UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

November 3, 2004

The Honorable Nils J. Diaz
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Diaz:

SUBJECT: SUMMARY REPORT - 516th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, OCTOBER 7-9, 2004, AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

During its 516th meeting, October 7-9, 2004, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following reports, letters, and memoranda:

REPORTS:

Reports to Nils J. Diaz, Chairman, NRC, from Mario V. Bonaca, Chairman, ACRS:

- Draft NUREG-XXXX, "The Report on the Independent Verification of the Mitigating Systems Performance Index (MSPI) Results for the Pilot Plants," dated October 14, 2004
- Safety Evaluation of the Industry Guidelines Related to Pressurized Water Reactor Sump Performance, dated October 18, 2004
- Proposed Resolution of Generic Safety Issue 185, "Control of Recriticality Following Small-Break LOCAs in PWRs," dated October 22, 2004
- Report on an "Overview of Differences in Nuclear Safety Regulatory Approaches and Requirements Between United States And Other Countries," dated November 3, 2004

LETTER:

Letter to Luis A. Reyes, Executive Director for Operations, NRC, from Mario V. Bonaca, Chairman, ACRS:

- Review of ACR-700 Pre-Application Safety Assessment Report dated October 14, 2004

MEMORANDA:

Memoranda to Luis A. Reyes, Executive Director for Operations, NRC, from John T. Larkins, Executive Director, ACRS:

- Draft Final Regulatory Guide DG-1085, "Standard Format and Content of Decommissioning Cost Estimates for Nuclear Power Reactors," and NUREG-1713, "Standard Review Plan for Decommissioning Cost Estimates for Nuclear Power Reactors," dated October 12, 2004
- Draft NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," dated October 12, 2004
- Seabrook Station, Unit No. 1 - Advisory Committee on Reactor Safeguards Review of Stretch Power Uprate Amendment (TAC No. MC2364), dated October 13, 2004
- Anonymous Letter Concerning the TRACE Computer Code Development and Review Practices, dated October 14, 2004

OTHER:

- Letter to David O'Brien, Commissioner, Vermont Department of Public Service, from Mario V. Bonaca, Chairman, ACRS, Subject: Vermont Yankee Extended Power Uprate Request, dated October 18, 2004

HIGHLIGHTS OF KEY ISSUES

1. Safety Evaluation of the Industry Guidelines Related to Pressurized Water Reactor (PWR) Sump Performance

The Committee met with representatives of the NRC staff, its contractors, and the Nuclear Energy Institute (NEI) to discuss the staff's draft safety evaluation (SE) of the industry guidelines associated with the resolution of Generic Safety Issue (GSI)191, "Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at Pressurized Water Reactors (PWRs)." The ACRS Subcommittee on Thermal-Hydraulic Phenomena reviewed this matter during a meeting on September 22-23, 2004. The Committee heard the staff's presentations, and it also heard comments from industry, and it recommended that the SE not be issued in its present form. Industry expressed concern about the implementation of the guidance report as modified by the staff's SE.

Committee Action

The Committee issued a report to the NRC Chairman on this matter, dated October 18, 2004. In its letter, the Committee concluded that both the SE and the NEI guidance report contain too many technical faults and limitations to provide the basis for a defensible and robust long-term solution to GSI 191. The Committee described some of the faults and limitations in the present technical knowledge base that need to be addressed so that acceptable guidance can be developed. The Committee also recommended that the staff clarify the level of quality and acceptability that it expects from industry-sponsored experiments to fill in the gaps in the analytical and experimental data base. The Committee recommended that the risk-informed approach that has been explored by the staff be extended to treat the entire sequence of phenomena during the scenario, and develop a quantitative assessment of the model uncertainties related to the phenomena that could be used in the application of the Regulatory Guide 1.174 process to this situation. Based on the above, the Committee recommended that the SE not be issued in its current form.

2. Pre-Application Safety Assessment Report for the Advanced CANDU 700 (ACR-700) Design

The Committee held discussions with representatives of the NRC staff and the Atomic Energy of Canada Limited (AECL) Technologies regarding the staff's pre-application safety assessment report (PASAR) for the 700 Mwe ACR-700 design.

The ACR-700 pre-application review is being conducted in two phases. Phase 1 was a series of familiarization meetings designed to provide the staff with a general overview of the ACR-700 design. Phase 1 of the ACR-700 pre-application review has been completed and the ACR-700 pre-application review is currently in Phase 2. The objective of Phase 2 is to provide more information about the ACR-700 design to facilitate the staff's review of 13 focus topics, and to provide feedback to AECL prior to their application for standard design certification.

AECL's view is that the ACR-700 design will meet the applicable NRC regulations with the overall objective is to achieve high confidence in the acceptability of the design certification application. Currently, the staff is concluding that based on the materials submitted by AECL, including responses to requests for additional information, the applicant will need to pursue a number of technical issues in more detail.

Committee Action

The Committee issued a letter to the EDO on this matter dated October 14, 2004, commending the staff on an excellent job on its pre-application review of the focus topics, for which the staff had identified technical, regulatory, and policy issues.

3. Proposed Recommendations for Resolving GSI-185, "Control of Recriticality Following Small-Break LOCAs in PWRs"

The Committee met with representatives of the NRC staff and its contractors regarding the proposed recommendations for resolving GSI-185. The Committee discussed the proposed draft report on this issue that was prepared by the Office of Nuclear Regulatory Research (RES) to provide the technical basis for closure of the GSI. Overall, the low probability of this event, coupled with the limited consequences that were calculated by RES, support the conclusion that GSI-185 can be considered resolved for all operating PWRs.

Committee Action

The Committee issued a report on this matter and concluded that GSI-185 can be closed without imposition of any new regulatory requirements for all existing PWRs.

4. Mitigating Systems Performance Index Program

The Committee held discussions with the staff members of RES and the Office of Nuclear Reactor Regulation (NRR) regarding the status of the Mitigating Systems Performance Index (MSPI) program.

The NRC presentation on the MSPI was made by Pat Baranowsky and Don Dube, RES, and Stuart Richards, NRR. The purpose of this project is to develop a risk-informed indicator that includes unreliability and safety system unavailability (SSU). The MSPI is a measure of the deviation of actual plant system unavailability and component unreliabilities from historical baseline values, where each element is weighted by plant-specific risk importance measures. The MSPI eliminates known problems with the existing SSU Indicator, accounts for unavailability and unreliability of a system, weighted relative to their risk-importance, uses plant PRA models, identifies changes in equipment performance and variations, and uses data consistent with other methods.

Committee Action

The Committee wrote a report, dated October 14, 2004, concluding that the MSPI is substantially superior to the group of safety system unavailability performance indicators, which it replaces. The report recommends that draft NUREG-XXXX be issued, its recommendations implemented, and the process for incorporating the MSPI into the Reactor Oversight Process continue.

5. Technology Neutral Framework for Future Plant Licensing

The Committee held discussions with the NRC staff regarding the regulatory structure for new plant licensing, Part 1: technology-neutral framework. Previously, the staff had identified seven policy issues fundamental to licensing non-light water reactor designs. Four of these issues would be included in the development of the framework document for future plant licensing. These four issues are the definition for defense-in-depth, the use of probabilistic approach to establish the licensing basis, the use of scenario-specific source terms for licensing decisions, and the advisability of revision of the emergency planning zone. The Commission has approved the staff's approach for these four issues.

Two issues for which the Commission requested the staff to provide further details are the issue of requiring modular reactor designs account for the integrated risk posed by multiple reactors, and the issue of functional containment performance standards. The Commission disapproved the staff's recommendation related to the issue of international codes and standards. Potential new policy issues could include the level of safety, security, and selective implementation. The staff plans to issue a draft SECY-paper that includes a draft framework with recommendations on the new policy issues by December 31, 2004.

Committee Action

The staff's briefing was provided for information only. The Committee plans to review the proposed draft SECY-paper during the December 2-4, 2004 ACRS meeting.

6. Assessment of the Quality of the NRC Research Projects

RES is required to have an independent evaluation of the quality of its research programs. This evaluation is mandated by the Government Performance and Results Act (GPRA) and needs to be in place during the next fiscal year. The Committee has agreed to assist RES in assessing the effectiveness and utility of the NRC research programs. The Committee has previously approved the strategy for the review of the quality of selected research projects. This strategy is to be tried during FY 2004 and refined in FY 2005. During the October 7-9, 2004 ACRS meeting, the Committee discussed the assessment of the quality of the NRC research projects on Sump Performance and on MACCS Code.

Committee Action

The Committee plans to discuss the draft report on assessment of the quality of the research projects on Sump Blockage and on MACCS Code during November 2-4, 2004 ACRS meeting.

7. Divergence in Regulatory Approaches and Requirements Between U.S. and Other Countries

In an April 28, 2003 Staff Requirements Memorandum (SRM), resulting from the April 11, 2003 meeting with Advisory Committee on Reactor Safeguards (ACRS), the Commission stated that "In the course of its routine activities of reviewing and advising the Commission on reactor issues, the Committee should explore and consider other international regulatory approaches. Where there are significant differences in regulatory approaches and requirements, The Commission should be informed." Dr. Nourbakhsh, ACRS Senior Staff Engineer has prepared a report to be used by the ACRS in responding to the Commission. During the October 7-9, 2004 ACRS meeting, the Committee discussed the report, prepared by Dr. Nourbakhsh, regarding divergence in regulatory approaches between U.S. and other Countries. The Committee approved a report to transmit this report to the Commission.

Committee Action

The Committee issued a report, dated November 3, 2004, transmitting the report on differences in regulatory approaches and requirements between U.S. and other countries to the Commission. The Committee will endeavor to keep the Commission informed of significant differences in regulatory requirements of other countries that comes to its attention.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS/EDO COMMITMENTS

- The Committee considered the EDO's October 1, 2004 response to the Committee's July 16, 2004 concerning Generic Letter 2004-02, "Proposed Draft Final Generic Letter on Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at PWRs".

The Committee had recommended issuance of the generic letter, and it was issued on September 14, 2004. The Committee also recommended that staff should continue to perform confirmatory research in areas where the technical basis of the guidance is uncertain, and on issues that are not directly addressed by the guidance proposed by the Nuclear Energy Institute (NEI). The EDO replied that the staff intends to "continue confirmatory research consistent with your recommendations." However, the EDO declined to initiate any new research to confirm the technical basis in the guidance because it "would not be timely for the resolution of GSI-191". As discussed in its letter regarding the draft SE for the NEI Guidance Report, the Committee continues to believe that additional research needs to be performed to support the technical basis for resolution of GSI-191.

OTHER RELATED ACTIVITIES OF THE COMMITTEE

During the period from September 8, 2004, through October 6, 2004, the following Subcommittee meetings were held:

- Thermal Hydraulic Phenomena Subcommittee - September 22-23, 2004

The Subcommittee reviewed the staff's final safety evaluation report on the industry guidelines related to resolution of GSI-191, "Assessment of Debris Accumulation on PWR Sump Performance." In addition, the Subcommittee reviewed the final staff resolution of GSI-185, "Control of Recriticality Following Small-Break LOCAs in PWRs."

- Planning and Procedures - October 6, 2004

The Subcommittee discussed proposed ACRS activities, practices, and procedures for conducting Committee business and organizational and personnel matters relating to ACRS and its staff.

LIST OF MATTERS FOR THE ATTENTION OF THE EDO

- The Committee forwarded to the EDO an anonymous letter that was sent to two ACRS members concerning the development, verification, and validation of the TRACE thermal-hydraulic system analysis code. The letter was very similar in tone to an anonymous email that was sent to one of these ACRS members in March, 2004. That email was also forwarded to the EDO for his information.

PROPOSED SCHEDULE FOR THE 517th ACRS MEETING

The Committee agreed to consider the following topics during the 517th ACRS meeting, to be held on November 4-6, 2004:

- Proposed Rule Language for Risk-Informing 10 CFR 50.46
- Proactive Materials Degradation Assessment Program
- Proposed Rule on Post-Fire Operator Manual Actions
- Grid Reliability Issues and Related Significant Operating Events
- Status of Early Site Permit Reviews
- Assessment of the Quality of Selected NRC Research Projects
- Plant License Renewal Subcommittee Report

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

CERTIFIED

Date Issued: 11/22/2004
Date Certified: 11/29/2004

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October 7-9, 2004

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- Review of ACR-700 Pre-Application Safety Assessment Report dated October 14, 2004

MEMORANDA:

Memoranda to Luis A. Reyes, Executive Director for Operations, NRC, from John T. Larkins, Executive Director, ACRS:

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APPENDICES

516th ACRS Meeting
OCTOBER 7-9, 2004

CERTIFIED

MINUTES OF THE 516th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
OCTOBER 7-9, 2004
ROCKVILLE, MARYLAND

The 516th meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on October 7-9, 2004. Notice of this meeting was published in the *Federal Register* on September 22, 2004 (65 FR 56800) (Appendix I). The purpose of this meeting was to discuss and take appropriate action on the items listed in the meeting schedule and outline (Appendix II). The meeting was open to public attendance. There were no written statements or requests for time to make oral statements from members of the public regarding the meeting.

A transcript of selected portions of the meeting is available in the NRC's Public Document Room at One White Flint North, Room 1F-19, 11555 Rockville Pike, Rockville, Maryland. Copies of the transcript are available for purchase from Neal R. Gross and Co., Inc. 1323 Rhode Island Avenue, NW, Washington, DC 20005. Transcripts are also available at no cost to download from, or review on, the Internet at http://www.nrc.gov/ACRS/ACNW.

ATTENDEES

ACRS Members: ACRS Members: Dr. Mario V. Bonaca (Chairman), Dr. Graham B. Wallis (Vice Chairman), Mr. Stephen L. Rosen, (Member-at-Large), Dr. George E. Apostolakis, Dr. F. Peter Ford, Dr. Thomas S. Kress, Dr. Dana A. Powers, Dr. Victor H. Ransom, Dr. William J. Shack, and Mr. John D. Sieber. For a list of other attendees, see Appendix III.

I. Chairman's Report (Open)

[Note: Dr. John T. Larkins was the Designated Federal Official for this portion of the meeting.]

Dr. Mario V. Bonaca, Committee Chairman, convened the meeting at 8:30 a.m. and reviewed the schedule for the meeting. He summarized the agenda topics for this meeting and discussed the administrative items for consideration by the full Committee.

II. Safety Evaluation of the Industry Guidelines Related to Pressurized Water Reactor (PWR) Sump Performance (Open)

[Note: Mr. Ralph Caruso was the Designated Federal Official for this portion of the meeting.]

The Committee met with representatives of the NRC staff, its contractors, and the Nuclear Energy Institute (NEI) to discuss the staff's draft safety evaluation (SE) of the industry guidelines associated with the resolution of Generic Safety Issue (GSI) 191, "Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at Pressurized Water Reactors (PWRs)." The Committee heard the staff's presentations, and it also heard comments from industry, and it recommended that the SE not be issued in its present form. Industry expressed concern about the implementation of the guidance report as modified by the staff's SE.

Committee Action

The Committee issued a report to the NRC Chairman on this matter, dated October 18, 2004. In its letter, the Committee concluded that both the SE and the NEI guidance report contained too many technical faults and limitations to provide the basis for a defensible and robust long-term solution to GSI 191. The Committee described some of the faults and limitations in the present technical knowledge base that needed to be addressed so that acceptable guidance can be developed. The Committee also recommended that the staff clarify the level of quality and acceptability that it expects from industry-sponsored experiments to fill in the gaps in the analytical and experimental data base. The Committee recommended that the risk-informed approach that has been explored by the staff be extended to treat the entire sequence of phenomena during the scenario, and develop a quantitative assessment of the model uncertainties related to the phenomena that could be used in the application of the Regulatory Guide 1.174 process to this situation. Based on the above, the Committee recommended that the SE not be issued in its current form.

III. Pre-Application Safety Assessment Report for the Advanced CANDU 700 (ACR-700) Design (Open)

[Note: Dr. Medhat El-Zeftawy was the Designated Federal Official for this portion of the meeting.]

Dr. Thomas Kress, Future Plant Designs Subcommittee Chairman, stated that the purpose of this meeting was to hold discussions with representatives of the NRC staff and Atomic Energy of Canada Limited (AECL) Technologies regarding the staff's pre-application safety assessment report (PASAR) for the 700 Mwe advanced CANDU reactor (ACR-700).

Ms. Belkys Sosa, Office of Nuclear Reactor Regulation (NRR), stated that the ACR-700 is an advanced CANDU (Canada Deuterium Uranium) design that utilizes horizontal fuel channels passing through a heavy-water moderator tank. As with other CANDU designs, the ACR-700 is designed to be refueled during power operation. Other features of the reactor system, coolant pumps, U-tube steam generators, and pressurizers are similar to PWR designs in the United States.

The ACR-700 will have features that make it significantly different from operating CANDU reactors. The ACR-700 utilizes light water as coolant within the fuel channels, whereas operating CANDU reactors utilize heavy water. The ACR-700 will be designed to have a negative void reactivity coefficient so that if boiling occurs within the fuel channels, the reactor power will decrease. The negative void coefficient for ACR-700 would be achieved by using slightly enriched uranium in the fuel and neutron-absorbing dysprosium elements in the fuel assemblies. Natural uranium fuel is used in operating CANDU reactors. The reactor core will be smaller than operating CANDU reactors with fewer fuel channels.

The ACR-700 pre-application review is being conducted in two phases. Phase 1 was a series of familiarization meetings designed to provide the staff with a general overview of the ACR-700 design. Phase 1 of the ACR-700 pre-application review has been completed and the ACR-700 pre-application review is currently in Phase 2. The objective of Phase 2 is to provide more information about the ACR-700 design to facilitate the staff's review of 13 focus topics (FTZ), and to provide feedback to AECL prior to their application for standard design certification.

AECL provided its proposed plan for NRC's pre-application review of the ACR-700 in a letter dated September 26, 2002. On December 18, 2002, AECL submitted an amended plan which expanded the scope of the NRC's pre-application review to include the following FTZ:

FT1, "Class 1 Pressure Boundary Design"
FT 2, "Design-Basis Accidents and Acceptance Criteria"
FT 3, "Computer Codes and Validation Adequacy"
FT 4, "Severe Accident Definition and Adequacy of Supporting Research and Development"
FT 5, "Design Philosophy and Safety-Related Systems"
FT 6, "Canadian Design Codes and Standards"
FT 7, "Distributed Control Systems and Safety Critical Software"
FT 8, "On-Power Fueling"
FT 9, "Confirmation of Negative Void Reactivity"
FT 10, "Preparation of Standard Design Certification Docketing"
FT 11, "ACR Probabilistic Risk Assessment Methodology"
FT 12, "ACR Technology Base"
FT 13, "ACR CANFLEX Fuel Design"

The NRC staff has just completed the ACR-700 Pre-Application Safety Assessment Report (PASAR). AECL requested that the NRC give priority to FTZ 1, 3, 8, and 9. The PASAR contains the staff's assessment of each focus topic, except for FTZ 5, 10, 12, and 13. Currently, the staff is concluding that based on the materials submitted by AECL, including responses to requests for additional information, the applicant will need to pursue a number of technical issues in more detail to reach a satisfactory conclusion for design certification. The staff expects that the review and evaluation of the ACR-700 design will be more challenging and will involve the expenditure of more resources.

Mr. Glenn Archinoff, AECL, stated that the ACR-700 design will meet the applicable NRC regulations with an overall objective to achieve high confidence in the acceptability of the design certification application. Currently, the staff is concluding that based on the materials submitted by AECL, including responses to requests for additional information, the applicant will need to pursue a number of technical issues in more detail. The purpose of the pre-application review is to determine if the ACR-700 design can be certified in the U.S. in a timely manner.

Mr. Peter Boczar, AECL, outlined some of the ACR design physics toolset. These include 2-dimensional transport, lattice cell calculations, multi-group cross sections generated for ACR-700, 3-dimensional transport incremental cross sections to represent reactivity devices between fuel channels. Others include theoretically rigorous treatment for detailed assessments of modeling accuracy.

Committee Action

The Committee issued a letter to the EDO on this matter dated October 14, 2004, commending the staff on an excellent job on its pre-application review of the focus topics for which the staff identified technical, regulatory, and policy issues.

IV. Proposed Recommendations for Resolving GSI-185, "Control of Recriticality Following Small-Break LOCAs in PWRs" (Open)

[Note: Mr. Ralph Caruso was the Designated Federal Official for this portion of the meeting.]

The Committee met with representatives of the NRC staff and its contractors regarding the proposed recommendations for resolving GSI-185. The Committee discussed the proposed draft report on this issue that was prepared by the Office of Nuclear Regulatory Research (RES) to provide the technical basis for closure of the GSI. Overall, the low probability of this event, coupled with the limited consequences that were calculated by RES, support the conclusion that GSI-185 can be considered resolved for all operating pressurized water reactors.

Committee Action

The Committee issued a report on this matter and concluded that GSI-185 can be closed without imposition of any new regulatory requirements for all existing pressurized water reactors.

V. Mitigating System Performance Index Program (Open)

[Note: Mrs. Maggalean W. Weston was the Designated Federal Official for this portion of the meeting.]

Mr. John D. Sieber, Chairman of the Plant Operations Subcommittee, introduced this topic to the Committee. The Committee heard presentations by the staff of RES and NRR regarding the status of the Mitigating System Performance Index (MSPI).

NRC Staff Presentation

The NRC presentation on the MSPI was made by Pat Baranowsky and Don Dube, RES, and Stuart Richards, NRR.

The purpose of this project was to develop a risk-informed indicator that includes unreliability and safety system unavailability (SSU). The MSPI is a measure of the deviation of actual plant system unavailability and component unreliabilities from historical baseline values, where each element is weighted by plant-specific risk importance measures. The MSPI was developed to address the use of fault exposure times in the SSU performance indicator (PI), the omission of unreliability elements from previous indicators, a definition of unavailability consistent with those of the Maintenance Rule and the INPO/WANO indicators, cascading of cooling water support systems failures, and thresholds that recognize plant-specific features.

The MSPI eliminates known problems with the existing SSU Indicator, accounts for unavailability and unreliability of a system, weighted relative to their risk-importance, uses plant PRA models, identifies changes in equipment performance and variations, and uses data consistent with other methods.

The NRC staff, industry, and NEI have agreed to move forward with the MSPI implementation. They will retain the frontstops and define the minimal set of PRA requirements and issues important to the MSPI, have created a staff-industry task force to identify important PRA issues that impact the MSPI, work to delineate implementation details in the guidance documents and implement the MSPI at all sites simultaneously.

The staff will brief the Committee in the future regarding additional work.

Committee Action

The Committee wrote a letter concluding that the MSPI is substantially superior to the group of safety system unavailability performance indicators, which it replaces. The letter recommends that draft NUREG-XXXX be issued, its recommendations implemented, and the process for incorporating the MSPI into the Reactor Oversight Process continue.

VI. Technology Neutral Framework for Future Plant Licensing (Open)

[Note: Dr. Medhat El-Zeftawy was the Designated Federal Official for this portion of the meeting.]

Dr. Thomas Kress, Chairman of the Future Plant Designs Subcommittee, stated that the purpose of this meeting was to hold discussions with the NRC staff regarding the regulatory structure for new plant licensing, Part 1: technology-neutral framework.

Ms. Mary Drouin, RES, stated that the NRC has over 40 years of nuclear power plant licensing and regulating experience. Such experience has been focused primarily on light-water reactors (LWRs) with limited application to gas-cooled and advanced reactors. Advanced reactors have design and operational issues that are different from current LWR issues. However, NRC LWR experience can contribute and provide insights. The most important insights is the recognition of the value of a licensing framework applicable to reactor designs that are different from currently operating plants.

The new framework would help ensure that a structured and systematic approach will instill uniformity and consistency in the licensing and regulation of advanced reactors, particularly when addressing the unique design and operational aspects of these reactors.

In addition, the framework for current LWRs has evolved without the benefit of insights from PRA and severe accident research. The NRC staff is proposing a structured approach for a regulatory structure for future generation reactors that provides guidance on how to use PRA results and insights to help ensure the safety by focusing the regulations on where the risk is most likely while maintaining basic safety principles such as defense-in-depth and safety margin.

RES has developed a DRAFT Framework document. The objective of this document is to develop and implement a risk-informed regulatory structure for licensing of future reactors. To meet this objective, four tasks are proposed:

- Development of a technology-neutral framework for the regulatory structure,
- Formulation of proposed content of technology-neutral requirements,
- Development of guidance for applying the framework on a technology-specific basis, and
- Formulation of technology-specific Regulatory Guides.

Previously, the staff had identified seven policy issues fundamental to licensing non-light water reactor designs. Four of these issues would be included in the development of the framework document for future plant licensing. These four issues are the definition for defense-in-depth, the use of probabilistic approach to establish the licensing basis, the use of scenario-specific source terms for licensing decisions, and the advisability of revision of the emergency planning zone. The Commission has approved the staff's approach for these four issues.

The Commission requested the staff to provide further details on two issues. One is the issue of requiring modular reactor designs to account for the integrated risk posed by multiple reactors, and the second issue is regarding functional containment performance standards. The Commission disapproved the staff's recommendation related to the issue of international codes and standards. Potential new policy issues could include the level of safety, security, and selective implementation. The staff plans to issue a draft SECY-paper that includes a framework with recommendations on the new policy issues by December 31, 2004.

Dr. Thomas King, RES, stated that the technical approach for the framework document is a risk-informed approach that blends probabilistic and deterministic criteria including consideration of uncertainties using a defense-in-depth philosophy. The framework establishes protective strategies which are fundamental to safe nuclear power plant design, construction, and operation.

Mr. James Riccio, Green Peace, presented his own views regarding the safety of nuclear power plants and public perception. He stated that, in the wake of September 11, 2001, the NRC staff, in developing the framework document, should not recommend licensing new reactors without the traditional containment structure. Mr. Riccio criticized the NRC staff for not considering terrorist attacks as a major element in the development of the framework document.

Committee Action

The staff's briefing was provided for information only. The Committee plans to review the proposed draft SECY-paper during the December 2-4, 2004 ACRS meeting.

VII. Assessment of the Quality of the NRC Research Projects (Open)

[Note: Dr. Hossein Nourbakhsh was the Designated Federal Official for this portion of the meeting.]

RES is required to have an independent evaluation of the quality of its research programs. This evaluation is mandated by the Government Performance and Results Act (GPRA) and needs to be in place during the next fiscal year. The Committee agreed to assist RES in assessing the effectiveness and utility of the NRC research programs. The Committee has previously approved the strategy for the review of the quality of selected research projects. This strategy is to be tried during FY 2004 and refined in FY 2005. During the October 7-9, 2004 ACRS meeting, the Committee discussed the assessment of the quality of the NRC research projects on Sump Performance and on MACCS Code.

Committee Action

The Committee plans to discuss the draft report on assessment of the quality of the research projects on Sump Blockage and on MACCS Code during the November 2004 ACRS meeting.

VIII. Divergence in Regulatory Approaches and Requirements Between U.S. and Other Countries (Open)

[Note: Dr. Hossein Nourbakhsh was the Designated Federal Official for this portion of the meeting.]

In an April 28, 2003 Staff Requirements Memorandum (SRM), resulting from the April 11, 2003 meeting with Advisory Committee on Reactor Safeguards (ACRS), the Commission stated that "In the course of its routine activities of reviewing and advising the Commission on reactor issues, the Committee should explore and consider other international regulatory approaches. Where there are significant differences in regulatory approaches and requirements the Commission should be informed." Dr. Nourbakhsh, ACRS Senior Staff Engineer prepared a report which will be used by the ACRS in responding to the Commission. During the October 2004 ACRS meeting, the Committee discussed the report regarding divergence in regulatory approaches between the U.S. and other countries. The Committee approved the report and to transmit it to the Commission.

Committee Action

The Committee issued a report, dated November 3, 2004, transmitting the report on differences in regulatory approaches and requirements between U.S. and other countries to the Commission. The Committee will endeavor to keep the Commission informed of significant differences in regulatory requirements of other countries that comes to its attention.

X. Executive Session (Open)

[Note: Dr. John T. Larkins was the Designated Federal Official for this portion of the meeting.]

A. Reconciliation of ACRS Comments and Recommendations/EDO Commitments

[Note: Mr. Sam Duraiswamy was the Designated Federal Official for this portion of the meeting.]

- The Committee considered the EDO's October 1, 2004 response to the Committee's July 16, 2004 concerning Generic Letter 2004-02, "Proposed Draft Final Generic Letter on Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at PWRs".

The Committee had recommended issuance of the generic letter, and it was issued on September 14, 2004. The Committee also recommended that staff should continue to perform confirmatory research in areas where the technical basis of the guidance is uncertain, and on issues that are not directly addressed by the guidance proposed by the Nuclear Energy Institute (NEI). The EDO replied that the staff intends to "continue confirmatory research consistent with your recommendations." However, the EDO declined to initiate any new research to confirm the technical basis in the guidance because it "would not be timely for the resolution of GSI-191". As discussed in its letter regarding the draft SE for the NEI Guidance Report, the Committee continues to believe that additional research needs to be performed to support the technical basis for resolution of GSI-191.

