NUREG-0869
For Comment

USI A-43 Resolution Positions

- Proposed Regulatory Guide 1.82, Rev. 1, "Sump for Emergency Core Cooling and Containment Spray Systems"
- Value/Impact Statement for USI A 43, Containment Emergency Sump Performance
- Meeting Minutes of Committee to Review Generic Requirements (CRGR) Regarding Unresolved Safety Issue (USI) A-43 Resolution (CRGR Meeting Nos. 26 and 28)

Office of Nuclear Reactor Regulation

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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 upon written request to the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

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ABSTRACT

NUREG-0869 is comprised of the following documents:

(1) Proposed Regulatory Guide 1.82, Revision 1, "Sump for Emergency Core Cooling and Containment Spray Systems"

(2) The Value-Impact Statement for USI A-43, "Containment Emergency Sump Performance"

(3) Background and Summary of Minutes of Meetings of the Committee to Review Generic Requirements Regarding Unresolved Safety Issue A-43 Resolution

and has been assembled to faciliate obtaining "For Comment" feedback on the position developed for resolution of USI A-43. There are no licensing requirements contained in NUREG-0869, and it should be clearly noted that this "For Comment" report will not be used as interim requirements.

CONTENTS

INTRODUCTION

NUREG-0869 is made up of the following documents:

(1) The proposed Regulatory Guide (RG) 1.82, Revision 1, "Sump for Emergency Core Cooling and Containment Spray Systems"
(2) The Value/Impact Statement for USI A-43, "Containment Emergency Sump Performance"
(3) Background and Summary of Minutes of Meetings of the to Review Generic Requirements (CRGR) Regarding the Resolution of Unresolved Safety Issue (USI) A-43 (CRGR Meeting Nos. 26 and 28).

This report has been assembled to facilitate obtaining "For Comment" feedback on the positions developed for resolution of USI A-43. There are no licensing requirements in NUREG-0869, and it should be clearly noted that this "For Comment" report will not be used to specify interim requirements.

A Federal Register Notice of Issuance and Availability regarding these "For Comment" documents that are contained in this NUREG, along with Standard Review Plan (SRP) Section 6.2.2, Revision 4, "Containment Heat Removal Systems" (to be part of NUREG-0800) and NUREG-0897, "Containment Emergency Sump Performance" (the staff's technical findings for USI A-43), has been issued. A copy of this Notice is enclosed.

At completion of the 60-day "For Comment" period, the staff will address comments received and will use such comments in developing the implementation (or final) versions of SRP 6.2.2, Revision 4 and RG 1.82, Revision 1. The implementation version of these documents will be reviewed with the CRGR before the staff proceeds with implementation.

The proposed changes to SRP Section 6.2.2 and RG 1.82 amend requirements that are subject to the Paperwork Reduction Act of 1980 (44 USC 3501 et seq.). The final proposed changes will be submitted to the Office of Management and Budget for review and approval of the paperwork requirements following receipt and review of "For Comment" responses.

PROPOSED REGULATORY GUIDE 1.82, REVISION 1

"SUMP FOR EMERGENCY CORE COOLING AND CONTAINMENT SPRAY SYSTEMS"

NUREG-0869

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REGULATORY RESEARCH

May 1983
Division 1
Task MS 203-4

DRAFT REGULATORY GUIDE AND VALUE/IMPACT STATEMENT

Contacts: L. Porse (301) 443-5860, A. Serkiz (301) 492-4217

PROPOSED REVISION 1 TO REGULATORY GUIDE 1.82

SUMPS FOR EMERGENCY CORE COOLING AND CONTAINMENT SPRAY SYSTEMS

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 a system be provided to remove the heat released to the containment following a postulated design basis accident (DBA) and that this system be designed to permit appropriate periodic inspection and testing to ensure its integrity, capability, 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 for the emergency core cooling and containment spray systems. This guide applies to light-water-cooled reactors.

Any guidance in this document related to information collection activities has been cleared under OMB Clearance No. 3150-0011.

This regulatory guide is being issued in draft form to involve the public in the early stages of the development of a regulatory position in this area. It has not received complete staff review and does not represent an official NRC staff position.

Public comments are being solicited on the guide, including its implementation. Comments should be sent to Mr. Karl Kniel, Division of Safety Technology, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, by July 11, 1983.

Requests for single copies of draft guides (which may be reproduced) or for placement on an automatic distribution list for single copies of future draft guides in specific divisions should be made in writing to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Technical Information and Document Control.

B. DISCUSSION

Sumps or pump intakes serve the emergency core cooling system (ECCS) and the containment spray system (CSS) by providing for the collection of reactor coolant and chemically reactive spray solution and allowing its recirculation for additional cooling and fission product removal.

Placement of the ECCS sumps at the lowest level practical ensures maximum utilization of available recirculation coolant. However, there may be places within containment where coolant could accumulate during the containment spray period; providing these areas with drains or flow paths to the sump location will minimize coolant holdup. This guide does not address the design of such drain paths. However, since debris generated by a loss-of-coolant accident (LOCA) can migrate to the sump via these pathways, these drains are best terminated in a manner that will prevent debris from being transported to and accumulating on the ECCS sump. Appendix A addresses concerns related to debris transport and the effects of attendant sump screen blockage.

Small drainage sumps that are used to collect and monitor normal leakage flow for leakage detection systems within containment are separate from the ECCS sump and are at a lower elevation than the ECCS sump to minimize inadvertent spillover into the ECCS sump due to minor leaks or spills within the containment. The floor adjacent to the ECCS sump would normally slope downward, away from the ECCS sump toward the drainage collection sumps. This downward slope away from the ECCS sump will minimize the collection of debris against the sump screens.

Debris resulting from a LOCA has the potential to block sump screens and result in a degradation of or loss of net positive suction head (NPSH) margin. The LOCA-generated 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 (4) debris that will float by virtue of low density or composition and will be transported to the sump scree Thus, debris generation due to the LOCA, early transport due to blowdown loads long-term transport, and attendant screen blockage effects must be analyzed to determine head loss effects. Appendix A provides guidelines for such evaluations; References 1 through 5 provide additional information.

It is necessary to protect pump intakes by screens and trash racks (coarse outer screens) of sufficient strength to resist impact loads that could be imposed by missiles that may be generated by the initial LOCA or by trash. Isolation of the ECCS sump from high-energy pipe lines is an important consideration in protection against missiles, and it is necessary to shield the screens and trash racks adequately from impacts of ruptured high-energy piping and associated jet loads from the break. When the screen and trash rack structures are located above floor level, the adverse effects from debris collecting on the screen structure will be at a minimum. Separating redundant ECCS sump screens and pump suction intakes to the extent practical will help to reduce the possibility that a partially clogged screen or missile damage to one screen would adversely affect other pump circuits. In addition, proper design of suction intakes will avoid flow degradation by air ingestion, swirl, or vortex formation.

The location of the pump suction intakes within the ECCS sump is important in order to minimize air ingestion that is a function of submergence level and sump outlet velocity. Other factors to consider are vortex formation (which can lead to air ingestion) and swirl effects at the suction inlet. It has been experimentally determined that air ingestion can be minimized or eliminated if the hydraulic design guidelines provided in Appendix A are followed. References 1, 3, 6, 7, 8, and 9 provide additional technical information relevant to sump hydraulic performance and design guidelines.

As noted above, the design of pump suction intakes includes consideration for avoiding air ingestion or other undesirable hydraulic effects (e.g., swirl, suction inlet design effects). However, for small amounts of air ingestion, the recirculation pumps can still be considered operable provided sufficient NPSH margin is demonstrated. Appendix A provides guidance for estimating NPSH margin if estimated levels of air ingestion are low (i.e., $\leq 2\%$). References 1 and 10 provide additional technical findings relevant to pump operation and NPSH effects.

It is expected that the water surface will be above the top of the screen structure after completion of the safety injection. However, the uncertainties about the extent of water coverage on the screen structure, the amount of floating debris that may accumulate, and the potential for early clogging do not favor the use of a horizontal top screen. Therefore, because of this uncertainty, no credit can be taken in computing the available surface area for any top horizontal screen, and the top of the screen structure would preferably be a solid deck designed to provide for the venting of any trapped air.

Slowly settling debris that is small enough to pass through the trash rack openings could block the inner screens if the coolant flow velocity is too great to permit the bulk of the debris to sink to the floor level during transport. A vertically mounted inner screen would minimize settling of debris on the screen surface, and if sufficient unblocked screen area is provided to keep the coolant flow velocity at the screen approximately 6 cm/sec (0.2 ft/sec), debris with a specific gravity of 1.05 or more will settle before reaching the screen surface.

The size of openings in the fine screens is dependent on the physical restrictions, including spray nozzles, that may exist in the systems that are supplied with coolant from the ECCS sump. The size of openings in the containment spray nozzles, coolant channel openings in the core fuel assemblies, and pump impeller running clearances will need to be considered in determining the size of the fine screen.

Potential blocking of the fine screen would reduce the free-flow area through the screen, and it is essential that sufficient flow area be provided to maintain adequate NPSH margin.

A significant consideration is the potential for degraded pump performance that could be caused by a number of factors, including the loss of NPSH margin. If the NPSH available to a pump is not sufficient, degraded pump performance will significantly reduce the capability of the system to accomplish its safety function. The pressure drop across partially (or completely) blocked screens will reduce available NPSH margin and can be calculated based on the debris blockage evaluation methods outlined in Appendix A.

To ensure the readiness and integrity of the rack and screens, access openings are necessary to permit inspection of the inside structures and pump suction inlet openings. Inservice inspection for trash racks, screens, and pump suction inlet openings, 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 ECCS sump components late in the refueling period would help ensure the absence of construction debris in the ECCS sump area.

C. REGULATORY POSITION

Reactor building sumps that are designed to be a source of water for the emergency core cooling system (ECCS) or the containment spray system (CSS) following a loss-of-coolant accident (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 systems.

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., screens and outlet suction pipes) 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 intake should be protected by at least two vertically mounted screens: (1) a fine inner screen and (2) an outer trash rack to prevent large debris from reaching the fine inner screen. The sump screens should not be depressed below the floor elevation. A curb should be provided around the periphery of the screens to prevent high-density debris (i.e., fine particulates) from being swept into the sump.

4. The floor in the vicinity of the ECCS sump should slope gradually down 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 filter assemblies.

6. The strength of the trash rack should be adequate to protect the inner screen from missiles and other large debris generated by the LOCA.

7. The design coolant velocity at the fine inner screen should be approximately 6 cm/sec (0.2 ft/sec). The available screen surface area used in determining the design coolant velocity should be calculated to conservatively account for sump screen blockage that might result from debris generation and transport. Only the vertical sump screens should be considered in determining available surface area.

8. Evaluations of (1) sump hydraulic performance (e.g., geometric effects and air ingestion), (2) LOCA-generated debris effects (e.g., debris transport and screen blockage), and (3) the impact on pump NPSH margin should be performed to ensure that long-term recirculation cooling can be accomplished. Sump hydraulic effects and debris blockage considerations that could have an adverse impact on NPSH margin should be considered in the evaluation of the ECCS pump performance.

9. The top of the deck should be a solid plate that is designed to be fully submerged after a LOCA and completion of the ECC injection. The solid deck should be designed to ensure the venting of any air trapped underneath.

10. The trash rack and screens should be designed to withstand the vibratory motion of seismic events without loss of structural integrity.

11. The size of openings in the fine screen should be based on the minimum restriction found in systems served by the pump. The minimum restriction should take into account the requirements of the systems served.

12. Pump intake locations within the sump should be carefully considered to prevent degradation of pump performance by the effects of such conditions as air ingestion and sump-induced swirl.

13. Materials for trash racks and screens 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 trash rack and screen structure should include access openings to facilitate inspection of the structure and pump suction intake.

