	95	22 @ 4, 7 @ 3, 66 @ 2
8	Main steam	20	19 @ 30, 1 @ 32
9	Pressurizer	16	16 @ 14
SUBTOTAL WELDS		=238 for loop analyzed	
		-16 for pressurizer	
		-13 for chemical volume and control	
		-33 for safety injection/hot leg	
		176 x 4 loops = 704	

Pressurizer loop welds = 16
2 chemical volume and control loop welds (13) = 26
2 safety injection and hot leg penetrations welds (33) = 66

TOTAL WELDS =812 for 4 loops

Diameter (inches)	No. of welds		
34	8	hot leg	
36.3	6	cold leg	→ 40 welds for 16 in to 34 in. pipe
32.3	6	crossover	(P_o = 3E-6)
32	1	Main steam	
30	19		
16	5		→ 29 welds for 10 in. to 16 in. pipe
14	24		(P_o = 3E-5)
10	15		→ 19 welds for 6 in. to 10 in. pipe
8	4		(P_o = 4E-5)
6	17		
4	22		→ 150 welds for 2 in. to 6 in. pipe
3	7		(P_o = 3E-4)
2	104		

Source: NUREG/CR-3394, Vol 2, Table A.1-1.

TABLE 3

Probability distributions as a function of diameter*

(1) for j = 1, P_j = 3E-4

Wn	Wa	We
0.7	0.15	0.15

k = 6", N_{k6} = 17
= 4", N_{k4} = 22
= 3", N_{k3} = 7
= 2", N_{k2} = 104
N_k = 150 welds for j=1

w/o Weld Specification	NUREG/CR-3394 Values
$P_{k=2"}$ = (3E-4)(104/150) = 2.07E-4	2.07E-4
$P_{k=3"}$ = (3E-4)(7/150) = 0.14E-4	0.13E-4
$P_{k=4"}$ = (3E-4)(22/150) = 0.44E-4	0.41E-4
$P_{k=6"}$ = (3E-4)(17/150) = 0.34E-4	0.38E-4
$P_{j=1}$ = 2.99E-4	2.99E-4

(2) for j = 2: k = 8" and 10", N_{k8} = 4 and N_{k10} = 15

w/o Weld Specification	NUREG/CR-3394 Values
$P_{k=8"}$ = (4E-5)(4/19) = 0.84E-5	0.71E-5
$P_{k=10"}$ = (4E-5)(15/19) = 3.16E-5	3.29E-5
$P_{j=2}$ = 4.00E-5	4.00E-5

(3) for j = 3: k = 14" and 16", N_{k14} = 24 and N_{k16} = 5

w/o Weld Specification	NUREG/CR-3394 Values
$P_{k=14"}$ = (3E-5)(24/29) = 2.48E-5	2.47E-5
$P_{k=16"}$ = (3E-5)(5/29) = 0.52E-5	0.53E-5
$P_{j=3}$ = 3.00E-5	3.00E-5

(4) for j = 4: There was no piping in this diameter range for Salem 1

(5) for j = 5: k = 28", N_{k32} = 7, N_{k34} = 8, N_{k36} = 6

w/o Weld Specification	NUREG/CR-3394 Values
$P_{k=30"}$ = (3E-6)(19/40) = 1.42E-6	0.93E-6
$P_{k=32"}$ = (3E-6)(7/40) = 0.53E-6	0.88E-6
$P_{k=34"}$ = (3E-6)(8/40) = 0.60E-6	0.72E-6
$P_{k=36"}$ = (3E-6)(6/40) = 0.45E-6	0.47E-6
$P_{j=5}$ = 3.00E-6	3.00E-6

*Based on 238 welds/loop, as in Salem 1.

DEBRIS GENERATION

Estimating debris generation is a function of break size, jet expansion model, and the break versus target locations. For the analyses reported in NUREG/CR-3394, a hemispherical jet expansion region model was selected with the zones of influence that are shown in Figure 1, because the decompression pressure field for a high pressure, subcooled jet can be approximated with a hemispherical model. The energy levels (stagnation pressure level) within this expanding jet are also a function of distance from the break (or length to diameter, L/D). Calculational models that have been correlated with experiments show that at L/D = 3, the jet stagnation pressure is very nearly the same as the stagnation pressure within the jet emanating from the break location. For a PWR, this means pressures on the order of 2200 psi and extreme insulation destruction would take place, particularly for fibrous insulation. At L/D = 7, the PWR subcooled break jet stagnation pressure has been calculated to reduce to 20 to 40 psi. Although this is still a very high velocity field, experiments have shown (see NUREG/CR-3170) that the fiberglass covering shreds rather than totally destructing at these reduced pressures.

For analysis purposes, three L/D ratios of 3, 5, and 7 were selected to assess debris generation effects parametrically. The lower bound (L/D = 3) represents the highest jet intensity, because the expanding jet dynamic pressure at that distance is nearly that at the break jet exit plane and no conservatism exists for survival. The outer bound (L/D = 7) represents an axial distance where jet stagnation pressures have decreased to 20 to 40 psia, and at this distance the assumption that total destruction takes place does carry some conservatism. (See NUREG-0897, Revision 1 for further discussion on the selection of these L/D ratios to represent the different zones of insulation destruction.) In addition, blowdown tests in the HDR facility have demonstrated the highly destructive capabilities of LOCA jets. The weight of evidence is strongly against conceptualizing a high pressure pipe break as a simple water jet model. Such a break can be better termed as "an explosion."

These flow rates investigated parametrically represent emergency core cooling system flow rates given in Final Safety Analysis Reports (FSARs) submitted by applicants for operating licenses. The range of debris screen areas evaluated is representative of PWR sump designs. Because there is no standard sump design or ECCS flow requirement, this range was used to scope the range of sump blockage probabilities for PWRs; it is discussed in detail in NUREG/CR-3394.

The break location (that is, weld locations) versus target combinations for Salem 1 were systematically evaluated utilizing the plant insulation distribution. (The insulation was approximately half reflective metallic and half encapsulated fibrous insulation.) Table 4 shows the level of detail employed to evaluate all possible break-to-target combinations. The debris volumes associated with 14-inch pipe breaks are used for illustration, because these medium size breaks contribute significantly to the calculated overall sump blockage probability.

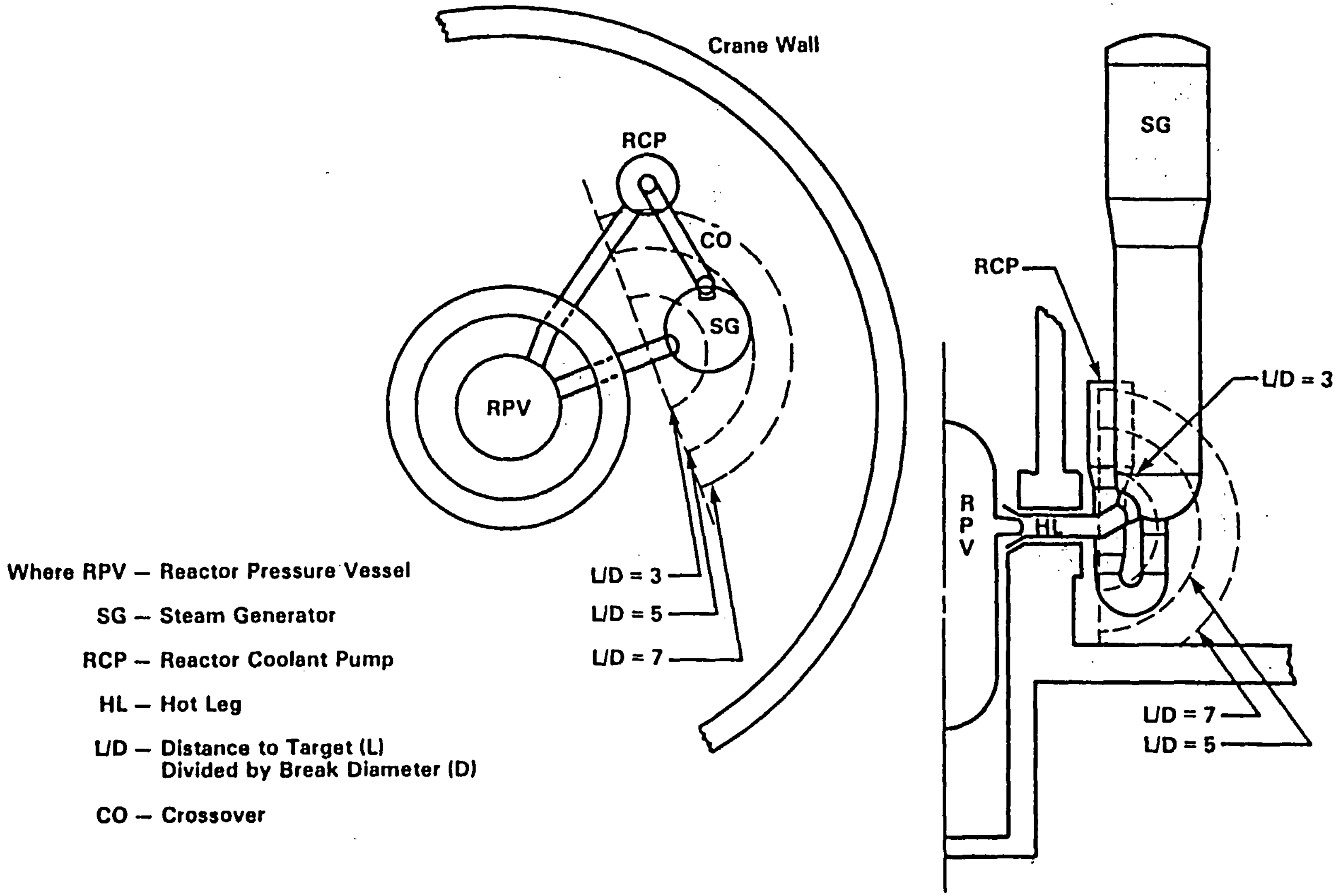


Figure 1 Zones of influence utilized for debris generation estimates in NUREG/CR-3394

TABLE 4

Illustration of break vs. target combinations and debris volumes calculated

BREAK NUMBER	TARGET DIAM. IN.	TARGET LENGTH FT.	INSUL. THICK. INCHES	INSUL. TYPE	DWG. REF.	INSUL. VOLUME FT **3	TARGET DIAM. IN.	TARGET LENGTH FT.	INSUL. THICK. INCHES	INSUL. TYPE	DWG. REF.	INSUL. VOLUME FT **3
96	14.00	1.50	3.00	SE	3	1.7	16.00	2.67	3.00	SE	3	3.3
	173.00	16.33	3.50	SE	3	220.1						
97	14.00	2.00	3.00	SE	3	2.2	16.00	2.67	3.00	SE	3	3.3
	173.00	16.33	3.50	SE	3	220.1						
98	14.00	3.60	3.00	SE	3	4.0	16.00	2.67	3.00	SE	3	3.3
	173.00	16.33	3.50	SE	3	220.1						
99	14.00	6.75	3.00	SE	3	7.5	16.00	8.50	3.00	SE	3	10.6
	173.00	18.67	3.50	SE	3	251.6						
100	14.00	10.75	3.00	SE	3	12.0	16.00	1.16	3.00	SE	3	1.4
	30.00	7.50	3.00	SE	20	18.2						
101	14.00	8.16	3.00	SE	3	9.1						
102	14.00	9.92	3.00	SE	3	11.0						
103	14.00	8.16	3.00	SE	3	9.1						
104	14.00	9.08	3.00	SE	3	10.1						
105	14.00	9.33	3.00	SE	3	10.4						
106	16.00	4.00	3.00	SE	3	5.0	14.00	7.33	3.00	SE	3	8.2
107	16.00	1.50	3.00	SE	3	1.9	173.00	18.67	3.50	SE	3	251.6
108	2.00	1.16	1.00	SE	33	0.1						
109	2.00	1.42	1.00	SE	33	0.1	1.00	2.33	1.00	SE	32	0.1

Chemical & Volume Control System

D-57

Source: Table B.5-3, NUREG/CR-3394 Volume 2

UNACCEPTABLE DEBRIS GENERATION

Unacceptable debris generation is defined as that volume of debris that, if transported to the sump screen, would result in a blockage head loss greater than NPSH requirements. Thus, the debris volume calculated through use of the break-to-target zones (or L/D ratios) shown in Figure 1 must be compared to the volume that could generate unacceptable screen blockage head losses that exceed the NPSH margins. This effect was analyzed parametrically and is reported in detail in NUREG/CR-3394.