B. Report on the Meeting of the Planning and Procedures Subcommittee (Open)

The Committee heard a report from the ACRS Chairman and the Executive Director, ACRS, regarding the Planning and Procedures Subcommittee meeting held on October 6, 2004. The following items were discussed:

- Review of the Member Assignments and Priorities for ACRS Reports and Letters for the October ACRS meeting

  Member assignments and priorities for ACRS reports and letters for the October ACRS meeting were discussed. Reports and letters that would benefit from additional consideration at a future ACRS meeting were also discussed.

- Anticipated Workload for ACRS Members

  The anticipated workload for the ACRS members through December 2004 was discussed. The objectives were:

  - Review the reasons for the scheduling of each activity and the expected work product and to make changes, as appropriate
  - Manage the members' workload for these meetings
  - Plan and schedule items for ACRS discussion of topical and emerging issues

- Proposed ACRS Meeting Dates for CY 2005

  Proposed ACRS meeting dates for CY 2005 are summarized below:

Meeting No.	Dates
---	January 2005 (No meeting)
519	February 10-12, 2005
520	March 3-5, 2005
521	April 7-9, 2005
522	May 5-7, 2005
523	June 1-3, 2005
524	July 6-8, 2005
---	August 2005 (No meeting)
525	September 8-10, 2005
526	October 6-8, 2005
527	November 3-5, 2005

528	December 8-10, 2005 (Changed)

These proposed meeting dates were provided to the members during the September ACRS meeting for feedback.

- ACRS Retreat in 2005

During its September 2004 meeting, the Committee discussed whether to have a retreat in 2005, and deferred its decision until after evaluating the topics proposed by the members for discussion at the retreat. Topics proposed by several members were discussed and are listed below under two categories (Process Issues and Technical Issues). The members who proposed the issues are identified in the parenthesis.

Process Issues

- Incoming Chairman's agenda (DAP)
- Alternative approach to review license renewal applications in view of Mr. Leitch's retirement (DAP)
- Should the ACRS take on fewer issues and spend more time on each? (DAP)
- Assessment of the Subcommittee structure and member assignments (DAP)
- Should the ACRS continue to defer voicing and/or documenting its concerns regarding the quality of science and engineering that goes into regulations, more importantly into the regulatory process? (DAP)
- Does the ACRS have a proactive role in nuclear safety issues? If so, how does it originate? (VHR)
- Can the ACRS better influence the NRC safety research (i.e., topics and quality) (VHR)
- Could the report writing be conducted so that a draft is available before the review starts? (This is done sometimes now). Perhaps, the review could begin by going around the table and hearing the main point of view of each member. (VHR)
- What will it take for the ACRS to recommend disapproval of a license renewal application? (TSK)
- Interactions with the NRC staff outside the formal Subcommittee and full Committee meetings (GBW)
- What are the limits to "working with the staff" (suggested by the Commission on certain issues) (GBW):
  - Acting like members of the staff, or its consultants, in developing solutions to the problems
  - Stepping into management gaps and trying to organize the staff and set objectives
  - Planning activities or strategies
  - Doing joint brainstorming or analysis of alternative approach to issues
  - Being peer reviewers
- Should we hold discussions with the staff only at formal meetings or through

formal communications, so that any criticism or disagreement is entirely out in the open (GBW)

- Should we workout anything that appears at all potentially disagreeable in private and have more apparent consensus in public? (GBW)
- Agenda for the Quadripartite meeting (should be a focused meeting - perhaps on risk-informed regulation or on divergence) (TSK)

Technical Issues

- Extended Power Uprate Issues (MVB)
- Power uprates - Is there any limit other than component capacities? Is there not a risk limit? (TSK)
- Is a design-basis accident a useful concept for future reactors? Should the concept be abandoned in favor of an examination of risk? Would not abandoning the design-basis concept lead to technology neutral regulations? (DAP)
- What should be the role (if any) of design-basis accidents in future regulations? (TSK)
- Why do we think redefining the LBLOCA size is O.K.? (TSK)
- What is the ACRS position on a technology neutral regulatory framework? (TSK):
    - risk acceptance criteria
    - appropriate F-C curves
    - defense in depth - in particular the question of whether a containment is always required
- Any preliminary concerns about design certification of ACR-700 (TSK)
- What should ACRS recommend on sump blockage? (TSK)
- What more should the ACRS do about security issues - particularly with respect to spent fuel? (TSK):
    - pool
    - dry cask storage
    - transportation
    - interim storage facility
- Shouldn't the ACRS do some more complaining about air oxidation? (TSK)
- Organizational factors (Safety Culture) (TSK/SLR)

Proposed Locations

- Baltimore, MD
- Salt Lake City, UT
- Missoula, MT
- Santa Barbara, CA
- San Diego, CA
- Datmouth, NH
- Cambridge, MA (MIT)

- St. Louis
- New Orleans
- Las Vegas, NV (Yucca Mountain)
- Phoenix
- Florida

Since we will be operating under a "continuing resolution," holding the retreat out of town will be subject to the availability of resources.

- <u>State of Vermont Request to the ACRS</u>

  On September 17, 2004, Mr. David O'Brien, NRC State Liaison Officer for Vermont, sent a letter to Dr. Bonaca, ACRS Chairman requesting, on behalf of the State of Vermont, that:

  - "The ACRS specifically review, as part of Entergy's request for extended power uprate for the Vermont Yankee Nuclear Plant, Entergy's request to change Vermont Yankee's design basis to take credit for containment overpressure to demonstrate the adequacy of its emergency core cooling and containment spray pumps, and the NRC staff's policy of granting such requests. (The reasons for the above request are included in the attachment.)

  - State of Vermont be allowed to present its case before the ACRS Subcommittee and full Committee, along with the Applicant and staff, in the Committees' deliberations of Vermont Yankee extended power uprate.

  - One or both of the Subcommittee and full Committee meetings regarding the Vermont Yankee extended power uprate be held in the vicinity of the nuclear plant."

- <u>TRACE Computer Code - Anonymous Letters</u>

  Drs. Wallis and Ransom each received a copy of an anonymous letter, alleging inconsistencies in the development of the TRACE code and the review of thermal-hydraulic codes in general by the ACRS. This letter is somewhat similar to the e-mail sent to Drs. Wallis and Ransom previously.

- <u>Member Issues</u>

  The Bethesda North Marriott Hotel (across from the White Flint Subway Station) will be opening soon. During the September meeting, there was a discussion about members staying at this hotel when they come to the ACRS meetings. However, no decision was made. There are several positive aspects to staying at this hotel, such as:

  - ACRS Office is walking distance from the hotel.

- Members do not have to depend on the subway train or other modes of transportation to come to the meeting especially in the event of inclement weather.

- If a meeting runs late (because of not completing certain agenda items), or within the allocated time, or if there is a need to start the next day meeting early, staying in a hotel close to the ACRS Office would be helpful.

- The full Committee meetings can start at 8:00 a.m., as needed, and recess early, if needed.

C. Future Meeting Agenda

Appendix IV summarizes the proposed items endorsed by the Committee for the 517th ACRS Meeting, November 4-6, 2004.

The 516th ACRS meeting was adjourned at 1:00 p.m. on October 9, 2004.

APPENDIX I

7:30 a.m. and 4:15 p.m. (e.t.) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted only during those portions of the meeting that are open to the public.

Further information regarding this meeting can be obtained by contacting the Designated Federal Official between 7:30 a.m. and 4:15 p.m. (e.t.). Persons planning to attend this meeting are urged to contact the above named individual at least two working days prior to the meeting to be advised of any potential changes in the agenda.

Dated: September 16, 2004.

Michael R. Snodderly,

Acting Associate Director for Technical Support, ACRS/ACNW.

[FR Doc. 04–21257 Filed 9–21–04; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Meeting Notice

In accordance with the purposes of Sections 29 and 182b of the Atomic Energy Act (42 U.S.C. 2039, 2232b), the Advisory Committee on Reactor Safeguards (ACRS) will hold a meeting on October 7–9, 2004, 11545 Rockville Pike, Rockville, Maryland. The date of this meeting was previously published in the **Federal Register** on Monday, November 21, 2003 (68 FR 65743).

Thursday, October 7, 2004, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

8:30 a.m.–8:35 a.m.: Opening Remarks by the ACRS Chairman (Open)—The ACRS Chairman will make opening remarks regarding the conduct of the meeting.

8:35 a.m.–10:45 a.m.: Safety Evaluation of the Industry Guidelines Related to Pressurized Water Reactor (PWR) Sump Performance (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and the Nuclear Energy Institute regarding the staff's evaluation of the industry guidelines associated with the resolution of Generic Safety Issue (GSI)–191, "Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at PWRs" and related matters.

11 a.m.–12:30 p.m.: Pre-Application Safety Assessment Report for the Advanced CANDU 700 (ACR–700) Design (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the staff's Safety Assessment Report related to the pre-application review of the ACR–700 design and related matters.

1:30 p.m.–3 p.m.: Proposed Recommendations for Resolving GSI–185, "Control of Recriticality Following Small-Break LOCAs in PWRs" (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and its contractors regarding the proposed recommendations for resolving GSI–185.

3:15 p.m.–4:45 p.m.: Mitigating System Performance Index Program (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the Mitigating System Performance Index Program.

5 p.m.–7 p.m.: Preparation of ACRS Reports (Open)—The Committee will discuss proposed ACRS reports on matters considered during this meeting. In addition, the Committee will discuss a proposed report responding to the August 25, 2004 EDO response to the May 21, 2004 ACRS letter on resolution of certain items identified by the ACRS in NUREG–1740, "Voltage-Based Alternative Repair Criteria."

Friday, October 8, 2004, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

8:30 a.m.–8:35 a.m.: Opening Remarks by the ACRS Chairman (Open)—The ACRS Chairman will make opening remarks regarding the conduct of the meeting.

8:35 a.m.–10 a.m.: Technology Neutral Framework for Future Plant Licensing (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the technology neutral framework for licensing of future plant designs.

10:15 a.m.–11:30 a.m.: Assessment of the Quality of the NRC Research Projects (Open)—The Committee will discuss the preliminary results of the cognizant ACRS members' assessment of the research projects on Sump Blockage and on MACCS code.

11:30 a.m.–12:15 p.m.: Divergence in Regulatory Approaches and Requirements Between U.S. and Other Countries (Open)—The Committee will discuss the draft Final White Paper prepared by Dr. Nourbakhsh, ACRS Senior Staff Engineer, regarding divergence in regulatory approaches and requirements between U.S. and other Countries.

1:15 p.m.–2:15 p.m.: Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open)—The Committee will discuss the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future meetings. Also, it will hear a report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, including anticipated workload and member assignments.

2:15 p.m.–2:30 p.m.: Reconciliation of ACRS Comments and Recommendations (Open)—The Committee will discuss the responses from the NRC Executive Director for Operations (EDO) to comments and recommendations included in recent ACRS reports and letters. The EDO responses are expected to be made available to the Committee prior to the meeting.

2:45 p.m.–7 p.m.: Preparation of ACRS Reports (Open)—The Committee will discuss proposed ACRS reports.

Saturday, October 10, 2004, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

8:30 a.m.–2 p.m.: Preparation of ACRS Reports (Open)—The Committee will continue its discussion of proposed ACRS reports.

2 p.m.–2:30 p.m.: Miscellaneous (Open)—The Committee will discuss matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

Procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on October 16, 2003 (68 FR 59644). In accordance with those procedures, oral or written views may be presented by members of the public, including representatives of the nuclear industry. Electronic recordings will be permitted only during the open portions of the meeting. Persons desiring to make oral statements should notify the Cognizant ACRS staff named below five days before the meeting, if possible, so that appropriate arrangements can be made to allow necessary time during the meeting for such statements. Use of still, motion picture, and television cameras during the meeting may be limited to selected portions of the meeting as determined by the Chairman. Information regarding the time to be set aside for this purpose may be obtained by contacting the Cognizant ACRS staff prior to the meeting. In view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with the Cognizant ACRS staff if such rescheduling would result in major inconvenience.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, as well as the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting Mr. Sam Duraiswamy, Cognizant ACRS staff (301–415–7364), between 7:30 a.m. and 4:15 p.m., ET.

ACRS meeting agenda, meeting transcripts, and letter reports are available through the NRC Public Document Room at *pdr@nrc.gov*, or by calling the PDR at 1–800–397–4209, or from the Publicly Available Records System (PARS) component of NRC's document system (ADAMS) which is accessible from the NRC Web site at *http://www.nrc.gov/reading-rm/adams.html* or *http://www.nrc.gov/reading-rm/doc-collections/* (ACRS & ACNW Mtg schedules/agendas).

Videoteleconferencing service is available for observing open sessions of ACRS meetings. Those wishing to use this service for observing ACRS meetings should contact Mr. Theron Brown, ACRS Audio Visual Technician (301–415–8066), between 7:30 a.m. and 3:45 p.m., ET, at least 10 days before the meeting to ensure the availability of this service. Individuals or organizations requesting this service will be responsible for telephone line charges and for providing the equipment and facilities that they use to establish the videoteleconferencing link. The availability of videoteleconferencing services is not guaranteed.

Dated: September 16, 2004.

Annette Vietti-Cook,

Secretary of the Commission.

[FR Doc. 04–21258 Filed 9–21–04; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting

AGENCY HOLDING THE MEETING: Nuclear Regulatory Commission.

DATE: Weeks of September 20, 27, October 4, 11, 18, 25, 2004.

PLACE: Commissioners' Conference Room, 11555 Rockville Pike, Rockville, Maryland.

STATUS: Public and closed.

MATTERS TO BE CONSIDERED:

Week of September 20, 2004

There are no meetings scheduled for the week of September 20, 2004.

Week of September 27, 2004—Tentative

There are no meetings scheduled for the week of September 27, 2004.

Week of October 4, 2004—Tentative

Thursday, October 7, 2004

10:30 a.m. Discussion of Security Issues (Closed—Ex. 1).

1 p.m. Discussion of Security Issues (Closed—Ex. 1).

Week of October 11, 2004—Tentative

Wednesday, October 13, 2004

9:30 a.m. Briefing on Decommissioning Activities and Status (Public Meeting) (Contact: Claudia Craig, (301) 415–7276).

This meeting will be webcast live at the Web address—*http://www.nrc.gov.*

1:30 p.m. Discussion of Intragovernmental Issues (Closed—Ex. 1 & 9).

Week of October 18, 2004—Tentative

There are no meetings scheduled for the week of October 18, 2004.

Week of October 25, 2004—Tentative

There are no meetings scheduled for the week of October 25, 2004.

*The schedule for Commission meetings is subject to change on short notice. To verify the status of meetings call (recording)—(301) 415–1292. Contact person for more information: Dave Gamberoni, (301) 415–1651.

* * * * *

The NRC Commission Meeting Schedule can be found on the Internet at: *http://www.nrc.gov/what-we-do/policy-making/schedule.html.*

* * * * *

The NRC provides reasonable accommodation to individuals with disabilities where appropriate. If you need a reasonable accommodation to participate in these public meetings, or need this meeting notice or the transcript or other information from the public meetings in another format (*e.g.* braille, large print), please notify the NRC's Disability Program Coordinator, August Spector, at (301) 415–7080, TDD: (301) 415–2100, or by e-mail at *aks@nrc.gov.* Determinations on requests for reasonable accommodation will be made on a case-by-case basis.

* * * * *

This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to the distribution, please contact the Office of the Secretary, Washington, DC 20555 (301) 415–1969). In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to *dkw@nrc.gov.*

Dated: September 17, 2004.

Dave Gamberoni,

Office of the Secretary.

[FR Doc. 04–21336 Filed 9–20–04; 9:34 am]

BILLING CODE 7590–01–M

SECURITIES AND EXCHANGE COMMISSION

Proposed Collection; Comment Request

Upon written request, copies available from: Securities and Exchange Commission Office of Filings and Information Services, Washington, DC 20549.

Extension: Form N–8F; SEC File No. 270–136; OMB Control No. 3235–0157.

Notice is hereby given that pursuant to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 *et seq.*) the Securities and Exchange Commission ("Commission") is soliciting comments on the collections of information summarized below. The Commission plans to submit this existing collection of information to the Office of Management and Budget ("OMB") for extension and approval.

Form N–8F (17 CFR 274.218) is the form prescribed for use by registered investment companies in certain circumstances to request orders of the Commission declaring that the registration of that investment company cease to be in effect. The form requests, from investment companies seeking a deregistration order, information about (i) the investment company's identity, (ii) the investment company's distributions, (iii) the investment company's assets and liabilities, (iv) the events leading to the request to deregister, and (v) the conclusion of business. The information is needed by the Commission to determine whether an order of deregistration is appropriate.

The Form takes approximately 3 hours on average to complete. It is estimated that approximately 261 investment companies file Form N–8F annually, so that the total annual burden for the form is estimated to be 783 hours. The estimate of average burden hours is made solely for the purposes of the Paperwork Reduction Act and is not derived from a

APPENDIX II

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

September 15, 2004

SCHEDULE AND OUTLINE FOR DISCUSSION
516th ACRS MEETING
OCTOBER 7-9, 2004

THURSDAY, OCTOBER 7, 2004, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

1) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (MVB/JTL/SD)
   1.1) Opening Statement
   1.2) Items of current interest

12:15 PM
2) 8:35 - ~~10:45~~ A.M. Safety Evaluation of the Industry Guidelines Related to Pressurized Water Reactor (PWR) Sump Performance (Open) (GBW/RC)
   2.1) Remarks by the Subcommittee Chairman
   2.2) Briefing by and discussions with representatives of the NRC staff and the Nuclear Energy Institute regarding the staff's evaluation of the industry guidelines associated with the resolution of Generic Safety Issue (GSI)-191, "Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at PWRs," and related matters.

11:00 - 11:15
~~10:45~~ - ~~11:00~~ A.M. ***BREAK***

1:00 - 2:55
3) ~~11:00~~ - ~~12:30~~ P.M. Pre-Application Safety Assessment Report for the Advanced CANDU 700 (ACR-700) Design (Open) (TSK/MME)
   3.1) Remarks by the Subcommittee Chairman
   3.2) Briefing by and discussions with representatives of the NRC staff regarding the staff's Safety Assessment Report related to the pre-application review of the ACR-700 design and related matters.

Representatives of the Atomic Energy of Canada Ltd. may provide their views, as appropriate.

2:55-3:10
~~12:30~~ - ~~1:30~~ P.M. ***LUNCH***

3:10 - 4:55
4) ~~1:30~~ - ~~3:00~~ P.M. Proposed Recommendations for Resolving GSI-185, "Control of Recriticality Following Small-Break LOCAs in PWRs" (Open) (VHR/RC/MRS)
   4.1) Remarks by the Subcommittee Chairman
   4.2) Briefing by and discussions with representatives of the NRC staff and its contractors regarding the proposed recommendations for resolving GSI-185.

Representatives of the nuclear industry may provide their views, as appropriate.

~~**3:00 - 3:15 P.M. ***BREAK*****~~

4:55-5:55

5) ~~3:15~~ - ~~4:45~~ P.M. <u>Mitigating System Performance Index Program</u> (Open) (JDS/MWW)

5.1) Remarks by the ACRS Chairman

5.2) Briefing by and discussions with representatives of the NRC staff regarding the Mitigating System Performance Index Program.

Representatives of the nuclear industry may provide their views, as appropriate.

4:45 - 5:00 P.M. ***BREAK***

7:30

6) 5:00 - ~~7:00~~ P.M. <u>Preparation of ACRS Reports</u> (Open)

Discussion of proposed ACRS reports on:

6.1) Pre-Application Safety Assessment Report for the ACR-700 Design (TSK/MME)

6:05-7:20 6.2) Safety Evaluation of the Industry Guidelines Related to PWR Sump Performance (GBW/RC)

6.3) Proposed Recommendations for Resolving GSI-185 (VHR/RC/MRS)

6.4) Mitigating System Performance Index Program (JDS/MWW)

6.5) Response to the August 25, 2004 EDO Response to the May 21, 2004 ACRS Letter on Resolution of Certain Items Identified by the ACRS in NUREG-1740, "Voltage-Based Alternative Repair Criteria" (GBW/FPF/CS)

<u>FRIDAY, OCTOBER 8, 2004, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND</u>

7) 8:30 - 8:35 A.M. <u>Opening Remarks by the ACRS Chairman</u> (Open) (MVB/JTL/SD)

10:30

8) 8:35 - ~~10:00~~ A.M. <u>Technology Neutral Framework for Future Plant Licensing</u> (Open) (TSK/MME)

8.1) Remarks by the Subcommittee Chairman

8.2) Briefing by and discussions with representatives of the NRC staff regarding the technology neutral framework for licensing of future plant designs.

Representatives of the nuclear industry may provide their views, as appropriate.

10:30 - 10:45

~~10:00~~ - ~~10:15~~ A.M. ***BREAK***

10:45- 12:00 PM

9) ~~10:15~~ - ~~11:30~~ A.M. <u>Assessment of the Quality of the NRC Research Projects</u> (Open) (DAP/SLR/TSK/RC/HPN)

9.1) Remarks by the Subcommittee Chairman

9.2) Discussion of the preliminary results of the cognizant ACRS members' assessment of the research projects on Sump Blockage and on MACCS code.

10) ~~11:30~~ 12:00 - 12:15 P.M. Divergence in Regulatory Approaches and Requirements Between U.S. and Other Countries (Open) (DAP/HPN/SD)
- 10.1) Remarks by the Subcommittee Chairman
- 10.2) Discussion of the draft Final White Paper prepared by Dr. Nourbakhsh, ACRS Senior Staff Engineer, regarding divergence in regulatory approaches and requirements between U.S. and other countries.

12:15 - 1:15 P.M. *****LUNCH*****

11) ~~1:15~~ 2:50 - 2:15 P.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (MVB/JTL/SD)
- 11.1) Discussion of the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings.
- 11.2) Report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, including anticipated workload and member assignments.

12) 2:15 - 2:30 P.M. Reconciliation of ACRS Comments and Recommendations (Open) (MVB, et al./SD, et al.)
Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

~~2:30~~ - ~~2:45~~ P.M. 2:40 - 2:50 *****BREAK*****

13) 2:45 - 7:00 P.M. Preparation of ACRS Reports (Open)
Discussion of the proposed ACRS reports on:
- 4:00-4:20 13.1) Pre-Application Safety Assessment Report for the ACR-700 Design (TSK/MME)
- 1:15-2:40 5:45-7:00 13.2) Safety Evaluation of the Industry Guidelines Related to PWR Sump Performance (GBW/RC)
- 4:45-5:45 13.3) Proposed Recommendations for Resolving GSI-185 (VHR/RC/MRS)
- 4:20-4:45 13.4) Mitigating System Performance Index Program (JDS/MWW)
- 13.5) Response to the August 25, 2004 EDO Response to the May 21, 2004 ACRS Letter on Resolution of Certain Items Identified by the ACRS in NUREG-1740, "Voltage-Based Alternative Repair Criteria" (GBW/FPF/CS)
- 12:00-12:15 13.6) Divergence in Regulatory Approaches and Requirements Between U.S. and Other Countries (DAP/HPN/SD)

SATURDAY, OCTOBER 9, 2004, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

14) 8:30 - ~~2:00~~ 1:00 P.M. Preparation of ACRS Reports (Open)
~~**(12:00-1:00 P.M. - LUNCH)**~~ Continue discussion of proposed ACRS reports listed under Item 13.

~~15) 2:00 - 2:30 P.M. Miscellaneous (Open) (MVB/JTL)~~
~~Discussion of matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.~~

NOTE:

- **Presentation time should not exceed 50 percent of the total time allocated for a specific item. The remaining 50 percent of the time is reserved for discussion.**

- **Thirty-Five (35) hard copies and (1) electronic copy of the presentation materials should be provided to the ACRS.**

APPENDIX III: MEETING ATTENDEES

516TH ACRS MEETING
OCTOBER 7-9, 2004

NRC STAFF (10/7/2004)

R. Elliott, NRR
J. Hannon, NRR
L. Lund, NRR
M. Mitchell, NRR
T. Mensah, NRR
D. Terao, NRR
A. Csantos, NMSS
F. Eltawila, RES
W. Jenson, NRR
L. Dudes, NRR
J. Fair, NRR
K, Kawoski, NRR
D. Carlson, RES
C. Jackson, OCM
S. Jones, NRR
M. Chiramal, NRR
S. Lui, NRR
M. Murphy, NRR
P. Klein, NRR
D. Harrison, NRR
S. Unikewicz, NRR
B. Parks, NRR
R. Architzel, NRR
M. Johnson, NRR
M. Giles, NRR
K. Kavanagh, NRR
T. Valentine, NRR
E. Throm, NRR
S. Burnell, OPA
R. Landry, NRR
A. Drozc, NRR
H. Vandermolen, RES
B.P. Jain, RES
W. Krotink, RES
S. Black, NRR
H. Wagage, NRR
M. Webb, NRR
J. Thompson, NRR
D. Dube, RES
S. Richards, NRR
J. Voglewede, RES
J. Rosenthal, RES
C. Ader, RES
W. Beckner, NRR
M. Kowal, NRR
A. Lavretta, NRR
J. Wilson, NRR
M. Dimarro, RES
D. Gilison, NRR
T. Attard, NRR
B. Sosa, NRR
J. Kim, NRR
F. Akstulewicz, NRR
M. Stutzke, NRR
M. Waterman, RES
S. Basin, RES
P. Clifford, NRR
R. Lee, RES
J. Ridgeley, RES
C. Greene, RES
J. Colaccino, NRR
J. Lee, NRR
M. Gamberoni, RES
J. Andersen, NRR

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC
D. Lochbaum, UCS
M. Kostelnik, Constellation Energy
S. Traiforos, LINK
T. Andreychek, Westinghouse
P. Negus, GE
J. Butler, NEI
B. Letellier, LANL
C. Shaffer, ARES
B. Smith, GE
A. Drake, Constellation Energy
J. Polcyn, AECL Tech
C. Reid, Bechtel
F. Archinoff, AECL Tech
P. Boczar, AECL
V. Snell, AECL
R. Ion, AECL
B. Rouben, AECL
C. Berger, Energetics
A. Wyete, SERCH
H. Ludewig, BNL
D. Diamond, BNL
S. Eicle, INEEL
C. Atwood, Statwood Consulting

NRC STAFF (10/8/2004)
M. Stutzke, NRR
M. Drouin, RES
M. Gamberoni, RES
J. Wilson, NRR
G. Parry, NRR
S. Basin, RES
J. Craig, RES
J. Mitchell, RES
N. Kadambi, RES
E. McKenna, NRR

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC
J. Riccio, Green Peace
C. Berger, Energetics
P. Negus, GE
D. C. Bley, Buttonwood
V. Mubayi, BNL
G. Archinoff, AECL Technologies

APPENDIX IV

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

October 18, 2004

SCHEDULE AND OUTLINE FOR DISCUSSION
517th ACRS MEETING
NOVEMBER 4-6, 2004

THURSDAY, NOVEMBER 4, 2004, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

1) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (MVB/JTL/SD)
    1.1) Opening statement
    1.2) Items of current interest

2) 8:35 - 10:30 A.M. Proposed Rule for Risk-Informing 10 CFR 50.46 (Open) (WJS/MRS)
    2.1) Remarks by the Subcommittee Chairman
    2.2) Briefing by and discussions with representatives of the NRC staff regarding the proposed rule for risk-informing 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," and related matters.

Representatives of the nuclear industry and members of the public may provide their views, as appropriate.

10:30 - 10:45 A.M. *****BREAK*****

3) 10:45 - 12:15 P.M. Proactive Materials Degradation Assessment Program (Open) (JDS/MWW)
    3.1) Remarks by the Cognizant ACRS Member
    3.2) Briefing by and discussions with representatives of the NRC staff regarding the status of the Proactive Materials Degradation Assessment Program.

Representatives of the nuclear industry may provide their views, as appropriate.

12:15 - 1:15 P.M. *****LUNCH*****

4) 1:15 - 2:45 P.M. Proposed Rule on Post-Fire Operator Manual Actions (Open) (SLR/MDS)
    4.1) Remarks by the Subcommittee Chairman
    4.2) Briefing by and discussions with representatives of the NRC staff regarding the proposed rule on post-fire operator manual actions and related matters.

Representatives of the nuclear industry and members of the public may provide their views, as appropriate.

2:45 - 3:00 P.M. ***BREAK***

5) 3:00 - 4:30 P.M. Grid Reliability Issues and Related Significant Operating Events (Open) (JDS/MWW)

- 5.1) Remarks by the Subcommittee Chairman
- 5.2) Briefing by and discussions with representatives of the NRC staff regarding their activities associated with grid reliability, significant operating events related to grid stability, and other related matters.