15. Inservice inspection requirements for ECCS sump components (trash racks, screens, and pump suction inlets) should include:

a. Inspection of ECCS sump components during every refueling period downtime, and

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

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 proposed revision to the regulatory guide has been published to encourage public participation in its development. 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 the revised active guide reflecting public comments will be used by the NRC staff in its evaluation of the design and construction of sumps for emergency core cooling and containment spray systems. In addition, the NRC staff intends to use this guide to evaluate the design and construction of sumps in plants for which an operating license has been issued; the implementation date will be specified in the active guide.

APPENDIX A

GUIDELINES FOR REVIEW OF ECCS SUMPS

1. General

The ECCS 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) effects of LOCA-induced debris, 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 under all postulated post-LOCA conditions is the final requirement.

2. 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 suction outlets) and required pumping capacity (or sump suction outlet velocity). The water depth (s) and suction pipe velocity (U) parameters can be expressed nondimensionally as the Froude number:

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

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

Sump hydraulic performance can be divided into three categories:

a. Zero air ingestion, thus avoiding pump cavitation,
b. Air ingestion $\leq$2%, a conservative level at which degradation of pumping capability is not expected,
c. Use of vortex suppressors to reduce air ingestion effects to a negligible level.

Zero air ingestion can be ensured by use of the design criteria in Table A-1. Determination of those designs having air ingestion levels 2% or less can be obtained using Table A-2 and the attendant sump envelope, placement, and screen guidelines contained in Tables A-3, A-4, and A-5. Table A-6 presents design guidelines for vortex suppression devices that have shown the capability to reduce air ingestion to zero. These guidelines (Tables A-1 through A-6) were developed from extensive full-scale sump hydraulic tests and provide a concise means of assessing sump hydraulic performance. If the sump design deviates significantly from the boundaries noted, similar performance data should be obtained for verification of adequate sump hydraulic performance.

3. Effects of LOCA-Induced Debris

Assessment of LOCA debris generation and determination of possible sump screen 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 sump screen. Thus estimates of sump screen blockage are specific to the material and the plant design and require consideration of such effects as are outlined in Table A-7.

Since evaluation of debris effects is dependent on the material type and also on recirculation flow velocities, a series of limiting evaluations can be performed; these are outlined in Tables A-8 and A-9. Table A-8 provides a concise means of evaluating the potential for debris transport at various flow velocities for three major types of insulation. Table A-9 provides a rapid means of assessing the transport of fibrous insulation debris and quantifying the volume of such debris that could result in loss of 50% of the NPSH requirement. Loose fibrous debris will be transported by velocities as low as 0.2 ft/sec (6 cm/sec).

If Table A-8 or Table A-9 does not readily identify negligible or conservatively low levels of sump screen blockage, the considerations outlined in Table A-7 must be evaluated on a plant-specific basis. Figure A-2 is provided for additional guidance in estimating such debris blockage effects, and results obtained from such an evaluation would be used to estimate the impact on NPSH margin. References 1, 2, 4, and 5 provide more detailed information relevant to assessment of debris effects.

4. 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 performance. The percentage has at times been arbitrary, but is generally in the range of 1-3%. A 2% limit on allowed air ingestion is recommended since higher levels have been shown to initiate degradation of pumping capacity.

The 2-volume-percent limit on sump air ingestion and the NPSH requirements act independently. However, air ingestion levels less than 2% can also affect NPSH conditions. Figure A-3 is therefore provided as a guide for evaluating conditions at the pump inlet, commencing at the sump. If air ingestion is indicated, the NPSH requirement from the pump curves should be corrected by the relationship:

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

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

5. Combined Effects

As shown in Figure A-1, three interdependent effects (i.e., sump hydraulic performance, LOCA debris effects, and pump operation under adverse conditions) require evaluation for determining long-term recirculation capability. Figure A-4 provides a logic diagram for combining these considerations to evaluate the ECCS sump design and expected performance.

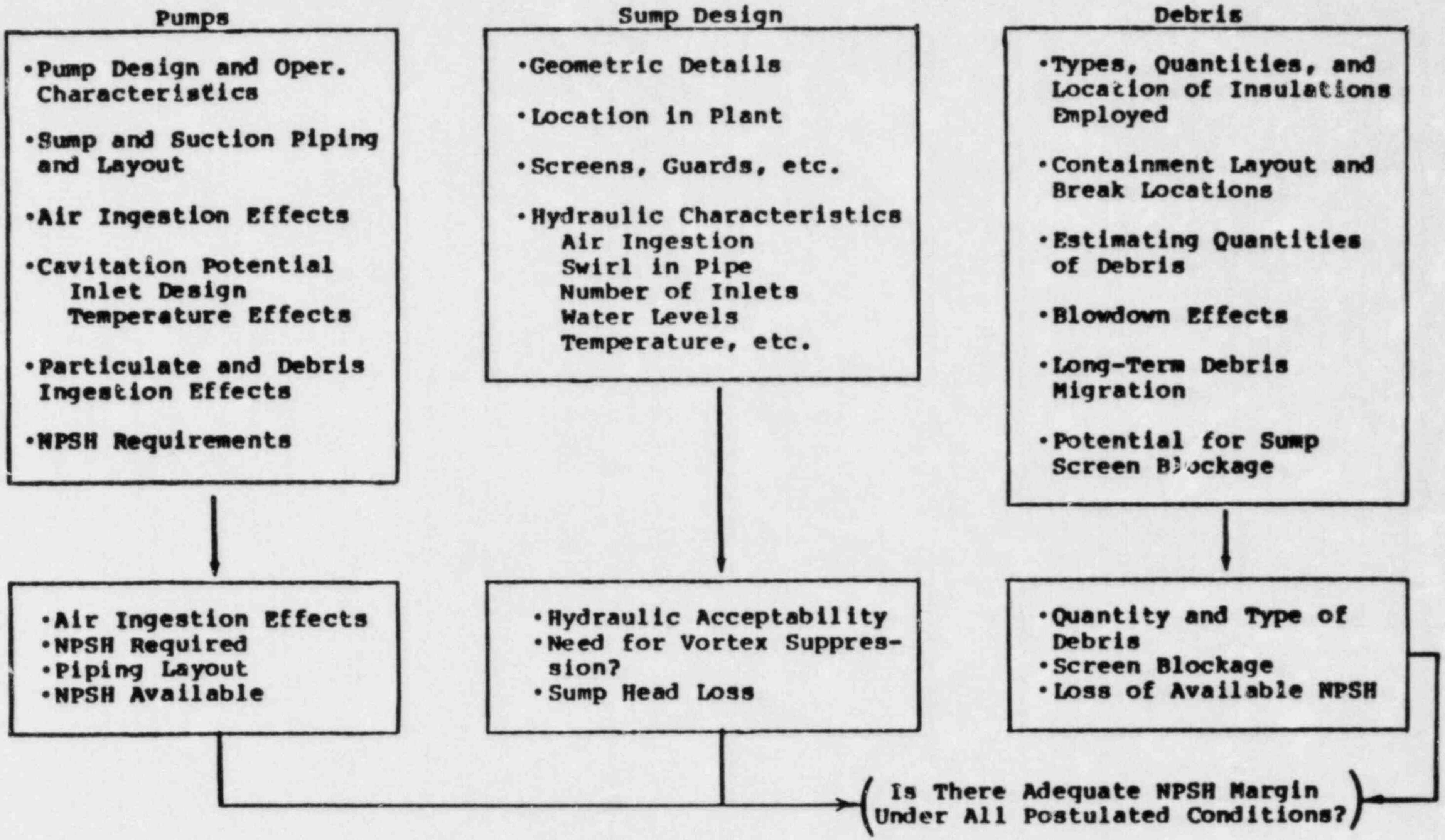
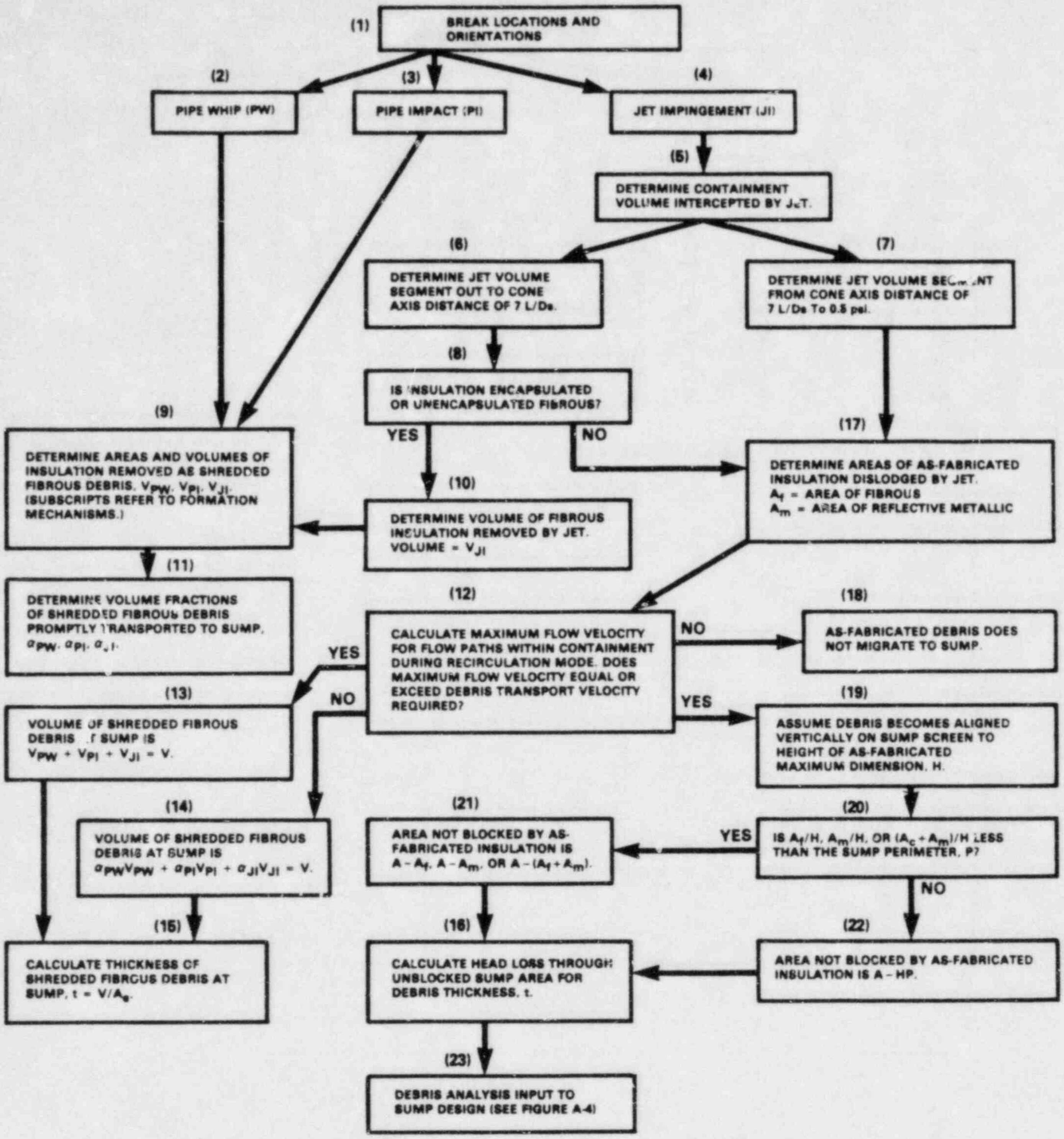


Figure A-1. Technical Considerations Relevant to ECCS Sump Performance



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 PIPE 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^2)
A_m - AREA OF AS FABRICATED REFLECTIVE METALLIC INSULATION DISLODGED BY JET (FT^2)
A - EFFECTIVE AREA OF SUMP SCREEN (FT^2)
A_e - EFFECTIVE UNBLOCKED SUMP SCREEN AREA (AREA AVAILABLE FOR FLOW) (FT^2)
H - 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 A-2. Estimation of Debris Generation, Transport to the ECCS Sump, and Screen Blockage. Calculational Methods Are Given in Reference 2.