Experiments have shown that the head loss for forced flow through fragmented fibrous insulations deposited on a debris screen can be expressed as an exponential relationship (NUREG/CR-2982, Rev. 1) and that such head losses are highly dependent on material type. The following empirical relationships for head loss (HL) have been obtained:*

$$HL = 1653\ (Q/A)^{1.84}\ (V/A)^{1.54} \text{ for high density fiberglass}$$

$$HL = 68.3\ (Q/A)^{1.79}\ (V/A)^{1.07} \text{ for NUKON}$$

$$HL = 123\ (Q/A)^{1.51}\ (V/A)^{1.36} \text{ for mineral wool}$$

In addition, these experiments have shown that shredded fibrous insulation materials distribute uniformly over debris screens. The data for NUKON were submitted by the Owens Corning Fiberglass Corporation during the For Comment period for USI A-43.

The analyses reported in NUREG/CR-3394 have parametrically investigated the following plant design and operational variables:

Variable	Range investigated
Pipe break-to-target distance	3 to 7 pipe diameters
Sump flow rate	6,000 to 10,000 gpm
Debris screen area	50 to 200 ft^2

*Q/A is the approach velocity as calculated from volumetric flow (ft^3/sec) divided by debris screen area (ft^2) and the equivalent debris thickness (V/A). V/A comes from the transported debris volume (V) divided by the debris screen area (A).

The analyses reported in NUREG/CR-3394 (as illustrated in Table 4) considered each singular break-target combination to determine debris generation volumes. Each possible combination was considered on a singular weld/target basis for determining if that particular break probability should be retained for estimating overall sump blockage probabilities. The criteria for maintaining a particular break were based on a calculated head loss of 1, 2, or 5 feet of water because these evaluations were performed parametrically. Table 5 shows the probability distribution as a function of pipe diameter for the following: sump screen area of 50 ft^2; recirculation flow rate of 8000 gpm; and an allowable head loss of 1 foot of water. The estimated sump blockage probabilities are 1.8E-5, 3.3E-5, and 4.5E-5/rx-yr for L/D = 3, 5, and 7.

The calculations reported in NUREG/CR-3394 considered insulation with debris associated both with piping and targeted compartments. They are admittedly conservative because shadowing effects attributable to piping supports and other structures are not included. To illustrate the sensitivity of such an assumption, a series of calculations have been performed to assess the significance of insulation only on failed piping, and as a function of diameter size, thereby ignoring other targets. These results are shown in Table 6. These estimated insulation volumes and associated head losses show that smaller diameter primary coolant system piping (6- to 10-inches), for the flow and debris screen area assumptions shown in Table 6, will not generate sufficient quantities of fibrous debris to result in head losses exceeding 1 to 2 feet of water, thus they support the conclusions that can be derived from Table 5. These findings would be most applicable to plants having small NPSH margins and small screen areas.

The analyses reported in NUREG/CR-3394 were run with the high density fiberglass head loss correlation, and it was assumed that the insulation (for L/D = 3, 5, and 7) was totally destroyed, was transported to, and was deposited on the sump screen. The assumption of total transport is an imbedded conservatism for plants that have recirculation flow velocities less than 0.2 ft/sec within the containment and for those plants where intervening structures would inhibit transport. On the other hand, there are PWRs where the primary coolant system piping and the sump location are not isolated by intervening structures.

SUMP FAILURE PROBABILITY

When the calculational methods described above were applied to the plant parameters investigated they produced the range of estimated sump failure probabilities shown in Table 7. (More detailed tables are provided in NUREG/CR-3394.)

For high flow rates (10,000 gpm), small debris screen area (50 ft^2), L/D = 7, and low allowed head loss (1 ft of water), the calculated sump failure probability was 5.4E-5/Rx-yr. For values of 6000 gpm, 200 ft^2, L/D = 3, and an allowed head loss of 2 ft. of water, a

TABLE 5
Pipe break and sump blockage probabilities vs. pipe diameter

TABLE 4.1.1- 26
SUMMARY OF PROBABILITIES DIAMETER BASIS WITH WELD TYPE WEIGHTING
SCREEN AREA= 50. FT SQ,FLOWRATE= 8000. GPM AND HEAD LOSS= 1. FEET OF H2O

DIA IN.	PO	PI	BLOCKAGE FREQ L OVER D = 3		BLOCKAGE FREQ L OVER D = 5		BLOCKAGE FREQ L OVER D = 7		BLOCKAGE PROB L OVER D = 3		BLOCKAGE PROB L OVER D = 5		BLOCKAGE PROB L OVER D = 7	
			PO BFW	PI BFW	PO BFW	PI BFW	PO BFW	PI BFW	PO BK	PI BK	PO BK	PI BK	PO BK	PI BK
2.	2.07E-04	7.08E-05	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
3.	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
3.	1.30E-05	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
4.	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
4.	4.10E-05	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
5.	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
6.	3.85E-05	3.85E-05	0.0	0.0	2.74E-01	2.74E-01	3.71E-01	3.71E-01	0.0	0.0	1.1E-05	1.1E-05	1.4E-05	1.4E-05
8.	7.11E-06	7.11E-06	0.0	0.0	0.0	0.0	2.50E-01	2.50E-01	0.0	0.0	0.0	0.0	1.8E-06	1.8E-06
10.	3.29E-05	3.29E-05	1.35E-01	1.35E-01	2.70E-01	2.70E-01	2.70E-01	2.70E-01	4.4E-06	4.4E-06	8.9E-06	8.9E-06	8.9E-06	8.9E-06
12.	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
14.	2.47E-05	2.30E-05	3.33E-01	2.86E-01	4.33E-01	3.93E-01	7.00E-01	6.79E-01	8.2E-06	6.6E-06	1.1E-05	9.0E-06	1.7E-05	1.6E-05
16.	5.34E-06	3.70E-06	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	5.3E-06	3.7E-06	5.3E-06	3.7E-06	5.3E-06	3.7E-06
18.	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
20.	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
22.	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
24.	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
26.	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
28.	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
30.	9.31E-07	0.0	1.00E+00	0.0	1.00E+00	0.0	1.00E+00	0.0	9.3E-07	0.0	9.3E-07	0.0	9.3E-07	0.0
32.	8.79E-07	6.21E-07	5.88E-01	4.17E-01	5.88E-01	4.17E-01	5.88E-01	4.17E-01	5.2E-07	3.6E-07	5.2E-07	3.6E-07	5.2E-07	3.6E-07
34.	7.24E-07	7.24E-07	5.00E-01	5.00E-01	5.00E-01	5.00E-01	5.00E-01	5.00E-01	3.6E-07	3.6E-07	3.6E-07	3.6E-07	3.6E-07	3.6E-07
36.	4.66E-07	4.66E-07	5.56E-01	5.56E-01	6.67E-01	6.67E-01	7.78E-01	7.78E-01	2.6E-07	2.6E-07	3.1E-07	3.1E-07	3.6E-07	3.6E-07
38.	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
40.	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
42.	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
44.	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
46.	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
48.	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
50.	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
52.	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
TOT	3.73E-04	1.78E-04	0.0	0.0	0.0	0.0	0.0	0.0	2.0E-05	1.6E-05	3.8E-05	3.3E-05	5.0E-05	4.5E-05

NOMENCLATURE

DIA -INITIATING EVENT DIAMETER-INCHES
PO -OVERALL PROBABILITY OF INIT. EVENT OCCURRENCE
PI -PROB.- INIT. EVENT-PRIM.-SYST-IN-CRANE WALL
BLOCKAGE FREQ-RATIO OF EVENTS CAUSING BLOCKS TO TOTAL EVENTS(WEIGHTED BASIS)
BLOCKAGE PROB-PROBABILITY OF UNACCEPTABLE ECCS SUMP BLOCKAGE
L OVER D - TARGET DISTANCE TO BREAK DIAMETER RATIO
BFW -BLOCKAGE FREQUENCY WELD TYPE WEIGHTING BASIS PO FOR OVERALL PROB AND PI FOR PRIM IN CW PROB
BFL -BLOCKAGE FREQ, SEGMENT LENGTH WEIGHTING BASIS PO FOR OVERALL PROB AND PI FOR PRIM IN CW PROB
BK -BLOCKAGE PROB PO FOR OVERALL PI FOR PRIM IN CW

4-37

Source: NUREG/CR-3394, which contains the complete calculated data sets

TABLE 6
Headloss as a function of pipe size, insulation type
and debris volume

PIPE DIAMETER INCHES	INSULATION THICKNESS INCHES	L/D	VOLUME DESTROYED Cu Ft	SCREEN AREA Sq Ft	RECIRCUL. FLOW Gpm	HEAD(1) LOSS Ft Water	HEAD(2) LOSS Ft Water
36.30	3.50	3.00	27.58	50.00	10000.00	149.45	8.50
36.30	3.50	5.00	45.97	50.00	10000.00	328.21	14.69
36.30	3.50	7.00	64.35	50.00	10000.00	551.04	21.06
34.00	3.50	3.00	24.34	50.00	10000.00	123.29	7.44
34.00	3.50	5.00	40.57	50.00	10000.00	270.74	12.85
34.00	3.50	7.00	56.79	50.00	10000.00	454.56	18.42
32.30	3.50	3.00	22.07	50.00	10000.00	106.07	6.70
32.30	3.50	5.00	36.79	50.00	10000.00	232.93	11.50
32.30	3.50	7.00	51.51	50.00	10000.00	391.08	16.59
16.00	3.00	3.00	4.97	50.00	10000.00	10.69	1.36
16.00	3.00	5.00	8.29	50.00	10000.00	23.47	2.35
16.00	3.00	7.00	11.61	50.00	10000.00	39.41	3.37
14.00	3.00	3.00	3.89	50.00	10000.00	7.33	1.05
14.00	3.00	5.00	6.49	50.00	10000.00	16.10	1.81
14.00	3.00	7.00	9.09	50.00	10000.00	27.04	2.59
10.00	3.00	3.00	2.13	50.00	10000.00	2.89	.55
10.00	3.00	5.00	3.55	50.00	10000.00	6.35	.95
10.00	3.00	7.00	4.96	50.00	10000.00	10.65	1.36
8.00	3.00	3.00	1.44	50.00	10000.00	1.58	.36
8.00	3.00	5.00	2.40	50.00	10000.00	3.48	.62
8.00	3.00	7.00	3.36	50.00	10000.00	5.84	.89
6.00	3.00	3.00	.88	50.00	10000.00	.75	.21
6.00	3.00	5.00	1.47	50.00	10000.00	1.64	.37
6.00	3.00	7.00	2.06	50.00	10000.00	2.75	.53
6.00	1.50	3.00	.37	50.00	10000.00	.19	.08
6.00	1.50	5.00	.61	50.00	10000.00	.43	.14
6.00	1.50	7.00	.86	50.00	10000.00	.72	.21
2.00	1.50	3.00	.06	50.00	10000.00	.01	.01
2.00	1.50	5.00	.10	50.00	10000.00	.02	.02
2.00	1.50	7.00	.13	50.00	10000.00	.04	.03