Representatives of the nuclear industry may provide their views, as appropriate.

4:30 - 4:45 P.M. ***BREAK***

6) 4:45 - 7:00 P.M. Preparation of ACRS Reports (Open/Closed)

Discussion of proposed ACRS reports on:

- 6.1) Proposed Rule for Risk-Informing 10 CFR 50.46 (WJS/MRS)
- 6.2) Proposed Rule on Post-Fire Operator Manual Actions (SLR/MDS)
- 6.3) Grid Reliability Issues and Related Significant Operating Events (JDS/MWW)
- 6.4) Response to the August 25, 2004 ACRS Letter on Resolution of Certain Items Identified by the ACRS in NUREG-1740, "Voltage-Based Alternative Repair Criteria" (GBW/FPF/CS)
- 6.5) AP1000 Lessons Learned Report (TSK/MME)
- 6.6) Safeguards and Security Matters (CLOSED) (MVB/RKM/RPS)

FRIDAY, NOVEMBER 5, 2004, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

7) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (MVB/JTL/SD)

8) 8:35 - 10:00 A.M. Status of Early Site Permit Reviews (Open) (TSK/MME)

- 8.1) Remarks by the Subcommittee Chairman
- 8.2) Briefing by and discussions with representatives of the NRC staff regarding the status of the staff's review of the early site permit applications.

10:00 - 10:15 A.M. ***BREAK***

9) 10:15 - 11:45 A.M. Assessment of the Quality of Selected NRC Research Projects (Open) (GEA/SLR/TSK/HPN/SD)

- 9.1) Remarks by the Cognizant ACRS Member
- 9.2) Discussion of the results of the Cognizant ACRS Members' assessment of the quality of the NRC research projects on Sump Blockage and on MACCS code.

11:45 - 12:45 P.M. ***LUNCH***

10) 12:45 - 1:00 P.M. Plant License Renewal Subcommittee Report (Open) (MVB/CS)
Report by the Chairman of the ACRS Subcommittee on Plant License Renewal regarding interim review of the license renewal application for the Farley Nuclear Plant.

11) 1:00 - 2:00 P.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (MVB/JTL/SD)
- 11.1) Discussion of the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings.
- 11.2) Report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, including anticipated workload and member assignments.

12) 2:00 - 2:15 P.M. Reconciliation of ACRS Comments and Recommendations (Open) (MVB, et al./SD, et al.)
Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

2:15 - 2:30 P.M. ***BREAK***

13) 2:30 - 7:00 P.M. Preparation of ACRS Reports (Open/Closed)
Discussion of the proposed ACRS reports on:
- 13.1) Proposed Rule for Risk-Informing 10 CFR 50.46 (WJS/MRS)
- 13.2) Proposed Rule on Post-Fire Operator Manual Actions (SLR/MDS)
- 13.3) Grid Reliability Issues and Related Significant Operating Events (JDS/MWW)
- 13.4) Response to the August 25, 2004 ACRS Letter on Resolution of Certain Items Identified by the ACRS in NUREG-1740, "Voltage-Based Alternative Repair Criteria" (GBW/FPF/CS)
- 13.5) AP1000 Lessons Learned Report (TSK/MME)
- 13.6) Assessment of the Quality of Selected NRC Research Projects (GEA/SLR/TSK/HPN/SD)
- 13.7) Safeguards and Security Matters (CLOSED) (MVB/RKM/RPS)

SATURDAY, NOVEMBER 6, 2004, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

14)	8:30 - 12:30 P.M.	Preparation of ACRS Reports (Open/Closed) Continue discussion of proposed ACRS reports listed under Item 13.
15)	12:30 - 1:00 P.M.	Miscellaneous (Open) (MVB/JTL) Discussion of matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

NOTE:

- **Presentation time should not exceed 50 percent of the total time allocated for a specific item. The remaining 50 percent of the time is reserved for discussion.**

- **Thirty-Five (35) hard copies and (1) electronic copy of the presentation materials should be provided to the ACRS.**

APPENDIX V
LIST OF DOCUMENTS PROVIDED TO THE COMMITTEE
516TH ACRS MEETING
OCTOBER 7-9, 2004

[Note: Some documents listed below may have been provided or prepared for Committee use only. These documents must be reviewed prior to release to the public.]

MEETING HANDOUTS

AGENDA ITEM NO.	DOCUMENTS
1	Opening Remarks by the ACRS Chairman 1. Items of Interest, dated October 7-9, 2004
2	Safety Evaluation of the Industry Guidelines Related to Pressurized Water Reactor (PWR) Sump Performance 2. PWR Plant-Specific Sump Evaluation Process (GSI-191) presentation by NRR [Viewgraphs] 3. Accident Progression presentation by NRR [Viewgraphs] 4. Safety Evaluation Report, GSI-191 PWR ECCS Sump Performance presentation by NRR [Viewgraphs] 5. Industry Activities to Address PWR ECCS Sump Performance presentation by the Nuclear Energy Institute [Viewgraphs] 6. Review of Head Loss Prediction Across Sump Screens review by S. Banerjee, ACRS Consultant [Handout] 7. Letter to John Hannon, NRR from NEI, Subject: Safety Evaluation for NEI Guidance Report “Pressurized Water Reactor Containment Sump Evaluation Methodology,” dated October 1, 2004 [Handout]
3	Pre-Application Safety Assessment Report for the Advanced CANDU 700 (ACR-700) Design 8. ACR-700 Pre-Application Review presentation by NRR [Viewgraphs] 9. ACR-700 Pre-Application Safety Assessment Report (PASAR), Focus Topic 9 “Confirmation of Negative Void Reactivity,” presentation by RES [Viewgraphs] 10. ACR-700 Technologies presentation by AECL [Viewgraphs]
4	Proposed Recommendations for Resolving GSI-185, “Control of Recritically Following Small-Break LOCAs in PWRs” 11. Resolution of GSI 185 presentation by J. Rosenthal, RES 12. Analysis of Boron Dilution Transients in PWRs presentation by D. Diamond, BNL 13. Tools: Deborate Mixing presentation by M. diMarzo, RES
5	Mitigating System Performance Index Program 14. Mitigating Systems Performance Index presentation by NRR and RES [Viewgraphs]

8 Technology Neutral Framework for Future Plant Licensing

15. Regulatory Structure for New Plant Licensing, Part 1: Technology-Neutral Framework presentation by RES, BNL, Buttonwood Consulting, Inc.

11 Future ACRS Activities/Report of the Planning and Procedures Subcommittee

16. Future ACRS Activities/Final Draft Minutes of Planning and Procedures Subcommittee Meeting - October 8, 2004 [Handout #11.1]

12 Reconciliation of ACRS Comments and Recommendations

17. Reconciliation of ACRS Comments and Recommendations [Handout #12.1]

Appendix V
516th ACRS Meeting

MEETING NOTEBOOK CONTENTS

TAB	DOCUMENTS
2	Safety Evaluation of Industry Guidelines Related to Pressurized Water Reactor (PWR) Sump Performance

1. Table of Contents
2. Proposed Schedule
3. Status Report
4. NRC Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors," June 9, 2003 (electronic copy only)
5. Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors," September 13, 2004 (electronic copy only)
6. Nuclear Energy Institute Guidance Report (Proposed Document Number NEI 04-07), "Pressurized Water Reactor Sump Performance Evaluation Methodology" (electronic copy of all portions provided only on disk - hard copy provided previously)
7. Letter from Suzanne C. Black to John Larkins, September 8, 2004, "Request for ACRS Review of the Draft Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-XX, Nuclear Energy Institute Guidance Report (Proposed Document Number NEI 04-07), "Pressurized Water Reactor Sump Performance Evaluation Methodology" (electronic copy on disk - revised version provided as attachment - see below)
8. SECY-04-0154, August 24, 2004, "Issuance of Nuclear Regulatory Commission Generic Letter 2004-XX, "Potential Impact of Debris Basis Accidents at Pressurized-Water Reactors" (electronic copy only)
9. SECY-04-0150, August 16, 2004, "Alternate Approaches for for Resolving the Pressurized Water Reactor Sump Blockage Issue (GSI-191), Including Realistic and Risk-Informed Considerations" (electronic copy only)

4 Control of Recriticality Following Small-Break LOCAs in PWRs

10. Table of Contents
11. Proposed Schedule
12. Status Report
13. Draft NUREG XXXX, "Resolution of Generic Safety Issue No. 185, "Control of Recriticality Following Small-Break LOCAs in PWRs" (electronic copy on disk, previously as hard copy to all Members)
14. Comments from V. Ransom, June 13, 2004 (electronic copy only)
15. Comments from V. Ransom, May 3, 2004 (electronic copy only)
16. Comments from H. Nourbakhsh, September 20, 2004 (electronic copy only)
17. Comments from G. Wallis, September 21, 2004 (electronic copy only)
18. Draft NUREG XXXX, Resolution of Generic Safety Issue No. 185, "Control of Recriticality Following Small-Break LOCAs in PWRs" (electronic and hard copy provided)

5 Mitigating Systems Performance Index

19. Table of Contents
20. Proposed Schedule
21. Status Report
22. Report on the Independent Verification of Mitigating Systems Performance Index Results for Pilot Plants

8 Technology-Neutral Framework for Future Plant Licensing

23. Table of Contents
24. Proposed Schedule
25. Status Report
26. Part 1: Technology-Neutral Framework (Draft), [electronic]
27. Dr. T. Kress comments dated June 19, 2004
28. RES slides for October 8, 2004 ACRS presentation
29. Staff Requirements Memorandum, dated June 26, 2003
30. ACRS Report dated April 22, 2004
31. SECY-03-0047 dated March 28, 2003
32. SECY-04-103 dated June 23, 2004
33. SECY-04-0157 dated August 30, 2004

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

516th FULL COMMITTEE MEETING
OCTOBER 7-9, 2004

<u>October 7, 2004 2004</u>
Today's Date

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NAME	NRC ORGANIZATION
ROB ELLIOTT	NRR/DSSA
John Hannon	NRR/DSSA
Louise Lund	NRR/DE/EMCB
Matthew A. Mitchell	ROPMS
T. Mensah	NRR/PMAS
D. Terao	NRR/DE/EMEB
Aladar Csontos	NMSS/HLWRS
FAROUK ELTAWILA	RES/DSARE
Walton Jensen	NRR/DSSA
Laura Dudes	NRR/DRIP
John Fair	NRR/DE
Ken Karwoski	NRR/DE
Don Carlson	RES/DSARE
Christopher Jackson	OCM
Steven Jones	NRR/DSSA
MATTHEW CHIRAMAL	NRR/DE

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

516th FULL COMMITTEE MEETING
OCTOBER 7-9, 2004

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Martin Murphy	NRR/DE/EMCB
Paul Klein	NRR/DE/EMEB
Donnie Harrison	NRR/DSSA/SPSB
STEVEN UNIKEWICZ	NRR/DE/EMEB
BENJAMIN PARKS	NRR/DE/EMEB
Ralph Architzel	NRR/DSSA/SPLB
Michael Johnson	NRR/DSSA
Mark Giles	NRC/DSSA
KERRI KAVANAGH	NRR/DIPM
Theresa Valentine	NRR/DSSA/SPSB
EDWARD THROM	NRR/DSSA/SPSB
Scott Burnell	OPA
RALPH LANDRY	NRR/DSSA/SRXB
B. P. Jain	RES/DET/ERAB
William Krotiuk	RES/DSARE/ARREB
SUZANNE BLACK	NRR/DSSA
Hanry Wagage	NRR/DSSA/SPLB
Michael Webb	NRR/DLPM

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

516th FULL COMMITTEE MEETING
OCTOBER 7-9, 2004

October 7, 2004 2004
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DON DUBE	RES/OERAB
STUART RICHARDS	NRR/DIPM
JOHN VOGLEWEDE	RES/DSARE
Jack Rosenthal	RES/SMSAB
Charles Ader	RES/DRAA

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516th FULL COMMITTEE MEETING
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WILLIAM BECKNER	NRR
Mark Kowal	NRR
Angie Lavretta	NRC/NRR/DSSA
JERRY Wilson	NRC/RNRP
MARINO DIMARZO	NRC/RES
Dave Cullison	NRR/DSSA
Tony Attard	NRR/DSSA
BELKYS Sosa	NRR/RNRP
JIM KIM	NRR/DRIP/RNRP
FRANK Akstulewicz	NRR/DSSA/SRXB
MARTIN STUTZKE	NRR/DSSA/SPSB
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516th FULL COMMITTEE MEETING
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October 7, 2004 2004
Today's Date

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NAME	NRC ORGANIZATION
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Harold Vandermolen	RES/DSARE/ARREB
James Andersen	NRR/DIPM/IIPB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

516th FULL COMMITTEE MEETING
OCTOBER 7-9, 2004

October 7, 2004
Today's Date

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NAME	AFFILIATION
Dave Lochbaum	Union of Concerned Scientists
MARK KOSTELNIK	CONSTELLATION ENERGY
SPYROS TRAIFOROS	LINK
TIM ANDREYCHEK	WESTINGHOUSE ELECTRIC
PAIGE NEGUS	GE
John BUTLER	NEI
Bruce Letellier	LANL
Clint Shaffer	ARES
Bob Coward	
BARRY SMITH	GE
Andre Drake	Constellation Energy
JOHN POLCYN	AECL TECHNOLOGIES
CAL REID	BECHTEL
GLENN ARCHINOFF	AECL TECHNOLOGIES
Peter Boczar	AECL
VICTOR SNELL	AECL.
ROBERT ION	AECL.
BENJAMIN ROUBEN	AECL
~~LAURA DUD~~	

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

516th FULL COMMITTEE MEETING
OCTOBER 7-9, 2004

October 7, 2004
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Carl Berger	Energetics
Althea Wyche	SERCH Licensing/Bechtel
Hans Ludewig	BNL
David Diamond	BNL
Summer B. Sun	NRR/DSSA/SRXB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

516th FULL COMMITTEE MEETING
OCTOBER 7-9, 2004

October 7, 2004
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Steve Eide	INEEL
Cory Atwood	Statwood Consulting

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

516th FULL COMMITTEE MEETING
OCTOBER 7-9, 2004

October 8, 2004 2004
Today's Date

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Marsha Gamberoni	RES/DSARE/
JERRY Wilson	NRR/RNRP
Gareth Parry	NRR/DSSA
Sud Basu	RES/DSARE
John Craig	RES/DOD
Jocelyn Mitchell	RES/DSARE
NPKADAMBI	RES/DSARE
E Mckenna	NRR/DRIP

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

516th FULL COMMITTEE MEETING
OCTOBER 7-9, 2004

October 8, 2004
Today's Date

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Jim Riccio	GREENPEACE
Carl Berger	Energetics
Paige Negus	GE
D C Bley	Buttonwood
Vinod Mubayi	BNL
GLENN ARCHINOFF	AECL Technologies

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ITEMS OF INTEREST

516th ACRS MEETING

OCTOBER 7-9, 2004

ITEMS OF INTEREST
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
515th MEETING
September 8-11, 2004

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SPEECHES

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CONGRESSIONAL TESTIMONY

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Operating Events

Preliminary Notification of Event or Unusual Occurrences:

NUCLEAR SAFETY RESEARCH CONFERENCE (NSRC)

INSIDE NRC ARTICLES

RUTHLAND HERALD

NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION

Office of Public Affairs
Washington, DC 20555-0001
Telephone: 301/415-8200
E-mail: opa@nrc.gov
www.nrc.gov

No. 04-125

October 05, 2004

NRC CHAIRMAN DISCUSSES NUCLEAR PLANT SECURITY AT AMERICAS NUCLEAR ENERGY SYMPOSIUM

MIAMI - Security at U.S. nuclear power plants has been substantially upgraded since Sept. 11, 2001, with higher federal requirements placed on utilities that operate nuclear facilities, Nuclear Regulatory Commission Chairman Nils J. Diaz said Tuesday.

"Nuclear power plants have been and are even more so now among the most well-protected elements of our national civilian infrastructure," Diaz said in remarks prepared for delivery at the Americas Nuclear Energy Symposium jointly sp[illegible]ored by the U.S. Department of Energy and the American Nuclear Society.

"[illegible]RC has further strengthened security requirements at nuclear power plants and enhanced our coordination with federal, state and local organizations since Sept. 11," Diaz said. "In addition, the NRC has conducted research-based studies which concluded that a significant radiological release affecting public health and safety is unlikely from a terrorist attack, including a large commercial aircraft. And time is available to protect the public in the unlikely event of a radiation release."

Diaz also said the NRC is:

- improving nuclear plant operating safety in the United States through the use of a new Reactor Oversight Program that provides a better inspection regime for plants. He said the NRC objective is to "provide the tools for inspecting and assessing licensee performance in a manner that was more risk-informed, objective, predictable and understandable than the previous oversight processes, and that ensures the agency's performance goals are being met."

- providing oversight in a way that corresponds to the actual, real world risk presented, rather than a theoretical worst-case scenario. He added, "Simply put, technical and regulatory decisions are informed by the real world – utilizing advancing scientific knowledge, improving technological capabilities and the lessons that have been learned through decades of operating experience."

Outlining security advances, Diaz said that the steps taken by the NRC include: ordering plants to take into account a more challenging adversarial threat; tighter access controls and vehicle checks at greater stand-off distances; significantly improved force-on-force exercises to test the capabilities of plant defenders; better readiness by plant security forces; and enhanced liaison with the intelligence community, and federal, state and local authorities responsible for protecting the national critical infrastructure through integrated response training.

Th[illegible]posium brought together members of the government, industry, and academic communities for presentations and di[illegible]ns on issues related to the future of nuclear energy in the Americas.

Last revised Wednesday, October 06, 2004

September 8, 2004

The Honorable Tom Ridge
Secretary
U.S. Department of Homeland Security
Washington, D.C. 20528

Dear Secretary Ridge:

On behalf of the U.S. Nuclear Regulatory Commission (NRC), I am writing to update you on recent actions the NRC has taken to continue enhancing the security of NRC-regulated nuclear facilities and radioactive materials.[1]

The NRC has exercised its statutory responsibility to ensure adequate protection of public health and safety, promote the common defense and security, and protect the environment from potential hazards involved in the civilian use of nuclear materials. In the weeks and months following the terrorist attacks on September 11, 2001, the NRC focused its efforts on improving security at the facilities it regulates, including nuclear power reactors and fuel manufacturing facilities that possess significant quantities of special nuclear material, and activities such as transportation of spent nuclear fuel. On February 25, 2002, the NRC issued Orders to all nuclear power plant licensees requiring that they increase their defensive capabilities to protect against the new threat. These enhancements to security included increased security patrols, augmented security forces, additional security posts, increased vehicle standoff distances, and improved coordination with law enforcement and intelligence communities, as well as strengthened safety-related mitigation procedures and strategies.

On January 7, 2003, the NRC required further enhancements to access controls for the power plants. On April 29, 2003, the NRC issued three additional Orders requiring additional security enhancements, including:

- Work-hour limitations on security personnel;
- Enhanced training and qualification requirements for security force personnel; and
- Revisions to licensee security, training and qualifications, and contingency plans to protect against the supplemented design basis threat of radiological sabotage.

The April 2003 Orders required that the licensees submit their revised plans to the NRC for review and approval by April 29, 2004, and that the plans be implemented at the nuclear power plants by October 29, 2004. The licensees submitted revised security plans in April 2004. The NRC staff is on schedule to complete its review of those plans and will work with licensees to implement them.

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[1] See previous annual letters on NRC's web site @ www.nrc.gov.

Over the past year, we have continued to enhance security for NRC-licensed facilities and activities. On October 23, 2003, the Commission issued an immediately effective Order imposing additional security measures to all power reactor licensees and research reactor licensees who transport spent nuclear fuel. On January 12, 2004, the Commission issued an immediately effective Order imposing additional security measures for source manufacturers and distributors of high risk radioactive sources. In July 2004, the Commission approved issuance of Orders imposing additional security measures for the Honeywell uranium conversion facility, independent spent fuel storage facilities, and all decommissioning nuclear power plants with spent fuel in the spent fuel pool. Some of the requirements set forth in these various Orders formalize a series of security measures that NRC licensees had taken in response to advisories issued by the NRC in the aftermath of the September 11, 2001 terrorist attacks. Additional security enhancements, developed during our ongoing security review, are also provided in these Orders. The specific security measures addressed by the Orders, which supplement existing regulatory requirements, are classified as Safeguards Information under Section 147 of the Atomic Energy Act, as amended. These Orders remain in effect until the Commission determines otherwise.

In addition to these enhancements, we have continued to improve our security performance evaluation program (our force-on-force evaluations), which we consider an important element for ensuring protection of the Nation's critical infrastructure. In February 2003, we resumed the force-on-force program in the form of a pilot program to test recent program enhancements. In February 2004, the NRC began a transition force-on-force program, incorporating the lessons learned during the pilot program. The transition program follows the same format as the pilot program; however, the "mock adversary" force now uses the characteristics of the Design Basis Threat (DBT), as enhanced and supplemented by our Orders, to prepare for resumption of the full security performance assessment program in November 2004. Under that program, we will conduct approximately 22 force-on-force exercises per year, so that each site's security will undergo an NRC evaluated exercise at least once every three years. This represents a significant increase in the exercise frequency; in addition, each plant is required to conduct independent exercises at least once each year.

During the pilot program, the NRC identified the need to improve the offensive capabilities, consistency, and effectiveness of the exercise adversary force. The Commission addressed this need by directing the staff to develop a training standard for a Composite Adversary Force (CAF). The CAF for a given NRC-evaluated force-on-force exercise will comprise security officers from various nuclear power facilities (excluding the licensee being evaluated) and will have been trained in offensive, rather than defensive, skills to perform the adversary function.

The NRC has conducted detailed, site-specific engineering studies of a limited number of typical plants to assess potential vulnerabilities of nuclear power plants to deliberate attacks involving large commercial aircraft. The results of these studies have confirmed the effectiveness of the required mitigative measures and have identified further enhancements to mitigative strategies. For the facilities analyzed, the studies confirm that the likelihood of both damaging the reactor core and releasing radioactivity that could affect public health and safety is low. Even in the unlikely event of a radiological release due to a terrorist use of a large aircraft against a nuclear power plant, the studies indicate that there would be time to implement the required on-site mitigating actions. These results have also validated the off-site emergency planning basis. Additional studies are being considered to further enhance mitigative capabilities, and we will continue to coordinate with the Department of Homeland Security on this initiative.

The studies to date also indicate that significant releases of radioactive material due to a terrorist attack on a spent fuel pool are very unlikely. The safety and security of spent fuel storage is ensured through many safety and security measures that provide protection against terrorist threats. In addition, the studies indicate it is highly unlikely that a significant release of radioactivity would occur from a dry spent fuel storage cask, and no release of radioactive material is expected from an aircraft attack on a transportation cask. Measures are in place to adequately protect the public from attacks on spent fuel, in either wet or dry configurations.

In terms of nuclear material security, the NRC has taken action with our domestic and international counterparts to ensure protection of radioactive sources that could pose significant hazards to public safety. In partnership with the U.S. Departments of State and Energy, we have made key contributions to revising the Code of Conduct for the Safety and Security of Radioactive Sources promulgated by the International Atomic Energy Agency (IAEA) in September 2003 at the 47th session of the General Conference. We have also worked with the U.S. Departments of State and Energy to hold consultations with supplier nations concerning export/import controls. The NRC has independently proposed an export/import rule to enhance controls on these radioactive sources consistent with the IAEA Code of Conduct. The NRC has also coordinated with State officials to enhance controls on radioactive sources and as such, NRC has completed an interim source database that will, in time, grow into the National Source Tracking System consistent with the IAEA Code of Conduct for the Safety and Security of Radioactive Sources.

Another noteworthy action taken in the past year to enhance the security and safety of nuclear materials was the issuance of a regulatory bulletin requesting about 1,100 licensees to perform verification of certain types of nuclear materials currently in their possession. This activity supports NRC's Nuclear Materials Management and Safeguards System (NMMSS), a national database used by the NRC and the Department of Energy (DOE) to track certain nuclear materials and other government-owned materials.

With respect to emergency preparedness and incident response, the NRC continues to work with the U.S. Department of Homeland Security and other Federal agencies to integrate Federal Response Plans into a unified National Response Plan and National Incident Management System, and to refine the National Preparedness Policy. We have also completed the development of the commercial Nuclear Reactors, Materials, and Waste Key Resource Plan for Critical Infrastructure Protection. This document serves as the Sector-Specific component of the National Infrastructure Protection Plan. In addition, we continue to coordinate protective strategies with various components of the U.S. Department of Defense, including NORTHCOM and NORAD, and have recently participated in exercises such as Unified Defense '04 and Amalgam Virgo '04. We have also conducted integrated response tabletop exercises, involving licensees, State and local responders, as well as multiple Federal agencies, to focus combined efforts and actions when responding to a possible terrorist event at a nuclear power plant. In June 2004, the NRC integrated emergency preparedness functions into the Office of Nuclear Security and Incident Response to handle preparedness and response activities more effectively and efficiently.

In summary, the NRC has made, and will continue to make, significant progress in supporting our Nation's efforts to enhance homeland security and preparedness. Please do not hesitate to contact me for additional information.

Sincerely,

/RA/

Nils J. Diaz

STATEMENT SUBMITTED

BY THE

UNITED STATES NUCLEAR REGULATORY COMMISSION

TO THE

SUBCOMMITTEE ON NATIONAL SECURITY, EMERGING THREATS,
AND INTERNATIONAL RELATIONS

COMMITTEE ON GOVERNMENT REFORM

UNITED STATES HOUSE OF REPRESENTATIVES

CONCERNING

HOMELAND SECURITY: MONITORING NUCLEAR POWER PLANT SECURITY

SUBMITTED BY

LUIS A. REYES
EXECUTIVE DIRECTOR FOR OPERATIONS

Submitted: September 14, 2004

Introduction

Mr. Chairman and Members of the Subcommittee, it is a pleasure to appear before you today to discuss some of the efforts by the Nuclear Regulatory Commission (NRC) and its licensees with respect to security at nuclear power plants.

Overview

The NRC's mission is to regulate nuclear reactors, materials and waste facilities in a manner that protects the health and safety of the public, promotes the common defense and security, and protects the environment. Nuclear power plants have maintained strong safety and security measures, and were designed to withstand catastrophic events including fire, flood, earthquakes, and tornadoes. These plants were also designed using a defense-in-depth strategy, with redundant safety systems and are operated and protected by highly trained staff. Multiple barriers protect the reactor and prevent or mitigate off-site releases of radioactive materials. Design features of the reactor facilities provide substantial protection against a malevolent attack.

Security at nuclear facilities across the country has long been the subject of NRC regulatory oversight, dating back to the 1970's, and nuclear power plants have been required to implement security programs that are capable of defending against violent assaults by well-armed, well-trained adversaries. With sophisticated surveillance equipment, stringent access controls, physical barriers, professional security forces, and well qualified armed response forces and partnership with the local law enforcement agencies (LLEA), the nuclear power facilities have likely been among the best protected commercial facilities in the Nation prior to September 11, 2001, and remain so today. Coupled with emergency plans that are tested on a regular basis and support from local government agencies, these facilities are designed, operated, and regulated to protect the public from a wide range of events, including potential terrorist attacks.

The terrorist attacks on the United States brought to light a new and more immediate threat to our country. All custodians of the Nation's critical infrastructure needed to reconsider decisions made earlier about the adequacy of security at the facilities under their charge. To cope with these changes in the threat environment, the NRC undertook a reassessment of its safeguards and security programs, to identify prompt actions and long-term enhancements that would raise the level of security at the nuclear facilities across the country.

Since the terrorists attacks, the NRC has ordered its licensees to take specific actions to improve security at their facilities and to augment the protection of the nuclear materials they possess. Additionally, we have made internal programmatic and organizational changes to enhance the effectiveness of NRC's regulation of the security of nuclear facilities and materials. We believe that these comprehensive acts also effectively address major Congressional concerns about the adequacy of security in the new threat environment. We recognize though that security would be further enhanced if five legislative proposals that the Commission has submitted to the Congress, which we discuss later in this testimony, are promptly enacted.

Orders

In the weeks and months following September 11, the NRC focused its efforts on improving security at the facilities and activities it regulates, including nuclear power plants, certain fuel cycle facilities possessing significant quantities of special nuclear material, and transportation of spent nuclear fuel. Compensatory measures were imposed and the NRC required licensees to make changes to their security programs to deal with a new level of threat.

On February 25, 2002, the NRC issued Orders to all nuclear power plant licensees requiring that they formally incorporate specific compensatory measures into their safeguards and security programs. These enhancements to security included increased security patrols, augmented security forces, additional security posts, increased vehicle standoff distances, and improved coordination with law enforcement. On January 7, 2003, another set of Orders, designed to enhance security by tightening the plant access authorization requirements, was issued. On April 29, 2003, the NRC issued additional Orders setting security officer work-hours limitations to minimize fatigue; new training and qualification requirements for security force members; and requiring licensees to revise their security and contingency plans to protect against a new level of threat.