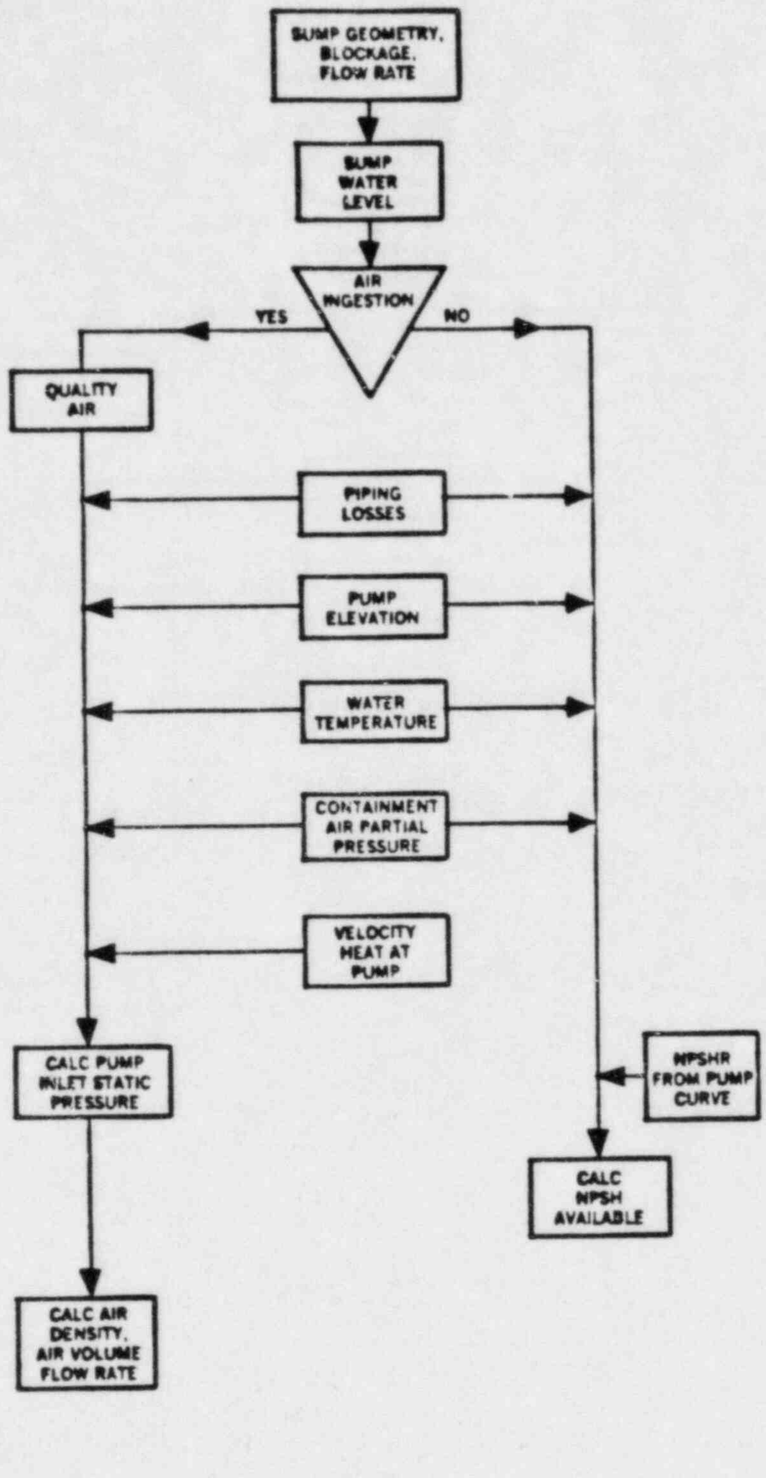


Figure A-3. Flowchart for Calculating Pump Inlet Conditions

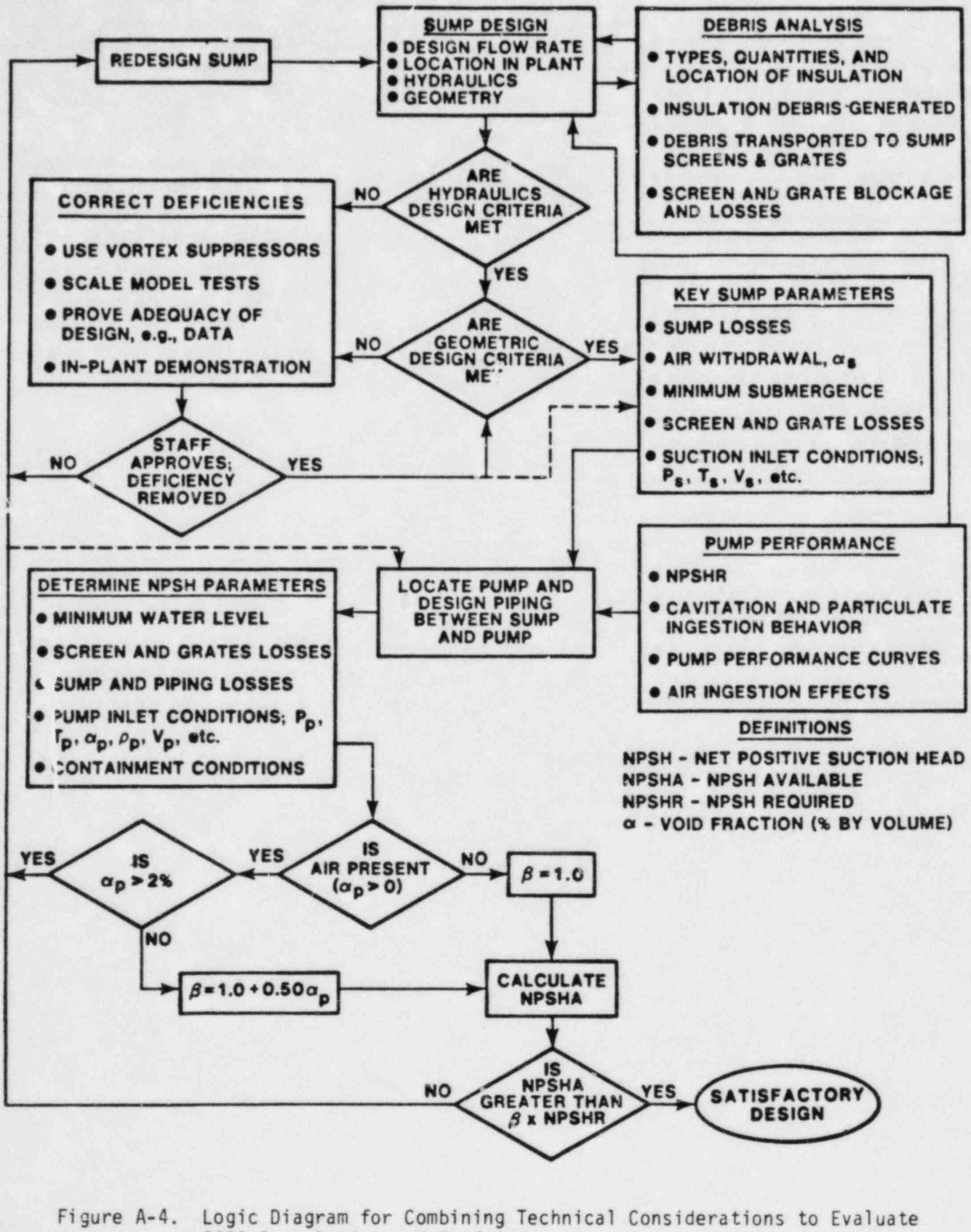


Figure A-4. Logic Diagram for Combining Technical Considerations to Evaluate ECCS Sump Design and Performance

Table A-1

Guidelines for Zero Air Ingestion

Hydraulic Design

Item	Horizontal Outlets	Vertical Outlets
Minimum submergence, s (ft) (m)	10 3.1	10 3.1
Maximum Froude Number, F	0.25	0.25
Maximum Pipe Velocity, U (ft/s) (m/s)	4 1.2	4 1.2

Geometric Design*

Item	Horizontal Outlets	Vertical Outlets
Aspect Ratio	1 to 5	1 to 5
Minimum Perimeter (ft) (m)	36 10.9	16 4.9
$(B - e_y)/d$	≧3	≧1
c/d	≧1.5	≧0; ≦1
e_y/d	≧0; ≧1	≧1.5
Minimum Screen Area (ft^2) (m^2)	75 7	35 3.3

*See Table A-3 for definitions. These guidelines were established using experimental results from tests conducted at the Alden Research Laboratory.

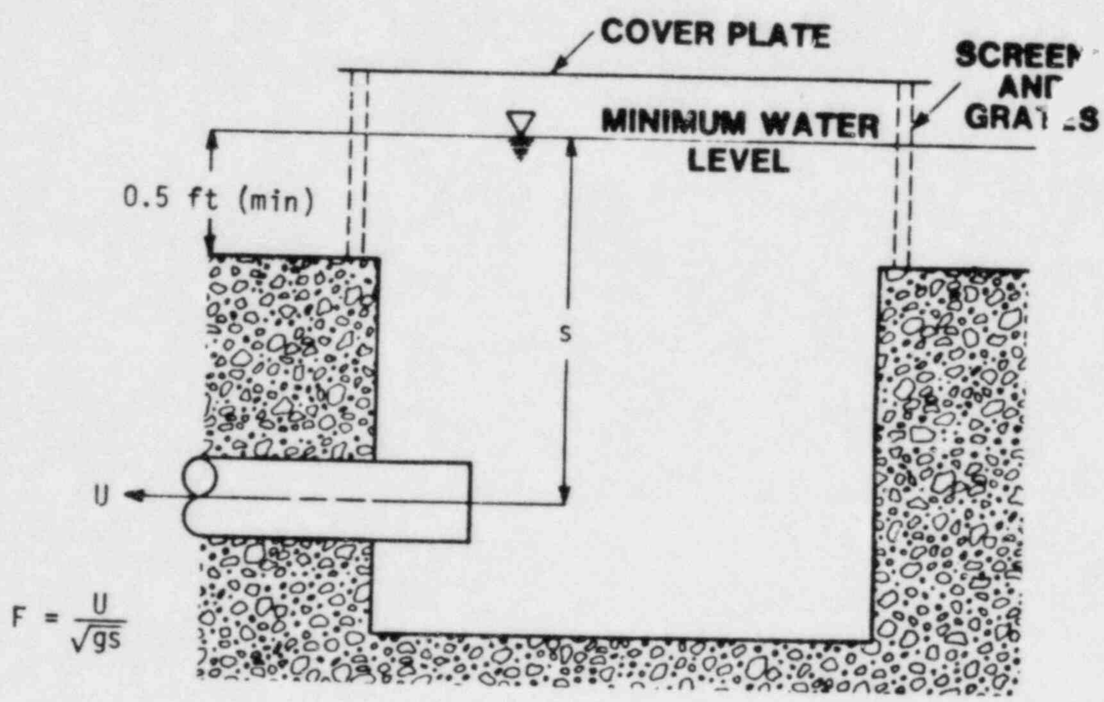


Table A-2

Hydraulic Design Guidelines
for Air Ingestion ≧2%

Item	Horizontal Outlets		Vertical Outlets	
	Dual	Single	Dual	Single
Minimum Submergence, s (ft) (m)	7.0 2.1	8.0 2.4	8.0 2.4	10 3.1
Maximum Froude Number, F	0.53	0.40	0.41	0.33
Maximum Pipe Velocity, U (ft/s) (m/s)	8.0 2.4	6.5 2.0	7.0 2.1	6.0 1.8
Maximum Screen Face Velocity (blocked and minimum submer gence) (ft/s) (m/s)	3.0 0.9	3.0 0.9	3.0 0.9	3.0 0.9
Minimum Water Level (inside screens and grates)	Sufficient to cover 1.5 ft (0.5 m) of open screen			
Maximum Approach Flow Velocity (ft/s) (m/s)	0.36 0.11	0.36 0.11	0.36 0.11	0.36 0.11
Sump Loss Coefficient, C_L	1.2	1.2	1.2	1.2