(1) High Density Fiberglass, H=1653((Q/A)^1.84*(V/A)^1.54

(2) Low Density Fiberglass, H=68.3((Q/A)^1.79*(V/A)^1.07

TABLE 7
Summary of the probability of sump failure

RECIRCULATION FLOW RATE (GPM)	SCREEN AREA (SQ FT)	ALLOWED HEAD LOSS (FT WATER)	EST'D FAILURE PROBABILITY (BASED ON WELD TYPES) L/D=3	L/D=5	L/D=7
6000	50	1.0	1.1e-5	1.7e-5	2.8e-5
6000	75	1.0	5.7e-6	1.2e-5	1.8e-5
6000	100	1.0	3.1e-6	6.7e-6	1.8e-5
6000	200	1.0	2.9e-6	5.9e-6	6.9e-6
6000	50	2.0	9.9e-6	1.2e-5	2.4e-5
6000	75	2.0	3.1e-6	9.2e-6	1.8e-5
6000	100	2.0	3.1e-6	5.9e-6	1.4e-5
6000	200	2.0	2.9e-6	5.8e-6	6.9e-6
6000	50	5.0	4.7e-6	1.2e-5	1.8e-5
6000	75	5.0	3.1e-6	5.9e-6	1.5e-5
6000	100	5.0	3.1e-6	5.9e-6	7.7e-6
6000	200	5.0	0	5.3e-6	6.9e-6
8000	50	1.0	1.6e-5	3.3e-5	4.5e-5
8000	75	1.0	9.1e-6	1.2e-5	2.4e-5
8000	100	1.0	3.9e-6	1.2e-5	1.8e-5
8000	200	1.0	2.9e-6	5.9e-6	7.7e-6
8000	50	2.0	1.1e-5	1.7e-5	2.7e-5
8000	75	2.0	4.7e-6	1.2e-5	1.8e-5
8000	100	2.0	3.1e-6	6.7e-6	1.6e-5
8000	200	2.0	2.9e-6	5.9e-6	6.9e-6
8000	50	5.0	7.4e-6	1.2e-5	1.9e-5
8000	75	5.0	3.1e-6	6.7e-6	1.7e-5
8000	100	5.0	3.1e-6	5.9e-6	1.3e-5
8000	200	5.0	2.9e-6	5.7e-6	6.9e-6
10000	50	1.0	1.6e-5	4.1e-5	5.4e-5
10000	75	1.0	9.9e-6	1.2e-5	2.4e-5
10000	100	1.0	4.8e-6	1.2e-5	1.8e-5
10000	200	1.0	3.1e-6	5.9e-6	9.4e-6
10000	50	2.0	1.1e-5	1.8e-5	4.3e-5
10000	75	2.0	7.4e-6	1.2e-5	1.8e-5
10000	100	2.0	3.1e-6	9.2e-6	1.8e-5
10000	200	2.0	2.9e-6	5.9e-6	6.9e-6
10000	50	5.0	9.9e-6	1.2e-5	2.4e-5
10000	75	5.0	3.1e-6	9.2e-6	1.8e-5
10000	100	5.0	3.1e-6	5.9e-6	1.4e-5
10000	200	5.0	2.9e-6	5.8e-6	6.9e-6

blockage probability of 2.9E-6/Rx-yr is calculated. Because insulation change out is an ongoing plant activity, the staff does not specifically know the types and quantities of insulation employed. Therefore, calculation of a generic value is not possible.

References

Bush, S., "Reliability of Piping in Light Water Reactors," International Atomic Energy Agency, IAEA-SM-218/11, October 1977.

U. S. Nuclear Regulatory Commission, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," July 1981.

---, NUREG/CR-2982, Revision 1, "Buoyancy, Transport, and Head Loss of Fibrous Reactor Insulation," D. N. Brocard, Alden Research Laboratory, July 1983 (also Sandia National Laboratory, SAND-82-7205).

---, NUREG/CR-3170, "The Susceptibility of Fibrous Insulation Pillows to Debris Formation Under Exposure to Energetic Jet Flows," W. W. Durgin, and J. Noreika, Alden Research Laboratory, January 1983 (also Sandia National Laboratory, SAND-83-7008).

---, NUREG/CR-3394, "Probabilistic Assessment of Recirculation Sump Blockage Due to Loss of Coolant Accident," J. J. Wysocki, Burns and Roe Inc., July 1983 (also Sandia National Laboratory, SAND-83-7116).

APPENDIX E

CONSEQUENCES OF LOSS OF RECIRCULATION CAPABILITY

CONSEQUENCES OF LOSS OF RECIRCULATION CAPABILITY

Risk analyses were utilized to assess the public health consequences associated with a loss of recirculation flow capability in the period after a loss-of-coolant accident (post-LOCA period) as a result of the effects of debris blockage. Blockage of a sump in pressurized water reactors (PWRs) and blockage of residual heat removal (RHR) suction inlets in a boiling water reactor (BWR) have similar consequences since loss of the redundant recirculation systems can result in core uncovery, which will lead to core melt. In addition, containment sprays will fail if a loss of suction takes place and containment overpressurization can occur. Loss of containment structural integrity leads to high public doses. To obtain initial estimates of public exposure, the staff referred to probabilistic risk assessments (PRAs) of five plants: Surry, Calvert Cliffs, and Crystal River-3, which are PWRs, and Peach Bottom and Grand Gulf, which are BWRs. The accident sequence of interest for the PWRs is designated AHF; a large break LOCA (A) followed by failure of recircula-tion to the core (H) and to the containment sprays (F). For the BWRs, analogous sequences designated AHI, AI, and AGHI were used as the basis for staff evaluations.

All five PRAs are based on the reactor safety study (RSS) methodology (NUREG-75/014). Consequently the conditional probabilities for various containment failure modes were approximately the same for all three PWRs. A synopsis of conditional probabilities of source terms is shown in Table 1. Table 1 shows that 70% to 80% of the AHF sequences lead to release category PWR-6 (basemat meltthrough), and 20% to 30% lead to PWR-2 (early overpressure failure). There are also small conditional probabilities of PWR-1 (steam explosions) and PWR-4 (failure to isolate containment).

TABLE 1
Conditional probabilities of release categories for large break LOCA with loss of ECCS recirculation, as estimated in plant-specific PRAs for the five reference plants, using RSS/RSSMAP methodology

	Plant release category probability			
Plant	PWR-1	PWR-2	PWR-4	PWR-6
Surry	0.01	0.207	---	0.78
Crystal River	0.01	0.2	0.007	0.8
Calvert Cliffs	0.01	0.3	0.007	0.7
	BWR-1	BWR-2	BWR-3	BWR-4
Peach Bottom	0.008	0.165	0.824	0.003
Grand Gulf	0.01	---	0.79	0.2

For the two BWR plants (using RSSMAP methodology), about 1% of the events are expected to result in BWR-1 releases (steam explosions), and about 80% would lead to BWR-3 (radiation release to the reactor building). For Peach Bottom, there is a 16.5% probability of BWR-2 (direct release to the environment) and less than 1% chance of BWR-4 (failure to isolate the drywell). For Grand Gulf, there is no chance of a BWR-2, but a 20% probability of containment isolation failure reported in the RSSMAP study.

The offsite doses associated with each release category were evaluated with the CRAC code, under the following assumptions: a uniform population distribution of 340 people per square mile, meteorological conditions typical of the Byron site, no evacuation, and integration of conditional consequences to a 50-mile radius from the plant. The resulting estimates of public exposure are shown in Table 2. It should be noted that the conditional consequences are not monotonically decreasing with release category, as would be expected considering the release category definitions. This is most likely a result of neglecting the dose beyond 50 miles, the higher release energy for BWR-1 deposits of fission products at greater distances.

TABLE 2
Calculated conditional public radiation exposures from each of the release categories for the A-43 reference site conditions using RSS methodology

Release category	Calculated consequences (person-rem)	Release category	Calculated consequences (person-rem)
PWR-1	5.4E+6	BWR-1	5.4E+6
PWR-2	4.8E+6	BWR-2	7.1E+6
PWR-3	5.4E+6	BWR-3	5.1E+6
PWR-4	2.7E+6	BWR-4	6.1E+5
PWR-5	1.0E+6		
PWR-6	1.5E+5		
PWR-7	2.3E+4		

The conditional consequences of a core melt due to sump failure are obtained by weighting the calculated public exposure for each release category (Table 2) with its associated conditional probability (Table 1), and then summing over release categories. The resulting conditional consequences for the five reactors are listed in Table 3.

The risk, in person-rem per reactor year, is the product of core melt frequency and conditional consequences.

TABLE 3
Preliminary estimate of conditional consequences for loss of recirculation (based on WASH-1400 methods of assessing severe accident risks, using RSS/RSSMAP methodology)

Plant	Conditional consequences (person-rem)
Surry	1.2E+6
Crystal River	1.1E+6
Calvert Cliffs	1.6E+6
Peach Bottom	5.4E+6
Grand Gulf	4.2E+6

Since the publication of the WASH-1400 study in 1975, a great deal of progress has been made in two areas relating to the calculation of severe accident consequences: (1) containment response characteristics and (2) radiological release fractions. When the preliminary version of the A-43 resolution package was presented to the Committee for Review of Generic Requirements (CRGR), members of the Committee inquired as to whether recent research results related to containment response would substantively alter the consequence evaluation presented above. The staff has performed such an assessment, and has examined the possible impact of the revised methodology for estimating radiological source terms.

Reassessment of Containment Response

The containment response assumed in estimating the consequences of sump failure in the reference PWRs may be characterized as follows: from the RSS/RSSMAP studies there are (1) about a 20% conditional probability of a serious radiological release to the environment (PWR-2) and (2) about an 80% probability of a benign release (PWR-6). If the estimates of societal risk are to be significantly reduced from those given above, the (conditional) probability of a serious release would have to be significantly lower than 20%. To state with confidence that the release rate is that low would require a great deal of confidence in the systems that prevent containment failure and reduce the resulting source term.

There is one containment type which meets this criterion, the PWR large dry containment with safety-grade fan coolers (a large majority of large dry containments safety-grade have fan coolers). The staff has a great deal of confidence that overpressure failure of these containments can be prevented either by operation of the fan coolers or by connecting the sprays to an alternative source of water. The probability that the fans will fail and that spray hookup to an alternative water source will not be made is judged to the small. Large dry containments are not susceptible to abrupt failure due to hydrogen burns and steam spikes, and the probability of steam explosions or failure to isolate containment is small.

For subatmospheric containments, prevention of eventual overpressure failure depends on connecting the sprays to an alternative source of water. Subatmospheric containments do not have adequately sized heating, ventilation, and air conditioning (HVAC) systems to meet the energy addition to containment associated with a large-break LOCA. Noncondensible gas production due to core-concrete reactions could overpressurize containment in a day or more. Hydrogen burns associated with invessel hydrogen production are not expected to cause the containment to fail, but late hydrogen burns with the containment at an elevated partial pressure of steam and/or noncondensible gases may threaten containment integrity. However, if inplant capability to detect flow degradation exists and corrective operational procedures have been developed, there is time to prevent eventual overpressure failure by operator action.

In an ice condenser containment, powered hydrogen ignitors would protect against a deflagration threat. However, because there would be water in the reactor cavity, containment failure due to steam production would occur at the time of vessel failure or within a few hours thereafter, depending on the amount of ice remaining at the time of core melt. The assumption is made in this analysis that refueling plugs have been removed in accordance with Technical Specifications and hence a wet cavity would exist at the time of vessel failure.

The staff's base case assessment of BWR consequences is equivalent to a 100% probability of severe radiological releases for Peach Bottom and an 80% probability for Grand Gulf. Current understanding of containment response indicates that lower probabilities are possible.

The Browns Ferry BWR Mark I containment has a specific design feature that allows the operator in the control room to connect the spray suction line to the condensate storage tank. However, it would have to be verified that all other Mark I containments have the same fature before generic credit for a radiological release reduction if this feature is taken. It would also have to be verified that procedures are in place to ensure that in the event of failure of recirculation sprays, the proper realignments and the necessary interlocks overridden within the available time.

BWR Mark II and III designs do not have the option of taking spray suction from the condensate storage tank. It is possible to align spray suction to the fuel storage pool, but the staff is not aware that the necessary procedures are in place for such plants. Furthermore, a source of makeup water to the fuel storage pool would have to be found.

The physical plant layout for Mark III design has the advantage that, for the most likely mode of failure, the fission products will bubble through a subcooled suppression pool, yielding a radionuclide release significantly lower than BWR-1. The resulting consequences (in terms of person-rem) would be reduced substantially. In addition, no credit is being given for scrubbing effects beyond that assumed in prior BWR analyses. A similar mechanism for release reduction for Mark I and II containments would result if the operator relieves pressure by venting the containment through existing wetwell penetrations to the atmosphere.

Current emergency procedure guidelines (EPGs) for BWRs describe containment venting. Forthcoming BWR owners group EPGs will discuss wetwell venting in greater depth, and it appears that controlled venting will be adopted by all BWR owners.

For each of the containment types, however, there are credible mechanisms that would allow fission products to bypass the suppression pool under certain circumstances. In the Mark I, drywell failure can occur either at the time of core melt as a result of overpressurization, or several hours later when penetration seals fail as a result of high temperatures. In some Mark II designs the molten core would be retained on the diaphram floor and containment overpressure failure would occur in the drywell. Hence a direct path for some radionuclides could occur. In the Mark III design, direct leakage from the drywell to the wetwell can result either from existing measured leakage paths or from failure to isolate the drywell.