In the months since those Orders were issued, there has been close coordination with the regulated industry and representatives of Federal, State, and local government agencies that would be called upon to support the licensees' response to a potential terrorist attack. The Orders of April 2003 required that the licensees submit their revised security plans to the NRC by April 29, 2004, for review, revision as appropriate, and approval. The NRC staff is on schedule to complete its review of these plans and will work with licensees to implement them.

The licensees are responsible for providing security for their plants and costs incurred in this process are funded by the licensees. Except for directly after September 11 when Congress specifically authorized money for regulatory costs for upgrading NRC security, almost all of the Federal regulatory costs are paid by the NRC licensees through payment of fees.

Design Basis Threat and New Threat Level

Security programs at NRC-licensed nuclear power plants and certain fuel cycle facilities are designed to protect against an NRC specified level of threat called the design basis threat. The NRC first promulgated its design basis threats (DBTs) for radiological sabotage -- applicable to nuclear power plants -- and theft or diversion of strategic special nuclear material -- applicable to certain fuel cycle facilities -- in the late-1970s. In general terms, DBTs describe the attributes of a hypothetical adversary that these facilities must defend against with high assurance, including numbers of adversaries, types of weapons, and offensive strategies that would be employed by the adversaries. The threat attributes enumerated in the DBTs are based on extensive analyses by the NRC and discussion with the Intelligence Community and law enforcement officials. The DBT received periodic reviews by the staff and Commission resulting in at least one significant upgrade to the DBT prior to the September 11, 2001, attacks. When approved by the Commission, the DBT represents the characteristics of an adversary that a private guard force for a commercial nuclear facility should reasonably be required to protect against.

Following the September 11, 2001, attacks, the NRC conducted a comprehensive review of NRC's safeguards and security programs. This included a reassessment of the DBTs. As a result, the threat characteristics set forth in NRC regulations were supplemented by Orders issued to power reactors and to certain fuel cycle facilities in April 2003.

During its comprehensive review, the NRC staff also participated in the multi-agency working group developing the DOE/DOD Postulated Threat which, although not intended to apply to the commercial nuclear sector, provided insights considered in the development of NRC's supplemented DBTs. Additional coordination was conducted between NRC and DOE to ensure a clear understanding of the differences between the agencies' DBTs. This resulted in a proposed revision to NRC's threat characteristics which were presented to the Commission for their consideration to impose on specific licensees. In January 2003, the NRC sought comments on the supplement to the DBT from State agencies, Federal agencies, and licensees who were authorized access to the information. The Commission considered this information in establishing the supplemental requirements to implement the DBT. In addition, meetings were held with stakeholders, including other Federal agencies, State authorities, and industry representatives. Comments developed during those meetings, as well as the NRC staff final views were provided to the Commission in April 2003.

The Orders of April 29, 2003, required nuclear power plant licensees to revise their security and contingency plans to defend against the supplemented DBT. The NRC is currently reviewing these revised plans for nuclear power plants and certain nuclear fuel cycle facilities, nearly 200 plans in all, and expects that all plans will be reviewed, revised, as appropriate, approved, and, with few exceptions, implemented by the October 29, 2004, deadline imposed by the Commission's April 29, 2003, Orders. As was the case prior to September 11, the

Commission will review the DBTs semiannually against changes in the threat environment to ensure their continuing validity.

The NRC has been working with DHS, the White House Homeland Security Council, and other agencies regarding an “integrated response” by government assets to help defend against threats that could exceed the DBTs. The concept of “integrated response” applies to both prevention of and response to a potential terrorist event. The NRC is participating in tabletop exercises involving a number of Federal, State, and local agencies at nuclear power plants and continues to support the Homeland Security Council and DHS, the FBI, DoD, and other Federal, State, and local authorities regarding integrated response capabilities.

Vulnerability Assessment and Mitigation Strategies

The NRC has completed an extensive set of vulnerability assessments and identified mitigation strategies for NRC-licensed activities involving radioactive materials and nuclear facilities. Thus far, the results of these studies have validated the actions NRC has taken to enhance security as well as shown areas needing improvement. These efforts have continued to affirm the robustness of these facilities, the effectiveness of redundant systems and defense-in-depth design principles, and the value of effective programs for operator training and emergency preparedness.

Our vulnerability studies confirm that it would be difficult for even determined adversaries to both damage the reactor core and release radioactivity that could affect public health and safety. Further, the studies confirm that even in the unlikely event of a radiological release due to terrorist use of a large aircraft, NRC's emergency planning basis remains valid. The aircraft vulnerability studies also indicate that significant damage to a spent fuel pool is not likely, that it is highly unlikely that the impact on a dry spent fuel storage cask would cause a significant release of radioactivity, and that the impact of a large aircraft on a transportation cask would not result in a release of radioactive material. Measures are in place to adequately protect the public from attacks on spent fuel, in either wet or dry configurations. Thus, we conclude that nuclear power plant safety, security, and emergency planning programs continue to provide reasonable assurance of adequate protection of the public health and safety and protection of the common defense and security.

Force-on-Force Exercises

We have continued to improve our security performance evaluation program (our force-on-force evaluations), which we consider an important element for ensuring protection of the Nation's critical infrastructure. In February 2003, we resumed the force-on-force program in the form of a pilot program to test recent program enhancements. In February 2004, the NRC began a transition force-on-force program, incorporating the lessons learned during the pilot program. The transition program follows the same format as the pilot program; however, the "mock adversary" force now uses the characteristics of the Design Basis Threat (DBT), as enhanced and supplemented by our Orders, to prepare for resumption of the full security performance assessment program in November 2004. Under that program, we will conduct approximately 22 force-on-force exercises per year, so that each site's security will undergo an NRC evaluated exercise at least once every three years. This represents a significant increase in the exercise frequency; in addition, each plant is required to conduct independent exercises at least once each year.

During the pilot program, the NRC identified the need to improve the offensive capabilities, consistency, and effectiveness of the exercise adversary force. The Commission addressed this need by directing the staff to develop a training standard for a Composite Adversary Force (CAF). The CAF for a given NRC-evaluated force-on-force exercise will comprise security officers from various nuclear power facilities (excluding the licensee being evaluated) and will have been trained in offensive, rather than defensive, skills to perform the adversary function.

Baseline Inspection Program

The NRC's oversight program for security is far broader than the force-on-force exercises, vulnerability assessments, strategic and tactical threat assessment, and security plan reviews. It also includes a comprehensive baseline inspection program to verify the continued effectiveness of security measures and confirm compliance. The baseline inspection program for power reactors is part of NRC's Reactor Oversight Program. Through a sampling of licensee security activities, the NRC assesses whether the licensee's security program complies with requirements and provides adequate protection against the DBT for radiological sabotage. Before September 11, the NRC security oversight program focused on four key areas. Shortly after the September 11 attacks, the NRC appropriately refocused portions of its inspection program on verifying licensee implementation of the upgrades specified in NRC-issued advisories and Orders.

Since then, the NRC has substantially revised the baseline inspection program and involved the NRC on-site resident inspectors to a greater degree than before in overseeing security at the plants, ensuring that the NRC has real-time assessments of the status of security on the sites. The NRC implemented a revised baseline inspection program in mid-February 2004 that focuses on an expanded set of key areas, including: (1) access authorization, (2) access controls, (3) security plan changes, (4) contingency response and force-on-force testing, (5) security equipment performance, testing, and maintenance, (6) security training, (7) fitness for duty program, (8) owner controlled area controls, (9) information technology security, (10) material control and accountability, and (11) physical protection of shipments of spent fuel.

The NRC is continuing to enhance and adjust the oversight program for security by developing and implementing more effective processes to assess the significance of inspection findings, more meaningful performance indicators, and revised inspection procedures as a result of the ongoing vulnerability assessment activities and related mitigating strategies.

These changes have allowed the NRC to enhance the effectiveness and efficiency of its oversight of the security measures deemed essential by the Commission after the September 11 attacks. Onsite security inspection hours per year have increased considerably since September 11. Through audits and inspections of the security programs, NRC inspectors confirm that the required security enhancements are implemented.

Status of Security Plan Reviews

As discussed above, in Orders issued on April 29, 2003, the NRC required that licensees take steps to increase protections against a new level of threat, including changes to the adversary force composition and characteristics in light of information gained from the Intelligence Community since September 11, 2001. As part of the Orders, we required nuclear power reactor and certain fuel cycle facilities to develop new security plans, safeguards contingency plans, and training and qualification plans, and to submit them to the NRC by April 29, 2004, for review and approval. The purpose of the Commission's action was to ensure that licensees' plans were revised to specifically describe how the requirements in the NRC's security regulations and post-September 11 Orders were or will be implemented.

When fully implemented, the measures described in the revised plans provide greater capability to respond to more robust attacks than previously required. The new plans also cover a broader spectrum of contingency actions, provide for better trained and qualified security force members, and ensure that more time is devoted to exercises and drills designed to improve the skills of the licensees' guard forces.

The NRC assembled a dedicated team of NRC staff members to review the plans submitted by the industry. As of today, the NRC staff is completing its technical reviews of these plans, and is now working to complete the necessary written safety evaluations and licensing documents to formalize analysis and conclusions associated with each plan review.

Emergency Preparedness

The NRC has long required that its licensees maintain and frequently exercise plans designed to deal with response to emergencies at their plants. State and local agencies, and sometimes Federal agencies, participate in these exercises. The scenarios developed for these plans include many catastrophic events, which are the result of equipment malfunctions, operator errors, or natural disasters. The NRC continues to work with the U.S. Department of Homeland Security and other Federal agencies to integrate Federal Response Plans into a unified National Response Plan and National Incident Management System, and to refine the National Preparedness Policy. We have also completed the development of the commercial Nuclear Reactors, Materials, and Waste Key Resource Plan for Critical Infrastructure Protection. This document serves as the Sector-Specific component of the National Infrastructure Protection Plan. In addition, we continue to coordinate protective strategies with various components of the U.S. Department of Defense, including NORTHCOM and NORAD, and have recently participated in exercises such as Unified Defense '04 and Amalgam Virgo '04. The NRC has revised its Strategic Plan to enhance the recognition of the importance of physical security and emergency preparedness, and licensees will be expected to maintain a high level of preparedness and performance in these areas.

Sharing of Information

The NRC has sought stakeholder input for the many actions taken since September 11. Due to the sensitive, non-public nature of most of the security information, there have been limitations on public access. In order to expand the sphere of the discussion as far as possible, the NRC has had State outreach meetings, a workshop attended by State and local government homeland security advisors, and public meetings to discuss security. The NRC also posts information on its web site to keep the public informed of actions taken and plans for the future.

In coordination with other Federal agencies, the NRC developed a database of reported security incidents, referred to as the Security Information Database (SID), which contains security reports issued by nuclear plant licensees as a result of advisories that NRC issues. Each report that NRC receives and adds to the SID provides details about a specific security incident that has occurred at a nuclear plant (e.g., suspicious person, suspicious activity, flyovers) and the actions that plant officials are taking to address the incident. SID reports are considered sensitive information and are handled accordingly. This information is posted on a protected web site and shared with authorized nuclear industry officials and Federal, State, and local government agencies.

The NRC is committed to ensuring openness in its regulatory programs and makes every attempt to make as much information as possible available to the public, as well as obtain public input in its decision making. At the same time, the NRC is necessarily interested in ensuring that sensitive information regarding nuclear facilities does not fall into the hands of those who wish to do us harm. After careful consideration, the Commission has decided that certain security information, previously released to the public, will no longer be publicly available and will no longer be updated on our web site. The NRC's public web site will continue to display performance indicators, inspection reports, and other information not related to plant security. The Commission's decision enhances the protection of information related to the security of licensed facilities, but will not diminish NRC's commitment to openness in carrying out our public health and safety responsibilities.

Addressing the desire of local officials to more frequently and directly communicate with NRC on emergency preparedness, we increased our interactions with State and local emergency preparedness officials. We have supported workshops, meetings and other activities addressing emergency planning issues such as potassium iodide use, radiological dose assessment, communications during event response, and the like. We will continue these efforts whenever important, specific issues are raised.

NRC Computer Security

The NRC recognizes the importance of providing a comprehensive framework for ensuring the effectiveness of information security controls over information resources that support Federal operations and assets and provides for development and maintenance of controls required to protect Federal information and information systems. The NRC has historically been focused on technical safety and security issues, and computer security is another facet of that overall concern. Congressional oversight and participation in Federal Chief Information Officer groups have helped focus our computer security efforts to more effectively protect our computer systems. NRC has had a computer security program since 1980 and our focus on computer security from project inception and throughout the project life cycle has enabled us to appropriately protect our computer systems.

The NRC received an "A" on the Federal computer security report card issued by the House Government Reform Subcommittee on Technology, Information Policy, Intergovernmental Relations and the Census. The NRC operates with offices across the Nation and interacts with the public in general informational, regulatory, and discovery interchanges. In each of these interchanges, we take the inherent computer security requirements very seriously and work toward a seamless integration of computer security in our day-to-day operations.

Legislative Needs

Over the years, the NRC has repeatedly expressed its support for enactment of legislation needed to strengthen the security of facilities regulated by the Commission. Although the Commission has used existing authority to ensure robust security for nuclear power plants and high risk radioactive materials, prompt enactment of these provisions would grant the statutory authority for steps that we believe should be taken to further enhance the protection of the country's nuclear infrastructure and prevent malevolent use of radioactive material.

The proposals that the Commission believes to be most important are: (1) authorization of security personnel at NRC-regulated facilities and activities to receive, possess, and, in appropriate circumstances, use more powerful weapons against terrorist attacks, (2) enlargement of the classes of NRC-regulated entities and activities whose employees are subject to fingerprinting and criminal history background checks, (3) Federal criminalization of unauthorized introduction of dangerous weapons into nuclear facilities, (4) Federal criminalization of sabotage of additional classes of nuclear facilities, fuel, and material, and (5) extension of NRC's regulatory oversight to discrete sources of accelerator-produced radioactive material and radium-226.

All but the last of these provisions are contained in H.R. 6, as approved by the conferees on that bill in the first session of this Congress, and in S. 2095, which has been introduced in this session. The major part of the last provision is contained in S. 1043, which was reported by the Senate Committee on Environment and Public Works in the first session of this Congress. Accelerator-produced radioactive material and radium-226 are not now covered by the Atomic Energy Act, and while there is other radioactive material that is not subject to the regulatory authority of the NRC, discrete sources of accelerator-produced radioactive material and radium-226 are of the greatest concern in our effort to develop uniform national standards to prevent malevolent use of nuclear material.

A copy of the five proposals listed above has been appended to this testimony and the Commission looks forward to working with you on their enactment in this session of Congress.

I appreciate the opportunity to appear before you today and look forward to answering any questions you may have.

NRC-REQUESTED NUCLEAR SECURITY LEGISLATION

SEC. 1. FINGERPRINTING FOR CRIMINAL HISTORY RECORD CHECKS.

(a) In General- Subsection a. of section 149 of the Atomic Energy Act of 1954 (42 U.S.C. 2169(a)) is amended--

(1) by striking `a. The Nuclear' and all that follows through `section 147.' and inserting the following:

`a. In General-

`(1) REQUIREMENTS-

`(A) IN GENERAL- The Commission shall require each individual or entity--

`(i) that is licensed or certified to engage in an activity subject to regulation by the Commission;

`(ii) that has filed an application for a license or certificate to engage in an activity subject to regulation by the Commission; or

`(iii) that has notified the Commission, in writing, of an intent to file an application for licensing, certification, permitting, or approval of a product or activity subject to regulation by the Commission,

to fingerprint each individual described in subparagraph (B) before the individual is permitted unescorted access or access, whichever is applicable, as described in subparagraph (B).

`(B) INDIVIDUALS REQUIRED TO BE FINGERPRINTED- The Commission shall require to be fingerprinted each individual who--

`(i) is permitted unescorted access to--

`(I) a utilization facility; or

`(II) radioactive material or other property subject to regulation by the Commission that the Commission determines to be of such significance to the public health and safety or the common defense and security as to warrant fingerprinting and background checks; or

`(ii) is permitted access to safeguards information under section 147.';

(2) by striking `All fingerprints obtained by a licensee or applicant as required in the preceding sentence' and inserting the following:

`(2) SUBMISSION TO THE ATTORNEY GENERAL- All fingerprints obtained by an individual or entity as required in paragraph (1)';

(3) by striking `The costs of any identification and records check conducted pursuant to the preceding sentence shall be paid by the licensee or applicant.' and inserting the following:

`(3) COSTS- The costs of any identification and records check conducted pursuant to paragraph (1) shall be paid by the individual or entity required to conduct the fingerprinting under paragraph (1)(A).'; and

(4) by striking `Notwithstanding any other provision of law, the Attorney General may provide all the results of the search to the Commission, and, in accordance with regulations prescribed under this section, the Commission may provide such results to licensee or applicant submitting such fingerprints.' and inserting the following:

`(4) PROVISION TO INDIVIDUAL OR ENTITY REQUIRED TO CONDUCT FINGERPRINTING- Notwithstanding any other provision of law, the Attorney General may provide all the results of the search to the Commission, and, in accordance with regulations prescribed under this section, the Commission may provide such results to the individual or entity required to conduct the fingerprinting under paragraph (1)(A).'.

(b) Administration- Subsection c. of section 149 of the Atomic Energy Act of 1954 (42 U.S.C. 2169(c)) is amended--

(1) by striking `, subject to public notice and comment, regulations--' and inserting `requirements--'; and

(2) by striking, in paragraph (2)(B), `unescorted access to the facility of a licensee or applicant' and inserting `unescorted access to a utilization facility, radioactive material, or other property described in subsection a.(1)(B)'.

(c) Biometric Methods- Subsection d. of section 149 of the Atomic Energy Act of 1954 (42 U.S.C. 2169(d)) is redesignated as subsection e., and the following is inserted after subsection c.:

`d. Use of Other Biometric Methods- The Commission may satisfy any requirement for a person to conduct fingerprinting under this section using any other biometric method for identification approved for use by the Attorney General, after the Commission has approved the alternative method by rule.'.

SEC.2. USE OF FIREARMS BY SECURITY PERSONNEL OF LICENSEES AND CERTIFICATE HOLDERS OF THE COMMISSION.

Section 161 of the Atomic Energy Act of 1954 (42 U.S.C. 2201) is amended by adding at the end the following subsection:

`(z)(1) notwithstanding section 922(o), (v), and (w) of title 18, United States Code, or any similar provision of any State law or any similar rule or regulation of a State or any political subdivision of a State prohibiting the transfer or possession of a handgun, a rifle or shotgun, a short-barreled shotgun, a short-barreled rifle, a machinegun, a semiautomatic assault weapon, ammunition for the foregoing, or a large capacity ammunition feeding device, authorize security personnel of licensees and certificate holders of the Commission (including employees of contractors of licensees and certificate holders) to receive, possess, transport, import, and use 1 or more of those weapons, ammunition, or devices, if the Commission determines that--

`(A) such authorization is necessary to the discharge of the security personnel's official duties; and

`(B) the security personnel--

`(i) are not otherwise prohibited from possessing or receiving a firearm under Federal or State laws pertaining to possession of firearms by certain categories of persons;

`(ii) have successfully completed requirements established through guidelines implementing this subsection for training in use of firearms and tactical maneuvers;

`(iii) are engaged in the protection of--

`(I) facilities owned or operated by a Commission licensee or certificate holder that are designated by the Commission; or

`(II) radioactive material or other property owned or possessed by a person that is a licensee or certificate holder of the Commission, or that is being transported to or from a facility owned or operated by such a licensee or certificate holder, and that has been determined by the Commission to be of significance to the common defense and security or public health and safety; and

`(iv) are discharging their official duties.

`(2) Such receipt, possession, transportation, importation, or use shall be subject to--

`(A) chapter 44 of title 18, United States Code, except for section 922(a)(4), (o), (v), and (w);

`(B) chapter 53 of title 26, United States Code, except for section 5844; and

`(C) a background check by the Attorney General, based on fingerprints and including a check of the system established under section 103(b) of the Brady Handgun Violence Prevention Act (18 U.S.C. 922 note) to determine whether the person applying for the authority is prohibited from possessing or receiving a firearm under Federal or State law.

`(3) This subsection shall become effective upon the issuance of guidelines by the Commission, with the approval of the Attorney General, to govern the implementation of this subsection.

`(4) In this subsection, the terms `handgun', `rifle', `shotgun', `firearm', `ammunition', `machinegun', `semiautomatic assault weapon', `large capacity ammunition feeding device', `short-barreled shotgun', and `short-barreled rifle' shall have the meanings given those terms in section 921(a) of title 18, United States Code.'.

SEC.3. UNAUTHORIZED INTRODUCTION OF DANGEROUS WEAPONS.

Section 229 a. of the Atomic Energy Act of 1954 (42 U.S.C. 2278a(a)) is amended in the first sentence by inserting `or subject to the licensing authority of the Commission or to certification by the Commission under this Act or any other Act' before the period at the end.

SEC. 4. SABOTAGE OF NUCLEAR FACILITIES OR FUEL.

(a) In General- Section 236 a. of the Atomic Energy Act of 1954 (42 U.S.C. 2284(a)) is amended--

(1) in paragraph (2), by striking `storage facility' and inserting `storage, treatment, or disposal facility';

(2) in paragraph (3)--

(A) by striking `such a utilization facility' and inserting `a utilization facility licensed under this Act'; and

(B) by striking `or' at the end;

(3) in paragraph (4)--

(A) by striking `facility licensed' and inserting `, uranium conversion, or nuclear fuel fabrication facility licensed or certified'; and

(B) by striking the comma at the end and inserting a semicolon; and

(4) by inserting after paragraph (4) the following:

`(5) any production, utilization, waste storage, waste treatment, waste disposal, uranium enrichment, uranium conversion, or nuclear fuel fabrication facility subject to licensing or certification under this Act during construction of the facility, if the destruction or damage caused or attempted to be caused could adversely affect public health and safety during the operation of the facility;

`(6) any primary facility or backup facility from which a radiological emergency preparedness alert and warning system is activated; or

`(7) any radioactive material or other property subject to regulation by the Nuclear Regulatory Commission that, before the date of the offense, the Nuclear Regulatory Commission determines, by order or regulation published in the Federal Register, is of significance to the public health and safety or to common defense and security,'.

(b) Penalties- Section 236 of the Atomic Energy Act of 1954 (42 U.S.C. 2284) is amended by striking `$10,000 or imprisoned for not more than 20 years, or both, and, if death results to any person, shall be imprisoned for any term of years or for life' both places it appears and inserting `$1,000,000 or imprisoned for up to life without parole'.

SEC.5. TREATMENT OF ACCELERATOR-PRODUCED AND OTHER RADIOACTIVE MATERIAL AS BYPRODUCT MATERIAL

(a) DEFINITION OF BYPRODUCT MATERIAL.--Section 11 e. of the Atomic Energy Act of 1954 (42 U.S.C. 2014 (e)) is amended--

(1) by striking "The term 'byproduct material' means" and inserting the following: "The term 'byproduct material' means–";

(2) by inserting on the line following "The term 'byproduct material' means–" the clause in section 11 e. that begins "(1) any radioactive material";

(3) by striking ", and" at the end of clause (1) of section 11 e. and inserting ";";

(4) by inserting on the line following the semicolon added by clause (3) the clause in section 11 e. that begins "(2) the tailings or wastes";

(5) by striking "content." at the end of clause (3) in section 11 e. and inserting "content; and"; and

(6) by inserting on the line following "content; and" the following:

"(3)(A) any discrete source of radium-226 that is produced, extracted, or converted after extraction, before, on, or after the date of enactment of this paragraph, for use in a commercial, medical, or research activity; or

"(B) any material that --

"(i) has been made radioactive by use of a particle accelerator; and

"(ii) is produced, extracted, or converted after extraction, before, on, or after the date of enactment of this paragraph, for use in a commercial, medical, or research activity; and

"(4) any discrete source of naturally occurring radioactive material, other than source material that --

"(A) the Nuclear Regulatory Commission determines (after consultation with the Administrator of the Environmental Protection Agency, the Secretary of Energy, the Secretary of Homeland Security, and the head of any other appropriate Federal agency), would pose a threat similar to that posed by a discrete source of radium-226 to the public health and safety or the common defense and security; and

"(B) before, on, or after the date of enactment of this paragraph, is extracted or converted after extraction, for use in a commercial, medical, or research activity.".

(b) AGREEMENTS.--Section 274 b. of the Atomic Energy Act of 1954 (42 U.S.C. 2021) is amended--

(1) by redesignating paragraphs (3) and (4) as paragraphs (5) and (6), respectively; and

(2) by inserting after paragraph (2) the following:

"(3) byproduct materials (as defined in section 11 e.(3));

"(4) byproduct materials (as defined in section 11 e.(4));".

(c) REGULATIONS.--

(1) IN GENERAL.--Not later than the effective date of this section, the Nuclear Regulatory Commission shall promulgate final regulations establishing such requirements and standards as the Commission considers necessary for the acquisition, possession, transfer, use, or disposal of byproduct material (as defined in paragraphs (3) and (4) of section 11 e. of The Atomic Energy Act of 1954 (as added by subsection (a))).

(2) COOPERATION.--The Commission shall cooperate with the States in formulating the regulations under paragraph (1).

(3) TRANSITION.--To ensure an orderly transition of regulatory authority with respect to byproduct material as defined in paragraphs (3) and (4) of section 11 e. of the Atomic Energy Act of 1954 (as added by subsection (a)), not later than 180 days before the effective date of this section, the Nuclear Regulatory Commission shall prepare and provide public notice of a transition plan developed in coordination with States that--

(A) have not, before the effective date of this section, entered into an agreement with the Commission under section 274 b. of the Atomic Energy Act of 1954 (42 U.S.C. 2021); or

(B) in the case of a State that has entered into such an agreement, has not, before the effective date of this section, applied for an amendment to the agreement that would permit assumption by the State of regulatory responsibility for such byproduct material.

(d) WASTE DISPOSAL.--

(1) Notwithstanding any other Federal or State law or any action that has been taken to implement such law, commencing with the effective date of subsection (a), byproduct material as defined in section 11 e.(3) and (4) of the Atomic Energy Act of 1954 may be transferred to and disposed of--

(A) in a disposal facility licensed by the Commission, if the disposal meets the requirements of the Commission, or

(B) in a disposal facility licensed by a State that has entered into an agreement with the Commission under section 274b. of the Atomic Energy Act of 1954, if the disposal meets requirements of the State that are equivalent to the requirements of the Commission.

(2) Notwithstanding the provisions of paragraph (1), byproduct material as defined in section 11 e.(3) and (4) of the Atomic Energy Act of 1954 may be disposed of under the provisions of Title II of the Solid Waste Disposal Act (42 U.S.C. 6901 et seq.), popularly known as the "Resource Conservation and Recovery Act," to the same extent as such material was subject to those provisions before the enactment of this section.

(3) Byproduct material as defined in section 11 e.(3) and (4) of the Atomic Energy Act of 1954 shall not be considered low-level radioactive waste as defined in title I of the Low-Level Radioactive Waste Policy Amendments Act of 1985, or in

implementing any Congressionally approved Compact entered into pursuant to the Low-Level Radioactive Policy Act of 1980 as amended.

(e) EFFECTIVE DATE.--Except with respect to matters that the Nuclear Regulatory Commission determines are required to be addressed earlier to protect the public health and safety or to promote the common defense and security, the amendments made by this section take effect on the date that is 4 years after the date of enactment of this Act.

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U.S. Nuclear Regulatory Commission

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April 22, 2004

Mr. James P. Riccio
Nuclear Policy Analyst
Greenpeace
702 H Street, NW, Suite 300
Washington, DC 20001

Dear Mr. Riccio:

This letter responds to the petition dated August 25, 2003, from you and the other Petitioners (the Nuclear Information & Resource Service and the Union of Concerned Scientists) regarding Davis- Besse Nuclear Power Station, Unit 1 (Davis- Besse), as supplemented by the additional information provided during the September 17, 2003, meeting between the Nuclear Regulatory Commission (NRC) staff and the Petitioners' representatives. Both the August 25 letter and the transcript from the September 17 meeting are available at http:// www. nrc. gov/ reactors/ [illegible]ing/ ops- experience/ vessel- head- degradation/ controlled- correspondence. html. [illegible]C was requested to take enforcement actions against FirstEnergy Nuclear Operating C[illegible]ny (FirstEnergy), the licensee for Davis- Besse, and to suspend the Davis- Besse license and prohibit plant restart until certain conditions have been met. As basis for the request to have the NRC take enforcement actions against the licensee, the Petitioners stated that FirstEnergy has failed to complete commitments related to the NRC's 50.54(f) design basis letter (issued on October 9, 1996), and referred to numerous design basis violations dating back to plant licensing (corresponding to Requests 1 and 2 in the Petitioners' August 25 letter). The Petitioners also requested that the NRC suspend the Davis- Besse license and prohibit plant restart until all design basis deficiencies identified in response to the NRC's 50.54(f) design basis letter are adequately addressed, the plant probabilistic risk assessment is updated to reflect design flaws, and no systems are in a " degraded but operable" condition (corresponding to Requests 3, 4, and 5 in the Petitioners' August 25 letter).