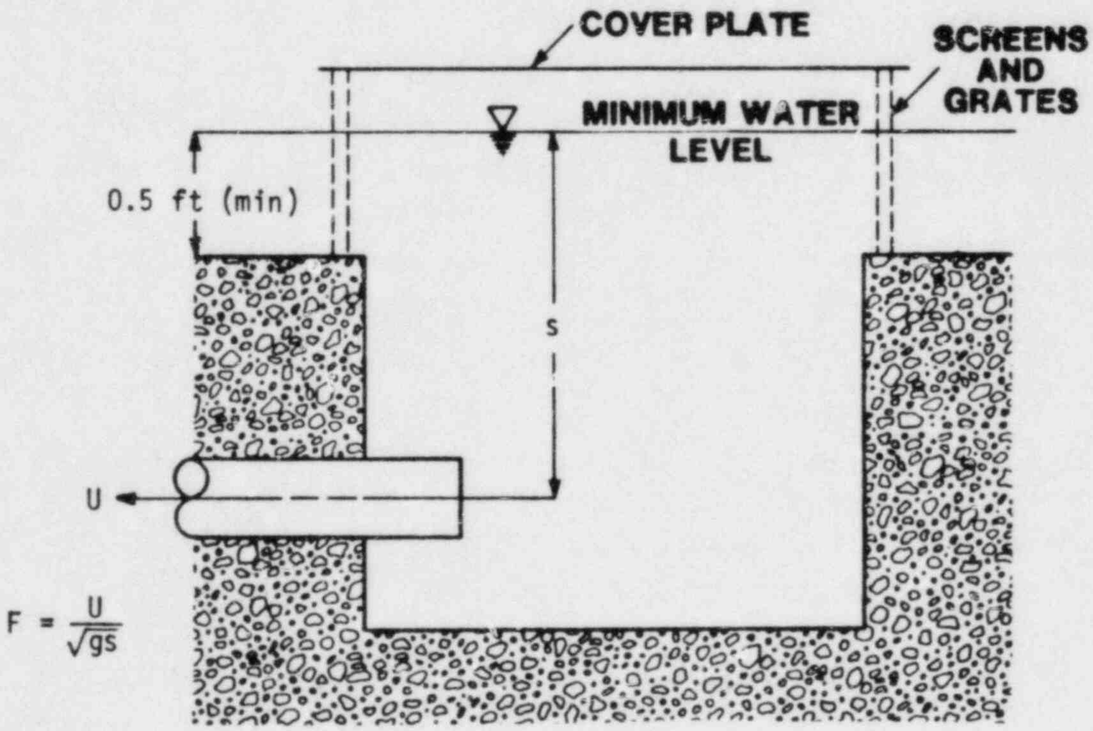


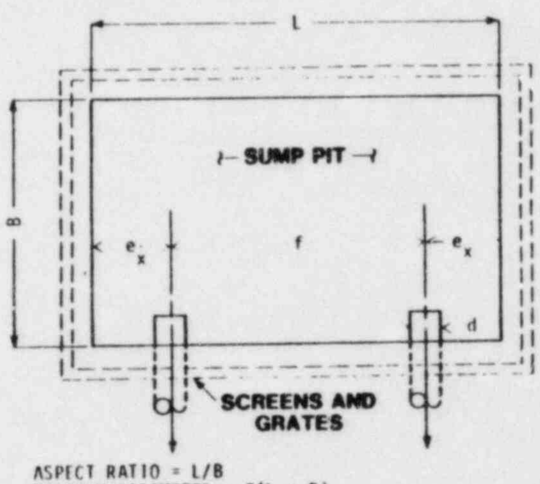
Table A-3

Geometric Design Envelope Guidelines

Sump Outlet		Size			Pump Inlet Position*						Screens and Grates	
		Aspect Ratio	Min. Perimeter (ft)	(m)	e_y/d	$(B - e_y)/d$	c/d	b/d	f/d	e_x/d	Min. Screen Area (Plane face) (ft^2)	(m^2)
Horiz	Dual	1 to 5	36	11	≥0	≥3	≥1.5	≥1	≥4	1.5** or	75	7
	Single	1 to 5	16	4.9	≤1				-	>1.5	35	3.3
Vert	Dual	1 to 5	36	11	1.5** or	≤1	≥0	≥1	≥4	1.5** or	75	7
	Single	1 to 5	16	4.9	>1.5		≤1		-	>1.5	35	3.3

* Preferred location

** Dimensions are always measured to pipe centerline.



ASPECT RATIO = L/B
MINIMUM PERIMETER = 2(L + B)

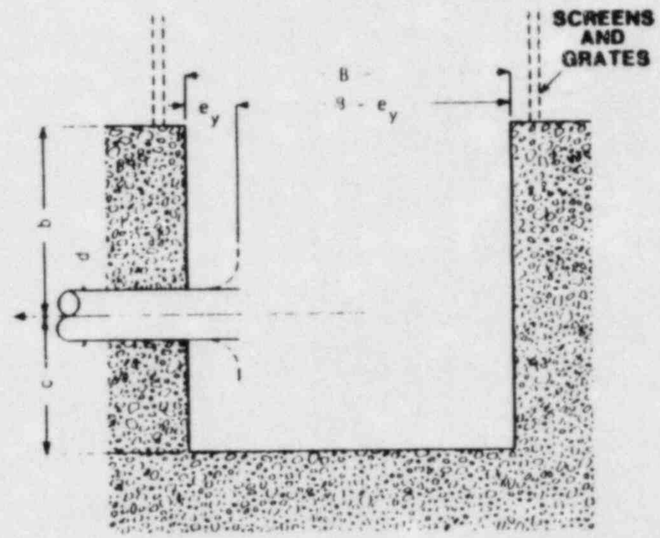


Table A-4

Additional Considerations Related to Sump Size and Placement*

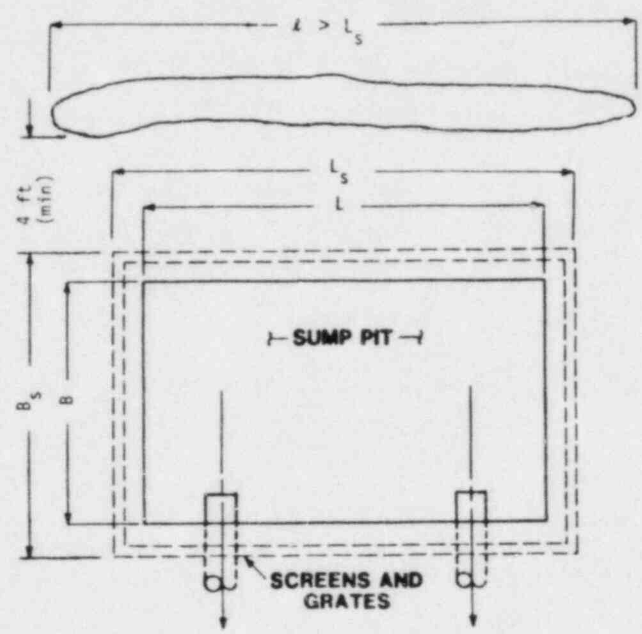
1. Aspect ratio (see Table A-3).

2. Minimum sump perimeter (see Table A-3).

3. The clearance between the screens/grates 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).

4. 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.

*These additional considerations are provided to ensure that the experimental data boundaries (upon which Tables A-2 and A-3 are based) resulting from the experimental studies at Alden Research Laboratory are noted.



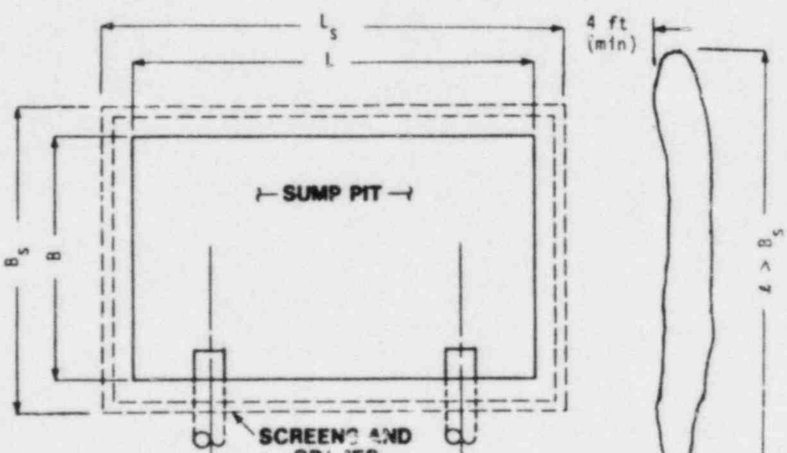


Table A-5

Design Guidelines* for Screen, Grate, and Cover Plate

1. Minimum plane face screen area should be obtained from Table A-3.

2. Minimum height of open screen 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. Gratings 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 grates should be 6 inches (15.2 cm) or less.

7. A solid cover plate should be mounted above the sump and should extend to the screens and grates. 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).

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

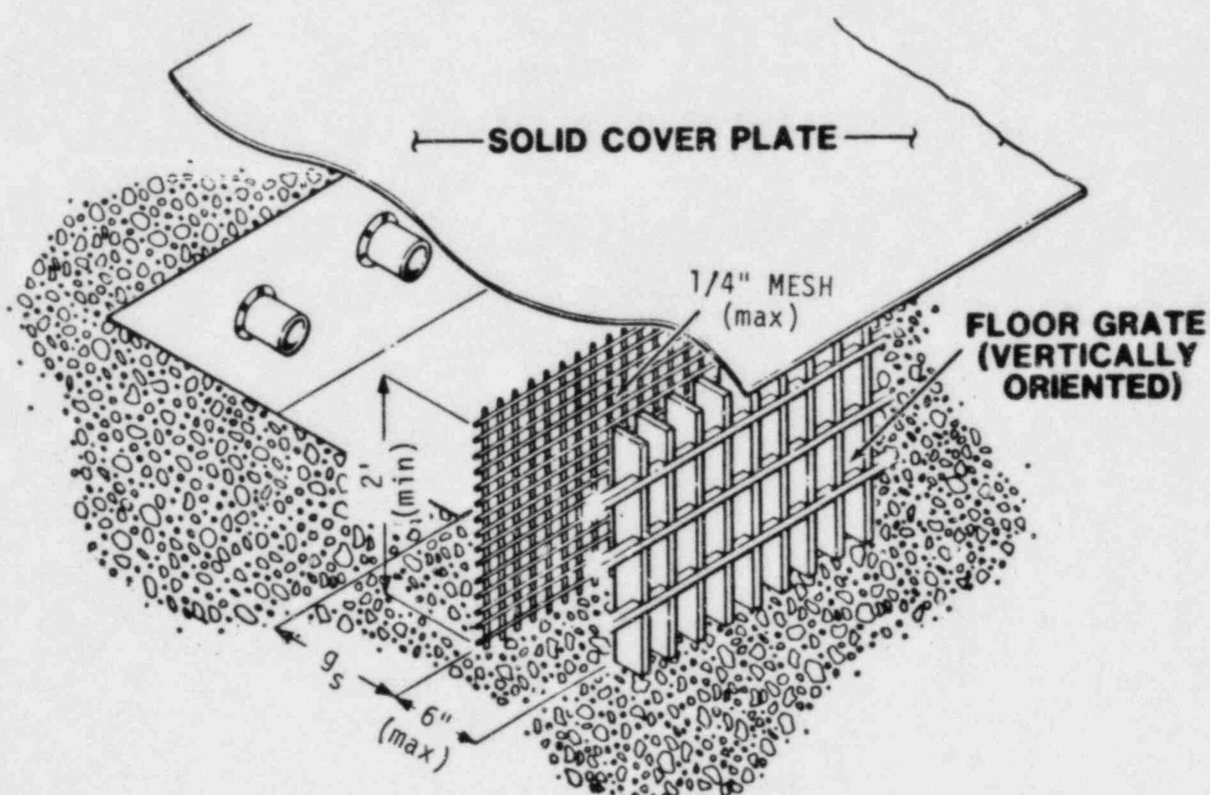

Table A-6

Guidelines 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 ≧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 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.