Reassessment of Radiological Releases

A wealth of experimental and theoretical research on radiological releases in severe accidents has been performed in the past decade. The NRC Office of Research (RES) has developed a revised source term methodology based on that research and has documented the results of sample accident sequences for several reference plants (BMI-2104). A committee of the American Physical Society has reviewed the methods and results, and committee members have concluded, among other things, that there is considerable uncertainty in some aspects of the methodology. They further stated that the radiological release estimates for some sequences could be higher than predicted by WASH-1400 (NUREG-75/014).

When the new results and their associated uncertainties are integrated into the analysis of specific sequences for specific plants, the calculated releases are generally lower than previously predicted, but the uncertainties are large. For the purpose of this assessment, the staff confines its consideration to those cases in which the new source term methods unambiguously predict lower releases of radionuclides.

For sequences in which containment failure is predicted to occur long after the core has melted, the new methods predict considerable decreases in the suspended aerosol concentrations in containment, because of enhanced agglomeration and gravitional settling. During the early and intermediate time period (less than 10 hours) after core melt, there are competing mechanisms that would tend to somewhat offset this reduction: most notably, enhanced releases of refractory fission products during core-concrete interactions in some reactor cavity designs, and the possible revaporization of fission products deposited in the primary system. The latter mechanism is not a substantive consideration for A-43, because there is very little primary system retention of fission products in large-break LOCA sequences.

For large dry containments with fan coolers, containment failure would occur more than a day after core melt, if at all. The revised source term methodology would predict significantly lower radiological releases and offsite consequences than the staff assumed in this base case and prior A-43 analyses.

Other large dry containments and subatmospherics fall into two categories: those in which the reactor cavity would be full of water following sump failure, and those in which it would be dry. In the wet cavity case, the core-concrete interaction would be suppressed, and the source term would be greatly reduced when containment fails as a result of steam overpressurization (estimated to occur at about 12 hours after core melt). For dry cavity designs, enhanced production of refractory metals during core-concrete interaction could occur, but the predicted containment failure time--if the containment fails--is much later because of the absence of steam production. Consequently, the staff also expects a greatly reduced source term for dry cavity cases.

Because the enhanced aerosol removal does not affect noble gases, organic iodine, or gaseous fission products, the staff would limit the predicted reduction in offsite person-rem within 50 miles to about a factor of 10 for large dry and subatmospheric PWR containments.

Because containment failure in an ice condenser is estimated to occur early after core melt, the staff cannot be confident that the radiological releases are any lower than predicted in the base case.

For BWRs, the containment failure times are generally much less than 10 hours, and the enhanced aerosol settling may not be as significant as estimated for PWRs. Furthermore, for early containment failure, the releases of refractory metals can be significantly higher than previously assumed in the Mark I containment. Consequently, the staff does not expect significant reductions in the predicted fission product releases for recirculation failure in BWRs as a result of the revised NRC source term methods.

Summary

On the basis of its reassessment of containment performance and radiological releases, the staff has developed revised estimates of the offsite consequences for each reactor type. Because of the very approximate nature of the review, the results are quoted with only order-of-magnitude accuracy. That is, the assessment determines whether the average consequences are about the same as a severe release (BWR-3 or PWR-3), an order of magnitude less (x 0.1), or two orders of magnitude less (x 0.01). The results for PWRs are shown in Table 4 and for BWRs in Table 5.

TABLE 4

Approximate consequence reduction factors* for PWR containments based on reassessment of containment response and source terms+

Containment type	Without spray recovery	With spray recovery
Large dry designs with fan coolers	----	0.01
Large dry designs without fan coolers and subatmospherics	0.1	0.01
Ice condensers	1	----

* Because of the approximate nature of the revised consequence estimates, they are quoted with only order-of-magnitude accuracy.

\+ Revised consequence values can be obtained by multiplying the reduction factors with the consequence estimates associated with PWR-3 (5.4×10^6 person-rem).

TABLE 5

Approximate consequence reduction factors* for BWR containments, based on reassessment of containment response and source terms+

Containment	Without wetwell venting	With wetwell venting
Mark I	1	0.1
Mark II	1	0.1
Mark III	0.1	0.1

* Because of the approximate nature of the revised consequence estimates, they are quoted with only order-of-magnitude accuracy.

\+ Revised consequence values can be obtained by multiplying the reduction factors with the consequence estimates associated with BWR-3 (5.1×10^6 person-rem)

Large dry containments with safety-grade fan coolers are not estimated to fail. Furthermore, in the event of a failure, radionuclide releases would be greatly reduced because of the long time to failure.

For other large dry and subatmospheric containments, failure could occur within a day after core melt, but the reduction in source terms due to the enhanced gravitational settling would lead to an order of magnitude (x 0.1) reduction in consequences. A further order-of-magnitude reduction (x 0.01) would result if containment failure were prevented by connecting the sprays to an alternativ water source.

Ice condensers are expected to fail at the time of vessel failure, or a few hours thereafter, depending on the amount of ice remaining at the time of vesse failure. While source terms could be lower than predicted in WASH-1400, this circumstance would not, lead to an order-of-magnitude reduction in the predicted person-rem. Furthermore, recovery of spray operation before containment failure could not be ensured, because containment failure may be rapid. Consequently, the staff concludes that the consequences of sump failure in this type of plant could be more severe than the estimates for other PWRs.

BWR Mark I and Mark II containments are expected to fail within a few hours after core melt, and the radionuclide releases are expected to be on the same order as for BWR-3 (Table 3). Failure could be later and releases could be lower for some Mark II plants, but because of the variability of the Mark II designs and the uncertainty of the phenomenology, the staff cannot draw this general conclusion. In both the Mark I and Mark II, successful venting of the wetwell prior to containment failure could lead to substantial source term retention in the suppression pool and a dramatic reduction in conditional consequences. A similar reduction would result if the drywell sprays could be connected to an alternative source of water.

The staff has limited its consequence reduction factor to one order-of-magnitud (x 0.1) because there is some possibility that the suppression pool may be bypassed because of early containment failure or leakage from the drywell.

The Mark III design fails in such a way that the fission products are channeled through the pool before release to the environment, with or without wetwell venting. As with the Mark I and Mark II plants, the reduction factor is limited to 0.1 because of the possibility of pool bypass.

The revised estimates of offsite consequences are shown in Table 6. They are obtained by multiplying the reduction factors in Table 4 for PWRs and Table 5 for BWRs by the consequences associated with the PWR-3 and BWR-3 releases, respectively.

TABLE 6
Revised estimates of offsite consequences based on current understanding of containment response and radiological source terms

PWR containment type	Conditional consequences (person-rem) No spray recovery	With spray recovery
Large dry (safety-grade fan coolers)	---	5×10^4
Other large dry and subatmospheric	5×10^5	5×10^4
Ice condenser	5×10^6	---

BWR containment type	Conditional Consequences (person-rem) No venting or spray recovery	With venting or spray recovery
Mark I	5×10^6	5×10^5
Mark II	5×10^6	5×10^5
Mark III	5×10^5	5×10^5

Estimation of Offsite Releases

Utilization of the estimated consequences for PWR ice condenser plants shown in Table 6, without consideration of plant-specific sump design features and recirculation flow requirements, could lead to the conclusion that consequences associated with a failed sump are 10% higher. Table 7 provides an overview of ice condenser plants; the significant factors are:

(1) All ice condenser plants utilize reflective metallic insulation on the primary coolant piping and major components (steam generators, pressurizer, reactor coolant pumps, etc.); thus debris blockage concerns associated with transport of fibrous insulation debris are not present (at least not in significant amounts).

(2) The majority of these plants have lower recirculation flow rates and larger debris screen areas. The net effect is that approach velocities are less than 0.2 ft/sec; therefore debris transport is not likely to occur.

(3) Net positive suction head (NPSH) margins are high, significantly higher for most plants than the 1 to 5 feet of available NPSH margins utilized in the sump failure probabilities discussed in Appendix D.

These factors all lead to the conclusion that sump failure probabilities developed for PWRs with a mix of fibrous and reflective metallic insulation and low NPSH margins should be reduced for applications to ice condenser plants. Interpolation within the values shown in Table 7 of Appendix D can be used to derive a sump failure probability (for ice condensers) in the range of 3 to 9 x 10^{-6}/Rx-yr. This value was used for estimating the consequences discussed below.

TABLE 7
Overview of ice condenser plant design an operational features

Plant	RHR flow (gpm)	Debris screen (Ft^2)	NPSH margin (ft water)	Insulation used	Approach velocity (ft/sec)
Catawba 1&2	6000	135	7.9 (RHR)*	Reflective Metallic	0.10
D.C. Cook 1&2	6000	90	21.9 (RHR)	Reflective Metallic	0.15
McGuire 1&2	6800	120	6.9 (CSS)* 16.9 (RHR)	Reflective Metallic	0.13
Sequoyah 1&2	9500	43	2.8 (RHR)	Reflective Metallic	0.49
Watts Bar 1&2	8000 8000	43 164**	11.5 (RHR)	Reflective Metallic	0.41 0.11

* RHR: residual heat removal; CSS: containment spray system.
**Trash rack area; this structure would intercept large size insulation debris before transport to the sump debris screen structure could occur.

Utilization of the consequence values in Table 6 and estimated blockage probabilities (from Appendix D) can be used to estimate averted risks. Table 8 contains such estimates and the blockage probabilities that were developed. These values were used to calculate the value-impact ratios discussed in Section 4.4 of the main body of NUREG-0869, Revision 1.

TABLE 8
Overview of consequences associated with sump blockage

Containment type	Estimated conditional consequences (person-rem)	Estimated blockage probability (1/Rx-yr)	Assumed core melt conditional probability*	Estimated risk averted** (ΔR person-rem/Rx)
PWR dry w/SGFCs+	--	3 to 50×10^{-6}	0.5	--
PWR ice condenser	5×10^{6}	1 to 9×10^{-6}	0.5	40 to 560
PWR dry w/o SGFCs and subatmospheric	5×10^{5}	3 to 50×10^{-6}	0.5	19 to 313
PWR dry w/o SGFCs and subatmospheric, w/spray recovery	5×10^{4}	3 to 50×10^{-6}	0.5	2 to 31
Mark I and II	5×10^{6}	4 to 20×10^{-6}	0.5	250 to 1250
Mark III	5×10^{5}	4 to 20×10^{-6}	0.5	25 to 125
Mark I and II w/venting or spray recovery	5×10^{5}	4 to 20×10^{-6}	0.5	25 to 125

* The assumption is made that 50% of the time that blockage occurs, core melt would occur. This assignment of a conditional core melt probability is realistic in view of potential operator dection and mitigating actions which could be taken.

**An outstanding reactor life span of 25 years has been assumed.
+ SGFC: safety-grade fan cooler

References

U.S. Nuclear Regulatory Commission, NUREG-75/014, "Reactor Safety Study," 1975 (formerly WASH-1400).

BMI-2104, "Radionuclide Release Under Specific LWR Accident Conditions", by Gieseke, Cybulskis, Denning, Kuhlman and Lee, Battelle Columbus Laboratories, Columbus, Ohio, July 1983.

APPENDIX F

CONTAINMENT SURVIVABILITY

CONTAINMENT SURVIVABILITY

The several different containment design concepts currently in use for the many operating plants can be grouped as follows:

(1) The dry containment structures for pressurized water reactors (PWRs) must absorb all loads from accident conditions. The results are characterized by large containment volumes (i.e. 2×10^6 ft^3) and high design pressures (i.e. 60 psig) with considerable margin beyond the design point.

(2) Containment structures which incorporate a means of passive steam condensation have taken advantage of this approach to design smaller and lower pressure containment buildings. Boiling water reactor (BWR) containments utilize a water pool for condensing steam and PWR ice condenser plants utilize an annular ice bed for condensing steam.

Containment Structural Capabilities

Containment failure modes and attendant releases were analyzed in WASH-1400 and related to a major loss of containment capability through: (1) steam explosion induced failures (α mode); (2) hydrogen burn induced failures (β mode); (3) overpressurization of the containment building resulting from steam generation (molten core interacting with water) and noncondensible gases (molten core interacting with basemat) (γ mode); and (4) basemat penetration (ε mode). These WASH-1400 studies also assumed a loss of containment without assessing the significance of any design margin available in the different containment structures as currently designed.

Risk assessments performed in recent years indicate that risk from nuclear power plants is dominated by severe core melt accidents. Typical containment loading pressures and temperatures associated with whole core melt scenarios are on the order of approximately 100 psia and approximately 300°F. More recently, mechanistic models for containment failure (see NUREG/CR-3653) have also been included in these assessments of severe accident scenarios.