In the NRC's October 7, 2003, acknowledgment letter to the Petitioners' August 25 letter, we stated that the NRC staff would provide its findings on the Petitioners' requests for " immediate action" before the Davis- Besse plant is allowed to restart. The staff considered the Petitioners' requests to suspend the Davis- Besse license and prohibit plant restart until certain conditions have been met to be equivalent to " immediate action" requests because the Davis- Besse licensee might complete all necessary restart activities, and the NRC staff might complete all necessary oversight activities, before the staff could finalize the Director's Decision on this Petition.

Requests 3, 4, and 5 in the Petitioners' August 25 letter were considered immediate action requests, and the staff's evaluation of each of these requests was provided in its letter to the Petitioners dated November 26, 2003. This evaluation is repeated in the enclosed Director's Decision. The October 7 and November 26 NRC letters are available at the Web site location referred to above.

J. Riccio - 2-

The staff's Director's Decision addresses the two remaining Petitioners' requests for enforcement actions (Requests 1 and 2 described in the Petitioners' August 25 letter). In Request 1, the Petitioners requested the NRC to " take enforcement actions against First En[illegible]y Nuclear Operating Company for failure to live up to their commitments made in r[illegible]e to the NRC's October 1996 10 CFR 50.54(f) letter. Since the 10 CFR 50.54(f) letter w[illegible]ued in direct response to the problems at Millstone that netted its owner a record $ 2.1 million fine from the NRC, failure to heed the Millstone warning should carry at least an equivalent sanction." In Request 2, the Petitioners requested the NRC to " take enforcement actions against First Energy Nuclear Operating Company for the numerous design basis violations dating back to the date of licensure with penalties for each day that the licensee was out of compliance with NRC regulations."

With respect to the first request for enforcement action, the NRC staff finds that the Petitioners' r[illegible]t for enforcement based solely on failure of the licensee to complete commitments r[illegible]nts a misinterpretation of the agency's enforcement policies regarding commitments. R[illegible]able assurance of adequate protection of public health and safety is, as a general matter, defined by the Commission's health and safety regulations themselves. In most cases, the agency cannot take formal enforcement actions solely on the basis of whether licensees fulfill commitments, as failure to meet a commitment in itself does not constitute a violation of a legally binding requirement. However, when failures to meet commitments result in violations of the Commission's health and safety regulations, the staff will take the appropriate enforcement actions. Although the staff has not taken any formal enforcement actions against FirstEnergy in direct response to any failures to meet commitments, the staff has taken formal enforcement actions, as discussed in the enclosed Director's Decision, against the licensee for noncompliance with NRC requirements.

Therefore, the Petitioners' request for enforcement actions based solely on any failures on the part of the licensee to not fully comply with commitments made in response to the 50.54(f) letter is denied. Formal enforcement actions are taken when there is a noncompliance with NRC requirements, and the severity of those actions are based in part on the degree of risk posed by that noncompliance.

With respect to the second request for enforcement action, the NRC staff finds that the Petitioners' request for enforcement based on numerous design basis violations is in effect being granted by the actions already taken by the staff, as detailed in the enclosed Director's Decision.

It is also important to note that the highest level of staff oversight of licensee activities for plants with performance problems or operational events is governed by NRC Inspection Manual Chapter (IMC) 0350, " Oversight of Operating Reactor Facilities in a Shutdown Condition With Performance Problems," and that the agency has been overseeing the licensee's activities using this process since May 3, 2002. The decision by the staff to place the Davis- Besse licensee in the highest level of staff oversight was based on the identified performance deficiencies, and was also intended to assure close coordination between NRC and licensee personnel on the corrective actions needed to assure safe plant restart. Any additional enforcement actions, as requested by the Petitioners, would not increase this level of staff oversight, which is directed at assuring that the plant is capable of safe operation in accordance with the Commission's rules and regulations.

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T[illegible]C sent a copy of the proposed Director's Decision to the Petitioners and to the licensee for comment on February 5, 2004. Neither the Petitioners nor the licensee provided comments on the proposed Director's Decision.

A copy of the enclosed Director's Decision (DD- 04- 01) providing the staff's conclusions on the Petition requests will be filed with the Secretary of the Commission for the Commission to review in accordance with 10 CFR 2.206(c). As provided by this regulation, the decision will constitute the final action of the Commission 25 days after the date of decision unless the Commission, on its own motion, institutes a review of the decision within that time.

I have also enclosed a copy of the notice of " Issuance of the Director's Decision Under 10 CFR 2.206" that has been filed with the Office of the Federal Register for publication.

Sincerely,
/ RA/
J. E. Dyer, Director
Office of Nuclear Reactor Regulation
Docket No. 50- 346
Enclosures: 1. Director's Decision
2. Notice of Issuance of Director's Decision
cc w/ encls: See next page

□

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Oak Harbor, OH 43449- 9760

☐
J. Riccio - 3-

The NRC sent a copy of the proposed Director's Decision to the Petitioners and to the licensee for comment on February 5, 2004. Neither the Petitioners nor the licensee provided comments on the proposed Director's Decision.

A copy of the enclosed Director's Decision (DD- 04- 01) providing the staff's conclusions on the Petition requests will be filed with the Secretary of the Commission for the Commission to [illegible] in accordance with 10 CFR 2.206(c). As provided by this regulation, the decision will [illegible]tute the final action of the Commission 25 days after the date of decision unless the Commission, on its own motion, institutes a review of the decision within that time.

I have also enclosed a copy of the notice of " Issuance of the Director's Decision Under 10 CFR 2.206" that has been filed with the Office of the Federal Register for publication.

Sincerely,
/ RA/
J. E. Dyer, Director
Office of Nuclear Reactor Regulation
Docket No. 50- 346
Enclosures: 1. Director's Decision
2. Notice of Issuance of Director's Decision
cc w/ encls: See next page
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Incoming GT20030508: ML032400435
Response: ML0 40980367 NRR- 106 * See previous concurrence ** Concurred by phone
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OPA
JCraig
JGoldberg
DB0350
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DD- 04- 01
U[illegible] STATES OF AMERICA
N[illegible] REGULATORY COMMISSION
OF[illegible]E OF NUCLEAR REACTOR REGULATION

J. E. Dyer, Director
In the Matter of) Docket No. 50- 346
)
FirstEnergy Nuclear Operating Company) License No. NPF- 3
)
[illegible]
([illegible]s- Besse Nuclear Power Station, Unit 1))
)

DIRECTOR'S DECISION UNDER 10 CFR 2.206

I. Introduction

By letter dated August 25, 2003, Greenpeace filed a Petition pursuant to Section 2.206 of Title 10 of the Code of Federal Regulations (10 CFR) on behalf of the Nuclear Information & Resource Service and the Union of Concerned Scientists (collectively, the Petitioners). The Petitioners requested that the Nuclear Regulatory Commission (NRC) take enforcement actions against FirstEnergy Nuclear Operating Company (FirstEnergy), the licensee for Davis- Besse Nuclear Power Station in Oak Harbor, Ohio, and also requested that NRC suspend the Davis-Besse license and prohibit plant restart until certain conditions have been met. As basis for the request to have the NRC take enforcement actions against the licensee, the Petitioners stated that FirstEnergy has failed to complete commitments related to the NRC's 50.54(f) design basis letter (issued on October 9, 1996), and referred to numerous design basis violations dating back to plant licensing (corresponding to Requests 1 and 2 in the Petitioners' August 25 letter). The Petitioners also requested that the NRC suspend the Davis- Besse license and prohibit plant restart until all design basis deficiencies identified in response to the NRC's 50.54(f) design basis letter are adequately addressed, the plant probabilistic risk assessment (PRA) is

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updated to reflect design flaws, and no systems are in a " degraded but operable" condition (corresponding to Requests 3, 4, and 5 in the Petitioners' August 25 letter).

In a letter dated October 7, 2003, the NRC informed the Petitioners that the issues in the Petition were accepted for review under 10 CFR 2.206 and had been referred to the Office of Nuclear Reactor Regulation for appropriate action. A copy of the acknowledgment letter is publicly available in the NRC's Agencywide Documents Access and Management System (ADAMS) under Accession No. ML032690314. A copy of the Petition is publicly available in ADAMS under the Accession No. ML032400435.

The Petitioners' representatives met with NRC staff on September 17, 2003, to provide additional details in support of this request. This meeting was transcribed and the transcript is publicly available on the NRC Web site as a supplement to the Petition (http:// www. nrc. gov/ [illegible]rs/ operating/ ops- experience/ vessel- head- degradation/ controlled- correspondence. html). [illegible]censee responded to the Petition on October 20, 2003 (ML033421458). This response was considered by the staff in its evaluation of the Petition.

In a letter dated November 26, 2003 (ML033010172), the NRC provided to the Petitioners its evaluation of their " immediate action" requests. The staff considered the Petitioners' requests to suspend the Davis- Besse license and prohibit plant restart until certain conditions have been met to be equivalent to " immediate action" requests because the Davis-Besse licensee might complete all necessary restart activities, and the NRC staff might complete all necessary oversight activities, before the staff could finalize the Director's Decision on this Petition. Requests 3, 4, and 5 in the Petitioners' August 25 letter were considered immediate action requests, and the staff's November 26 evaluation is repeated in Section II. D for completeness.

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The NRC sent a copy of the proposed Director's Decision to the Petitioners and to the licensee for comment on February 5, 2004 (ML040280003). Neither the Petitioners nor the licensee provided comments on the proposed Director's Decision.

II. Discussion

This section contains a discussion of the agency's regulatory oversight process, relevant NRC enforcement policies, the NRC staff's response to the Petitioners' requests for enforcement action (Requests 1 and 2 in the Petitioners' August 25, 2003, letter), and for completeness, the staff's November 26, 2003, response to the Petitioners' immediate action requests (Requests 3, 4, and 5 in the Petitioners' August 25, 2003, letter).

The objective of the descriptive information presented below on the agency's processes and policies is to provide a clear understanding of the basis for the staff's findings with respect to the Petitioners' two requests for enforcement action. These findings are summarized below.

- With respect to the first request for enforcement action, the NRC staff finds that the Petitioners' request for enforcement based solely on failure of the licensee to complete commitments represents a misinterpretation of the agency's enforcement policies regarding commitments. As will be discussed later in this Director's Decision, reasonable assurance of adequate protection of public health and safety is, as a general ma[illegible] defined by the Commission's health and safety regulations themselves. In most c[illegible] the agency cannot take formal enforcement actions solely on the basis of w[illegible] licensees fulfill commitments, as failure to meet a commitment in itself does not constitute a violation of a legally binding requirement. However, when failures to meet commitments result in violations of the Commission's health and safety regulations, the staff will take the appropriate enforcement actions. Although the staff has not taken any formal enforcement actions against FirstEnergy solely for failure to meet commitments, the staff has taken formal enforcement actions against the licensee for noncompliance with NRC requirements, including enforcement actions for failure to meet design- related requirements.

- With respect to the second request for enforcement action, the NRC staff finds that the Petitioners' request for enforcement based on numerous design basis violations (i. e., the licensee event reports [LERs] submitted by the licensee) is in effect being granted by the actions already taken by the staff, as will be evident from the discussions of our processes for reviewing and evaluating LERs presented later in this Director's Decision.

A. Reactor Oversight Process

This section provides a brief overview of the process by which the staff inspects and assesses licensees' compliance with the Commission's rules and regulations. This overview is intended to provide an understanding of what the current regulatory oversight of the Davis- Besse plant consists of, and why this oversight was imposed on this licensee. It is important to note that the agency has been utilizing its process for the highest level of staff oversight for plants with performance problems or operational events in inspecting and assessing the Davis- Besse licensee activities since May 3, 2002. Any additional enforcement actions, as requested by the Petitioners, would not increase this level of staff oversight, which is directed at assuring that the plant is capable of safe operation in accordance with the Commission's rules and regulations.

The fundamental building blocks of the framework for the regulatory oversight process are the seven cornerstones of safety: initiating events, mitigating systems, barrier integrity, emergency preparedness, occupational radiation safety, public radiation safety, and physical protection. These cornerstones are grouped into three strategic areas: reactor safety, radiation safety, and safeguards. This framework is based on the principle that the agency's mission of assuring public health and safety is met when the agency has reasonable assurance that licensees are meeting the objectives of the seven cornerstones of safety.

The Reactor Oversight Process (ROP) integrates the NRC's inspection, assessment, and enforcement programs. Along with performance indicators (PIs), assessment, and enforcement, the reactor inspection program is an integral part of the ROP. Acceptable performance in the cornerstones, as measured by the PIs and the risk- informed baseline inspection program, is indicative of overall performance that provides for adequate protection of public health and safety.

Another principle of the framework is that there is a level of performance for which the NRC does not need to engage the licensee beyond some baseline level of oversight. Performance indicators reported by power reactor licensees and the NRC's inspection program provide the information used in comparing licensee performance against the cornerstones of safety. The risk- informed baseline inspection program is designed to be the inspection oversight that provides indications of performance within areas of the cornerstones of safety that are not measured by the PIs or not adequately measured by PIs.

The Operating Reactor Assessment Program evaluates the overall safety performance of operating commercial nuclear reactors and communicates the results to licensee management, members of the public, and other government agencies.

This assessment program collects information from inspections and PIs to enable the agency to arrive at objective conclusions about the licensee's safety performance. Based on this assessment information, the NRC determines the appropriate level of agency response, including supplemental inspection and pertinent regulatory actions ranging from management meetings up to and including orders for plant shutdown. The assessment information and agency response are then communicated to the public. Followup agency actions, as applicable, are conducted to ensure that the corrective actions designed to address performance weaknesses were effective.

In general, when significant performance problems are identified in one or more of the seven cornerstones in the areas of reactor safety, radiation safety, or security, as defined by NRC Inspection Manual Chapter (IMC) 0305, " Operating Reactor Assessment Program," the level of NRC actions is governed by the Action Matrix (Exhibit 5 of IMC 0305). The Action Matrix was developed with the philosophy that, within a certain level of safety performance (e. g., the licensee response band), licensees would address their performance issues without additional NRC engagement beyond the baseline inspection program. Agency action beyond the baseline inspection program will occur only if assessment input thresholds are exceeded.

The Action Matrix identifies the range of NRC and licensee actions and the appropriate level of communication for varying levels of licensee performance. The Action Matrix describes a graded approach in addressing performance issues. The possible approaches could include additional supplemental inspection, a demand for information, a confirmatory action letter, or issuance of an order, up to and including a plant shutdown. The highest level of staff oversight of licensee activities for plants with performance problems or operational events is governed by IMC 0350, " Oversight of Operating Reactor Facilities in a Shutdown Condition With Performance Problems."

B[illegible]tter dated April 29, 2002, the NRC informed FirstEnergy that its corrective actions [illegible]is- Besse would receive enhanced NRC oversight as described in IMC 0350. The [illegible]on by the staff to place the Davis- Besse licensee in the highest level of staff oversight was based on the identified performance deficiencies, and was also intended to assure close coordination between NRC and licensee personnel on the corrective actions needed to assure safe plant restart. That enhanced monitoring began on May 3, 2002, and included the creation of an oversight panel to provide the required oversight during the plant shutdown, during the startup of the plant following the NRC's letter dated March 8, 2004, which removed the

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restriction the NRC had placed on plant restart, and following restart until a determination is made that the plant is ready for return to the NRC's normal ROP.

When a plant is under the IMC 0350 process, the routine ROP is suspended. However, the ROP continues to be used as guidance. The oversight panel will assess inspection findings and other performance data to determine the required level and focus of followup inspection activities and any other appropriate regulatory actions. The focus of this manual chapter is to provide oversight of the licensee's performance until a return to the routine oversight under the ROP is warranted.

All of the documents referenced in Section II. A are available at the NRC Web site, www. nrc. gov.

B. Relevant Enforcement Policies

This section provides a brief overview of the NRC's scope and authority relative to the enforcement policy, and the processes by which the staff takes enforcement actions relative to licensees' compliance with the Commission's rules and regulations. This overview is intended to provide a general understanding of how and why enforcement actions are taken against licensees, as well as an understanding of the appropriate enforcement actions relative to the specific requests from the Petitioners.

Background

The Atomic Energy Act of 1954, as amended, establishes " adequate protection" as the standard of safety on which NRC regulations are based. In the context of NRC regulations, safety means avoiding undue risk or, stated another way, providing reasonable assurance of adequate protection of workers and the public in connection with the use of source, byproduct, and special nuclear materials.

[illegible] safety is the fundamental regulatory objective, compliance with NRC requirements [illegible] an important role in giving the NRC confidence that safety is being maintained. Under

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Atomic Energy Commission and NRC case law, reasonable assurance of adequate protection of public health and safety is, as a general matter, defined by the Commission's health and safety regulations themselves. That is, unless otherwise provided, there is reasonable assurance of adequate protection of public health and safety when the applicant or licensee demonstrates compliance with the Commission's regulations. NRC requirements, including technical specifications, other license conditions, orders, and regulations, have been designed to ensure adequate protection — which corresponds to " no undue risk to public health and safety" — through acceptable design, construction, operation, maintenance, modification, and quality assurance measures. The regulations were established using defense- in- depth principles and conservative practices which provide a degree of margin to unsafe levels. In the context of risk- informed regulation, compliance plays a very important role in ensuring that key assumptions used in underlying risk and engineering analyses remain valid.

While adequate protection is presumptively assured by compliance with NRC requirements, circumstances may arise where new information reveals that an unforeseen hazard exists or that there is a substantially greater potential for a known hazard to occur. In such situations, the NRC has the statutory authority to require licensee action above and beyond existing regulations to maintain the level of protection necessary to avoid undue risk to public health and safety.

The NRC also has the authority to exercise discretion to permit continued operations — despite the existence of a noncompliance — where the noncompliance is not significant from a risk perspective and does not, in the particular circumstances, pose an undue risk to public health and safety. When noncompliance occurs, the NRC must evaluate the degree of risk posed by that noncompliance to determine if specific immediate action is required. Where needed to ensure adequate protection of public health and safety, the NRC

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ma[illegible]emand immediate licensee action, up to and including a shutdown or cessation of [illegible]ed activities.

B[illegible]n the NRC's evaluation of noncompliance, the appropriate action could include refraining from taking any action, taking specific enforcement action, issuing orders, or providing input to other regulatory actions or assessments, such as increased oversight (e. g., increased inspection). Since some requirements are more important to safety than others, the NRC endeavors to use a risk- informed approach when applying NRC resources to the oversight of licensed activities, including enforcement activities.

The primary purpose of the NRC's Enforcement Policy is to support the NRC's overall [illegible] mission in protecting the public health and safety and the environment. Consistent with [illegible] purpose, the policy endeavors to:

° deter noncompliance by emphasizing the importance of compliance with NRC requirements, and
° encourage prompt identification and prompt, comprehensive correction of violations of NRC requirements.

Therefore, licensees, contractors, and their employees who do not achieve the high standard of compliance which the NRC expects will be subject to enforcement sanctions. Each enforcement action is dependent on the circumstances of the case. However, in no case will licensees who cannot achieve and maintain adequate levels of safety be permitted to continue to conduct licensed activities.

Relevant Enforcement Policies

The Petitioners' requests for enforcement actions against the Davis- Besse licensee are related to commitments made by the licensee in response to the NRC's 1996 50.54(f) letter and to the LERs submitted by the licensee (these two requests are explained more fully in Section II. C).

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With respect to the issue of enforcing commitments, the agency in most cases cannot take formal enforcement actions solely on the basis of whether licensees fulfill commitments, as failure to meet a commitment in itself does not constitute a violation of a legally binding requirement such as a rule, order, license condition, or technical specification. However, when failures to meet commitments result in violations of the Commission's health and safety regulations, the staff will take the appropriate enforcement actions.

With respect to the issue of taking enforcement related to LERs, the staff has processes in place for reviewing LERs submitted by nuclear power plant licensees, and that process includes determining appropriate enforcement actions if violations are identified. A brief description is provided below of how the staff reviews and dispositions LERs. This process is part of the Reactor Oversight Process, described in Section II. A.

NRC inspectors conduct inspections of licensed nuclear power plants following guidance in the NRC Inspection Manual, which contains objectives and procedures to use for each type of inspection. Inspection Procedure 71153, " Event Followup," requires inspectors to [illegible] LERs and related documents for accuracy of the LER, appropriateness of corrective [illegible]s, violations of requirements, and generic issues.

[illegible] LER involves a finding or noncompliance which the licensee entered into its corrective action program, IMC 0612, " Power Reactor Inspection Reports," directs the inspectors to include in the inspection report a description of the safety significance of the event and any appropriate enforcement actions.

The safety significance of LER findings is determined by using the Significance Determination Process (SDP) as defined in IMC 0609, " Significance Determination Process." Each SDP analysis supports a cornerstone associated with the strategic performance areas as defined in IMC 2515, " Light- Water Reactor Inspection Program - Operations Phase." The SDP

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is primarily used to assess the significance of NRC inspection findings, but is also used for other purposes, including the assessment of LER findings.

Depending on their significance, LER findings are assigned colors of:

° green (very low safety significance),
° white (low to moderate safety significance),
° yellow (substantial safety significance), or
° red (high safety significance).

If the LER findings are associated with violations of regulatory requirements, enforcement actions are processed in accordance with the current revision of NUREG- 1600, " General Statement of Policy and Procedures for NRC Enforcement Actions." The significance of the LER findings is considered in the determination of the appropriate enforcement action.

All of the documents referenced in Section II. B are available at the NRC Web site, www. nrc. gov.

C. Staff Response to Petitioners' Requests To Take Enforcement Action

Response to First Request for Enforcement

The first of the two specific requests for enforcement action by the Petitioners was for the NRC to " take enforcement actions against First Energy Nuclear Operating Company for failure to live up to their commitments made in response to the NRC's October 1996 10 CFR 50.54(f) letter. Since the 10 CFR 50.54(f) letter was issued in direct response to the problems at Millstone that netted its owner a record $ 2.1 million fine from the NRC, failure to [illegible] Millstone warning should carry at least an equivalent sanction."

T[illegible]rpose of the 1996 10 CFR 50.54(f) letter was to require information that would provide the NRC added confidence and assurance that U. S. nuclear power plants are operated and maintained within the design bases and any deviations are reconciled in a timely manner.

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As[illegible]ated in the staff's response to the Petitioners' requests for immediate action (see S[illegible]n II. D), the adequacy of safety- significant structures, systems, and components would be a[illegible]ed by the staff's oversight activities and would be adequately addressed before the NRC al[illegible]ed the plant to restart. The staff's oversight activities while the Davis- Besse plant was shut down provided the level of confidence and assurance that this plant meets the objectives stated in the 10 CFR 50.54(f) letter. These oversight activities included system health assurance inspections, inspections of design basis issues, and an inspection of the licensee's actions associated with the completeness and accuracy of required records and submittals to the NRC (Inspection Report 50- 346/ 03- 19, dated January 28, 2004).

In Inspection Report 50- 346/ 03- 19, the staff stated that, based on the documents and corrective actions reviewed and the results of previous NRC inspections of licensee activities under the Davis- Besse Return- to- Service Plan, the NRC has reasonable confidence that important docketed information is complete and accurate in all material respects and that future submittals will be complete and accurate. This inspection identified no widespread noncompliances with regulatory requirements or current programmatic concerns associated with the completeness and accuracy of submittals to the NRC. The inspection report identified three findings, including a noncited violation and an apparent violation, which is being considered for escalated enforcement. The apparent violation involves failure to provide the NRC complete and accurate information as required by 10 CFR 50.9 in the licensee's November 11, 1998, response to NRC Generic Letter 98- 04, " Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss- of- Coolant- Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment." Specifically, information pertaining to unqualified protective coatings and the likelihood of clogging of the containment emergency sump screen was not provided to the NRC in a complete and accurate manner. The licensee submitted a written

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response to the apparent violation on February 27, 2004. The NRC is in the process of making its final enforcement decision.

As stated in the previous section, the agency in most cases cannot take formal enforcement actions solely on the basis of whether licensees fulfill commitments, as failure to meet a commitment in itself does not constitute a violation of a legally binding requirement. Although the staff has not taken any formal enforcement actions against FirstEnergy [illegible] for failure to meet commitments, the staff has taken formal enforcement actions against [illegible]censee for noncompliance with NRC requirements, including enforcement actions for f[illegible]re to meet design- related requirements.

Two recent enforcement actions taken against the Davis- Besse licensee were for systems, structures, and components (SSCs) not configured or maintained in accordance with the plant design and licensing basis. On October 7, 2003, the NRC issued a final significance determination for a yellow finding associated with potential clogging of the emergency sump following a loss- of- coolant accident. In addition, on March 4, 2004, the NRC issued its final significance determination for a white finding for a design issue involving the high- pressure injection pumps. The licensee submitted LERs to the NRC on both of these issues and more detailed discussions of these two issues are included in the staff's response below to the Petitioners' second request for enforcement action.

In response to the Petitioners' reference to the enforcement actions taken against the Millstone licensee, those enforcement actions were for noncompliance with NRC requirements, not solely related to any failures to fulfill commitments. With respect to the civil penalty assessed to the Millstone licensee, it should be noted that the agency's Enforcement Policy has changed since that time in conjunction with adopting the ROP.

Instead of using civil penalties as a deterrent, the NRC uses enforcement actions under the ROP as but one part of the agency's overall regulatory response. The ROP's Action Matrix

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will cause the staff to consider specific regulatory actions based on the risk significance of the issue. Actions might include increased inspections, demands for information, or orders. However, civil penalties (and the use of severity levels) will be considered for issues with actual consequences, such as an overexposure to the public or plant personnel above regulatory limits, failure to make the required notifications, impacting the ability of Federal, State and local agencies to respond to an actual emergency preparedness (site area or general emergency), a transportation event, or a substantial release of radioactive material. Civil penalties and severity levels will also be used to address violations that are willful or that have the potential for impacting the regulatory process.

The use of civil penalties in these instances remains appropriate as a deterrent for these t[illegible] of issues. To the extent that the SDP can provide an assessment of the significance of [illegible]derlying violation or issue, it will be used as a first step in determining the significance of [illegible]olation. This will ensure a consistent approach for significance determinations. The staff co[illegible]ders the SDP output in conjunction with the guiding principles for assessing significance and the guidance included in the supplements to the Enforcement Policy to determine the appropriate severity level. For example, a procedural violation associated with an inspection finding characterized by the SDP as green may be categorized at Severity Level IV based on the risk significance and ultimately assigned a Severity Level III categorization because the violation was willful.

[illegible] are ongoing NRC activities that may lead to civil and/ or criminal proceedings [illegible]st the Davis- Besse licensee. NRC's Office of Investigations (OI) conducted an investigation to determine whether the Davis- Besse licensee willfully violated NRC requirements and whether the licensee willfully misled the NRC. The results of the OI investigation were provided to the U. S. Department of Justice (DOJ) in accordance with the Memorandum of Understanding (MOU) between the NRC and DOJ. The Federal investigation into these

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matters continues under the leadership of the U. S. Attorney in Cleveland supported by the NRC Office of Investigations and the DOJ.

In accordance with Section III. C. of the MOU, after notifying DOJ, the NRC may take immediate actions necessary to protect the public health and safety. Absent such circumstances, the NRC shall normally defer actions such as civil penalties until DOJ concludes its activities. The staff concluded that immediate actions to protect the health and safety of the public are not necessary at this time. A senior NRC manager is monitoring the ongoing Federal investigation for any emerging safety concerns.

Response to Second Request for Enforcement

The second specific request for enforcement action by the Petitioners was for the NRC to " take enforcement actions against First Energy Nuclear Operating Company for the numerous design basis violations dating back to the date of licensure with penalties for each day that the licensee was out of compliance with NRC regulations." As basis for this request, the Petitioners cite the LERs submitted by the licensee since the plant was licensed, and they specifically cite seven LERs that have been submitted by the licensee since mid- 2002.

Based on the NRC's evaluation of each noncompliance reported in the LERs (and other sources such as NRC inspection reports), the appropriate action could include refraining from taking any action, taking specific enforcement action, issuing orders, or providing input to other regulatory actions or assessments, such as increased oversight (e. g., increased inspection).

The NRC endeavors to use a risk- informed approach when applying NRC resources to the oversight of licensed activities, including enforcement activities. As described in the Commission's Enforcement Policy, varying levels of significance using either one of four severity levels or one of four risk levels derived from the ROP are applied to documented violations. Civil penalties can be applied to Severity Levels III, II, and I, but are not normally

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applied to ROP findings that constitute violations. The ROP utilizes other mechanisms, such as increased inspection oversight, to motivate compliance and corrective actions.