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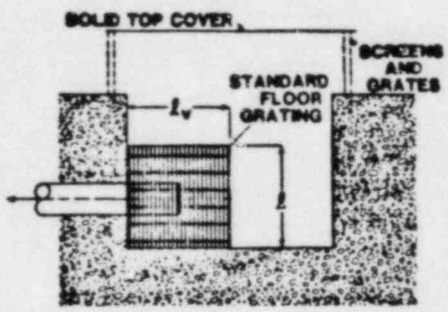
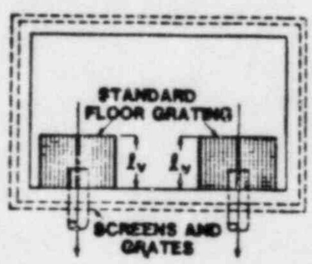
*Tests on these types of vortex suppressors at Alden Research Laboratory have demonstrated their capability to reduce air ingestion to zero even under the most adverse conditions simulated.





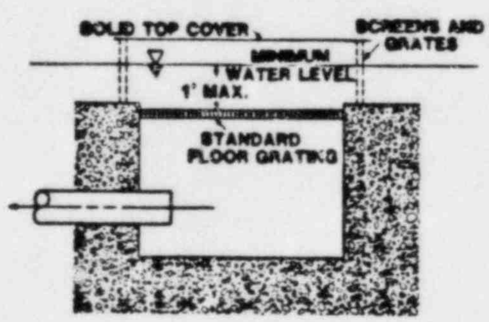


Table A-7

Debris Assessment

CONSIDERATION	EVALUATE
1. Debris generator (Pipe breaks & location as identified in SRP Section 3.6.2)	· Major pipe breaks & location · whip & pipe impact · Break jet expansion envelope (This is the <u>major</u> debris generator)
2. Expanding jets	· Jet expansion envelope · Piping & plant components targeted (i.e., steam generators) · Jet forces on insulation · Insulation that can be destroyed or dislodged by blowdown jets · Sump screen 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 & sump location · Heavy (or "sinking") debris · Floating debris · Neutral buoyancy debris
5. Screen blockage effects (impairment of flow or NPSH margin)	· Screen design · Sump location · Water level under post-LOCA conditions · Flow requirements
Key elements for assessment of debris effects	· Estimated amount of debris that can reach sump · Screen blockage · ΔP across blocked screens

Table A-8

Criteria for Zero Potential for Screen Blockage

	Criterion 1	Criterion 2	Criterion 3
V_{fb}	0	0	any value
V_{rm}	0	any value	any value
V_{cc}	any value	any value	any value
V_{hg}	0	0	0
U_f	any value	$\leqq$2.0 ft/sec (60 cm/sec)	$\leqq$0.15 ft/sec (4.5 cm/sec)
H_w	$\geqq H_s$	$\geqq H_s$	$\geqq H_s$

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 $H_w \geqq H_s$, 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 on the smaller of (1) the screen area that is shielded from prompt transport of insulation below the minimum water level or (2) the smallest immediate, total approach-flow area to the screens/grates below the minimum water level

Table A-9

Criteria for an Acceptable Potential for Screen Blockage when Fluid Transport of Insulation Debris is Possible

Criterion 1	Criterion 2	Criterion 3
$V_{fb} = 0$ $V_{rm} > 0$ V_{cc} = any value $V_{hg} = 0$	V_{fb} = any value V_{rm} = any value V_{cc} = any value $V_{hg} = 0$	V_{fb} = any value V_{rm} = any value V_{cc} = any value $V_{hg} = 0$
$U_f > 2.0$ ft/sec (61 cm/sec)	$U_f > 2.0$ ft/sec (61 cm/sec)	0.15 ft/sec $\leq U_f \leq$ 2.0 ft/sec (4.5 cm/sec) (61 cm/sec)
$H_w \geq H_s$	$H_w \geq H_s$	$H_w \geq H_s$
$A_m = L_m \times P_s \leq 0.75A'_s$	$A_m = L_m \times P_s \leq 0.75A'_s$	$A_m = 0$
	$A_e = A'_s - A_m$ $U_e = Q/A_e$	$A_e = A'_s$ $U_e = Q/A_e$
	$\Delta P = C_o t^m U_e^{n*}$ $\tilde{V}_{fb} = A_e \left(\dfrac{0.5\Delta P_{pm}}{C_o U_e^n} \right)^{\frac{1}{m}}$	$\Delta P = C_o t^m U_e^{n*}$ $\tilde{V}_{fb} = A_e \left(\dfrac{0.5\Delta P_{pm}}{C_o U_e^n} \right)^{\frac{1}{m}}$
	$V_{fb} \leq \tilde{V}_{fb}$	$V_{fb} \leq \tilde{V}_{fb}$

V_{fb}, V_{rm}, V_{cc}, V_{hg}, H_w, H_s, U_f (see Table A-8)

A_m = screen area blocked by metallic insulation

A_e = effective unblocked screen area (area available to flow, Q)

A'_s = screen area shielded from prompt transport of insulation and below the minimum water level

U_e = effective flow velocity

Q = volume flow rate

L_m = largest linear dimension for reflective metallic insulation

P_s = sump screen perimeter or its equivalent

t = nominal thickness of shredded insulation on sump screen; t is the volume of as-fabricated fibrous insulation dislodged and shredded divided by the effective unblocked screen area

ΔP_{pm} = net positive suction head margin, the difference between the net positive suction head available and the net positive suction head required.

$\tilde{V}_{fb}$ = volume of fibrous insulation that leads to a head loss of 0.5 ΔP_{pm}

$\Delta P = C_o t^m U_e^n$ head loss equation for shredded insulation with nominal thickness t; ΔP in ft, t in ft, U_e in ft/sec.

fiberglass: $C_o = 1080$, $m = 1.3$, $n = 2.3$

mineral wool: $C_o = 230$, $m = 1.4$, $n = 2.0$

* This head loss equation was suggested by the results of experiments reported in Reference 4. Other expressions developed from adequately conducted experiments can be used.

REFERENCES

1. U.S. Nuclear Regulatory Commission, "Containment Emergency Sump Performance (Technical Findings Related to USI A-43)," NUREG-0897, May 1983.

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, "Buoyancy, Transport, and Head Loss of Fibrous Reactor Insulation," NUREG/CR-2982 (SAND82-7205), November 1982.

5. 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.

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

7. 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.

8. 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.

9. 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.

10. 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.

NOTE: NUREG-series documents are available from the NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

VALUE/IMPACT STATEMENT

A separate value/impact 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 value/impact analysis 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 the proposed revision to Regulatory Guide 1.82.

VALUE/IMPACT STATEMENT FOR USI A-43,

CONTAINMENT EMERGENCY SUMP PERFORMANCE

VALUE/IMPACT STATEMENT FOR USI A-43,

CONTAINMENT EMERGENCY SUMP PERFORMANCE

NUREG-0869

CONTENTS

VALUE-IMPACT ANALYSIS

USI A-43, "CONTAINMENT EMERGENCY

SUMP PERFORMANCE"

I. The Proposed Action(s)

A. Summmary of Problem and Proposed Action

Unresolved Safety Issue (USI) A-43 deals with safety concerns related to containment emergency sump performance during the post-loss of coolant accident (LOCA) period wherein long-term recirculation cooling must be maintained to prevent core melt. These safety concerns can be summarized in the following question:

> "In the recirculation mode, will the sump design provide water to the residual heat removal (RHR) pumps in sufficient quantity, and will this water be sufficiently free of LOCA-generated debris and air ingestion so as not to impair pump performance, while providing adequate net positive suction head (NPSH)?"

These concerns have been addressed in three parts, namely:

a. Sump hydraulic performance under post-LOCA adverse conditions such as air ingestion, elevated temperatures, break and drain flow, etc.

b. LOCA-generated debris arising from the break jet destroying large quantities of insulation, this insulation debris being transported to the sump screen(s), and the resulting screen blockage being sufficient to reduce NPSH significantly below that required to maintain adequate pumping.

c. The performance capability of RHR and containment spray system (CSS) pumps to continue pumping when subjected to air ingestion, debris ingestion and effects of particulates.

These concerns have been investigated on a generic basis, and the findings can be summarized as follows:

a. Measurements in extensive, full-scale sump hydraulic tests have shown low levels of air ingestion (i.e., 1-2%) and demonstrated that vortex observations cannot be used to quantify sump performance. These experimental results have been used to develop sump hydraulic design guidelines and acceptance criteria.

b. Generic plant insulation surveys and development of debris calculational methods have shown that debris effects are dependent on the type and quantities of insulation employed and plant layout. The results also show that the 50% screen blockage guidance provided in the current 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.

c. 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 ($\leq$ 2%) will not significantly degrade pumping performance, and that the types of pumps employed will tolerate ingestion of insulation debris and other types of post-LOCA particulates, which can pass through sump screens.

These results reveal a significantly lesser safety concern with respect to vortex formation and sump hydraulic effects than previously hypothesized but a greater concern for loss of recirculation cooling capability from debris effects. Thus, the results warrant the recommendations set forth. The following actions are proposed:

1. Revise the NRC Standard Review Plan (SRP), Section 6.2.2, "Containment Heat Removal Systems," and Section 6.3, "Emergency Core Cooling Systems" to incorporate the technical findings and sump design review guidelines set forth in NUREG-0897. This action will provide for review consistency based on the extension data base acquired for the resolution of USI A-43, and can remove the need for "in-plant" sump tests or sump model tests.

2. Revise RG 1.82, to reflect the findings contained in NUREG-0897, "Containment Emergency Sump Performance," July 1982. In particular, the 50% screen blockage guidance should be removed and replaced with a requirement for plant-specific debris evaluations based on the technical findings described in NUREG-0897.

3. Operating plants should be assessed for determination of the extent of debris blockage potential and based on the outcome of those plant analyses, action should be taken to correct unacceptable conditions.

   The debris blockage concern stems from use of certain insulations such as mineral wool and/or fiberglass which can lead to excessive sump screen blockages with attendant loss of recirculation pump NPSH margin. The USI A-43 surveys (for 19 plants) have shown that some older plants employ such insulations, and plant-specific calculations reveal (i.e., Maine Yankee) that excessive screen blockage could occur. Thus operating plants [Pressurized Water Reactors (PWRs) in particular] should be required to provide their assessment of debris induced screen blockage utilizing the criteria and guidelines set forth in Appendix A of the revised RG 1.82.

Generally speaking, it is not expected that Boiling Water Reactors (BWRs) will encounter a debris blockage problem, nor will PWRs that extensively use reflective metallic insulations. The unencapsulated fibrous insulations are believed to present the principal debris problem and it is estimated that six to ten PWRs may require some type of corrective action. BWR insulation debris problems are not expected to arise since BWRs make extensive use of reflective metallic insulation and the design of the suppression pool vent's missile cover is such that it will block insulation migration to the pool.

B. Need for Proposed Action(s)

The need for the proposed actions is as follows:

1. Issuance of the proposed revision to SRP Section 6.2.2 is needed to correct previous sump review criteria which are not supported by current findings (i.e., judgment of sump hydraulic acceptability principally on vortex formation). SRP Section 6.2.2 has been revised to reflect findings from full-scale sump tests and generic plant studies, the net result is the clear identification of the need to assess sump

hydraulic performance, LOCA generated debris effects (i.e., sump screen blockage) and recirculation pump performance under post-LOCA conditions. Current findings do not support the need for continued in-plant sump tests (per RG 1.79) or more sump model tests (w/o measurement of air ingestion)

2. RG 1.82 requires revision to incorporate the results of two years of sump testing and generic plant studies. There is also the need to correct deficiences in the current RG 1.82, such as the 50% screen blockage rule. Generic plant calculations addressing LOCA-generated debris effects have shown that the 50% blockage rule can be excessive in some plants, and non-conservative for other plants. Continued use (without revision) of this regulatory guideline 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 revised RG 1.82 to provide guidance and acceptance 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. An assessment of the possible extent of debris blockage effects is needed since previous reviews have been based on the current RG 1.82 50% blockage guidance and (as noted above) this has been shown to result in non-conservative assessments in some cases. Based on USI A-43 evaluations, it is concluded that the debris blockage question is dependent on the type of insulation employed (i.e., unencapsulated fibrous insulations

transport and block screens) and containment design, or layout. Although these A-43 evaluations show plant-specific concerns (i.e., the Maine Yankee* plant insulation debris assessment), they do not suggest the existence of a widespread problem warranting immediate action. Newer plants employ mostly reflective metallic, or encapsulated insulations--some of the older plants employ a higher percentage of unencapsulated, or fibrous type insulations (see Table 1). BWRs appear to use predominantly reflective metallic insulation.