These severe accident and containment structural capability studies provide the following insights:

(1) Because of structural design margins, containments have inherent capabilities beyond their design basis. This provides a capability to contain or mitigate a wide spectrum of severe accidents.

(2) Best estimate analyses of containment performance indicate that containments can retain structural integrity at pressures as high as 2.2 to 2.5 times the design pressure. "Extensive yielding" is the term used in NUREG/CR-3653 to describe loss-of-structural integrity for these best estimate analyses.

(3) Although it is possible that leaks through penetrations could occur before loss of structural integrity, the risks from such leaks would be considerably smaller than from the gross containment failures assumed in WASH-1400 studies.

Containment Heat Removal Systems

Makeup water flow is needed to absorb the energy released into the containment atmosphere or to the pools as a result of the long-term decay heat and to keep the core covered. Although this flow is small compared to the design-basis accident flow, if recirculation flow to the vessel is reduced, boil-off will occur. The steam released into the containment will raise the pressure and temperature. This energy must be removed from the containment so the containment pressure is maintained below its failure pressure.

The containment heat removal systems (CHRS) are sized to remove the energy released into the containment due to decay heat. Depending on the containment type, any one or combination of the following systems might be used as the CHRS: suppression pool cooling systems, containment atmosphere spray systems, and air cooling systems.

BWR plants (Mark I, II and III containments) consider the pressure suppression pool as the short-term heat sink for both the blowdown energy and the decay heat. In the long term, the pool cooling system is the most important CHRS in limiting the maximum pool temperature (and pressure for Mark III). Reducing the flow to the suppression pool water cooling system because of debris breakage would result in increasing the pool temperature. If the reduction of the flow rate is small (some few percent), the temperature increase would be small because of heat exchanger capacity margins. However, a major reduction (a factor of 5 to 10) in flow rate would very soon lead to a temperature increase in the pool beyond desigr limits. This increase would result because of the strong connection between mass flow rate through the heat exchanger and overall heat transfer coefficient of the heat exchanger, together with the reduced flow rate.

Containment atmosphere spray systems are used in all types of containments for both fission product and energy removal from the atmosphere in accident situation: The relative importance of the spray system is dependent on the containment type. In ice condenser containments the spray system is the only means to remove the decay heat released as steam into the containment atmosphere after the ice beds have melted. A reduction of the containment spray flow rate would immediately lead to higher containment pressure and temperature in addition to a reduction in the atmospheric fission product removal effectiveness. This change would result because of several factors: (1) smaller total amount of water flow, (2) bigger droplet sizes because of smaller pressure drop over the spray nozzles, (3) shorte stay time in the containment atmosphere, (4) smaller total heat transfer area of the droplets, (5) loss of thermodynamic equilibrium between containment atmosphere and droplets, and (6) smaller coverage by spray in the containment. It is expected that a reduction of containment spray flow rate would cause a significant reduction of spray cooling efficiency.

For Mark I and II designs, the spray systems are not required to mitigate the course of the transient. Mark III containments do however rely on sprays, primarily for the short-term mitigation. As a result, loss of spray system effects can be considered to be of secondary importance for BWR designs. Dry containments usually employ both sprays and fan coolers. Each system is sized for 100% capacity. Therefore, the coolers could perform the same heat removal function without the sprays being operational.

Safety-grade air coolers, typical for dry containment designs, are not affected by a reduction of residual heat removal (RHR) system flow rate and therefore would not be affected by loss of this system. However, the availability of containment air coolers, which are not safety grade and are not designed to operate during post-LOCA accident situations, cannot be independently relied upon. Because of the elevated pressure and steam-air mixture density in the containment, such fans would be expected to trip because of overload. Another factor would be, of course, the very questionable survivability of the related electrical components not qualified for the post-accident environment.

Expected Containment Design Response Due to Loss of Recirculation Water Sources

The influence of reduced RHR-system flow rate upon the overall containment behavior is a plant-specific matter, depending on the containment design (Mark I, II, III, ice condenser, or dry containment).

Mark I and II containments are both small and quite similar in their pressure responses following an accident. The peak pressure is reached very early, and after one hour the pressure is already low compared to the design pressure. However, the pool temperature is high and still increasing, and it would probably be the first limiting design parameter associated with reducing the RHR system flow rate that would be exceeded. This elevated water temperature to the RHR pumps could jeopardize their operation. Thus, a total loss of the RHR system, or a major reduction of the flow rate, would result in a slow overpressurization and potential failure of the containment. This failure could be expected in less than 12 hours, if the containment is not vented as recommended by the emergency procedure guidelines (EPG). Based on current engineering judgment, a rupture of the containment can be expected to occur when pressure reaches 2 to 3 times the design pressure.

In Mark III containment, the pressure and pool temperature are still high and increasing after 1 hour. A major reduction of RHR system flow could mean that both the containment pressure and pool temperature would exceed the design values and would result in the same consequences discussed above for Mark I and II plants. Without venting, containment failure could be expected during the first 24 hours following loss of recirculation flow capability.

The ice of an ice condenser containment will melt in about 1 to 2 hours. After the ice is melted, the containment spray system is needed to keep the pressure below the design limit. A reduction in the RHR system flow rate could also cause the containment design pressure to be exceeded in this type plant. Reduction of the RHR system flow rate would occur in the same time period when the ice would be totally melted and would result in rapid overpressurization and early containment failure. This could be expected in about 3 hours.

The peak pressure in a dry containment (normal atmospheric or subatmospheric) is reached early in the transient, and, after 1 hour, the pressure is expected to be very low. The dry containments also have safety-grade air coolers, which could maintain containment pressure and temperature below the design values and, therefore, a reduction of RHR system flow will not adversely affect the dry containment. Thus, containment failure for large dry containments with safety-grade fan coolers (SGFCs) would not be expected.

Severe Accident Study Insights

For large dry PWR containments with safety grade fan coolers, the staff can conclude that the containment response assumed previously in the A-43 consequence analysis is conservative because the probability of containment failure as a result of overpressure has been overestimated. Reaching a similar conclusion for BWR Mark I containments would be contingent on the licensee or applicant demonstrating that there is a high likelihood of the operator successfully exercising the option to align the containment sprays to take suction from the condensate storage tank.

The staff is not aware of a generic design option for PWR subatmospheric and ice condenser designs and BWR Mark II and Mark III plants that allows the operator to readily switch spray suction to an alternative water source. Although alternate water sources exist, and in some cases the piping appears to be in place, it remains to be demonstrated that the proper connection can be made, valves aligned, and safety system interlocks overridden within a reasonable time under accident conditions. Without the sprays, the probability of overpressure (or overtemperature) failure for these containments could be higher than previously assumed. The PWR subatmospheric containment design has an advantage over the ice condenser, Mark II and Mark III containments because its predicted failure time is much later than the failure time for the others. This additional time allows more time for operators to recover the sprays.

The BWR Mark III containment stands out because its principal failure mode leads to significant scrubbing of fission products in the suppression pool. The reduction in conditional consequences that would result is offset somewhat by the staff's conclusion that the probability of early failure for Mark III plants is higher than prior estimates. A similar mechanism for source term reductions for Mark I and Mark II containments would result if the operator can successfully relieve pressure by opening a wetwell vent. Without a plant-specific analysis, the staff cannot say with confidence that the containment response estimates developed for resolution of USI A-43 are conservative for BWR containments (see also Gieske, 1984).

Conclusions

In summary, the following situations appear to exist:

(1) PWR dry containments (with safety-grade fan coolers) are likely to survive a core melt situation, even with a loss of the containment emergency sump. Even without safety-grade fan coolers, large dry containments will not overpressurize for 6 to 12 hours following a LOCA, leaving time for the operator to take corrective action to provide an alternate water source for containment spray (provided a loss-of-sump condition is detected and acted upon). The availability of safety-grade fan coolers to minimize containment pressure, thereby enhancing containment survival, is a significant factor.

(2) Mark I containments have a high containment overpressure survivability if alternate containment spray suction can be provided should loss of RHR suction occur. However, Mark I containments are most susceptible to containment failures that lead to release consequences comparable to WASH-1400 (NUREG-75/014) releases.

(3) Controlled venting of any of the BWR containment designs will preserve structural integrity for all BWR containment designs, if such venting is implemented correctly.

(4) Subatmospheric PWR containments (without safety grade fan coolers) and PWR ice condenser plants are most susceptible to loss of integrity if the containment emergency sump is lost and core sprays cannot be recovered through use of alternate water sources.

References

Gieske, J. A., et al., "Radionuclide Release Under Specific LWR Accident Conditions," BM-2104 (Volumes 1-7) Battelle Memorial Institute, July 1984.

U.S. Nuclear Regulatory Commission, NUREG-75/014, "Reactor Safety Study," (originally published in October 1975 as U.S. Atomic Energy Commission report WASH-1400.)

....., NUREG/CR-3653, "Final Report, Containment Analysis Techniques, a State-of-the-Art Summary," March 1984.

APPENDIX G

ESTIMATION OF COSTS TO REPLACE INSULATION

ESTIMATION OF COSTS TO REPLACE INSULATION

The monetary costs of insulation replacement are dependent on plant design, material, and preplanning efforts. Table 1 shows the insulation replacement costs and estimated exposures that were used in the For Comment version of the USI A-43 value-impact analysis (NUREG-0869), which was published in April 1983. Cost information received from the industry (Atomic Industrial Forum) and insulation vendors (Owens-Corning Fiberglas and Diamond Power Company) is summarized in Table 2. This information was used to develop the reinsulation costs shown in Table 3. The variability in estimates from industry sources is large and further illustrates the variances in plants and installation experience. The composite average estimates of $153/ft^2 (Table 3) for insulation replacement can be compared to the earlier USI A-43 estimate of $75/ft^2. Table 3 also shows an estimated insulation replacement cost range of $92/ft^2 to $244/ft^2.

The most severe monetary impact would result from a decision to replace all of the problem insulation in a given plant. The PWR sump failure probability study showed (NUREG/CR-3394) that insulation on the primary system piping and the lower one-third of the steam generator are the principal sources of debris that leads to unacceptable sump blockage conditions. Appendix D of this report discusses the significance of insulation on the primary system piping ($\geq$ 10-inch diameter or greater), steam generators, reactor coolant pumps, and pressurizer that could lead to unacceptable debris screen blockage. Table 4 illustrates the variability of fibrous debris as a function of pipe size and was derived from the Salem plant analysis (NUREG/CR-3394). Plant insulation variability (as installed) is illustrated for the Salem 1 and Maine Yankee plants as shown in Table 5. Approximately 2400 ft^3 (covering an area of 8200 ft^2) of potentially troublesome insulations are identified. On the basis of these two illustrative plants, it is estimated that only 4400 ft^2 to 5235 ft^2 of insulation might have to be replaced rather than replacing all plant insulation. In the cost impacts developed in Section 2 of this report, these amounts are used as the upper bounds of the amount of insulation that would have to be replaced for determining insulation replacement costs for a typical PWR.

TABLE 1
Estimates of costs to replace insulation and associated exposures illustrating plant variability

Plant	Unencapsulated insulation (ft^2)	Cost est* 1 ($\$ \times 10^3$)	Cost est** 2 ($\$ \times 10^3$)	Cost est+ 3 ($\$ \times 10^3$)	Estimated++ exposure (person-rem)
Salem 1	13,200	281	238	660	99
Maine Yankee	6,700	142	121	335	47
Ginna	1,000	21	18	50	8
Millstone 2	1,300	28	23	65	10

*These costs are derived from Surry 1 and 2 steam generator removal and reinstallation data, and discussions with onsite staff. A per-unit rate of 0.85 hr/ft^2 for replacing insulation was derived, and labor costs of $25.00/hr were assumed.

**Telephone estimates from New England Insulation Company (Maine Yankee has employed this firm) were: $3/$ft^2$ to remove insulation, $11/$ft^2$ to fabricate new panels, and $3 to $5/$ft^2$ to install the new panel.

+Telephone estimates of $35 to $50/$ft^2$ for MIRROR™ insulation fabrication and installation were obtained from Diamond Power, which supplies such insulation. The value of $50/$ft^2$ was used.

++Exposure data were derived from Surry 1 and 2 data. Discussions with the Surry site staff indicate that a 50-person-rem exposure level for insulation replacement is realistic if the job is adequately preplanned. An equivalent dose of 7×10^{-3} person-rem/ft^2 of insulation to be replaced can be derived.