As stated in Section II. B, the staff's findings on individual LERs are discussed in resident inspection reports. Of the seven LERs specifically cited by the Petitioners in support of their request for enforcement action, the staff had published inspection reports providing its findings on four at the time the proposed Director's Decision was issued. To illustrate how the staff implements the agency's Enforcement Policy in regard to LER findings, summaries from these published inspection reports for the four LERs are provided below. Of the remaining three LERs specifically mentioned by the Petitioners, two were closed in Inspection Report 50- 346/ 03- 10, dated March 05, 2004 (ML040680070), which documented the NRC's special corrective action team inspection to assess the effectiveness of the implementation of the licensee's corrective action program. The two LERs closed by this report, LER 2002- 006, " EDG [Emergency Diesel Generator] Exhaust Piping Not Adequately Protected From Potential Tornado- Generated Missiles," and LER 2003- 003, " Potential Inadequate High Pressure Injection Pump Minimum Recirculation Flow Following a Small Break Loss of Coolant Accident," were identified as noncited violations, having very low safety significance. The remaining LER, 2003- 007, " AC System Analysis Shows Potential Loss of Offsite Power Following Design Basis Event," is currently open and will be addressed by the same process used to disposition the closed LERs.

LER 2002- 004, " Containment Isolation Closure Requirements for Reactor Coolant Pump (RCP) Seal Injection Valves MU66A- D"

This LER documented a condition where the pressure regulating valve setpoint for the RCP seal injection valves was inadequate to ensure closure of the valves upon receipt of a containment isolation signal. This condition represented a potential common- mode failure. As a result of this condition, during postulated accident conditions, a potential for uncontrolled

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radioactive leakage outside containment could be created. This condition had apparently existed since original plant construction, and is a violation of Technical Specification ([illegible].6.3.1 for Modes 1- 4. For Modes 1, 2, 3, and 4, this TS states, " All containment isolation v[illegible] shall be operable with isolation times less than or equal to required isolation times." Contrary to this requirement, the pressure regulating setpoint for the RCP seal injection valves was inadequate to ensure closure of the valves upon receipt of a containment isolation signal. In addition, the valves were determined to be installed inconsistent with design assumptions. However, downstream of these isolation valves are check valves that are designed to prevent flow out of the reactor coolant system, thereby isolating the flow path regardless of w[illegible]er the RCP seal injection valves are closed. The reliability of the check valves was [illegible]ined to be high based on test history (no test failures in the past 10 years had occurred). [illegible]egional senior reactor analyst performed a Phase 3 assessment in accordance with IMC 0609 and determined that the issue had very low safety significance (green). This determination was due to the low initiating event frequency of an interfacing system loss- ofcoolant accident (ISLOCA), 1E- 7, coupled with the check valve probability of failure to prevent a potential ISLOCA if the RCP seal injection valve failed. The senior reactor analyst also reviewed the licensee's risk assessment and determined that the calculation was conservative given the assumptions used. The licensee's analysis determined that the change in core damage frequency was in the 1E- 8 per year range.

Based on the above evaluation of risk, this LER was closed in Inspection Report 50- 346/ 02- 17 as a licensee- identified noncited violation of TS 3.6.3.1.

LER 2002- 005, " Potential Clogging of the Emergency Sump Due to Debris in Containment"

On September 4, 2002, with the reactor defueled, FirstEnergy determined that a gap in the sump screen larger than allowed by design basis (greater- than- 1/ 4 inch openings) existed.

Also, the existing amount of unqualified containment coatings and other debris (e. g., insulation) inside containment could have potentially blocked the emergency sump intake screen, rendering the sump inoperable following a loss- of- coolant accident. The unqualified coatings and existence of other debris had existed since original construction. FirstEnergy declared the emergency sump inoperable and entered the deficiency into its corrective action program. With the emergency sump inoperable, both independent emergency core cooling system (ECCS) trains and both containment spray (CS) system trains were inoperable, due to both requiring suction from the emergency sump during the recirculation phase of operation. This could prevent both trains of ECCS from removing residual heat from the reactor and could prevent CS from removing heat and fission product iodine from the containment atmosphere.

FirstEnergy reported this information in LER 2002- 05 on November 4, 2002. On December 11, 2002, FirstEnergy submitted Supplement 1, which provided additional information regarding corrective actions for the sump strainer and coatings issues. In this supplement, FirstEnergy stated that a debris generation and transport analysis would be performed. Supplement 2, dated May 21, 2003, provided additional information regarding a[illegible]ional corrective actions. On May 28, 2003, FirstEnergy informed the NRC that a further [illegible] of the past significance of these issues would not be performed.

[illegible]nergy obtained information on at least two occasions prior to issuance of the LER th[illegible] should have alerted them to the unqualified coatings. First, a 1976 letter from Babcock and Wilcox (B& W) informed the Davis- Besse licensee that B& W had no data regarding design basis accident testing for particular coatings. The equipment coated with unqualified paint identified in the letter included the RCP motors, reactor vessel, steam generators, pressurizer, and reactor coolant system piping. Second, NRC Generic Letter 98- 04, " Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss- of- Coolant Accident Because of Construction and Protective Coating Deficiencies and

Foreign Material in Containment," dated July 14, 1998, was issued to operating reactors specifically requesting information about the potential effects of containment coating deficiencies.

On July 3, 2003, a Significance and Enforcement Review Panel meeting was held regarding the significance of the failure to effectively implement corrective actions for design control deficiencies regarding containment coatings, uncontrolled fibrous material, and other debris inside containment. This deficiency resulted in the inability of the ECCS sump to perform its safety function under certain accident scenarios due to clogging of the sump screen. The NRC staff determined that several combinations of factors led to core damage frequency increases in the 1E- 5 (yellow) range.

On July 30, 2003, the NRC issued the preliminary yellow finding in Inspection Report 50- 346/ 03- 15. FirstEnergy provided a written response dated August 29, 2003, acknowledging the performance deficiency. FirstEnergy did not contest the finding and its response provided no new information to change the NRC's preliminary conclusion. On October 7, 2003, the NRC issued the Yellow Final Significance Determination, which included a Notice of Violation of 10 CFR Part 50, Appendix B, Criterion XVI, " Corrective Actions," for the failure to promptly identify and correct significant conditions adverse to quality involving the potential to clog the emergency core cooling and CS system sump with debris following a lossof- coolant accident.

As corrective actions, FirstEnergy performed extensive modifications on the sump du[illegible] the recent extended outage. FirstEnergy replaced the previous emergency sump s[illegible]r with a much larger strainer. The unqualified coatings and other debris, including f[illegible]s insulation remaining in containment, have been walked down, verified, and documented. Much of the fibrous insulation has been removed from containment, and most of the containment internal surfaces and surfaces of equipment inside containment have been re-

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c[illegible]d with qualified paint. Debris generation, transport, strainer head loss, and strainer [illegible]ty analyses were performed for the emergency sump to return the emergency sump to f[illegible]ualification and operability. The NRC inspection of FirstEnergy's new sump is documented in Inspection Report 50- 346/ 03- 06. The NRC identified no significant issues with the new sump. This LER will be closed following the NRC staff's review of the past significance of the gap in the sump screen.

LER 2003- 002, " Potential Degradation of High- Pressure Injection Pumps Due to Debris in Emergency Sump Fluid Post Accident"

On October 22, 2002, with the reactor defueled and in an extended outage, FirstEnergy identified a design deficiency regarding internal clearances of the high- pressure injection (HPI) pumps. This deficiency resulted in operation of the HPI pumps being affected by debris that may be entrained in the process fluid during some post- accident scenarios. Specifically, it was determined that small ports in the HPI pumps that supply lubricating water to the hydrostatic bearing in the pump were smaller than the designed openings in the emergency sump screen. During certain accidents when the reactor coolant system is at high pressure, the HPI pumps are needed to maintain the core cooled by operating in the high- pressure sump recirculation mode of operation and taking suction from the containment sump via the low- pressure injection pumps. It was during this mode of operation that the potential existed for debris from the sump (fibrous insulation, paint chips, and smaller debris such as containment floor dirt) to be transported to the HPI pumps and cause blockage of the ports and loss of lubricating water to the hydrostatic bearing. This could result in failure of the pumps due to excessive vibration/ overheating.

This deficiency was an original design flaw that had existed since initial plant operation. On April 7, 2003, FirstEnergy reported this issue to the NRC in accordance with 10 CFR 50.72. Subsequently, on May 5, 2003, FirstEnergy submitted LER 2003- 02. FirstEnergy modified both

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HPI pumps during the recent extended outage to eliminate the potential for blockage of the ports.

On September 4, 2003, a Significance and Enforcement Review Panel determined the issue to be greater than green because of the large uncertainty in determining the most likely failure probability of the HPI pumps and the contribution to risk from fires. On October 8, 2003, t[illegible]RC issued Inspection Report 50- 346/ 03- 21 transmitting the preliminary greater- than- green [illegible]g to the licensee. On November 7, 2003, FirstEnergy requested an extension on the [illegible]se to the preliminary significance determination. On December 5, 2003, FirstEnergy pr[illegible]ided its analysis of this issue. On March 5, 2004, the NRC issued a white Final Significance Determination, which included a Notice of Violation of 10 CFR Part 50, Appendix B, Criterion III, " Design Control," for the failure to adequately implement design control measures for verifying and checking the adequacy of the original design of the HPI pumps to mitigate all postulated accidents.

Regarding corrective actions, the licensee performed extensive analysis, pump modifications, qualification testing, in- plant testing, and reduction of fibrous insulation in the containment to ensure adequate HPI pump performance during the recirculation mode. The staff conducted a review of the analysis, testing, and modifications performed by the licensee and concluded that the licensee's overall approach to the modification of the HPI pumps was acceptable and provided reasonable assurance that the pumps would perform their required functions when called upon. This review is detailed in the Task Interface Agreement 2003- 04 response dated February 11, 2004, which is included as an attachment to NRC Inspection Report 50- 346/ 04- 02.

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LER 2003- 005, " Containment Gas Analyzer Heat Exchanger Valves Found Closed Rendering the Containment Gas Analyzer Inoperable"

This LER reported the failure by the licensee to establish an appropriate operational test from the original plant startup until May 2003 to ensure that sufficient cooling water flow is provided to the hydrogen analyzer heat exchangers during operational modes that require the hydrogen analyzers to be operable. The hydrogen analyzers are part of the containment hydrogen control system, which is designed to control the concentration of hydrogen which may be released into containment following a LOCA. In accordance with IMC 0609, Appendix A, Attachment 1, the inspectors performed a SDP Phase 1 screening and determined that the issue affected the Reactor Safety Strategic Performance Area. The finding was more than minor because (1) it involved the configuration control attribute of the barrier integrity cornerstone, and (2) it affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents o[illegible]nts.

T[illegible]nding is unrelated to SSCs that are needed to prevent accidents from leading to c[illegible]amage. To determine if this finding had an effect on large early release frequency (LERF), the inspectors used IMC 0609, Appendix H, " Containment SDP." The finding was

characterized as a Type B finding (having no impact on core damage frequency) and was then compared to Table 3 in Appendix H. The inspectors determined that the hydrogen analyzer had no impact on the containment- related SSCs listed in Table 3 (i. e., containment penetration seals, containment isolation valves, or purge and vent lines) and would not influence LERF. On t[illegible]basis, the finding has very low safety significance.
[illegible]e of the very low safety significance and because the issue was entered into the [illegible]see's corrective action program, it was treated as a noncited violation, consistent with

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Section VI. A of the NRC Enforcement Policy. The details of the staff's evaluation are contained in Inspection Report 50- 346/ 03- 17, dated September 29, 2003.
This LER also discussed a second issue which involved a condition that would potentially render the moisture trap on the gas analyzer sample line inoperable. The licensee provided a January 23, 2004, supplement to this LER describing the modifications made to prevent the moisture trap from becoming inoperable. The staff determined that the potential containment bypass pathway caused by the improper trap condensate drain path was a licensee- identified minor violation of 10 CFR 50, Appendix B, Criterion III, " Design Control." This issue was determined to be of minor significance because there was no evidence that the trap had ever functioned, and since pressure of the air supplied to the trap was insufficient to operate the trap, it was highly unlikely that the trap would have ever functioned. This LER was closed in Inspection Report 50- 346/ 04- 02, dated March 22, 2004.
Although the above discussion on the individual LERs is meant to demonstrate how the staff implements the agency's inspection process and Enforcement Policy in regard to LER findings, the staff's first priority is to assure that the issues involved will not adversely impact future plant safety. The staff then reviews the licensee's analysis for accuracy and completeness, and conducts its own risk assessment of the condition reported by the licensee. Once the safety implications are well understood, the staff imposes the appropriate enforcement actions in accordance with the Enforcement Policy.
D. Staff Response to Petitioners' Immediate Action Requests.
The NRC staff provided its findings on the Petitioners' requests for " immediate action" in a letter dated November 26, 2003. The staff considered the Petitioners' requests to suspend the Davis- Besse license and prohibit plant restart until certain conditions have been met to be equivalent to " immediate action" requests because the Davis- Besse licensee might complete all necessary restart activities, and the NRC staff might complete all necessary oversight activities,

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[illegible] the staff could finalize the Director's Decision on this Petition. Requests 3, 4, and 5 [illegible]ned in the Petitioners' August 25 letter are considered to be the immediate action requests, and the staff's evaluation of each of these requests contained in its November 26 letter is repeated verbatim below.
In Request 3, the Petitioners requested the NRC, " suspend the license and prohibit restart of the Davis Besse reactor unless and until First Energy Nuclear Operating Company has addressed all 1000 design basis deficiencies identified in 1997." The NRC staff agrees that design basis issues need to be addressed before plant restart. The NRC's oversight activities of the licensee's ongoing programs related to the design adequacy of the Davis- Besse plant are focused on plant safety. The licensee has initiated, and is still implementing, extensive corrective actions to address hardware, programmatic, and human performance issues to assure compliance with its license and NRC regulations. Compliance includes evaluating, testing, or inspecting safety- related systems to ensure that they are able to perform their design basis functions as defined in the plant's technical specifications (TS) and updated final safety analysis report. The staff's oversight activities include independent NRC inspections and NRC reviews of the licensee's evaluations to ensure conformance of safety systems and programs to the design and licensing bases. The adequacy of safety- significant structures, systems, and components is being tracked under NRC Restart Checklist Item 5. b, □ Systems Readiness for Restart" and must be adequately addressed before the NRC will allow the plant to restart.
The Petitioners' Request 3 is based on information contained in the NRC's February 26, 2003, inspection report on Davis- Besse design- related activities, which reported that approximately 200 of the more than 1000 design basis deficiencies identified in response to the NRC's 50.54(f) design basis letter had not been corrected. The licensee had agreed, prior to the Petitioners' August 25, 2003, letter, to place all remaining unresolved design basis deficiencies identified in response to the NRC's 50.54(f) design basis letter in its corrective action program. Information on how the remaining unresolved design basis deficiencies will be dispositioned can be found in the licensee's October 20, 2003, letter responding to this Petition, and in the licensee's letter dated N[illegible]ber 20, 2003, providing supplemental information related to the NRC's 5[illegible] f) design basis letter. In these letters, FirstEnergy stated that, while it had b[illegible]low to implement corrective actions for those issues identified in response to the NRC's 50.54(f) design basis letter, FirstEnergy has determined that these

issues either were corrected or have been documented in condition reports and entered into the Davis- Besse corrective action program. Each condition report generated by FirstEnergy was evaluated for potential impact on the operability of systems, structures, or components (SSCs). Those conditions classified as r[illegible]rt action items require evaluation for needed corrective actions prior to [illegible]t. Conditions that are not classified as restart action items will remain in [illegible]icensee's corrective action program and will be prioritized for resolution, which may occur after plant restart. The licensee stated in these letters that the

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number of open items has been reduced to approximately 100, with only a small number designated as restart items.

Appendix B of 10 CFR Part 50 requires operators of nuclear power plants to maintain an effective corrective action program. The process described above by the licensee to evaluate and disposition the remaining design basis deficiencies conforms with this regulatory requirement. The NRC's oversight of the licensee's activities includes specific inspections of the corrective action program to assure that this process is being followed correctly. The NRC will not allow the plant to restart until the licensee has demonstrated the capability to adequately manage the resolution of unresolved design basis deficiencies.

Therefore, the staff considers these activities, initiated prior to receiving the Petition, to completely satisfy the Petitioners' immediate action request to prohibit plant restart until the licensee has addressed all 1000 design basis deficiencies. The staff also concludes that the Petitioners immediate action request to suspend the plant license until the licensee has addressed all 1000 design basis deficiencies is in effect being granted by the actions already taken by the staff. These actions include our confirmatory action letter of March 13, 2002 (which confirmed the licensee's agreement that NRC approval is required for restart of the Davis- Besse plant), the enhanced NRC oversight as described in NRC Inspection Manual Chapter 0350, and compliance with the regulatory requirements imposed on all U. S. nuclear power plants. If the licensee had not agreed to obtain NRC approval before restarting the Davis- Besse plant, the NRC would have taken appropriate regulatory actions to assure restart would not occur unless NRC approval was received.

In Request 4, the Petitioners requested the NRC, " suspend the license and prohibit restart of the Davis Besse reactor unless and until First Energy has u[illegible]ated its Probabilistic Risk Assessment to reflect the flaws in it[s] design and [illegible]sing basis." The Petitioners provided clarifying information related to this [illegible]t during the September 17, 2003, meeting. The Petitioners are requesting th[illegible] the Davis- Besse PRA be revised to include the known design flaws, which will be corrected before the plant is allowed to restart, to account for unknown design flaws that may currently exist or may exist in the future.

The NRC's policy statement on PRA encourages greater use of this analysis technique to improve safety decisionmaking and improve regulatory efficiency in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense- in- depth philosophy. However, for the specific purpose of assuring that all restart issues have been satisfactorily addressed, the staff does not intend to rely on the Davis- Besse PRA to determine if there is reasonable assurance of adequate protection of the public health and safety.

On a more general level, while the staff recognizes that a PRA is a useful analysis tool, there are currently no regulatory requirements for licensees to develop a plant PRA, nor are there requirements to maintain or update a plant PRA. As explained in Management Directive 8.11, requests for changes to existing NRC regulations should be submitted as a petition for rulemaking and

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are not considered valid requests under 10 CFR 2.206. Therefore, the staff will not consider taking any action under Section 2.206 in regard to the Petitioners' request that the NRC suspend the license and prohibit restart of the Davis- Besse reactor until the plant PRA is updated to reflect design flaws, and this request is therefore denied.

In Request 5, the Petitioners requested the NRC, " suspend the license and prohibit restart of the Davis Besse reactor with any systems in a ' degraded but operable' condition." It is the staff's judgement that the processes and programs in place for the Davis- Besse restart effort (described above in the staff's response to Request 3) will provide reasonable assurance that all safety- related systems will be capable of performing their intended safety function and will be in compliance with the plant license and TS. The NRC has issued generic g[illegible]ce (Generic Letter 91- 18, " Information to Licensees Regarding Two NRC [illegible]tion Manual Sections on Resolution of Degraded and Nonconforming C[illegible]ions and on Operability") which provides a process for licensees to develop a basis to continue operation or to place the plant in a safe condition and take prompt corrective action. This process assures that issues affecting the operability of SSCs that are subject both to 10 CFR Part 50, Appendix B, and 10 CFR Part 50.59 are corrected promptly, and that SSCs in degraded but operable conditions are returned to full functional capability in a timely fashion.

Each licensee is authorized to operate its plant in accordance with the NRC's regulations and the plant license. If an SSC is degraded or nonconforming but operable, the licensee must establish an acceptable basis to continue to operate. The licensee must, however, promptly identify and correct the condition adverse to safety or quality in accordance with 10 CFR Part 50, Appendix B, Criterion XVI. The basis for this authority to continue to operate is that the plant license and TS contain the specific characteristics and conditions of operation necessary to ensure that an abnormal situation or event does not pose an undo risk to public health and safety. Thus, if the TS are satisfied and required equipment is operable, and the licensee is correcting any degraded conditions in a timely manner, allowing a plant to restart or to continue operation does not pose an undue risk to public health and safety. This generic guidance applies to all U. S. nuclear power plants, including Davis- Besse, and the NRC will continue to monitor licensees to assure that this guidance is followed appropriately.

Therefore, the Petitioners Request 5 is not needed to assure plant safety nor is it consistent with established staff regulatory requirements, and is therefore denied.

III Conclusion

The NRC staff has carefully considered the Petitioners' arguments regarding why the NRC should take enforcement actions against FirstEnergy. In summary, the Petitioners stated that FirstEnergy has failed to complete commitments related to the NRC's 50.54(f) design basis letter (issued on October 9, 1996) and referred to numerous design basis violations dating back

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- 27 -

to plant licensing (corresponding to Requests 1 and 2 in the Petitioners' August 25 letter). As noted earlier, the Petitioners' requests for immediate actions (corresponding to Requests 3, 4, and 5 in the Petitioners' August 25 letter) were evaluated in the staff's November 26, 2003, letter and this evaluation is repeated in Section II. D of this Director's Decision for completeness.

With respect to the first request for enforcement action, the NRC staff finds that the Petitioners' request for enforcement based solely on failure of the licensee to complete commitments represents a misinterpretation of the agency's enforcement policies regarding commitments. As stated earlier, reasonable assurance of adequate protection of public health and safety is, as a general matter, defined by the Commission's health and safety regulations themselves. In most cases, the agency cannot take formal enforcement actions solely on the basis of whether licensees fulfill commitments, as failure to meet a commitment in itself does not constitute a violation of a legally binding requirement. However, when failures to meet commitments result in violations of the Commission's health and safety regulations, the staff will take the appropriate enforcement actions. Although the staff has not taken any formal enforcement actions against FirstEnergy in direct response to any failures to meet commitments, the staff has taken formal enforcement actions, as discussed in the previous section, against the licensee for noncompliance with NRC requirements.

Therefore, I deny the Petitioners' request for enforcement actions based solely on any failures on the part of the licensee to not fully comply with commitments made in response to the 50.54(f) letter. Formal enforcement actions are taken when there is a noncompliance with NRC requirements, and the severity of those actions is based in part on the degree of risk posed by that noncompliance.

With respect to the second request for enforcement action, the NRC staff finds that the Petitioners' request for enforcement based on numerous design basis violations (i. e., the LERs submitted by the licensee) is in effect being granted by the actions already taken by the staff, as shown by the earlier discussion of our processes for reviewing and evaluating LERs.

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It is also important to note that the highest level of staff oversight of licensee activities for plants with performance problems or operational events is governed by IMC 0350, and that the agency has been overseeing the licensee's activities using this process since May 3, 2002. The decision by the staff to place the Davis- Besse licensee in the highest level of staff oversight was based on the identified performance deficiencies, and was also intended to assure close coordination between NRC and licensee personnel on the corrective actions needed to assure safe plant restart. Any additional enforcement actions, as requested by the Petitioners, would not increase this level of staff oversight, which is directed at assuring that the plant is capable of safe operation in accordance with the Commission's rules and regulations.

As provided in 10 CFR 2.206(c), a copy of this director's decision will be filed with the Secretary of the Commission for the Commission to review. As provided for by this regulation, the decision will constitute the final action of the Commission 25 days after the date of the decision unless the Commission, on its own motion, institutes a review of the decision within that time.

Dated at Rockville, Maryland, this 22nd day of April 2004.

FOR THE NUCLEAR REGULATORY COMMISSION

/ RA/
J. E. Dyer, Director
Office of Nuclear Reactor Regulation

□
7[illegible]01- P
U[illegible]NUCLEAR REGULATORY COMMISSION
DO[illegible] NO. 50- 346
LICENSE NO. NPF- 03
FIRSTENERGY NUCLEAR OPERATING COMPANY
NOTICE OF ISSUANCE OF DIRECTOR'S DECISION UNDER 10 CFR 2.206

Notice is hereby given that the Director, Office of Nuclear Reactor Regulation, has issued a Director's Decision with regard to a letter dated August 25, 2003, filed by Greenpeace pursuant to Section 2.206 of Title 10 of the Code of Federal Regulations (10 CFR) on behalf of the Nuclear Information & Resource Service and the Union of Concerned Scientists (collectively, the Petitioners). The Petitioners requested that the Nuclear Regulatory Commission (NRC) take enforcement actions against FirstEnergy Nuclear Operating Company (FirstEnergy), the licensee for Davis- Besse Nuclear Power Station in Oak Harbor, Ohio, and also requested that NRC suspend the Davis- Besse license and prohibit plant restart until certain conditions have been met. As basis for the request to have the NRC take enforcement actions against the licensee, the Petitioners stated that FirstEnergy has failed to complete commitments related to the NRC's 50.54(f) design basis letter (issued on October 9, 1996), and referred to numerous design basis violations dating back to plant licensing (corresponding to Requests 1 and 2 in the Petitioners' August 25 letter). The Petitioners also requested that the NRC suspend the Davis- Besse license and prohibit plant restart until all design basis deficiencies identified in response to the NRC's 50.54(f) design basis letter are adequately addressed, the plant probabilistic risk assessment (PRA) is updated to reflect design flaws, and no systems are in a " degraded but operable" condition (corresponding to Requests 3, 4, and 5 in the Petitioners' August 25 letter).

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- 2 -

In a letter dated October 7, 2003, the NRC informed the Petitioners that the issues in the Petition were accepted for review under 10 CFR 2.206 and had been referred to the Office of Nuclear Reactor Regulation for appropriate action. A copy of the acknowledgment letter is publicly available in the NRC's Agencywide Documents Access and Management System (ADAMS) under Accession No. ML032690314. A copy of the Petition is publicly available in ADAMS under the Accession No. ML032400435.

T[illegible]titioners' representatives met with NRC staff on September 17, 2003, to provide [illegible]onal details in support of this request. This meeting was transcribed and the transcript is p[illegible]cly available on the NRC Web site as a supplement to the Petition (http:// www. nrc. gov/ reactors/ operating/ ops- experience/ vessel- head- degradation/ controlled- correspondence. html).

The licensee responded to the Petition on October 20, 2003 (ML033421458). This response was considered by the staff in its evaluation of the Petition.

In a letter dated November 26, 2003 (ML033010172), the NRC provided to the Petitioners its evaluation of their " immediate action" requests. The staff considered the Petitioners' requests to suspend the Davis- Besse license and prohibit plant restart until certain conditions have been met to be equivalent to " immediate action" requests because the Davis- Besse licensee might complete all necessary restart activities, and the NRC staff might complete all necessary oversight activities, before the staff could finalize the Director's Decision on this Petition. Requests 3, 4, and 5 in the Petitioners' August 25 letter were considered immediate action requests, and the staff's November 26 evaluation is repeated in Section II. D of the Director's Decision for completeness.

The NRC sent a copy of the proposed Director's Decision to the Petitioners and to the licensee for comment on February 5, 2004 (ML040280003). Neither the Petitioners nor the licensee provided comments on the proposed Director's Decision.

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- 3 -

The Director of the Office of Nuclear Reactor Regulation has determined that the Petitioners' first request for enforcement based solely on failure of the licensee to complete commitments represents a misinterpretation of the agency's enforcement policies regarding commitments and therefore is denied. The Director of the Office of Nuclear Reactor Regulation has also determined that the Petitioners' second request for enforcement based on numerous design basis violations is in effect being granted by the actions already taken by the staff. The reasons for these decisions are explained in Director's Decision DD- 04- 01, the complete text of which is available in ADAMS, or is available for inspection at the Commission's Public Document Room (PDR), located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records are accessible from the ADAMS Public Electronic Reading Room on the NRC Web site, http:// www. nrc. gov/ reading- rm/ adams. html.

Pe[illegible]s who do not have access to ADAMS or who encounter problems in accessing the d[illegible]ts located in ADAMS should contact the NRC PDR reference staff at 1- 800- 397- 4209 o[illegible] 415- 4737, or by e- mail to pdr@ nrc. gov.

A [illegible]y of the Director's Decision will be filed with the Secretary of the Commission for the Commission's review in accordance with 10 CFR 2.206 of the Commission's regulations. As provided for by this regulation, the Director's Decision will constitute the final action of the Commission 25 days after the date of the decision, unless the Commission, on its own motion, institutes a review of the Director's Decision in that time.

Dated at Rockville, Maryland, this 22nd day of April 2004.

F[illegible] NUCLEAR REGULATORY COMMISSION
/[illegible]
J. [illegible] Dyer, Director
Office of Nuclear Reactor Regulation

EA-04-005 - Surry 1 & 2 (Virginia Electric and Power Co.)