Since it is not clear which of the operating plants [or near term operating licenses (NTOLs)] have addressed the debris blockage question adequately, it is recommended that a systematic plant evaluation for all operating reactors be undertaken utilizing the guidance provided in Appendix A of the revised RG 1.82. If such evaluations reveal plants where corrective actions should be undertaken, then such cases should be pursued accordingly. A generic letter requesting such evaluations would be used for such implementation.

C. Value-Impact of the Proposed Actions

1. Risk Analysis Results

A risk analysis was performed to assess the effects of loss of the containment emergency sump; for example: due to LOCA debris blockage. Three plants and their corresponding probabilistic risk assessments (PRAs) were selected, these being: Crystal River,

*It should be noted that Maine Yankee staff have indicated that some insulation replacement was planned and also the possibility of installing additional debris capture screens is being considered.

Table 1

Types of Insulation Used Within the Primary Coolant[1] Systems Shield Wall in Plants Surveyed

Plant	Types of Insulation and Quantity in Ft[2]					
	Reflective Metallic	Totally Encapsulated	Mineral Fiber/Wool Blanket	Calcium Silicate Block	Unibestos Block	Fiberglass
Oconee 3	14,500	--	--	--	--	300
Crystal River 3	12,500	715	150	--	--	--
Midland 2	15,750	--	--	--	--	4,400
Haddam Neck	450	--	--	--	14,200[3]	150
R. E. Ginna	--	--	1,000	22,300	2,800	--
H. B. Robinson	--	--	--	3,800	21,800	--
Prairie Isl 1&2	19,200	--	--	--	--	500
Kewaunee	5,200	--	--	--	4,500	--
Salem 1	6,700	13,200[2]	--	--	--	--
McGuire 1&2	18,000	--	--	--	--	--
Sequoyah 2	18,500	--	--	--	--	--
Maine Yankee	2,900	--	6,700	5,500	--	100
Millstone 2	6,300	9,100	1,300	7,200	--	--
St. Lucie 1	1,500	--	--	17,300	--	--
Calvert Cliffs 1&2	4,400	7,300	--	--	--	--
Arkansas 2	6,300	7,400	--	--	--	200
Waterford 3	2,300	15,500	--	--	--	--
Cooper	30%	70%	--	--	--	--
WPPSS 2	100%	--	--	--	--	--

[1]Tolerance is ±20%

[2]Both totally and semi-encapsulated Corablanket is used; however, inside containment only totally encapsulated is employed.

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

IREP-PRA; Calvert Cliffs, RSSMAP-PRA; and Surry, RSS-PRA. The PRA event trees were reanalyzed to determine the effects of sump loss following a large LOCA. Whereas previously these event trees assumed availability of the sump, this analysis assumes total sump failure for 50% of the large LOCAs; the resulting core melt frequencies and release category frequencies were then computed. The 50% assumption reflects the fact that not all large LOCAs will result in total sump failure. Table 2 summarizes results obtained.

The release category frequencies were converted to public dose via the airborne pathway utilizing the following values:

Release Category	Core Melt Release (man-rem)
PWR 1	5,400,000
PWR 2	4,800,000
PWR 3	5,400,000
PWR 4	2,700,000
PWR 5	1,000,000
PWR 6	150,000
PWR 7	2,300

These values were derived using the CRAC code and assuming the guidelines and quantities of radioactive isotopes used in the Reactor Safety Study (WASH-1400), the meteorology at a typical mid-West site (Byron-Braidwood), a uniform population density of 340 people per square-mile (which is an average of all U. S. nuclear power plant sites) and no evacuation of population and are based on a 50 mile release radius model.

The release values used are similar to the those shown in WASH-1400, but with some modifications to arrive at a reference plant value. Generally speaking, release categories 2 and 3 were the major contributors to public dose. Averaging the change in calculated public dose (or change between w/o sump loss and w/sump loss) results in an average increase of public dose of 65 man-rem/plant year due to loss of the sump (see also Table 2).

TABLE 2, SUMMARY OF RISK ASSESSMENT CALCULATIONS

Calculated Core Melt Frequency $(\text{plant-yrs})^{-1}$:

	Base Case w/o Sump Loss	Adjusted Case w/Sump Loss[3]	Increase in Core Melt Frequency
Crystal River	3.7×10^{-4}	4.2×10^{-4}	5×10^{-5}
Calvert Cliffs[1]	2×10^{-3}	2.05×10^{-3}	5×10^{-5}
Calvert Cliffs[2]	4.0×10^{-4}	4.5×10^{-4}	5×10^{-5}
Surry	5×10^{-5}	1×10^{-4}	5×10^{-5}

Calculated Public Dose (man-rem/plant-year):

	Base Case w/o Sump Loss	Adjusted Case w/Sump Loss[3]	Calculated Increase In Public Dose
Crystal River	926	983	57
Calvert Cliffs[1]	7,617	7,698	81
Calvert Cliffs[2]	653	734	81
Surry	52	108	56
		Average =	65

(1) Calvert Cliffs w/o AFW improvement

(2) Calvert Cliffs w/AFW improvement

(3) These values are based on the assumption that only 50% of the large LOCAs lead to sump loss.

Reference: Probabilistic Risk Assessment of Unresolved Safety Issue A-43, September 1982, by Science Applications Inc. (Ref. 12)

Given the results of the risk analyses summarized above, and utilizing "averaged" numbers, the following quantities can be calculated:

$$\text{Public Dose Averted} = \left(65 \frac{\text{man-rem}}{\text{plant-yr}}\right) \left(\frac{23 \text{ years outstanding}}{\text{plant life}}\right)$$
$$= 1487 \text{ man-rem/plant}$$

The avoided on-site dose (due to core melt) can also be calculated as:

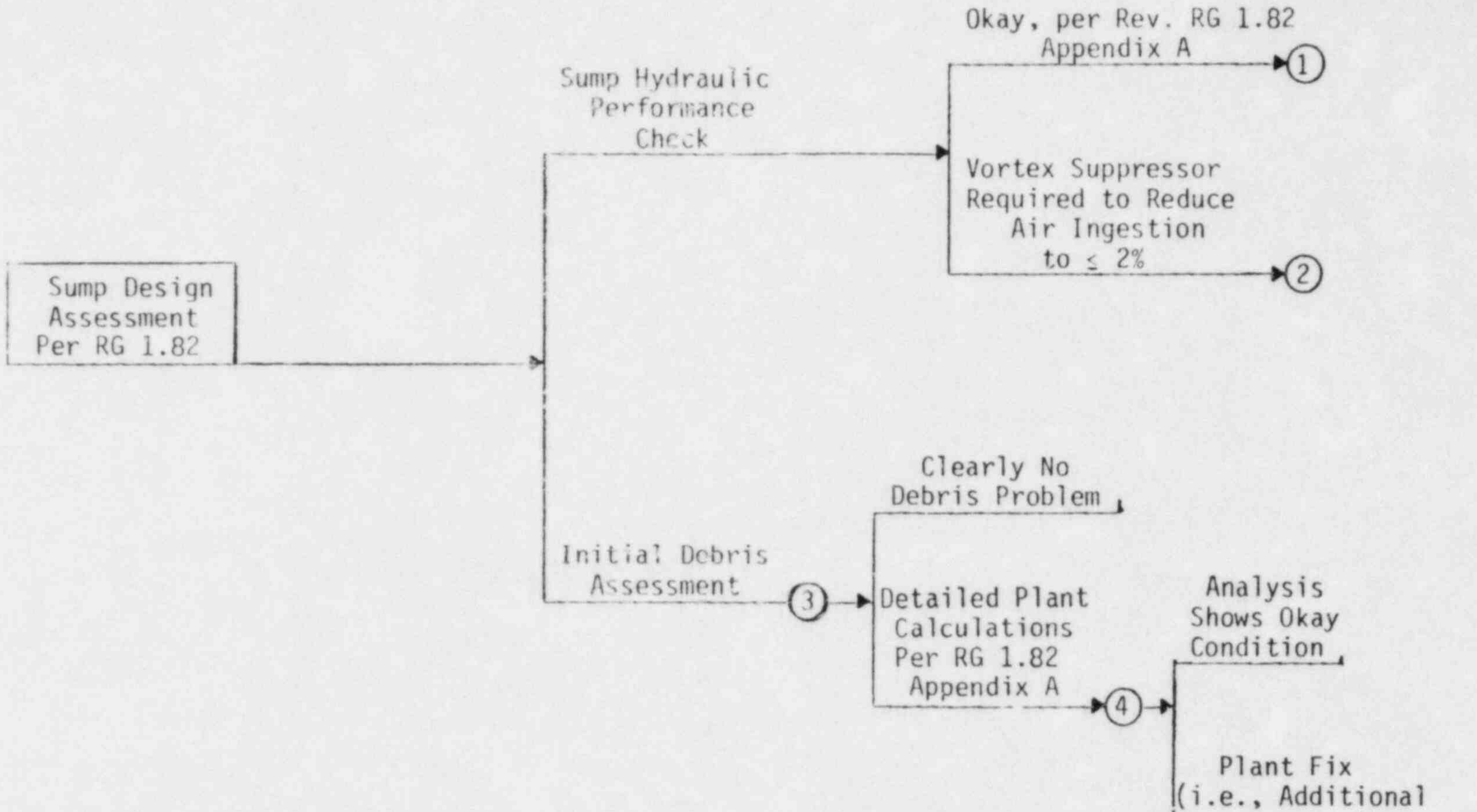
$$\text{Avoided On-site Dose} = \left(19{,}860 \frac{\text{man-rem}}{\text{accident-yr}}\right) (\Delta f \text{ core-melt}) (23 \text{ yrs})$$
$$= 23 \text{ man-rem/plant}$$

The potential to avoid a public dose of 1500 man-rem/plant is a significant risk/consequence finding.