TABLE 2
Insulation fabrication and installation cost estimates received during the A-43 (NUREG-0897) for comment period

Commentor	Comment/Estimate
Atomic Industrial Forum	Productivity: Industry experience shows an average rate of 2.0 to 2.5 hr/ft^2 for installation. Labor: Industry labor costs are \$30 to \$45/hr. Total: The averaged cost of \$550,000 per plant is off by at least a factor of 2; material costs are not included.
Diamond Power Company (manufacturer of MIRROR™ insulation)	MIRROR™: The cost of material supplied at Cooper was approximately \$40/ft^2. Productivity: Installation rates averaged 1.24 hr/ft^2 Labor: The estimate of \$25/hr used previously seems reasonable.
Owens Corning Fiberglass Company (manufacturer of NUKON™ insulation)	NUKON™: Material cost is estimated to be \$90/ft^2. Reflective: An assumption of \$100/ft^2 would be very reasonable. Labor: Current labor costs of \$40 to \$50/hr are common. Productivity: Installation rates of 7½ to 12 ft^2/hr are typical. Total: This leads to a cost of \$112/ft^2 for labor and \$100/ft^2 for material, for a total cost of \$212/ft^2 to remove existing insulation and replace it with reflective insulation.

TABLE 3
Insulation replacement cost estimates developed from public comments received

Commentor	Cost: Estimate
Atomic Industrial Forum	Reinsulation labor costs: \$60 to \$112/ft^2
	Insulation fabrication cost: \$75/ft^2
	Combined total: \$135 to \$185/ft^2
	Average combined total: \$160/ft^2
Diamond Power Company	Installation labor costs: \$31/ft^2
	Insulation fabrication costs: \$30 to 50/ft^2
	Subtotal: \$61 to \$81/ft^2
	Assumed removal costs: \$31/ft^2
	Combined total: \$92 to \$112/ft^2
	Average combined total: \$102/ft^2
Owens-Corning Fiberglass Company	Reinsulation labor costs: \$60 to \$144/ft^2
	Insulation fabrication cost: \$90 to \$100/ft^2
	Combined total: \$150 to \$244/ft^2
	Averaged combined total: \$197/ft^2
Composite of all three estimates	Averaged total: (\$160 + \$102 + \$197)/3 = \$153/ft^2

TABLE 4
Maximum LOCA-generated debris summarized by break size

Pipe diameter (in.)	Total fibrous debris (ft^3)	Total all type (ft^3)
2	1	1
6	2	22
8	2	3
10	4	31
14	227	227
16	270	270
32	144	295
34	315	726
36	118	402

Notes:

(1) These values correspond to break locations in the primary system within the crane wall and represent the largest quantity of debris generated by a single break of a given pipe diameter.

(2) The insulation types and distribution within containment are those used in Salem 1. All insulation within 7 L/Ds of a break location is assumed to be destroyed and released as fragmented debris.

Source: NUREG/CR-3394

TABLE 5
Typical volumes of primary system insulation employed*

Component	Salem		Maine Yankee	
	Volume (ft^3)	Type of insulation	Volume (ft^3)	Type of insulation
Steam generator	1284	reflective metallic/ fibrous	1144	calcium silicate fibrous
Hot leg	160	reflective metallic	149	fibrous
Cold leg	140	reflective metallic	156	fibrous
Crossover	60	reflective metallic	279	fibrous
Pressurizer	160	reflective metallic	302	calcium silicate fibrous
Pressurizer surge line	129	reflective metallic	57	calcium silicate fibrous
Reactor coolant	570	reflective metallic	149	calcium silicate fibrous
Bypass	N/A	N/A	88	fibrous
TOTAL**	2503		2324	
SUBTOTAL*** (excluding reflective metallic and calcium silicate)	1284 (4402 ft^2)		1527 (5234 ft^2)	

*This table is based on information provided by the operators in 1981. Plant changes since 1981 have made the data less accurate for these specific reactors. However, as representative data for reactors in general, the table is still valid.

**This volume includes all of the insulation that could be hit by a water jet from a LOCA pipe break (in pipes ≧10-inch diameter). If the volume were restricted to insulation within L/D = 7 of a break, it might be significantly smaller.

***For conservatism, Salem's steam generator is assumed to be covered entirely with fibrous insulation; 50% of the insulation on Maine Yankee's steam generator, pressurizer, and reactor coolant pump is assumed to be fibrous.

Using these revised insulation replacement costs and the revised estimates of the amount of insulation that must be replaced results in the cost estimates work sheet shown in Table 6. A cost range of $300,000 to $1,300,000 (average cost = $750,000) is arrived at and is used in the value-impact calculations contained in Section 2.2 of this report.

Other estimated costs associated with plant evaluations and potential backfits are shown in Figure 1. These estimated costs were derived as follows:

(1) Utilization of information provided in RG 1.82, Revision 1 and NUREG-0897, Revision 1 provides a rapid means for estimating sump air ingestion and debris potential. An evaluation impact of $10,000 (see 1 in Figure 1) is estimated for plants having high post-LOCA water levels (which prevent air ingestion) and low containment recirculation flow velocities (i.e., $\leq$ 0.15 ft/sec) (which precludes debris transport).

(2) Should air ingestion pose a problem, the use of vortex suppressors has been shown experimentally to reduce air ingestion to zero; the relevant information is provided in RG 1.82, Revision 1 and NUREG-0897, Revision 1. The design, fabrication, and installation of a vortex suppressor (consisting of commercially available floor grating materials, either installed horizontally or formed into a cage) are estimated to cost $35,000 to $50,000 (see 2 in Figure 1).

(3) Debris blockage problems can be assessed in two steps. The initial step--based on limiting calculations as described in RG 1.82, Appendix A and NUREG-0897, Revision 1--is estimated to cost $15,000. Should a second step--a detailed debris-generation, transport, and screen blockage analysis--be required, the cost would be higher. Plant-specific analyses in USI A-43 studies were on the order of $35,000 to $50,000 per plant analyzed. Thus, this cost impact is estimated to range from $25,000 to $65,000 (see 3 in Figure 1).

TABLE 6

Estimated insulation replacement costs

Plant	Fibrous insulation employed (ft^2)	Amount requiring replacement (ft^2)	Estimated cost* ($ thousands) High	Avg	Low
Salem 2	13,200	4,400**	880	675	440
Maine Yankee	6,700	5,235**	1050	815	525
Millstone 2	1,300	1,300 (assumed)	260	200	130
	AVERAGED COST		730	565	365
	AVERAGED COST (w/o Millstone 2)		965	745	485

Estimated cost variance ($ thousands)

Case 1 (averaged cost of three plants)

940 ← 730(+30%) → 475

↑

730 ← $\boxed{565}$ → 365

↓

475 ← 365(-35%) → 240

Case 2 (averaged cost without Millstone 2)

1255 ← 965(+30%) → 630

↑

965 ← $\boxed{745}$ → 485

↓

630 ← 485(-35%) → 315

Case 3 (rounded values from Case 2, used for value-impact ratio calculations)

1300 1000 650

↑ ↑ ↑

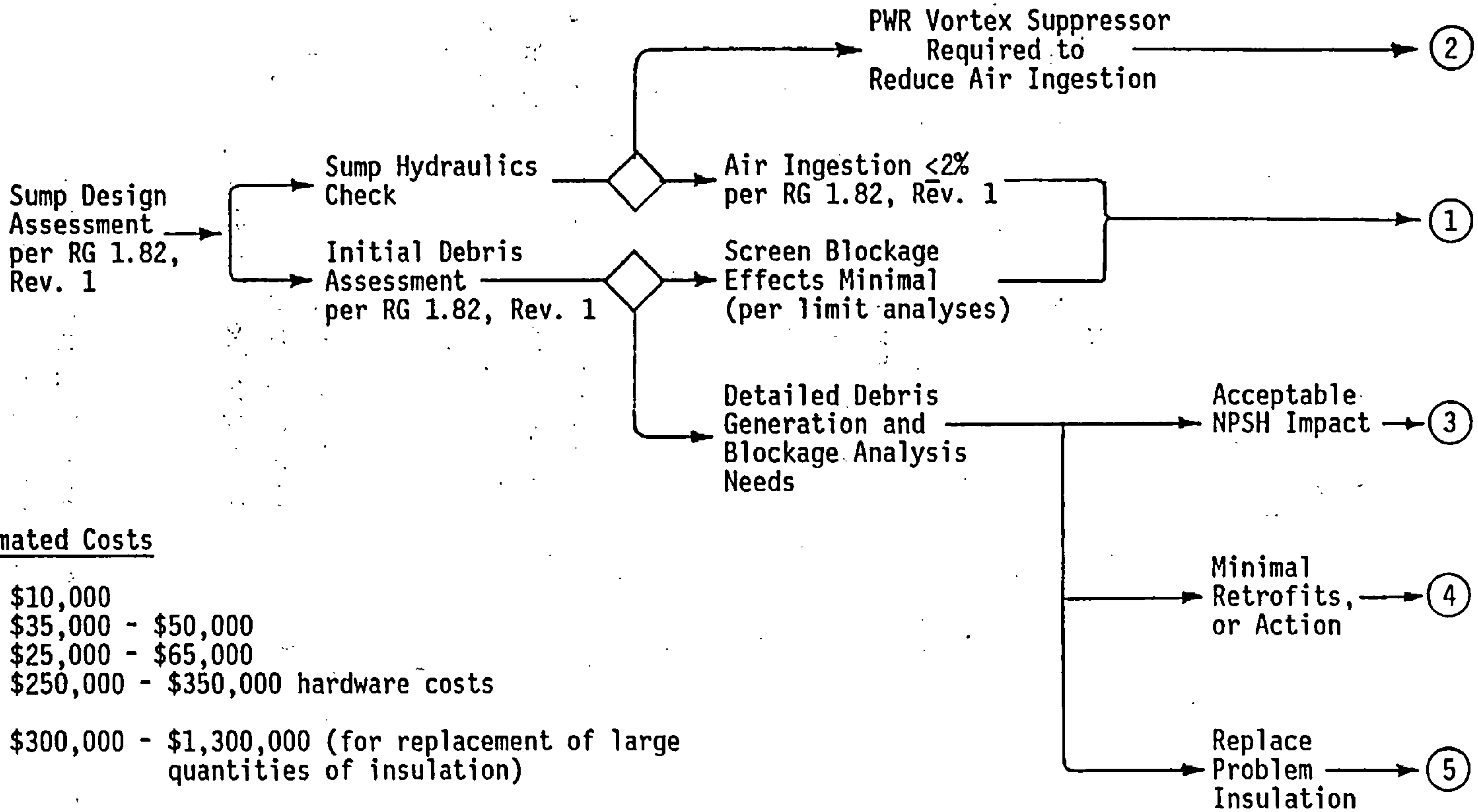
1000 ← $\boxed{750}$ → 500

↓ ↓ ↓

650 500 300

*Utilizes cost range shown in Table 3.

**See Table 5.



<u>Estimated Costs</u>

1 $10,000
2 $35,000 - $50,000
3 $25,000 - $65,000
4 $250,000 - $350,000 hardware costs

5 $300,000 - $1,300,000 (for replacement of large quantities of insulation)

Figure 1 Actions required to determine sump design acceptability

(4) Should the debris assessment calculations show a need for plant modifications, consideration logically should be given to alternatives that would be less costly than replacing large quantities of insulation. Because preservation of the net positive suction head margin is the key criterion, two ways the plant could be modified are

- Increasing debris screen area to reduce the impact of a loss of pump suction head caused by blockage. This could be done by enlarging the sump or intake screen. Along the same lines, use of screens upstream of the currently installed screen would have a two-fold benefit: it would intercept undesirable debris at some distance from the sump location and it would reduce the impact of head loss because of the reduced approach velocities associated with the enlarged upstream screen area. Such a hardware backfit is estimated to cost \$250,000 to \$350,000 (see 4 in Figure 1).

- Re-examining the recirculation flow rates required for the long-term recirculation mode, possibly reducing the currently established flow rates (which are set by immediate post-LOCA flow requirements), and submitting the re-analysis of long-term recirculation needs. This option can be considered an analytical backfit, and the cost of such an analysis is estimated to range from \$25,000 to \$65,000.

(5) The estimated costs for replacement of insulation are \$300,000 to \$1,300,000 (see 5 in Figure 1); these represent the major cost impact and are discussed at the beginning of this section.

References

U. S. Nuclear Regulatory Commission, NUREG/CR-3394, "Probabilistic Assessment of Recirculation Sump Blockage Due to Loss-of-Coolant Accidents," July 1983.