ATTACHMENTS 2 AND 3 CONTAIN PROPRIETARY INFORMATION
PROPRIETARY INFORMATION REMOVED

September 15, 2004

EA-04-005

Virginia Electric and Power Company
ATTN: Mr. David A. Christian
Senior Vice President and
Chief Nuclear Officer
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR WHITE FINDINGS AND NOTICE OF VIOLATION (NRC INSPECTION REPORT NO. 05000280/2004008 AND 05000281/2004008 SURRY POWER STATION)

[illegible]r. Christian:

The purpose of this letter is to provide you with the Nuclear Regulatory Commission's (NRC's) final significance determination for a finding regarding Surry fire response procedures that were not effective in ensuring a safe shutdown of Unit 1 during a postulated severe fire in Emergency Switchgear and Relay Room (ESGR) Number (No.) 1. Specifically, these procedures may not preclude an extended loss of reactor coolant pump (RCP) seal injection flow, resulting in an RCP seal loss of coolant accident. As a result, in the event of such a severe fire, there would not be reasonable assurance the facility would be able to maintain pressurizer level within the indicating range, as required by 10 CFR 50, Appendix R. The finding was documented in NRC Inspection Report 05000280/2003008 and 05000281/2003008, issued on February 2, 2004 (ML040490131), and was assessed using the significance determination process as a preliminary White issue (i.e., an issue of low to moderate safety significance, which may require additional NRC inspection). The cover letter to the inspection report informed Virginia Electric and Power Company (VEPCO) of the NRC's preliminary conclusion, provided VEPCO an opportunity to request a regulatory conference on this matter, and forwarded the details of the NRC's preliminary estimate of the change in core damage frequency (CDF) for this finding.

At VEPCO's request, an open regulatory conference was conducted on April 1, 2004, to discuss VEPCO's position on this issue. A letter summarizing this meeting dated April 21, 2004 (ML041180603), included the list of attendees at the regulatory conference and copies of the materials presented by VEPCO and the NRC at the conference.

During the conference, VEPCO presented its fire strategy and an overview of the ESGR and local fire protection features. VEPCO also presented an assessment of portions of the NRC's Phase 3 analysis, the results of VEPCO's analysis of the increase in CDF due to the performance deficiency, a discussion on the applicability of the Unit 1 ESGR finding to the Unit 2 ESGR, and a summary of corrective actions that have been initiated. VEPCO's estimate of the increase in CDF was approximately one order of magnitude lower than the NRC's preliminary estimate. This lower estimate was due, in part, to the identification of plant specific features, which VEPCO contended were not appropriately reflected in the NRC's Phase 3 S[illegible]alysis. These included the generic probability of non-suppression; generic severity factors and non-suppression p[illegible]ilities for ESGR welding fires; and generic reactor coolant pump seal leakage probabilities. At the conclusion of the co[illegible]ference, the NRC requested that VEPCO provide additional information related to the design and testing of Surry's RCP floating ring seals, the frequency of welding and the probability of a welding fire in the ESGR, and the fire suppression capability of the Halon system for a fire in the ESGR. VEPCO forwarded this information to the NRC by letter dated May 7, 2004. Based on the above information, VEPCO concluded that the finding was of very low safety significance for both units. VEPCO did not contest the violation.

After considering the information developed during the inspection, the information VEPCO provided at the conference, and the additional information submitted by VEPCO subsequent to the conference, the NRC has concluded that the final inspection finding is appropriately characterized as White for Unit 1 in the mitigating systems cornerstone. As discussed in the final phase 3 analysis (Enclosure 2), we concluded that the information VEPCO provided on the ESGR welding activities and the Halon system did not warrant a change in the generic values used in our initial analysis. Based on the information VEPCO provided for the RCP floating ring seal, the probability of initial seal failure following the loss of forced cooling was reduced in the appropriate phase 3 analysis event tree. However, this reduction was offset by an increase in the severity factor for electrical cabinet fires. This severity factor was revised, using plant-specific information and the most current methodology (see Reference 17 of Enclosure 2, which was revised in July 2004), in the final phase 3 analysis. As a result, the final CDF was determined to be essentially unchanged.

Additionally, we evaluated the information provided by VEPCO on the applicability of this finding to the Unit 2 ESGR. Based on the critical cables in Unit 2 being located in the fifth cable tray above the 2J electrical cabinet, the additional elapsed time for cable damage was determined and the probability of non-suppression was recalculated to account for fire brigade response. The dominant accident sequence was analyzed using these adjusted values and a CDF of greater than 1E-6/year was determined (Enclosure 3). As a result, the NRC has concluded that the final inspection finding is applicable to Unit 2 and is appropriately characterized as White in the mitigating systems cornerstone.

You have 10 business days from the date of this letter to appeal the staff's determination of significance for the identified White finding. Such appeals will be considered to have merit only if they meet the criteria given in NRC Inspection Manual Chapter 0609, Attachment 2.

The NRC also determined that a violation occurred involving the requirements of 10 CFR 50.48 and 10 CFR 50, Appendix R, Section III.L. Specifically, the alternative shutdown capability and response procedures specified for a fire in ESGR No. 1 or in ESGR No. 2 may not preclude an extended loss of reactor coolant pump seal injection flow and may initiate a reactor coolant pump seal loss of coolant accident which could result in pressurizer level failing to be maintained within the operating range. Accordingly, a Notice of Violation is included as Enclosure 1 to this letter. In accordance with the NRC Enforcement Policy, NUREG-1600, the Notice of Violation is considered escalated enforcement action because it is associated with a White finding.

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation and prevent recurrence, and the date when full compliance was achieved is adequately addressed on the docket in the information provided by VEPCO at the conference. Therefore, you are not required to respond to this letter unless the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to provide additional information, you should follow the instructions specified in the enclosed Notice.

Based on Inspection Manual Chapter 0305 guidance, the performance consideration start date for this finding is the first quarter of 2004 (i.e., when the preliminary significance determination was made known, via Inspection Reports 05000280/2003008 and 05000281/2003008). For the Mitigating Systems Cornerstone during the first quarter of 2004, both Units 1 and 2 are in the Degraded Cornerstone Column of the NRC Action Matrix because of this White Finding and a White Performance Indicator, Safety System Unavailability - Emergency AC Power. In accordance with the NRC Action Matrix, we will conduct a Supplemental Inspection using Inspection Procedure 95002. We will notify you of the date of this inspection by separate correspondence.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and your response (should you choose to provide one) will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), which is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. However, the NRC is continuing to review the appropriate classification of the Summary of Phase 2 SDP Risk Analysis and Phase 3 SDP Analysis (Enclosures 2 and 3) within our records management program, considering changes in our practices following the events of September 11, 2001. Using our interim guidance, the attached analyses have been marked as Proprietary Information or Sensitive Information in accordance with Section 2.390 ([illegible] Title 10 of the Code of Federal Regulations. Please control the document accordingly (i.e., treat the document as if you had determined that it contained trade secrets and commercial or financial information that you considered privileged or confidential). We will inform you if the classification of these documents changes as a result of our ongoing assessment. To the extent possible, any response should not include any personal privacy, proprietary, classified, or safeguards information so that it can be made available to the Public without redaction. The NRC also includes significant enforcement actions on its Web site at www.nrc.gov; select **What We Do, Enforcement, then Significant Enforcement Actions**.

S[illegible]d you have any questions regarding this letter, please contact Charles Ogle, Chief, Engineering Branch 1, 404-562-4[illegible]

Sincerely,

/RA/

William D. Travers
Regional Administrator

Docket Nos.: 50-280, 50-281
License Nos.: DPR-32, DPR-37

Enclosures:
1. Notice of Violation
2. Surry Unit 1 Final Phase 3 SDP Analysis
3. Surry Unit 2 Final Phase 3 SDP Analysis

cc w/encls:
Chris L. Funderburk, Director
Nuclear Licensing and
Operations Support
Virginia Electric & Power Company
Electronic Mail Distribution

Richard H. Blount, II
S[illegible]ice President
[illegible]Power Station
V[illegible]ia Electric & Power Company
Electronic Mail Distribution

Virginia State Corporation Commission
Division of Energy Regulation
P. O. Box 1197
Richmond, VA 23209

Lillian M. Cuoco, Esq.
Senior Counsel
Dominion Resources Services, Inc.
Electronic Mail Distribution

Attorney General
Supreme Court Building
900 East Main Street
Richmond, VA 23219

NOTICE OF VIOLATION

Virginia Electric and Power Company
S[illegible] Power Station
U[illegible] & 2

Docket Nos.: 50-280, 50-281
License Nos.: DPR-32, DPR-37
EA-04-005

During an NRC inspection completed on January 7, 2004, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedures for NRC Enforcement Actions," (Enforcement Policy), the violation is listed below:

> 10 CFR 50.48 states, in part, "Each operating nuclear power plant must have a fire protection program that satisfies Criterion 3 of Appendix A to this part." Surry Unit 1 Operating License DPR-32, and Surry Unit 2 Operating License DPR-37 Condition 3.I, specifies, in part, that the licensee implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report (UFSAR) and as approved in the SER dated September 19, 1979, and subsequent supplements.
>
> UFSAR Section 9.10.1 specifies that the facility meets 10 CFR 50, Appendix R, Sections III.G and III.L. Section III.G.3 states that alternative shutdown capability should be provided where the protection of systems whose function is required for hot shutdown, does not satisfy the requirements of III.G.2. Section III.L of Appendix R specifies the requirements to be met by alternative shutdown methods. Section III.L.2.b states, in part, that "The reactor coolant makeup function shall be capable of maintaining the reactor coolant level . . . within the level indication in the pressurizer in PWRs." Section III.L.3 specifies that "procedures shall be in effect to implement this capability."
>
> Contrary to the above, on or about February 13, 2003, the alternative shutdown capability and response procedures specified for a fire in Emergency Switchgear Room Number 1 or 2 were not effective and did not meet this requirement. Specifically, the licensee's procedures would not preclude an extended loss of reactor coolant pump seal injection flow or a reactor coolant pump seal loss of coolant accident under certain fire scenarios, which could result in pressurizer level failing to be maintained within the indicating range.

This violation is associated with White Significance Determination Process findings for Units 1 and 2.

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation and prevent recurrence and the date when full compliance was achieved is already adequately addressed on the docket in the information provided by VEPCO at the conference. However, you are required to submit a written statement or explanation pursuant to 10 CFR 2.201 if the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to respond, clearly mark your response as a "Reply to a [illegible] of Violation - EA-04-005," and send it to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555 with a copy to the Regional Administrator, Region RII, within 30 days of the date of the letter transmitting this Notice of Violation (Notice).

If you contest this enforcement action, you should also provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

If you choose to respond, your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

In accordance with 10 CFR 19.11, you may be required to post this Notice within two working days.

Dated this 15th day of September 2004

**ATTACHMENTS 2 AND 3 CONTAIN PROPRIETARY INFORMATION
PROPRIETARY INFORMATION REMOVED**

EA-04-115 - Oconee 1, 2, 3 (Duke Energy Corporation)

September 24, 2004

EA-04-115

Duke Energy Corporation
ATTN: Mr. Ronald A. Jones
Vice President
Oconee Site
7800 Rochester Highway
Seneca, SC 29672

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR A WHITE FINDING AND NOTICE OF VIOLATION (NRC INSPECTION REPORT 05000269,270,287/2004013, OCONEE NUCLEAR STATION)

Dear Mr. Jones:

The purpose of this letter is to provide you with the Nuclear Regulatory Commission's (NRC's) final significance [illegible]ination for a finding at Duke Energy Corporation's (DEC's) Oconee Nuclear Station involving fire response procedures [illegible]ere not consistent with the licensing basis in regards to the criteria for manning of the Standby Shutdown Facility (SSF). In some scenarios, this could result in a delay of transfer of control to the SSF that could challenge the capability of the installed SSF makeup pump. This condition could result in the failure to maintain pressurizer level within the indicating range as required by 10 CFR 50, Appendix R.

The finding was documented in NRC Inspection Report 05000269,270,287/2004012, dated July 20, 2004, and was assessed under the significance determination process as a preliminary greater than Green issue for all three Oconee units (i.e., an issue of at least low to moderate safety significance, which may require additional NRC inspection). The cover letter to the inspection report informed DEC of the NRC's preliminary conclusion, provided DEC an opportunity to request a regulatory conference on this matter, and forwarded the details of the NRC's preliminary estimate of the change in core damage frequency (CDF) for this finding.

At your request, an open regulatory conference was conducted with members of your staff on September 13, 2004, to discuss DEC's position on this issue. The enclosures to this letter include the list of attendees at the regulatory conference, and copies of the material presented by your staff and the NRC at the regulatory conference. During the conference, DEC provided the results of its review of the safety significance of the finding and highlighted the modeling and assumption differences between its analysis of the change in CDF and that of the NRC's preliminary estimate. In addition, DEC agreed with the NRC's characterization of the finding as a violation of regulatory requirements, and stated that DEC strategy and procedures have been revised to man the SSF upon identification of a confirmed fire in the specified fire areas of concern.

A particular focus of DEC's presentation was a substantive difference in the assumed failure probability of Oconee's primary safety relief valves (PSVs). DEC stated at the conference that the NRC's PSV failure probability used in its preliminary estimate was very conservative for Oconee's scenario. To determine a PSV failure probability that was specific to Oconee, DEC convened an Expert Elicitation Panel, commissioned with the Electric Power Research Institute. DEC explained the Expert Elicitation Panel Process in detail, and stated that its goal was to obtain a PSV "failure to reseat" probability based on te[illegible]a, plant experience, and expert judgment. As described by DEC, the failure probability also considered the range of f[illegible]unique to Oconee's PSVs that may affect valve performance, such as lift type, inlet piping configuration, and fluid conditions.

Based on the efforts of the Expert Elicitation Panel, DEC concluded that the important factors in determining PSV failure rate were inlet piping configuration, fluid conditions, and the number of cycles. Regarding the factor of inlet piping configuration, DEC concluded that because the Oconee configuration is a short inlet pipe with no loop seal, its physical configuration is the m[illegible]eliable relative to other piping configurations. Secondly, DEC concluded that PSV reliability is highest when relieving s[illegible] Because the relieving fluid conditions at Oconee, for these scenarios, are expected to be steam, DEC stated that this fa[illegible] would result in a higher PSV reliability relative to other fluid conditions such as water and/or subcooled liquid. Finally, DEC concluded that the PSV failure probability for cycles two through five would be substantially less than the failure probability on the initial cycle, for reasons as discussed at the conference.

Based on the above, DEC concluded that the failure probability of Oconee's PSVs was approximately one order of magnitude less than that assumed by the NRC in its preliminary estimate. As a result, DEC concluded that the finding should be characterized as Green for all three Oconee units.

After considering the information developed during the inspection and the information DEC provided at the conference, the NRC has concluded that the final inspection finding is appropriately characterized as White for all three Oconee units, in the mitigating systems cornerstone. In summary, the NRC concluded that the factors discussed at the conference are not well known with respect to their influence on PSV failure probability. The analytical techniques and risk analysis of DEC's proposal are novel and unverified with respect to the PSV failure probability following the initial lift. Additionally, DEC did not provide specific testing data to support the conclusion presented at the conference. Absent any additional specific operational, empirical, or testing data, the NRC concluded that the information provided by DEC at the conference was insufficient to warrant a change in the NRC's preliminary estimate.

You have 10 business days from the date of this letter to appeal the staff's determination of significance for the identified White finding. Such appeals will be considered to have merit only if they meet the criteria given in NRC Inspection Manual Chapter 0609, Attachment 2.

The NRC also determined that a violation occurred involving the requirements of 10 CFR 50, Appendix R, Section III.G.3, in that procedures for a fire requiring SSF manning and activation would not assure that the reactor coolant makeup function would be capable of maintaining reactor coolant level within the indicated range of the pressurizer. Accordingly, a Notice of Violation is included as an enclosure to this letter. In accordance with the NRC Enforcement Policy, NUREG-1600, the Notice [illegible]ation is considered escalated enforcement action because it is associated with a White finding.

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken to correct the violation and prevent recurrence, and the date when full compliance was achieved is adequately addressed on the docket in the information provided by DEC at the conference (Enclosure 3). Therefore, you are not required to respond to this letter unless the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to provide additional information, you should follow the instructions specified in the enclosed Notice.

Based on NRC Inspection Manual Chapter 0305 guidance, the performance consideration start date for this issue is the third quarter of 2004 (i.e., when the preliminary significance determination was made known via Inspection Report 05000269,270,287/2004012, dated July 20, 2004). Consequently, as a result of this White finding, plant performance has been determined to be in the Degraded Cornerstone Column for Units 1, 2, and 3, because of a previously identified White finding in the Mitigating Systems Cornerstone (EA-03-145). We will use the NRC Action Matrix to determine the most appropriate NRC response for this finding and will notify you of that determination by separate correspondence.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosures, and your response (should you choose to provide one), will be available electronically for public inspection in the NRC Public Document Room (PDR) or from the NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

For administrative purposes, this letter is issued as a separate NRC Inspection Report, No. 05000269,270,287/2004013, and the above violation is identified as VIO 05000269, 270,287/2004013-01: Failure to Meet Licensing Basis for Staffing the SSF in the Event of a Confirmed Plant Fire. Accordingly, the associated apparent violation, AV 05000269,270, 287/2004012-01, is closed.

Sh[illegible] you have any questions regarding this letter, please contact Charles Ogle, Chief, Division of Reactor Safety, E[illegible]ring Branch 1, at 404-562-4605.

Sincerely,

/RA/

[redacted]

William D. Travers
Regional Administrator

Docket Nos.: 50-269, 50-270, 50-287
License Nos.: DPR-38, DPR-47, DPR-55

Enclosures:
1. Notice of Violation
2. List of Attendees
3. Material presented by DEC
4. Material presented by NRC

cc w/ encls:
B. G. Davenport
Compliance Manager (ONS)
Duke Energy Corporation
Electronic Mail Distribution

Lisa Vaughn
Legal Department (PB05E)
Duke Energy Corporation
422 South Church Street
P. O. Box 1244
Charlotte, NC 28201-1244

[redacted]Cottingham
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Beverly Hall, Acting Director
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Henry J. Porter, Director
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County Supervisor of
Oconee County
415 S. Pine Street
Walhalla, SC 29691-2145

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R. L. Gill, Jr., Manager
Nuclear Regulatory Licensing
Duke Energy Corporation
526 S. Church Street
Charlotte, NC 28201-0006

Peggy Force
Assistant Attorney General
N. C. Department of Justice
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NOTICE OF VIOLATION

Duke Energy Corporation
Oconee Nuclear Station
Units 1, 2 and 3

Docket No. 50-269, 50-270, 50-287
License No. DPR-38, DPR-47, DPR-55
EA-04-115

During an NRC inspection completed on February 18, 2004, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," (Enforcement Policy), the violation is listed below:

> Oconee Unit 1 Operating License DPR-38, Oconee Unit 2 Operating License DPR-47, and Oconee Unit 3 Operating License DPR-55 Condition D provide, in part, that the licensee implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report (UFSAR) and as approved in the Safety Evaluation Report (SER) dated April 28, 1983 and subsequent supplements.
>
> The licensee's UFSAR commits to 10 CFR 50, Appendix R, Sections III.G and III.L. Section III.G.3 states that alternative shutdown capability should be provided where the protection of systems whose function is required for hot shutdown, does not satisfy the requirements of III.G.2. Section III.L of Appendix R provides requirements to be met by alternative shutdown methods. Section III.L.2.b states, in part, that "The reactor coolant makeup function shall be capable of maintaining the reactor coolant level. . . within the level indication in the pressurizer in PWRs." Section III.L.3 specifies that "procedures shall be in effect to implement this capability."
>
> Contrary to the above, on February 8, 2004, the licensee's procedures for a fire requiring SSF manning and activation would not assure that the reactor coolant makeup function would be capable of maintaining reactor coolant level within the indicated range of the pressurizer. Specifically, delaying the manning of the SSF until after the occurrence of a loss of function of the high pressure injection and component cooling or feedwater rather than manning the SSF immediately upon confirmation of a fire in the areas of concern may not preclude an extended loss of reactor coolant system inventory.

This violation is associated with a White Significance Determination Process finding for Units 1, 2 and 3.

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken to correct the violation and prevent recurrence and the date when full compliance was achieved is already adequately addressed on the docket in the information provided by Duke Energy Corporation at the conference (Enclosure 3) and in NRC Inspection Report 05000269,270,287/2004012. However, you are required to submit a written statement or explanation pursuant to 10 CFR 2.201 if the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to respond, clearly mark your response as a "Reply to a Notice of Violation - EA-04-115," and send it to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555 with a copy to the Regional Administrator, Region RII, within 30 days of the date of the letter transmitting this Notice of Violation (Notice).

If you contest this enforcement action, you should also provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

If you choose to respond, your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

In accordance with 10 CFR 19.11, you may be required to post this Notice within two working days.

Dated this 24th day of September 2004

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EA-04-109 - D.C. Cook (American Electric Power Company)

September 29, 2004

EA-04-109

Mr. M. Nazar
Senior Vice President and
Chief Nuclear Officer
Nuclear Generation Group
American Electric Power Company
500 Circle Drive
Buchanan, MI 49107

SUBJECT: NOTICE OF VIOLATION
[INSPECTION REPORT 0500315/2004007(DRS); 0500316/2004007(DRS)]

Dear Mr. Nazar:

[illegible]fers to information provided to the U.S. Nuclear Regulatory Commission (NRC) by the American Electric Power C[illegible]any (AEP) on March 24, 2004, concerning the permanent physical condition of a licensed Senior Reactor Operator (SRO) at the D.C. Cook Nuclear Plant. During a February 2004 review of medical information for licensed operators at the D.C. Cook Nuclear Plant, the new Medical Review Officer for Licensed Operators (MRO) determined that the NRC had not been informed of a cardiac condition experienced by an SRO during December 1996. The failure to provide the NRC with complete and accurate information concerning an SRO's permanent medical condition is an apparent violation of 10 CFR 50.9. A copy of the inspection report concerning this issue was provided to you on July 2, 2004.

In the letter transmitting the inspection report, we provided you the opportunity to address the apparent violation identified in the report by either attending a predecisional enforcement conference or providing a written response before we made our enforcement decision. You responded to the apparent violation in a letter dated August 2, 2004.

Based on the information developed during the inspection and the information you provided in your correspondence on March 24 and August 2, 2004, and during a telephone conversation on August 25, 2004, between Roger D. Lanksbury, Chief, Operator Licensing Branch, and Helen Etheridge of your staff, the NRC has determined that a violation of NRC requirements occurred. The violation is cited in the enclosed Notice of Violation (Notice) and the circumstances surrounding it are described in detail in the subject inspection report. In summary, the NRC issued an SRO license to the individual on February 1, 1994. On December 28, 1999, AEP submitted information to the NRC to renew the SRO license prior to its expiration on January 31, 2000. Included in the submission for renewal of the SRO license was a December 28, 1999, Form NRC - 396, "Certification of Medical Examination by Facility Licensee." Information on that Form NRC - 396 indicated that your prior MRO recommended only one condition be added to the SRO's license to require the SRO to wear corrective lenses when performing licensed duties. No other medical restriction was recommended by either AEP or the MRO in the December 28, 1999, renewal application. Based on the information submitted by AEP on December 28, 1999, the NRC renewed the SRO license on February 1, 2000, with the requirement that the SRO wear corrective lenses when performing licensed duties. The NRC placed no other medical restrictions on the SRO license based on the information submitted by AEP in the application for renewal.

T[illegible]O who provided the certification discussed above, retired during September 2001 and a new MRO was appointed. During February 2004, your new MRO reviewed medical records for licensed operators at the D.C. Cook Nuclear Plant.

Included in the new MRO's review were documents indicating that your prior MRO had been informed on January 15, 1997, that the SRO had experienced a myocardial infarction during December 1996. On February 23, 2004, the new MRO notified AEP that the SRO should no longer be allowed to continue to work as a solo operator and the NRC should be notified. That notification was provided to the NRC by AEP on March 24, 2004.

Licensed operators are entrusted with the safe operations of a nuclear reactor and must be capable of performing their assigned duties under normal, abnormal and emergency operating conditions of the plant. The physical condition and the general health of an operator is a significant concern of the NRC so that any sudden incapacitation of an operator due to an existing medical condition does not pose undo risk to the facility. Therefore, the NRC places restrictions for certain medical conditions on an operator's license to ensure that other licensed personnel are on duty and can immediately compensate for an operator who may be suddenly incapacitated because of an existing medical condition. By not informing the NRC of an operator's physical condition, such restrictions cannot be put in place and additional personnel may not be available to replace an operator who is suddenly incapacitated from an existing medical condition.

Furthermore, the information about the SRO's cardiac condition had been known to AEP's MRO since January 15, 1997, and the failure to provide accurate and complete information to the NRC regarding the pre-existing medical condition of a licensed SRO within 30 days, as required by 10 CFR 50.74(c), is a regulatory concern. Moreover, had the medical information been complete and accurate at the time the license renewal was sought by AEP on December 28, 1999, the NRC would have taken a different regulatory position by applying the appropriate restriction to the SRO license. Therefore, the information submitted to the NRC on December 28, 1999, was material to the licensing of an SRO on February 1, 2000, and the violation has been categorized in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600, at Severity Level III.

In accordance with the Enforcement Policy, a civil penalty in the base amount of $55,000 would be considered for a Severity Level III violation at the time the violation occurred. Because your facility has not been the subject of escalated enforcement actions evaluated in accordance with the civil penalty assessment process described in Section VI.C.2 of the Enforcement Policy within the last two years, the NRC considered whether credit was warranted for Corrective Action in accordance with the civil penalty assessment process in Section VI.C.2 of the Enforcement Policy. Credit was warranted for the Corrective Action factor. Corrective actions included: preventing activation of the SRO license until the medical status of the operator was resolved; discussing the requirements of ANSI 3.4-1983 with the current MRO and personnel in Operations Training and Regulatory Affairs; requiring all completed physical examination forms and recommendations from physicians be submitted to Regulatory Affairs for inclusion in license applications; and including an overview of the requirements for reporting changes in medical conditions in the operator requalification training program. Other corrective actions included: performing a 100% self-assessment review of licensed operator medical records; revising the procedure to require that all recent physical examination records be submitted to the NRC when requesting an initial or renewal reactor operator or SRO license; and planning by September 30, 2004, to revise the procedure for biennial self-assessment of medical records to discuss the requirements of ANSI 3.4-1983 with the designated MRO.

Therefore, to encourage prompt comprehensive correction of violations and in recognition of the absence of previous escalated enforcement action, I have been authorized, after consultation with the Director, Office of Enforcement, not to propose a civil penalty in this case. However, significant violations in the future could result in a civil penalty.

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation and prevent recurrence, and the date when full compliance was achieved, is already adequately addressed on the docket in a letter from AEP dated August 2, 2004. Therefore, you are not required to respond to this letter unless the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to provide additional information, you should follow the instructions specified in the enclosed Notice.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (should you choose to respond) will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the Public without redaction. The NRC also includes significant enforcement actions on its Web site at www.nrc.gov; select What We Do, Enforcement, then Significant Enforcement Actions.

Sincerely,

/RA/

James L. Caldwell
Regional Administrator

Docket Nos. 50-315; 50-316
License Nos. DPR-58; DPR-74

Enclosure: Notice of Violation

cc w/encl:
J. Jensen, Site Vice President
M. Finissi, Plant Manager
G. White, Michigan Public Service Commission
Michigan Department of Environmental Quality
Emergency Management Division
MI Department of State Police
D. Lochbaum, Union of Concerned Scientists

NOTICE OF VIOLATION

American Electric Power Company
D.C. Cook Nuclear Power Plant

Docket Nos. 50-315; 50-316
License Nos. DPR-58; DPR-74
EA-04-109

During an NRC inspection that was completed on June 4, 2004, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the violation is listed below:

> 10 CFR 50.9 requires that information provided to the Commission by an applicant for a license or by a licensee or information required by statute or by the Commission's regulations, orders, or license conditions to be maintained by the applicant or the licensee shall be complete and accurate in all material respects.
>
> 10 CFR 55.23 requires, in part, that to certify the medical fitness of the applicant, an authorized representative of the facility licensee shall complete and sign Form NRC - 396, "Certification of Medical Examination by Facility Licensee."
>
> Form NRC - 396, when signed by an authorized representative of the facility licensee, certifies that a physician conducted a medical examination of the applicant (as required in 10 CFR 55.21), and that the guidance contained in American National Standards Institute/American Nuclear Society (ANSI/ANS) - 3.4 -1983, "Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power Plants" was followed in conducting the examination and making the determination of medical qualification.
>
> ANSI/ANS 3.4-1983, Section 5.3.2(1), provides, in part, that certain cardiovascular conditions, including myocardial infarction, preclude solo operation of a nuclear power plant.
>
> 10 CFR 55.25 requires, in part, that if, during the term of the license, the licensee develops a permanent physical condition that causes the licensee to fail to meet the requirements of 10 CFR 55.21, the facility licensee shall notify the Commission within 30 days of learning the diagnosis of the condition, in accordance with 10 CFR 50.74(c). 10 CFR 50.74(c) provides, in part, that each licensee shall notify the appropriate Regional Administrator within 30 days of the permanent disability or illness of a licensed operator or senior operator as described in 10 CFR 55.25.
>
> Contrary to the above, on December 28, 1999, the licensee submitted to the NRC a Form NRC 396, an application for renewal of a Senior Reactor Operator (SRO) license, that was not complete and accurate in all material respects. Specifically, the Form NRC 396 certified that the applicant met the medical requirements of ANSI/ANS 3.4 -1983 and that the applicant's only restriction was to require corrective lenses be worn when

performing licensed duties. During December 1996, the SRO developed a permanent physical condition which did not meet the minimum cardiovascular standards specified in ANSI/ANS -3.4 -1983, Section 5.3.2(1) and which precluded the SRO from " solo" operation of a nuclear power plant. This information was material to the NRC because the NRC relied on the information contained in the Form NRC 396 dated December 28, 1999, to determine whether the applicant met the requirements of 10 CFR Part 55 to operate the controls of a nuclear power plant and to renew the SRO's license on February 1, 2000. In addition, the facility licensee was provided on January 15, 1997, with information about the SRO's December 1996 myocardial infarction, but did not notify NRC of the SRO's physical condition until March 24, 2004, a period in excess of 30 days after learning of the SRO's physical condition.