2. Industry Impact

Industry impact will vary from plant-to-plant. As stated previously, not all plants will be found to have large quantities of non-encapsulated fibrous insulations (the type which could lead to severe screen blockage and loss of NPSH). To facilitate understanding of potential impacts on industry, Figure 1 is provided. Also shown on Figure 1 are the estimated costs which might be incurred, depending on the extent of the problem. The major impact would result if the determination is made that large quantities of insulation must be replaced (e.g., 2,000-7,000 ft^2 of insulation). Actual determination of quantities and location of insulation requiring replacement would reduce the impact; also use of alternative methods such as intermediate screens should be

FIGURE 1. IDENTIFICATION OF IMPACTS RESULTING FROM PROPOSED CHANGES TO RG 1.82 AND SRP 6.2.2



ESTIMATED COSTS (per plant)
1 = $2,500
2 = $10,000 to $30,000
3 = $5,000
4 = $10,000 to $15,000
5 = $50,000 to $80,000
6 = $25,000 to $700,000

evaluated. The sections which follow provide more insight into the expected impacts.

a. Given the guidelines set forth in revised RG 1.82, the initial sump hydraulic design evaluation will take very little time through use of acceptance criteria tables. If sump design and operating conditions show less than 2% air ingestion potential, and if predominantly reflective metallic insulation is employed, the methods and tables as contained in Appendix A of RG 1.82 will allow a sump design review in less than 1 man-day. A conservative impact would be 1 man-week of professional effort (est. $2,500), see also 1 in Figure 1.

b. If the results of the sump hydraulics evaluation show a need for fixes (i.e., the need for vortex suppressors to reduce the estimated air ingestion), an additional impact occurs. Design, fabrication and installation of a vortex suppressor consisting of floor grating materials (either horizontally mounted, or formed into a cage) is estimated to cost $10,000 to $30,000 depending on installation complexity, see also 2 in Figure 1.

c. The initial debris assessment will need to consider the types, quantities, methods of fabrication and installation, mechanical attachments, and hygroscopic characteristics of the insulation employed on primary and secondary system piping, reactor pressure vessel, and major components (e.g., steam generators, reactor coolant pumps, pressurizer, tanks, etc.) that can become targets of expanding "break" jet(s) occurring in the primary coolant system. For plants employing essentially all reflective metallic insulations [which can better survive break jet loads and will transport only at high water velocities ($\geq$ 2.5 ft/sec)], this assessment can be done quickly. Assuming that the licensee knows what insulations are within containment, such an evaluation should not require over 1 man-week's effort. Reporting the results to NRC might require another week. An impact of $5,000 is estimated, see also (3) in Figure 3.

d. If Step "c" indicates a need for detailed plant calculations to determine quantities of debris generated, what fraction gets to the sump, screen blockage effects, etc., an estimated time of 3 to 4 man-weeks is projected based on the level of effort expended for the generic plant-specific studies carried out for USI A-43 resolution.

An impact of $10,000 to $15,000 per detailed plant analysis is projected, see also (4) in Figure 1. Since it is expected that this debris related analysis will be required for some of the older plants employing unencapsulated mineral wool, or fibrous insulations. A four to six plant estimate is projected, which would result in a total industry impact of $40,000 to $90,000.

e. If plant-specific calculations reveal unacceptable sump screen debris blockage, design modifications then need to be considered. Possible solutions include utilization of intermediate screens which would intercept the debris deposition on the local sump screen occurs, encapsulation of insulation, shielding structures to prevent break jet impingement, etc. Use of interception screens is estimated to cost $50,000 to $80,000 (see also (5) in Figure 1) and is based on a potential fix discussed that four to six plants may require corrective actions and this would place the total industry impact at $200,000 to $480,000.

It should be noted that the detailed plant calculations (per RG 1.82, Appendix A guidance) will reveal location and quantities of insulation requiring attention. The existence of such problem areas does not imply the need to replace all the insulation. A more cost effective alternative would be selective insulation replacement.

f. The most severe impact would result if it were found necessary to replace all fibrous insulation, see also 6 in Figure 1. This case is considered in this value-impact analysis since it represents the severest fiscal impact.

Table 3 illustrates cost estimates for insulation replacement for several plants to illustrate plant dependency and is based on cost and exposure data derived from actual man-hours and exposures for steam generator replacement at the Surry Units 1 and 2, plus follow-up discussions with onsite staff. Two additional cost estimates were developed from contacts with the insulation suppliers noted. Estimated cost impacts can range from $25,000 to $700,000 depending on insulation quantities requiring total replacement for the plant in question. Given the costs shown in Table 3, an "averaged" cost impact of $550,000/plant will be assumed for value-impact calculations which follow.

In addition to labor costs, the radiological exposure impact must be considered and is derived from the values shown in Table 2. The dependence on level of insulation replace must required is evident, with a range of 10 to 100 man-rem being forecast. An insulation replacement exposure impact of 50 man-rem/plant was therefore assumed for the value-impact analyses which follows.

TABLE 3

ESTIMATES OF INSULATION REPLACEMENT COSTS AND ASSOCIATED EXPOSURES

Plant	Unencapsulated Insulation (Ft^2)	Cost Est.[1] No. 1 ($\$ \times 10^3$)	Cost Est.[2] No. 2 ($\$ \times 10^3$)	Cost Est.[3] No. 3 ($\$ \times 10^3$)	Estimated[4] Exposure (man-rem)
Salem Unit 1	13,200	281	238	660	99
Maine Yankee	6,700	142	121	335	47
Ginna	1,000	21	18	50	8
Millstone Unit 2	1,300	28	23	65	10

[1]These costs are derived from Surry Units 1 and 2 steam generator removal and reinstallation data, and discussions with onsite staff. A "per-unit" cost of $\$0.85/ft^2$ for replaced insulation was derived and labor costs of $25.00/hr were assumed.

[2]Telephone estimates from New England Insulation Company (Maine Yankee has employed this firm) were: $\$3/ft^2$ to remove, $\$11/ft^2$ to fabricate new panels, $\$3\text{-}5/ft^2$ to install.

[3]Telephone estimates of $\$35\text{-}50/ft^2$ for mirror-insulation fabrication and installation were obtained from Diamond Power who supplies such insulation. A value of $\$50/ft^2$ was employed.

[4]Exposure data were derived from Surry 1 and Surry 2 data. Discussions with Surry site staff indicates that a 50 man-rem exposure level for insulation replacement is realistic if the job is pre-planned. An equivalent dose of 7×10^{-3} man-rem/ft^2 of insulation replaced can be derived.

In addition, the assumption is made that plant shutdowns solely to replace undesirable insulations will not be required (thus purchase of replacement power has not been included) since the risk/consequence calculations do not support shutting down operating plants. Based on discussions with Maine Yankee staff, the plant owner indicated corrective actions (e.g., installation of additional screens and selective removal of mineral wool insulation) could be carried out during scheduled refueling outages. If necessary, the work involving replacement of insulation could be performed at two or more refueling outages.

With respect to new plants, or those applicants in the Operating License (OL) review cycle, the sump hydraulic performance data contained in NUREG-0897 and related references are a "value" since: (a) the extensive sump hydraulics data base (which has been incorporated into the revised RG 1.82) can remove the need for additional sump model tests which have previously cost $50,000 to $150,000 per plant, and (b) can remove the need for "in-plant" tests designed to demonstrate sump hydraulic design adequacy by visual observations for air-entraining vortex formation.

3. NRC Operation

The "impact" of proposed changes with respect to staff review time will be minimal making use of the guidelines contained in Appendix A of the revised RG 1.82. NUREG-0897 and supporting references provide additional technical information which will assist the staff reviewer. It is estimated that less than 1 man-week of staff review time would be required (Estimated cost = $1500/plant).

The experimental data and generic plant information and calculations contained in NUREG-0897 (and supporting references) represents a funding investment of approximately $3.0 million on the part of the NRC and the Department of Energy and this information is a "value" to both the NRC and industry. This extensive data base provides a basis for eliminating unnecessary in-plant testing, or sump model tests.

4. Other Government Agencies

Since sump design review and acceptance are carried out solely by NRC staff, no impact on other government agencies is projected.

5. Public

The "value" to the public would be avoidance of public dose from addditional core melts, due to sump failure, if the recommendations are adopted. Based on the PRA results noted in Table 1, the calculated average public dose which could be averted is 1,500 man-rem/plant. Given the projection of six to ten plants which may have a debris blockage problem, the total public "value" is 9,000 to 15,000 man-rem potential reduction.

6. Overall Value-Impact of the Proposed Actions

These value-impact results can be summarized as follows:

Avoided Public Dose = 1,500 man-rem/plant
Avoided Plant Site Dose = 23 man-rem/plant
Estimated Implementation Dose = 50 man-rem/plant
Core Melt Frequency Decrease = 5×10^{5}/(plant-yrs)
Core Melt Reduction = 11.5×10^{-4} accidents/plant remaining life

The estimated present-worth of plant cost due to a core melt accident is \$1.65 billion. Therefore, the proposed changes provide a means to avoid an accident cost of:

$$\begin{aligned}\text{Avoided Accident Cost} &= \text{(Core Melt Reduction) (Plant Cost)} \\ &= (11.5 \times 10^{-4})\ (\$1.65 \times 10^{9}) \\ &= \$1.9 \times 10^{6}/\text{plant}\end{aligned}$$

or nearly \$2 million per plant.

These "values" can be compared with estimated "impacts" of \$100,000 for plant fixes (such as supplemental debris screens) to \$400,000 to \$700,000 per plant for replacement of large quantities of troublesome insulation.

The overall impact on operating reactors is shown in Figure 2, which follows the same implementation actions and costs identified in Figure 1. Assuming 75 OLs, the estimated impact for determining the extent of the screen debris blockage problem is \$0.7 million; another \$3.0 million is projected for plant fixes (or retrofits).

The above value-impact data can be viewed as a ratio of value gained versus cost to implement (or a V-I ratio), which is defined as:

$$\text{V-I} = \frac{\text{Avoided Public Dose}}{\text{Cost of Implementation}}$$

For operating plants, this ratio computes to:

$$\text{V-I} = \frac{\text{(1500 man-rem/plant) (5 plants)}}{\$(.7 + 2.5)\text{M}} = 2344\ \frac{\text{man-rem}}{\$\text{Million}}$$

FIGURE 2. ESTIMATE OF OVERALL INDUSTRY COST IMPACT FOR OLs

Determines Extent of Debris Problem ← → Plant Backfits Required

75 OLs

→ 25 BWRs = (25)($5000/plant) = $125,000

→ 50 PWRs

→ (40)($7500/plant) = $300,000

→ (10)($22,500/plant) = $225,000

Total = $650,000

→ (5)($80,000/plant) = $400,000

(Alternate fixes, Rg-additional screens)

→ (5)($500,000/plant)* = $2,500,000

Total = $2.9 Million

*for replacement of large amounts of troublesome insulations.

$$\text{or} = (1500)(5)/(.7 + .4) = 6818 \frac{\text{man-rem}}{\$\text{Million}}$$

The reader is cautioned against over optimism regarding values to be gained versus impacts from these V-I's. There are uncertainties attributable to costs and avoided doses. However, the V-I computed value supports moving forward with the proposed actions. Generic studies have already identified one plant having potential debris blockage problems. A systematic determination of the extent of the problem is warranted from safety consideration aspects. The V-I ratio, based on a single problem plant assumption would be:

$$\text{V-I} = \frac{(1500 \text{ man-rem/plant})}{(\$.525\text{M})} = 2857 \frac{\text{man-rem}}{\$\text{Million}}$$

The radiological impact versus local plant radiological gain (50 man-rem incurred versus 23 man-rem avoided) should be considered offsetting due to the averaging methods used in these analyses and associated uncertainties.

II. Technical Approach

A. Technical Alternatives

a. Proceed with the proposed recommendation, including backfit correction to operating plants, only where plant-specific analysis reveal a change is needed.

b. Issue NUREG-0897 and the proposed changes to SRP Section 6.2.2 and RG 1.82, but with implementation required only on new plants.

c. Issue NUREG-0897 and associated references for information only, but take no other action.

B. Discussion and Comparison of Technical Alternatives

a Proceeding with the proposed recommendations will incur the values and impacts discussed in Section I.C and as summarized in Section I.C.6. A value-impact ratio of 2,300-6,800 man-rem avoided per million dollars to backfit has been computed. It is clear (with the exception of massive insulation backfits) that the benefits outweigh the impacts. Maintaining the current versions of RG 1.82 and SRP Section 6.2.2 runs contrary to technical findings presented in NUREG-0897 and associated references which reveal a much less severe sump air ingestion picture, but also reveal a deficiency in current assessments of debris blockage effects on sump operation.

b. Accepting the proposed changes to RG 1.82 and SRP Section 6.2.2, for implementation on those plants where a Safety Evaluation Report (SER) will be issued following implementation of the proposed changes is the minimal route which should be considered. The technical findings presented in NUREG-0897 and references, reveal a significantly different picture than previously hypothesized and show that the prior accepted levels of risk may not exist in some plants. However, ignoring the implications of the results of the A-43 debris blockage effects with respect to OLs and NTOLs is not acceptable. Analyses of emergency core cooling systems (ECCS) have assumed an operable sump based on the 50% blockage assumption, current evaluations indicate screen blockage potential for plants using unencapsulated fibrous insulations.

c. To continue to use the current RG 1.82 and SRP Section 6.2.2, would ignore the experimental data base and plant analyses which clearly point out the need for these recommended changes. This is not an acceptable alternative since A-43 plant-specific calculations have shown that the 50% screen blockage guidance in the current RG 1.82 can result in erroneous and non-conservative plant results.