APPENDIX H

DRAFT GENERIC LETTER

DRAFT

TO ALL LICENSEES OF OPERATING REACTORS, APPLICANTS FOR OPERATING LICENSEES, AND HOLDERS OF CONSTRUCTION PERMITS.

Gentlemen:

SUBJECT: POTENTIAL FOR LOSS OF POST-LOCA RECIRCULATION CAPABILITY DUE TO INSULATION DEBRIS BLOCKAGE (Generic Letter 85-)

This letter is to inform you about a generic safety concern regarding LOCA - generated debris that could block PWR containment emergency sump screens or BWR RHR suction strainers, thus resulting in a loss of recirculation or containment spray pump net positive suction head (NPSH) margin.

The potential exists for a primary coolant pipe break to damage thermal insulation on the piping as well as that on nearby components. Insulation debris could be transported to water sources used for long-term post-LOCA recirculation and containment sprays (i.e., PWR containment emergency sumps and BWR suction intakes in the suppression pools) and deposited on debris screens or suction strainers. This could reduce the NPSH margin below that required for recirculation pumps to maintain long-term cooling.

This concern has been addressed as part of the efforts undertaken to resolve USI A-43, "Containment Emergency Sump Performance." The staff's technical findings contain the following main points.

- Plant insulation surveys, development of methods for estimating debris generation and transport, debris transport experiments, and information provided as public comments on the findings have shown that debris blockage effects are dependent on the types and quantities of insulation employed, the primary system layout within containment, post-LOCA recirculation patterns and velocities, and the post-LOCA recirculation flow rates. It was concluded that a single generic solution is not possible, but rather that debris blockage effects are governed by plant specific design features and post-loca recirculation flow requirement.

- The current 50% screen blockage assumption identified in Regulatory Guide (RG) 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems," should be replaced with a more comprehensive requirement to assess debris effects on a plant-specific basis. The 50% screen blockage assumption does not require a plant-specific evaluation of the debris-blockage potential and usually will result in a non-conservative analysis for screen blockage effects.

The staff has revised Regulatory Guide (RG) 1.82, Revision 0, "Sumps for Emergency Core Cooling and Containment Spray Systems" and the Standard Review Plan Section 6.2.2, "Containment Heat Removal Systems" based on the above technical findings. However, the staff's regulatory analysis (NUREG-0869, Revision 1, "USI A-43 Regulatory Analysis") evaluated (1) containment designs and their survivability should loss of recirculation occur, (2) alternate means to remove decay heat, (3) release consequences (which were based on pipe break probabilities which did not incorporate insights gained from recent pipe fracture mechanics analyses), and (4) cost estimates for backfits considered (i.e., reinsulating). This regulatory analysis did not support a generic backfit action and resulted in the decision that this revised regulatory guidance will not be applied to any plant now licensed to operate or that is under construction. The revised guidance will be used on Construction Permit Applications, Preliminary Design Approval (PDA) applications, and applications for licenses to manufacture that are docketed after six (6) months following issuance of RG 1.82, Revision 1, and Final Design Approval (FDA) applications, for standardized designs which are intended for referencing in future Construction Permit Applications, that have not received approval at six (6) months following issuance of the RG 1.82, Revision 1.

Although the staff has concluded that no new requirements need be imposed on licensees and construction permit holders as a result of our concluding analyses dealing with the resolution of USI A-43, we do recommend that RG 1.82, Revision 1 be used as guidance for the conduct of 10 CFR 50.59 reviews dealing with the changeout and/or modification of thermal insulation installed on primary coolant system piping and components. RG 1.82, Revision 1 provides guidance for estimating potential debris blockage effects. If, as a result of NRC staff review of licensee actions associated with the changeout or modification of thermal insulation, the staff decides that Standard Review Plan Section 6.2.2, Revision 4 and/or RG 1.82, Revision 1 should be (or should have been) applied to the rework by the licensee, and the staff seeks to impose these criteria, then the NRC will treat such an action as a plant-specific backfit pursuant to 10 CFR 50.109. It is expected that those plants with small debris screen areas (less than 100 ft^2), high ECCS recirculation pumping requirements (greater than 8000 gpm), and small NPSH margins (less than 1 to 2 ft of water) would benefit the most from this type of assessment in the event of a future insulation change. RG 1.82, Revision 0 with its 50% blockage criteria does not adequately address this issue and is inconsistent with the technical findings developed for the resolution of USI A-43.

This information letter along with enclosed copies of NUREG-0897, Revision 1, RG 1.82, Revision 1 and SRP Section 6.2.2, Revision 4 should be directed to the appropriate groups within your organization who are responsible for conducting 10 CFR 50.59 reviews.

DRAFT

No written response or specific action is required by this letter. Therefore, no clearance from the Office of Management and Budget is required. If you have any questions on this matter, please contact your project manager.

Hugh L. Thompson, Jr., Director
Division of Licensing

Enclosure:
NUREG-0897, Revision 1
RG 1.82, Revision 1
SRP Section 6.2.2, Revision 4

NRC FORM 335
(2-84)
NRCM 1102,
3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE.

1. REPORT NUMBER (Assigned by TIDC, add Vol No., if any)
NUREG-0869, Revision 1

2. TITLE AND SUBTITLE
USI A-43 Regulatory Analysis

3. LEAVE BLANK

4. DATE REPORT COMPLETED

MONTH	YEAR
October	1985

5. AUTHOR(S)
A. W. Serkiz

6. DATE REPORT ISSUED

MONTH	YEAR
October	1985

7. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Division of Safety Technology
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, DC 20555

8. PROJECT/TASK/WORK UNIT NUMBER

9. FIN OR GRANT NUMBER

10. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Division of Safety Technology
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, DC 20555

11a. TYPE OF REPORT
Final Regulatory Analysis

b. PERIOD COVERED (Inclusive dates)

12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)
This report consists of: (1) the regulatory analysis for Unresolved Safety Issue (USI) A-43, "Containment Emergency Sump Performance"; (2) the proposed resolution; (3) a summary of public comments received and action taken; (4) the Committee to Review Generic Requirements (CRGR) minutes related to this USI; and (5) appendices that summarize assumptions, calculational methods, consequence analyses, and cost estimates used in this regulatory analysis.

14. DOCUMENT ANALYSIS – a. KEYWORDS/DESCRIPTORS
USI A-43
Regulatory Analysis for USI A-43

b. IDENTIFIERS/OPEN-ENDED TERMS

15. AVAILABILITY STATEMENT
Unlimited

16. SECURITY CLASSIFICATION
(This page)
Unclassified
(This report)
Unclassified

17. NUMBER OF PAGES

18. PRICE

U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
OFFICE OF NUCLEAR REACTOR REGULATION

6.2.2 CONTAINMENT HEAT REMOVAL SYSTEMS

REVIEW RESPONSIBILITIES

Primary - Containment Systems Branch (CSB)

Secondary - None

I. AREAS OF REVIEW

The CSB reviews the information in the applicant's safety analysis report (SAR) concerning containment heat removal under post-accident conditions to assure conformance with the requirements of General Design Criteria 38, 39, and 40 (Ref. 1, 2 and 3). The types of systems provided to remove heat from the containment include fan cooler systems, spray systems, and residual heat removal systems. These systems remove heat from the containment atmosphere and the containment sump water, or the water in the containment wetwell. The CSB review includes the following analyses and aspects of containment heat removal system designs:

1. Analyses of the consequences of single component malfunctions.

2. Analyses of the available net positive suction head (NPSH) to the containment heat removal system pumps.

3. Analyses of the heat removal capability of the spray water system.

4. Analyses of the heat removal capability of fan cooler heat exchangers.

5. The potential for surface fouling of fan cooler, recirculation, and residual heat removal heat exchangers, and the effect on heat exchanger performance.

6. The design provisions and proposed program for periodic inservice inspection and operability testing of each system or component.

7. The design of sumps and water sources for emergency core cooling and containment spray systems, including an assessment for potential loss of

d. Leak detection, isolation and containment capabilities being incorporated in the design of the containment heat removal system.

2. General Design Criterion 39, as it relates to the containment heat removal system being designed to permit periodic inspecton of components.

3. General Design Criterion 40, as it relates to the containment heat removal system being designed to permit periodic testing to assure system integrity, and the operability of the system, and active components.

Specific acceptance criteria necessary to meet the relevant requirement of GDC 38, 39, and 40 are as follows:

1. The containment heat removal systems should meet the redundancy and power source requirements for an engineered safety feature; i.e., the systems should be designed to accommodate a single active failure. The results of failure modes and effects analyses of each system should assure that the system is capable of withstanding a single failure without loss of function. This is conformance with the requirements of General Design Criterion 38.

2. With regard to General Design Criterion 38 as it relates to the capability of containment system to accomplish its safety function, the spray system should be designed to accomplish this without pump cavitation occurring. Therefore. the net positive suction head available to the pumps in both the injection and recirculation phases of operation should be greater than the required NPSH. A supporting analysis should be presented in sufficient detail to permit the staff to determine the adequacy of the analysis and should show that the available NPSH is greater than the required NPSH. Regulatory Guide 1.82, Rev. 1 (Ref. 5) describes methods acceptable to the staff for evaluating the NPSH margin.

   In the recirculation phase; i.e., in the long term (after about one hour) following a LOCA, the containment spray system is required to circulate the water in the containment. The NPSH analysis will be acceptable if (1) it is done in accordance to the guidance in Regulatory Guide 1.82, Rev. 1 (Ref. 5) and (2) it is done in accordance with the guidelines of Regulatory Guide 1.1 (Ref. 4), i.e., is based on maximum expected temperature of the pumped fluid and with atmospheric pressure in the containment. For clarification, the analysis should be based on the assumption that the containment pressure equals the vapor pressure of the sump water. This ensures that credit is not taken for containment pressurization during the transient.

   The recirculation spray system for a subatmospheric containment is designed to start about five minutes after a loss-of-coolant accident, i.e., during the injection phase of spray system operation. For subatmospheric containments, the guidelines of Regulatory Guide 1.1 as defined above will apply after the injection phase has terminated, which occurs about one hour after the accident. Prior to termination of the injection phase the NPSH analyses should include conservative predictions of the containment atmosphere pressure and sump water temperature transients.

3. In evaluating the performance capability of the containment spray system, to satisfy GDC 38, analyses of its heat removal capability should be based on the following considerations:

   a. The locations of the spray headers relative to the internal structures.

   b. The arrangement of the spray nozzles on the spray headers and the expected spray pattern.

   c. The type of spray nozzles used and the nozzle atomizing capability, i.e., the spray drop size spectrum and mean drop size emitted from each type of nozzle as a function of differential pressure across the nozzle.

   d. The effect of drop residence time and drop size on the heat removal effectiveness of the spray droplets.

   The spray systems should be designed to assure that the spray header and nozzle arrangements produce spray patterns which maximize the containment volume covered and minimize the overlapping of the sprays.

4. In evaluating the performance capability of the fan cooler system, to satisfy GDC 38, the design heat removal capability (i.e., heat removal rate vs. containment temperature) of fan coolers should be established on the basis of qualification tests on production units or acceptable analyses that take into account the expected post-accident environmental conditions and variations in major operating parameters such as the containment atmosphere steam-air ratio, condensation on finned surfaces, and cooling water temperature and flow rate. The equipment housing and ducting associated with the fan cooler system should be analyzed to determine that the design is adequate to withstand the effects of containment pressure following a loss-of-coolant accident (see SRP Section 6.2.5). Fan cooler system designs that contain components which do not have a post-accident safety function should be designed such that a failure of nonsafety-related equipment will not prevent the fan cooler system from accomplishing its safety function.

5. In evaluating the heat removal capability of the containment heat removal system, to satisfy GDC 38, the potential for surface fouling of the secondary sides of fan cooler, recirculation, and residual heat removal heat exchangers by the cooling water over the life of the plant and the effect of surface fouling on the heat removal capacity of the heat exchangers should be analyzed and the results discussed in the SAR. The analysis will be acceptable if it is shown that provisions such as closed cooling water systems are provided to prevent surface fouling or surface fouling has been accounted for in establishing the heat removal capability of the heat exchangers.

6. To satisfy the requirement of GDC 38 regarding the long-term spray system(s) and emergency core cooling system(s), the containment emergency sump(s) should be designed to provide a reliable, long-term water source for ECCS and CSS recirculation pumps. Provision should be made in the containment design to allow drainage of spray and emergency core cooling

long-term cooling capability due to LOCA generated debris effects such as debris screen blockage and pump seal failure.