C. Decision on Technical Approach

Given the positive-finding from the value-impact analysis (see Section I.C.6) and the need to correct current regulatory technical deficiencies, the recommendation is therefore made to revise SRP Section 6.2.2 and RG 1.82 which reflect the technical findings contained in NUREG-0897, and also backfit the licensing positions set forth in Appendix A of RG 1.82 to operating plants and NTOLs which have received a SER.

III. Plan for Implementation

A. Safety or Environmental Significance of Proposed Action

As noted previously, the estimated avoided public dose is approximately 1500 man-rem/plant. Since it is projected that six to ten PWRs may be found to have debris blockage potential that requires corrective action, proposed changes have the potential for avoiding a 9,000 to 15,000 man-rem public dose due to a blocked (or failed) sump.

B. Decision on Plan for Implementation

Given the technical findings and these value-impact assessments, the recommendation is made to proceed with the recommended changes to SRP Section 6.2.2 and RG 1.82, both of which incorporate the technical findings contained in NUREG-0897 and related references. This will provide the necessary safety assurance for new plant designs, and as a "forward fit" would represent a minimum impact route.

With respect to operating plants, and NTOLs for which an SER has already been issued, the applicant or licensee should be required to show an acceptable sump design utilizing the guidelines and criteria set forth in Appendix A of the revised RG 1.82. In particular, the applicant/licensee should demonstrate that potential LOCA generated debris effects do not result in excessive screen blockage leading to loss of NPSH margin for the recirculation pumps. It is expected that a few of the older plants employing unencapsulated fibrous insulations will require follow-up on corrective measures which may be submitted. It is also expected that PWRs would incur the major impact of reanalysis via Appendix A of the revised RG 1.82.

Implementation would follow issuance of the revised SRP Section 6.2.2 and RG 1.82 following receipt and consideration of public comments on the proposed revisions, and resubmitted to the CRGR for review prior to implementation. The generic letter would result in a two step operation which:

a. Identifies the extent and severity of the problem, and proposed fixes if required.

b. Establishes a schedule for implementation which minimizes impact on plant operation.

Although BWRs are not expected to incur insulation debris problems, operating BWRs should be required to show that plant insulations employed will not result in unacceptable debris blockages for the RHR suction intakes utilizing the methods outlined in the revised RG 1.82, Appendix A, or an equivalent alternate.

A "draft" generic letter which would implement these requirements for OLs and NTOLs is provided at the end of this value-impact analysis.

IV. Procedural Approach

A. Procedural Alternatives

a. Issue NUREG-0897, for information only; take no other action.

b. Implement use of the revised SRP Section 6.2.2 and RG 1.82, for only those plants not having a SER at time of implementation, or a "forward-fit" only 1983. Issue NUREG-0897.

c. Require that all plants (including operating plants and NTOL's) evlauate sump design adequacy per Appendix A of the revised RG 1.82, and in particular assess the sump screen blockage effects associated with LOCA generated debris.

B. Value-Impact of Procedural Atlernatives

a. The "impact" of alternative (a) is zero since no changes are implemented. There is a "value" associated with the information provided in NUREG-0897 and related references. This option is, however, unacceptable since deficiences have been identified in the current version of RG 1.82 with respect to debris assessment.

b. The "value" associated with alternative (b) is related to the data contained in NUREG-0897 (and references) which can replace in-plant and sump model tests. The "impacts" are associated with designing for avoidance of sump air ingestion, use of less troublesome insulations, etc. Since option (b) is a forward fit, plant cost impacts should be minimal. An "impact" of $10,000 to $15,000/plant is estimated (see also Section I.C.2).

c. Alternative "c", which is the recommended action, would have a "value" of an avoided accident dose of 1,500 man-rem/plant (over remaining plant life) with an attendant impact of $100,000 to 550,000/plant (see again Section I.C.6). In addition, avoidance of any accident situation which could lead to core melt should be pursued. Failure of the sump for those accidents requiring long-term recirculation capability can lead to core melt. The calculated reduction in core melt frequency attributables to sump failure was 5×10^{-5}/reactor-year.

C. Decision on Procedural Approach

Given the results of this "value-impact" assessment on the procedural approaches, the recommendation is made to proceed with Alternative "c"; namely, require that plants show by analysis that sump design is adequate and that debris blockage effects do not lead to excessive sump screen blockage per Appendix A of the revised RG 1.82. As noted previously, the severity of the identified problem should be reviewed by both applicant and staff prior to embarking in extensive fixes.

V. Statutory Considerations

A. NRC Authority

Since 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. Furthermore, the recommendation to require applicants to demonstrate adequate sump performance falls within the statutory authority of the NRC to regulate and assure the safe operation of nuclear power plants.

B. Need for National Environmental Policy Act (NEPA) Statement

The proposed changes and potential plant retrofits do not warrant a NEPA statement.

VI. Summary and Conclusions

1. Issue the revised SRP Section 6.2.2 and RG 1.82 for public comment.

2. Issue NUREG-0897 for public comment. This staff report summarizes USI A-43 technical findings.

3. After resolution of public comments and the Committee to Review Generic Requirements approval to proceed issue the revised RG 1.82 and SRP Section 6.2.2 and require that new plants, operating plants, NTOL applicants* assess their sump design and debris blockage potential as outlined in Appendix A of the Revised RG 1.82, or by other equivalent methods.

4. Upon receipt of the findings submitted under Item 3, and staff evaluations, determine what (if any) corrective plant actions may be required.

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*If the staff Safety Evaluation Report has already been issued at the time the RG 1.82 revision is issued in effective form, the assessment for a NTOL would be made after issuance of the OL.

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DRAFT GENERIC LETTER

TO: All Licensees of Operating Reactor Plants and Holders of Construction Permits

Gentlemen:

SUBJECT: ASSESSMENT OF CONTAINMENT EMERGENCY SUMP PERFORMANCE DURING THE RECIRCULATION MODE (GENERIC LETTER)

This letter is to provide you with the latest information and methodology developed by the staff in the process of resolving questions related to containment sump performance (USI A-43 "Containment Emergency Sump Performance"). We request that you review the containment sump design and installation in your facility(ies) in order to reconfirm containment emergency sump operability in the post-LOCA period wherein recirculation must be maintained. Our principal concern relates to LOCA generated debris which could lead to severe screen blockage and result in loss of pump net positive suction head (NPSH). The technical aspects of this issue (namely sump hydraulic performance, debris effects and pump operation under adverse conditions) have been extensively studied and the results are contained in NUREG-0897, "Containment Emergency Sump Performance." Non-encapsulated fibrous insulations appear to pose the potential for excessive screen blockage. These technical findings have been incorporated into NRC's Standard Review Plan, Section 6.2.2, Revision 4 and RG 1.82, Revision 1. Appendix A of RG 1.82 provides evaluation guidelines which can be used to evaluate sump performance. These revised documents form the criteria for licensing reviews and will be applied to plants for which the NRC Safety

Evaluation Report is not yet issued. Copies of these documents are provided for your use.

For operating reactors and for those plants where the NRC Safety Evaluation Report (SER) has already been issued, an assessment of the containment emergency sump performance must be made in accordance with the requirements enumerated below.

(1) Sump hydraulic performance, including an assessment of levels of air ingestion. (i.e., ≤ 2 volume % is considered acceptable.)

(2) The amount of insulation debris that might be generated by the postulated pipe break(s), the transport of such debris to the sump screen and attendant screen blockage which might occur. The resulting screen blockage calculated must be used to determine estimated head loss for estimating NPSH impact. The previously employed 50% blockage guidance no longer applies and should not be used as an assumption in your calculations.

(3) The available NPSH margin for the recirculation pumps when the combined effects of Items (a) and (b) are considered, must be sufficient to assure acceptable pump performance during the required period of operation.

Appendix A of RG 1.82, Revision 1 provides an acceptable method, or guidelines, for carrying out the analyses requested above.

As indicated above, the primary purpose of your evaluation is to reconfirm using the latest available information and methodology that NPSH margins are consistent with the NPSH requirements established in your previous safety analysis. If you calculations identify deficiencies in performance or operability, your response must identify corrective measures or plant modifications that are needed in order to assure acceptable sump performance.

Accordingly, licensees of operating plants and applicants who have received an OLSER, should submit their evaluation of sump performance and available NPSH within 150 days from the date of this letter, or submit within 30 days an alternate schedule for your analysis and response to this generic letter. If corrective measures or modifications are identified as being needed, your proposed schedule for implementing any modifications should also be provided.

This request for information has been approved by the Office of Management and Budget under clearance number ___________ which expires ___________.

Sincerely,

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

BACKGROUND AND SUMMARY OF MINUTES OF MEETINGS OF

COMMITTEE TO REVIEW GENERIC REQUIREMENTS (CRGR)

REGARDING UNRESOLVED SAFETY ISSUE (USI) A-43 RESOLUTION

(CRGR MEETING NOS. 26 AND 28)

BACKGROUND AND SUMMARY OF MINUTES OF MEETINGS OF

COMMITTEE TO REVIEW GENERIC REQUIREMENTS (CRGR)

REGARDING UNRESOLVED SAFETY ISSUE (USI) A-43 RESOLUTION

(CRGR MEETING NOS. 26 AND 28)

NUREG-0869

BACKGROUND AND SUMMARY OF MINUTES OF MEETINGS OF COMMITTEE TO REVIEW GENERIC REQUIREMENTS REGARDING UNRESOLVED SAFETY ISSUE A-43 RESOLUTION (REF. CRGR MEETING NOS. 26 AND 28)

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 the 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 entitled, "Containment Heat Removal Systems" and Regulatory Guide 1.82 entitled, "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 which question four key assumptions in the Office of Nuclear Reactor Regulation (NRR) calculations of averted public dose and state that DEDROGR staff feels that the dose is 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 have another meeting with the CRGR.

The staff is proceeding with additional calculations to estimate large loss of coolant accident frequencies based on a detailed piping and break probability analysis and estimating the percentage of these breaks which could lead to sump screen blockage. These calculations are being done in parallel with obtaining public comment on the USI A-43 proposed resolution and will be factored into the final evaluations. Interim feedback was provided to the CRGR on February 28, 1983 (Ref. 1).

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 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 which 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 (R.G.) 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 R.G. 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.

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 man-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 unencapusulated 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-rem) that could be averted from the estimated six to ten plants that are expected to need modifications. Key assumptions in this NRR analysis are:

(1) The expected value of large LOCA (greater than 6" diameter pipe) incidence is 10^{-4} per reactor-year.

(2) For those plants having sufficient 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}/RY.

(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-rem per plant per year was to 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 their 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.

REFERENCES

(1) H. R. Denton to V. Stello, Jr. memorandum dated February 28, 1983, "Response to CRGR Comments on USI A-43."

(2) V. Stello, Jr. to W. J. Dircks memorandum dated January 11, 1983, "Minutes of CRGR Meeting Number 28."

(3) 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)."

(4) V. Stello, Jr. to W. J. Dircks memorandum dated December 10, 1982, "Minutes of CRGR Meeting Number 26."

(5) 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."

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NUREG-0869, is comprised of the following documents:

(a) The proposed RG 1.82, Revision 1
(b) The Value/Impact Analysis for USI A-43, "Containment Emergency Sump Performance"
(c) Background and Summary of CRGR Meeting Minutes Regarding USI A-43 Resolution (CRGR Meeting Nos. 26 and 28)

and has been assembled to facilitate obtaining "For Comment" feedback on the positions developed for resolution of USI A-43. There are no licensing requirements contained in NUREG-0869, and it should be clearly noted that this "For Comment" report will not be used as interim requirements.

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