8. The effects of debris such as thermal insulation on recirculating fluid systems.

The CSB will coordinate other branch evaluations that interface with the overall review of the containment heat removal systems as follows: the Auxiliary Systems Branch (ASB) wll review the secondary cooling systems, which provide cooling water to the heat exchangers in the containment heat removal systems, as part of its primary review responsibility for SRP Section 9.2.2. The Instrumentation and Control Systems Branch (ICSB) will review the sensing and actuation instrumentation provided for the containment heat removal systems as part of its primary review responsibility for SRP Section 7.3. The Equipment Qualification Branch (EQB) will review the qualification test program for the active components of the fan cooler system, and the sensing and actuation instrumentation for the containment heat removal system as part of its primary review responsibility for SRP Section 3.11. The Chemical Engineering Branch (CMEB) will evaluate the quantity of unqualified paint that can potentially reach the emergency sump(s) under design basis pipe break accident review responsibility for SRP Section 6.1.2. The Accident Evaluation Branch (AEB) will review fission product control features of containment heat removal systems as part of its primary review responsibility for SRP Section 6.5.2. The Mechanical Engineering Branch (MEB) will review the system seismic design and quality group classification as part of its primary review responsibility for SRP Section 3.2.1 and SRP Section 3.2.2, respectively. The Licensing Guidance Branch (LGB) will review the proposed technical specifications for each system at the operating license stage of review as part of the primary review responsibility for SRP Section 16.0.

For those areas of review identified above being reviewed as part of the primary review responsibility of other branches, the acceptance criteria necessary for the review and their methods of application are contained in the referenced SRP section of the corresponding primary branch.

II. ACCEPTANCE CRITERIA

CSB acceptance criteria for the design of the containment heat removal system is based on meeting the relevant requirements of General Design Criterion 38, 39, and 40. The relevant requirements are as indicated below.

1. General Design Criterion 38 as it relates to:

   a. Containment heat removal system being capable of reducing rapidly the containment pressure and temperature following a LOCA, and maintaining them at acceptably low levels.

   b. The containment heat removal system performance being consistent with the function of other systems.

   c. The containment heat removal system being safety-grade design; i.e., have suitable redundancy of components and features, and interconnections, to assure that for either a loss of onsite as a loss of offsite power, the system function can be accomplished assuming a single failure.

water to the emergency sump(s), and for recirculation of this water through the containment sprays and emergency core cooling systems. The design of the sumps, and the protective screen assemblies is a critical element in assuring long-term recirculation cooling capability. Therefore, adequate design consideration of: a) sump hydraulic performance, b) evaluation of potential debris generation and associated effects including debris screen blockage, c) RHR and CSS pump performance under postulated post-LOCA conditions is necessary. These design considerations are addressed in Regulatory Guide 1.82, Rev. 1 (Ref. 5) and NUREG-0897, Rev. 1 (Ref. 7).

7. In meeting the requirements of GDC 39 and 40, regarding inspection and testing, provisions should be made in the design of containment heat removal systems for periodic inspection and operability testing of the systems and system components such as pumps, valves, duct pressure-relieving devices, and spray nozzles.

8. To satisfy the system design requirements of GDC 38, instrumentation should be provided to monitor containment heat removal system and system component performance under normal and accident conditions. The instrumentation should be capable of determining whether a system is performing its intended function, or a system train or component is malfunctioning and should be isolated.

III. REVIEW PROCEDURES

The procedures described below provide guidance for the review of containment heat removal systems. The reviewer selects and emphasizes material from the review procedures as may be appropriate for a particular case. Portions of the review may be done on a generic basis for aspects of heat removal systems common to a class of containments, or by adopting the results of previous reviews of plants with essentially the same system.

Upon request from CSB, the secondary review branches will provide input for the areas of review stated in subsection I of this SRP section. CSB obtains and uses such input as required to assure that this review procedure is complete. CSB assures that the design and functional capability of the containment heat removal system conform to the requirements of General Design Criteria 38, 39 and 40.

CSB determines the acceptability of a containment heat removal system design by reviewing failure modes and effects analyses of the system to be sure that all potential single failures have been identified and no single failure could incapacitate the entire system; verifying that engineered safety feature design standards have been applied; reviewing the system design provisions for periodic inservice inspection and operability testing to ensure that the system and components are accessible for inspection and all active components can be tested; and reviewing the capability to monitor system performance and control active components from the control room so that the operator can exercise control over system functions or isolate a malfunctioning system component.

CSB reviews analyses of the net positive suction head available to the spray system pumps. CSB assures that the analyses for the recirculation phase are done in accordance with the guidelines of Regulatory Guide 1.1, i.e., are based on maximum expected temperature of the pumped fluid and with atmospheric pressure in the containment. For clarification, the analyses should be based on the assumption that the containment pressure equals the vapor pressure of the sump water. This ensures that credit is not taken for containment pressurization during the transient. CSB assures that calculations of the available NPSH are based on transient values of the suction head and the friction head. The CSB reviews information provided by the applicant to identify and justify the conservatisms applied in determining the water level in the containment and the friction losses in the recirculation system suction piping. For example, the uncertainty in determining the free volume in the lower part of the containment that may be occupied by water, and the quantity of water that may be trapped by the reactor cavity and the refueling canal, should be factored into the calculation of the suction head.

The CSB reviews analyses of the available NPSH for subatmospheric containments for the period prior to termination of the injection phase of containment spray to determine that containment pressure and sump water temperature transients have been conservatively used in the NPSH calculations. The CSB reviews information provided by the applicant to identify and justify the conservatisms in the analysis of the containment atmosphere pressure and sump water temperature transients. The CSB also reviews the conservatisms used in determining the water level in the containment and the friction losses in the recirculation system piping.

The CSB compares the NPSH requirements for the containment heat removal system pumps to the minimum calculated NPSH available to the pumps to assure that a positive margin is maintained. The CSB also reviews the preoperational test programs, and periodic inservice inspection and test programs, to verify that adequate NPSH is available to the pumps and the continuing operability of the pumps during the lifetime of the plant.

If in the judgment of the CSB, the NPSH analyses were not done in a sufficiently conservative manner, confirmatory analyses are performed using the CONTEMPT-LT computer code.

The CSB also reviews the evaluation of the volume of the containment covered by the sprays and the extent of overlapping of the sprays with respect to heat removal capabilities. A judgment will be made regarding the acceptability of the spray coverage and extent of overlapping; the volume of the containment covered by the sprays should be maximized and the extent of overlapping kept to a minimum. Elevation and plan drawings of the containment showing the spray patterns are used to determine coverage and overlapping.

In general, the design requirements for the spray systems with respect to spray drop size spectrum and mean drop size, spray drop residence time in the containment atmosphere, containment coverage by the sprays, and extent of overlapping of the sprays are more stringent when the acceptability of the system is being considered from an iodine removal capability standpoint rather than from a heat removal capability standpoint. Consequently, when the iodine removal capability of the system is satisfied, the heat removal capability will be found acceptable. The Accident Evaluation Branch is responsible for determining the acceptability of the iodine removal effectiveness of the sprays

(See Standard Review Plan Section 6.5.2). Since all plants do not use the containment sprays as a fission product removal system, the CSB reviews the system for cases where the system is used only as a heat removal system.

CSB reviews analyses of the heat removal capability of the spray system. This capability is a function of the degree of thermal equilibrium attained by the spray water and the volume of the containment covered by the spray water. The spray drop size and residence time in the containment atmosphere determine the degree of thermal equilibrium attained by the spray water. The CSB confirms the validity of the degree of thermal equilibrium attained using the following information: an elevation drawing of the containment showing the locations of the spray headers relative to the internal structures, including fall heights, and the results of the spray nozzle test program to determine the spectrum of drop sizes and mean drop size emitted from the nozzles as a function of pressure drop across the nozzles.

Reference 6 contains information regarding the heating of spray drops in air-steam atmospheres which can be used to determine the validity of the degree of thermal equilibrium of the spray water used in the analyses.

CSB reviews the adequacy of provisions made to prevent overpressurization of fan cooler ducting following a loss-of-coolant accident (Standard Review Plan Section 6.2.5). CSB reviews the heat removal capability of the fan coolers. The test programs and calculation models used to determine the performance capability of fan coolers are reviewed for acceptability. If the secondary side of a fan cooler heat exchanger is not a closed system, the CSB reviews the potential for surface fouling. The CSB determines whether or not surface fouling impairs the heat removal capability of a fan cooler.

CSB reviews the system provided to allow drainage of containment spray water and emergency core cooling water to the recirculation suction points (sumps). CSB reviews the design of the protective screen assemblies around the suction points. CSB reviews plan and elevation drawings of the protective screen assemblies, showing the relative positions and orientations of the trash bars or grating and the stages of screening, to determine that the potential for debris clogging the screening is minimized. CSB also reviews the drawings to determine that suction points do not share the same screened enclosure. The effectiveness of the protective screen assembly will be determined by comparing the smallest mesh size of screening provided to the clogging potential of pumps, heat exchangers, valves, and spray nozzles. The methods of attachment of the trash bars or grating and the screening to the protective screen assembly structure should be discussed in the SAR and shown on drawings. A discussion of the adequacy of the surface area of screening with respect to assuring a low velocity of approach of the water to minimize the potential for debris in the water being sucked against the screening should be presented. Regulatory Guide 1.82, Rev. 1 (Ref. 5) provides guidelines for the acceptability of the design of PWR sumps and BWR RHR suction inlets. NUREG-0897, Rev. 1 (Ref. 7) details technical considerations pertinent to these matters.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

6.2.2 Containment Heat Removal Systems

The containment heat removal systems include (identify the systems).

The scope of review of the containment heat removal systems for the (plant name) has included system drawings and descriptive information. The review has included the applicant's proposed design bases for the containment heat removal systems, and the analyses of the functional capability of the systems.

The staff concludes that the design of the containment heat removal systems is acceptable and meets the requirements of General Design Criteria 38, 39 and 40.

The conclusion is based on the following: [The reviewer should discuss each item of the regulations or related set of regulations as indicated.]

1. The applicant has met the requirements of (cite regulation) with respect to (state limits of review in relation to regulation) by (for each item that is applicable to the review state how it was met and why acceptable with respect to the regulation being discussed):

   a. meeting the regulatory positions in Regulatory Guide _____ or Guides;

   b. providing and meeting an alternative method to regulatory positions in Regulatory Guide ____, that the staff has reviewed and found to be acceptable;

   c. meeting the regulatory position in BTP _____;

   d. using calculational methods for (state what was evaluated) that has been previously reviewed by the staff and found acceptable; the staff has reviewed the impact parameters in this case and found them to be suitably conservative or performed independent calculations to verify acceptability of their analysis; and/or

   e. meeting the provisions of (industry standard number and title) that has been reviewed by the staff and determined to be appropriate for this application.

2. Repeat discussion for each regulation cited above.

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plan for using this SRP section.

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guides.

The PWR sump and BWR RHR suction inlet design and evaluation guidance provided in Subsection II.6 of this SRP section, RG 1.82, Rev. 1, and as further detailed in NUREG-0897, Rev. 1B, is applicable to:

1) construction permit applications and preliminary design approvals (PDAs) that are docketed after [2];
2) applications for Final Design Approval (FDA), for standardized designs which are intended for referencing in future construction permit applications that have not received approval at[2].
3) applications for licenses to manufacture that are docketed after [2].

The other portions of SRP Section 6.2.2 remain unchanged and are applicable to all CP and OL plants.

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 38, "Containment Heat Removal."

2. 10 CFR Part 50, Appendix A, General Design Criterion 39, "Inspection of Containment Heat Removal System."

3. 10 CFR Part 50, Appendix A, General Design Criterion 40, "Testing of Containment Heat Removal System."

4. Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps."

5. Regulatory Guide 1.82, Rev. 1, "Water Sources for Long Term Reicrculation Cooling Following a Loss of Coolant Accident," October 15, 1985.

6. L. F. Parsly, "Design Considerations of Reactor Containment Spray Systems - Part VI, The Heating of Spray Drops In Air-Steam Atmospheres," ORNL-TM-2412, Oak Ridge National Laboratory, January 1970.

7. NUREG-0897, Rev. 1, "Containment Emergency Sump Performance - Technical Findings Related to USI A-43," October 1985.

[2]Six (6) months after issuance of this SRP Section (Ref. 4, October 1985) and Regulatory Guide 1.82, Rev. 1.