This is a Severity Level III violation (Supplement VII).

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation and prevent recurrence, and the date when full compliance will be achieved, is already adequately addressed on the docket in letter from the Licensee dated August 2, 2004. However, you are required to submit a written statement or explanation pursuant to 10 CFR 2.201 if the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to respond, clearly mark your response as a "Reply to a Notice of Violation, EA-04-109," and send it to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555 with a copy to the Regional Administrator and the Enforcement Officer, Region III, and a copy to the NRC Resident Inspector at the D.C. Cook Nuclear Plant, within 30 days of the date of the letter transmitting this Notice of Violation (Notice).

If you contest this enforcement action, you should also provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

If you choose to respond, your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html, to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that [illegible]ies the information that should be protected and a redacted copy of your response that deletes such information. If y[illegible]equest withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

In accordance with 10 CFR 19.11, you may be required to post this Notice within two working days.

Dated this 29th day of September 2004.

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PNO-IV-04-027: River Bend Station - Shutdown Greater than 72 Hours.

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October 4, 2004
PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE -- PNO- IV- 04- 027
This preliminary notification constitutes EARLY notice of events of POSSIBLE safety or public interest significance. The information is as initially received without verification or evaluation, and is basically all that is known by the Region IV staff on this date.
Facility
Entergy Operations, Inc.
River Bend Station
St. Francisville, Louisiana
Docket: 50- 458
License No. NPF- 47
[illegible]ee Emergency Classification
[illegible]ation of Unusual Event
Alert
Site Area Emergency
General Emergency
X Not Applicable
SUBJECT: Shutdown Greater than 72 Hours
DESCRIPTION: On October 1, 2004, at approximately 7: 19 a. m. (CDT) a fault occurred on Reserve Station Service (RSS) Line # 1, causing a loss of the Division I emergency bus. The Division I emergency diesel generator started and loaded as designed.
At 7: 30 a. m. a fault also occurred on the main transformer output line causing a generator lock- out, turbine trip, and subsequent reactor trip. The loss of the main transformer resulted in a loss of main feedwater and two circulating water pumps. The Main Steam Isolation Valves were closed by the operators due to lowering main steam line pressure, as a result of plant steam load isolation valves that could not be closed due to the loss of power, and difficulty in maintaining main condenser vacuum.
The reactor was stabilized using the High Pressure Core Spray system and the Safety Relief Valves to control reactor level and pressure. Main feedwater was reestablished to control reactor water level.
The plant was subsequently cooled down, and was in Mode 4 with reactor temperature less than 200 degrees F at 11: 35 p. m. on October 1. The licensee is investigating the cause of the faults on RSS Line # 1 and the main unit transformer output line.
The Senior Resident Inspector was in the control room at the time of the reactor trip and continued to monitor the licensee's actions. The Senior Resident Inspector from Waterford was dispatched to River Bend Station to assist in the event response. A special inspection will be conducted in response to this event.
Region IV has informed the offices of the Executive Director for Operations, Nuclear Reactor Regulation, and Public Affairs.
The state of Louisiana will be informed.
T[illegible]ormation has been discussed with the licensee and is current as of 4: 30 p. m. on O[illegible] 3, 2004.
ADAMS ACCESSION **ML04278052** [?]

CONTACTS: David Graves, 817- 860- 8141
Peter Alter, 225- 635- 3193

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PNO-IV-04-026 Diablo Canyon Power Plants, Units 1 & 2 - Response to California Earthquake.

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September 28, 2004
PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE -- PNO- IV- 04- 026
This preliminary notification constitutes EARLY notice of events of POSSIBLE safety or public interest significance. The information is as initially received without verification or evaluation, and is basically all that is known by the Region IV, Arlington, Texas, staff on this date.
Facility
Pacific Gas and Electric Company
Diablo Canyon Power Plant Units 1 and 2
Avila Beach, California 93424
Dockets: 50- 275; 50- 323
License Nos.: DPR- 80; DPR- 82
[illegible]see Emergency Classification
[illegible]fication of Unusual Event
Alert
Site Area Emergency
General Emergency
Not Applicable
SUBJECT: RESPONSE TO CALIFORNIA EARTHQUAKE
DESCRIPTION:

On September 28, 2004, at 10: 16 a. m. PDT, a magnitude 6.0 earthquake occurred approximately 45 miles northeast of Diablo Canyon Power Plant near Parkfield, California. The licensee declared a notification of unusual event (NOUE) at 10: 30 a. m. PDT for the ground motion felt by the control room operators and measuring greater than 0.01g acceleration (0.012g was recorded on the Earthquake Force Monitor). Diablo Canyon Power Plant, Units 1 and 2, remained at 100 percent power following the earthquake. The licensee is performing walkdowns of the fire protection systems, both containment buildings, auxiliary buildings, the turbine building, and other areas. While these impact assessments are still in progress, preliminary reports indicate no structural damage has been identified. An assessment of any inputs of the offsite power lines will also be conducted. Emergency evacuation routes remained available throughout the emergency planning zone. Two of the 131 emergency sirens were unavailable immediately following the earthquake, with one emergency siren within the emergency planning zone being unavailable because of planned maintenance and the second siren being unavailable due to a loss of power stemming from the earthquake. The licensee restored power to the second siren at 11: 40 a. m. PDT. The licensee decided to remain in the NOUE for a minimum of 24 hours because of the occurrence of aftershocks and notification from the United Sates Geological Survey of probable aftershocks for the first 24 hours following the initial earthquake. The resident inspectors responded to the control room at the time of the earthquake, independently monitored the plant response to the earthquake, and are independently assessing plant structures and equipment. Additionally, a specialist inspector is being dispatched to the site, and the NRC staff is m[illegible]ring the licensee's response activities from its Incident Response Center in Arlington, Texas. An N[illegible]spector will continue to remain on site throughout the period the licensee remains in the NOUE. Th[illegible]tate of California has been informed.

The licensee made an announcement on local radio stations indicating no damage occurred at the plant.
Region IV received notification of this occurrence by telephone from the NRC senior resident inspector a[illegible] Diablo Canyon Power Plant at 10: 20 a. m. PDT, on September 28, 2004.
[illegible] IV has informed EDO/ NRR/ PA.
T[illegible]nformation has been discussed with the licensee and is current as of 1: 55 p. m. PDT, September 28, 2004.
ADAMS ACCESSION NUMBER: **ML042720618** [?]
CONTACTS: William B. Jones (817) 860- 8147

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PNO-IV-04-026A: Diablo Canyon Power Plant Units 1 and 2, Response to California Earthquake - Update.

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September 29, 2004
PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE -- PNO- IV- 04- 026A
This preliminary notification constitutes EARLY notice of events of POSSIBLE safety or public interest significance. The information is as initially received without verification or evaluation, and is basically all that is known by the Region IV, Arlington, Texas, staff on this date.
Facility
Pacific Gas and Electric Company
Diablo Canyon Power Plant Units 1 and 2
Avila Beach, California 93424
Dockets: 50- 275; 50- 323
[illegible]e Nos.: DPR- 80; DPR- 82
[illegible]ee Emergency Classification
Notification of Unusual Event
Alert
Site Area Emergency
General Emergency
X Not Applicable
SUBJECT: RESPONSE TO CALIFORNIA EARTHQUAKE - UPDATE
DESCRIPTION:

On September 29, 2004, at 12: 15 p. m. PDT, the Diablo Canyon Power Plant exited the Notification of Unusual Event (NOUE). The NOUE was declared at 10: 30 a. m. PDT on September 28, 2004, due to a magnitude 6.0 earthquake which occurred approximately 45 miles northeast of Diablo Canyon Power Plant near Parkfield, California. The Nuclear Regulatory Commission Region IV office exited the monitoring mode on September 29, 2004, at 3: 00 p. m. CDT. Region IV activated its incident response center at 12: 47 p. m. CDT on September 28, 2004.

Diablo Canyon Power Plant, Units 1 and 2, remained at 100 percent power during the NOUE. The licensee and the NRC resident inspectors and a regional inspector performed walkdowns of the fire protection systems, containment buildings, auxiliary buildings, the turbine building, and other areas. No damage to any structures, systems, or components was identified. Emergency evacuation routes remained available throughout the emergency planning zone. The licensee remained in the NOUE to monitor the occurrence of aftershocks following the initial earthquake.

The resident inspectors responded to the control room at the time of the earthquake, independently monitored the plant response to the earthquake, and assessed plant structures and equipment. An NRC inspector was on site throughout the period the licensee was in the NOUE. A Region IV inspector was dispatched to the site on September 28, 2004, to augment the resident inspector staff and was onsite later that evening. The NRC staff monitored the licensee's response activities from its Incident Response Center in Arlington, Texas.

T[illegible]te of California has been informed.

R[illegible] IV received notification of exit from the NOUE by telephone from the NRC senior resident inspector at the Diablo Canyon Power Plant at 12: 32 p. m. PDT, on September 29, 2004.

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PNO-II-04-009B, Region II Response to Hurricane Jeanne.

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PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE PNO- II- 04- 009B
This preliminary notification constitutes EARLY notice of events of possible safety or public interest significance. This information is as initially received without verification or evaluation, and is basically all that is known by Region II staff (Atlanta, Georgia) on this date.
Facility Licensee Emergency Classification
Florida Power Corporation X Notification of Unusual Event
Crystal River Unit 1 Alert
Crystal River, FL Site Area Emergency
Docket/ License: 050- 302/ DPR 72 General Emergency
Not Applicable
SUBJECT: REGION II RESPONSE TO HURRICANE JEANNE
[illegible]l River declared a NOUE at 5: 05 a. m. (EDT) due to a hurricane warning being issued for the Crystal [illegible] area. Crystal River is currently at 100% power with no operational issues. Based on the projected weather conditions, Crystal River has no plans or requirements to shutdown the unit. The Technical Support Center is manned and operational. The Emergency Operation Facility has not been activated. Current wind speeds are 26 mph with 33 mph gusts. Peak wind speeds of up to 70 mph are expected at 2: 00 p. m. (EDT).
St. Lucie remains in a loss of offsite power to both units since 11: 56 p. m. (EDT) on September 25. All 4 emergency diesel generators started and are providing power to vital loads. On site diesel fuel oil supplies are estimated at seven days. The licensee remains in a Notice of Unusual Event for the loss of offsite power and the hurricane warning. Both St. Lucie units are on shutdown cooling. The cause of the loss of offsite power has been identified. Offsite power is estimated to be restored by 12 Noon (EDT), September 26, 2004.
Current access to the site is limited due to damage to bridges.
Turkey Point exited their NOUE at 7: 15 a. m. (EDT) on September 26 based on lifting of the Hurricane warning in the area. Turkey Point did not experience any significant high winds from the storm. The shutdown of Turkey Point Unit 3 for a scheduled refueling outage is continuing. Unit 4 remains at full power.
Hurricane Jeanne made landfall on the east coast of Florida, very close to the St. Lucie site. St. Lucie experienced maximum sustained wind speeds of 100 mph and gusts as high as 120 mph. As of 9: 00 a. m., (EDT), the storm is a Category 1 hurricane at winds approximately 85 mph.
The NRC remains in the Monitoring Mode with inspectors on site at St. Lucie and Crystal River.
Region II staff members continue to monitor activities from the Incident Response Center and the Florida Power and Light Emergency Operations Facility in Miami, Florida.
The information presented herein was discussed with both licensees (FPL and FPC/ Progress Energy) and is current as of 10: 00 a. m. (EDT) September 26, 2004.
CONTACT: Robert Haag
404- 562- 4480

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PNO-II-04-009C, Tropical Storm/Hurricane Jeanne.

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September 27, 2004
PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE PNO- II- 04- 009C
This preliminary notification constitutes EARLY notice of events of possible safety or public interest significance. The information is as initially received without verification or evaluation, and is basically all that is known by Region II staff (Atlanta, Georgia) on this date.
Facilities Licensee Emergency Classification
Florida Power and Light Notification of Unusual Event
St. Lucie, Units 1 and 2 Alert
Jensen Beach, Florida Site Area Emergency
Dockets/ Licenses: 05000335, DPR- 67, 05000389 General Emergency
N[illegible]16 X Not Applicable
[illegible]ss Energy - Florida Power Corporation
C[illegible] River Unit 3
Crystal River, Florida
Docket/ License: 05000302, DPR- 72
SUBJECT: TROPICAL STORM/ HURRICANE JEANNE
The NRC Region II office entered the Monitoring Mode and activated its Incident Response Center (IRC) on September 25, 2004, in preparation for Hurricane Jeanne's impact on Region II nuclear plants. Region II staff members were also dispatched to the Florida Power and Light Company Emergency Operations Facility in Miami, FL, and the State Emergency Operations Center in Tallahassee, FL. The NRC determined that the IRC would secure from the Monitoring Mode for this storm and deactivate the IRC in Region II on September 27, 2004 as of 0400 a. m.
St. Lucie
Both units at St. Lucie remain in Mode 4 on shutdown cooling. The licensee exited the Notice of Unusual Event (NOUE) at 1412 (EDT) on September 26 due to the lifting of the hurricane warning and restoration of offsite power. The resident inspectors maintained continual coverage through the hurricane and continue to monitor licensee actions. Region II has coordinated with FEMA to commence assessment of the offsite infrastructure damage.
Crystal River
Crystal River is currently at 97% power and remained in Mode 1 throughout the storm. The licensee exited the Notice of Unusual Event (NOUE) at 0345 (EDT) on September 27 once the storms winds had subsided. The licensee stayed in a NOUE and maintained Technical Support Center (TSC) staffing as a precautionary measure although the hurricane warning had been lifted for the area earlier on September 26. The resident inspectors maintained coverage throughout the hurricane and continued to assess plant conditions until the lifting of the NOUE. Maximum sustained winds at the site were approximately 50 to 60 mph with gusts up to 70 mph.
This information presented herein has been discussed with the licensees, and is current as of 4: 00 a. m., September 27, 2004.
C[illegible]CT: Steve Cahill
4[illegible]2- 4480 (Region II IRC)

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U.S. Nuclear Regulatory Commission

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PNO-II-04-009, Region II Response to Hurricane Jeanne.
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PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE PNO- II- 04- 009
This preliminary notification constitutes EARLY notice of events of possible safety or public interest significance. The information is as initially received without verification or evaluation, and is basically all that is known by Region II staff (Atlanta, Georgia) on this date.
Facility Licensee Emergency Classification
Florida Power and Light Co. X Notification of Unusual Event
St. Lucie Units 1 & 2 Alert
Jensen Beach, FL Site Area Emergency
Docket/ License: 050- 335/ DPR 67 (Unit 1), General Emergency
050- 389/ NPF 16 (Unit 2) Not Applicable
S[illegible]CT: REGION II RESPONSE TO HURRICANE JEANNE
T[illegible]clear Regulatory Commission Region II office is currently tracking Hurricane Jeanne which is currently a[illegible]egory 3 storm with winds exceeding 110 miles per hour. Region II activated its incident response center on Saturday, September 25, 2004, as Hurricane Jeanne neared the east coast of Florida. The agency entered the Monitoring Mode at 3: 15 p. m. Region II has NRC inspectors on site at St. Lucie, and inspectors are monitoring activities at Crystal River and Turkey Point. Region II staff members were also dispatched to the Florida Power and Light Company Emergency Operations Facility in Miami, Florida.
The St. Lucie and Turkey Point sites declared a Notice of Unusual Event on September 24 at 17: 00 (EDT) due to the issuance of a Hurricane Warning for the east coast of Florida. St. Lucie Units 1 and 2 shutdown at 11: 30 and 12: 00 (EDT), respectively, on September 25. Turkey Point Unit 3 is at 50 percent power in preparation for its refueling outage and Turkey Point Unit 4 is operating at 100 percent power.
The Federal Emergency Management Agency Region IV has also activated its Regional Operations Center to monitor the hurricane.
The State of Florida has been notified.
The information presented herein was discussed with the licensee and is current as of 7: 30 p. m. (EDT) September 25, 2004.
CONTACT: Len Wert
404- 562- 4480

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PNO-II-04-009A, Region II Response to Hurricane Jeanne.

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PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE PNO- II- 04- 009A
This preliminary notification constitutes EARLY notice of events of possible safety or public interest significance. This information is as initially received without verification or evaluation, and is basically all that is known by Region II staff (Atlanta, Georgia) on this date.
Facility Licensee Emergency Classification
Florida Power and Light Co. X Notification of Unusual Event
St. Lucie Units 1 & 2 Alert
Jensen Beach, FL Site Area Emergency
Docket/ License: 050- 335/ DPR 67 (Unit 1), General Emergency
050- 389/ NPF 16 (Unit 2) Not Applicable
SUBJECT: REGION II RESPONSE TO HURRICANE JEANNE
[illegible]cane Jeanne made landfall on the east coast of Florida, very close to the St. Lucie site. St. Lucie [illegible]enced maximum sustained wind speeds of 100 mph and gusts as high as 120 mph. Both St. Lucie and Turkey Point remain in a Notice of Unusual Event due to issuance of a hurricane warning.
At 11: 56 p. m. (EDT), offsite power was lost to both units at St. Lucie. All 4 emergency diesel generators started and provided power to vital loads. The 1B Intake Cooling Water Pump failed to automatically start, but was manually started from the control room. The licensee declared a Notice of Unusual Event for the loss of offsite power. Both St. Lucie units are on shutdown cooling. The cause of the loss of offsite power has not been identified.
Turkey Point did not experience any significant high winds from the storm. The shutdown of Turkey Point Unit 3 for a scheduled refueling outage is continuing. Unit 4 remains at full power.
Crystal River Unit 3 remains at full power. Winds near 70 mph are projected to be experienced at the site for about 1 hour beginning at 9: 00 a. m. (EDT) on September 26.
The NRC remains in the Monitoring Mode with inspectors on site at St. Lucie and Crystal River.
Region II staff members continue to monitor activities from the Incident Response Center and the Florida Power and Light Emergency Operations Facility in Miami, Florida.
The information presented herein was discussed with the licensee (FPL) and is current as of 5: 00 a. m. (EDT) September 26, 2004.
CONTACT: Len Wert
404- 562- 4480

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PNO-IV-04-025: Entergy Operations, Inc. - River Bend Station and Waterford-3: Region IV Response to Hurricane Ivan.

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September 15, 2004
PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE -- PNO- IV- 04- 025
This preliminary notification constitutes EARLY notice of events of POSSIBLE safety or public interest significance. The information is as initially received without verification or evaluation, and is basically all that is known by the Region IV, Arlington, Texas, staff on this date.
Facilities
Entergy Operations, Inc.
River Bend Station
Docket: 50- 458
License: NPF- 47
W[illegible]ford 3
[illegible]: 50- 382
License: NPF- 38
Licensee Emergency Classification
X Notification of Unusual Event for Waterford
Alert
Site Area Emergency
General Emergency
X Not Applicable for River Bend
SUBJECT: REGION IV RESPONSE TO HURRICANE IVAN
DESCRIPTION:
The Nuclear Regulatory Commission Region IV office is taking several steps to monitor the effects of Hurricane Ivan on nuclear facilities. The Hurricane is currently a Category 4 storm with winds exceeding 140 miles per hour. Region IV is planning to activate its incident response center by 1 p. m. CDT, Wednesday, September 15, 2004, as Hurricane Ivan nears the Gulf Coast with a projected landfall between Grand Isle, Louisiana, and Apalachicola, Florida. Region IV has additional NRC inspectors onsite at the Waterford and River Bend nuclear power plant sites in Louisiana to monitor licensee activities and to review preparations for severe weather. Region IV staff members were also dispatched to the Federal Emergency Management Agency Region VI office in Denton, Texas, and the State of Louisiana emergency operations center in Baton Rouge. Each of the Agencies have activated their operations centers in preparation for the Hurricane.
The Waterford 3 and River Bend Nuclear Stations are currently operating at 100 percent power. The Waterford 3 Steam Electric Station declared a Notice of Unusual Event on September 14, 2004, at 4: 04 p. m. (CDT) because of the issuance of a Hurricane Warning for Saint Charles Parish, Louisiana. Waterford 3 is currently not experiencing any high winds or rain.
The states of Louisiana and Mississippi have been notified.
The information presented herein is current as of 11 a. m. CDT, September 15, 2004.
AD[illegible] ACCESSION NUMBER: **ML042590359** [?]
C[illegible]T: W. B. Jones, 817/ 860- 8147
wbj@nrc. gov

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PNO-IV-04-025A: Entergy Operations, Inc.: River Bend Station and Waterford-3 - Update to Region IV Response to Hurricane Ivan.

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September 16, 2004
PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE -- PNO- IV- 04- 025A
This preliminary notification constitutes EARLY notice of events of POSSIBLE safety or public interest significance. The information is as initially received without verification or evaluation, and is basically all that is known by the Region IV, Arlington, Texas, staff on this date.
Facilities
Entergy Operations, Inc.
River Bend Station
Docket: 50- 458
License: NPF- 47
Waterford 3
Docket: 50- 382
License: NPF- 38
Licensee Emergency Classification
Notification of Unusual Event
Alert
Site Area Emergency
General Emergency
Not Applicable
SUBJECT: UPDATE TO REGION IV RESPONSE TO HURRICANE IVAN
DESCRIPTION:
The Nuclear Regulatory Commission Region IV office exited the monitoring mode on September 16, 2004, at 9: 00 a. m. Central Daylight Time (CDT). Region IV activated its incident response center at 1 p. m. CDT, Wednesday, September 15, 2004, Region IV augmented the NRC inspectors onsite at the Waterford and River Bend nuclear power plant sites in Louisiana to monitor licensee activities and to review preparations for severe weather. Region IV staff members were dispatched to the Federal Emergency Management Agency (FEMA) Region VI office in Denton, Texas, and the State of Louisiana emergency operations center in Baton Rouge.
The Waterford 3 and River Bend Nuclear Stations are currently operating at 100 percent power. The Waterford 3 Steam Electric Station exited the Notice of Unusual Event on September 16, 2004, at 5: 02 a. m. CDT. The nuclear reactors at both stations sustained no damage to safety systems from the storm and remained at 100 percent power. Maximum sustained winds near 30 mph, with gusts up to 44 mph were experienced at the Waterford 3 station. Winds began to diminish at the site shortly before Hurricane Ivan made landfall. At 4: 00 a. m. CDT, the National Weather Service cancelled the hurricane warning for Saint Charles Parish and other surrounding areas. Maximum sustained winds at the River Bend Station were 14 mph, with gusts up to 20 mph. Based on communication with the FEMA's and the state of Louisiana, the emergency evacuation routes surrounding the stations were not adversely affected by Hurricane Ivan.
The State of Louisiana has been notified.
The information presented herein is current as of 1 p. m. CDT, September 16, 2004.

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PNO-II-04-08C; Florida Power & Light & Progress Energy- Florida Power Corporation, Tropical Storm Frances.

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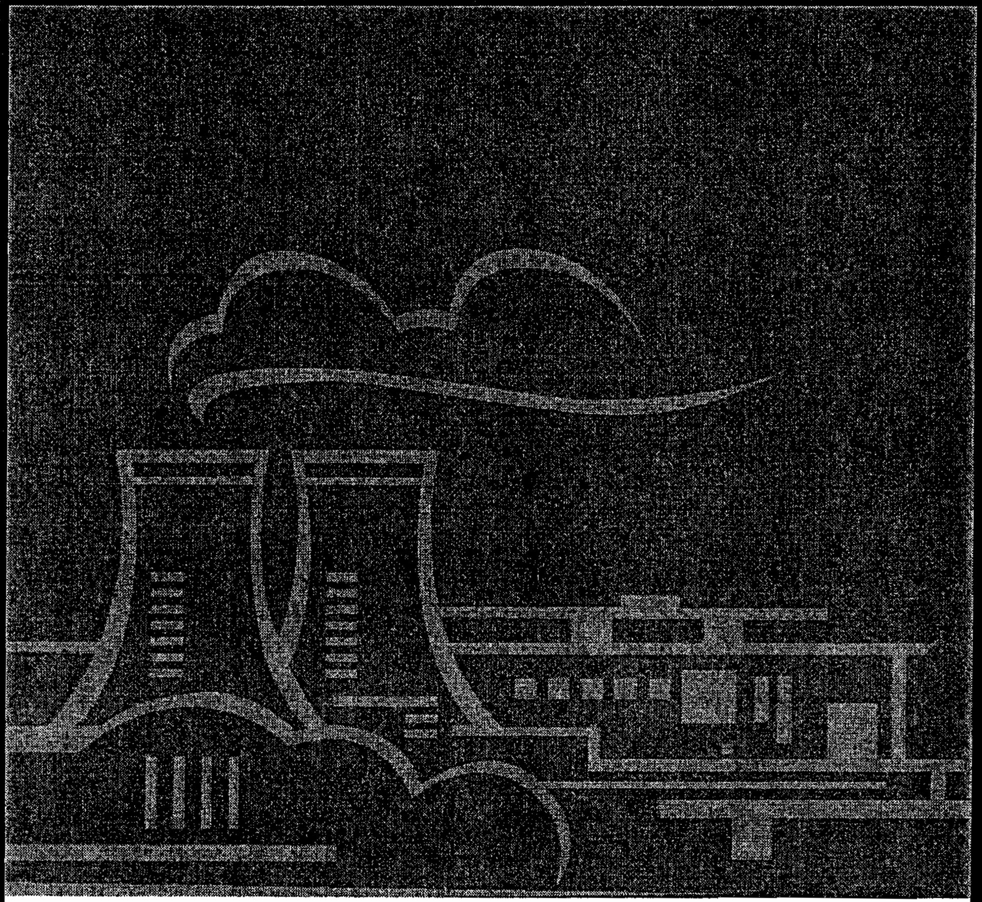
September 7. 2004
PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE PNO- II- 04- 08C
This preliminary notification constitutes EARLY notice of events of possible safety or public interest significance. The information is as initially received without verification or evaluation, and is basically all that is known by Region II staff (Atlanta, Georgia) on this date.
Facilities Licensee Emergency Classification
Florida Power and Light Notification of Unusual Event
St. Lucie, Units 1 and 2 Alert
Jensen Beach, Florida Site Area Emergency
Dockets/ Licenses: 05000335, DPR- 67, 05000389 General Emergency
[illegible]6 X Not Applicable
Progress Energy - Florida Power Corporation
Crystal River Unit 3
Crystal River, Florida
Docket/ License: 05000302, DPR- 72
Subject: Tropical Storm Frances
The Region II office entered the Monitoring Mode and activated its Incident Response Center (IRC) on September 3, 2004, in preparation for Hurricane Frances' impact on Region II nuclear plants. Region II staff members were also dispatched to the Florida Power and Light Company Emergency Operations Facility in Miami, FL, and the State Emergency Operations Center in Tallahassee, FL. The following is a brief synopsis of the events that transpired from September 3 to cessation of the Monitoring Mode and deactivation of the IRC in Region II on September 7.
St. Lucie
The licensee issued a Notice of Unusual Event (NOUE) at 10: 43 a. m.(EDT) on September 2 due to the declaration of a Hurricane Warning for the east coast of Florida. The site made preparations for the impending severe weather by reducing plant power and eventually shutting down both reactor plants on September 4. All necessary shutdown safety systems responded as required and were utilized in the shutdown of the reactor plants. The impact of Hurricane Frances was most severe at the site in the early morning of September 5, as the eye of Hurricane Frances passed over the site and subjected the plant to high winds exceeding 80 mph and heavy rain. The site did not suffer a loss of offsite power (LOOP), but did lose dedicated communications with the IRC. Alternative methods of communicating were established and maintained throughout the remainder of the hurricane. The licensee exited the NOUE on September 5 at 5: 20 p. m. An evaluation, after the hurricane passed revealed that the site had suffered some structural damage to warehouses and outbuildings and the city water supply had been isolated to the plant. On September 6, the licensee observed an increase in Unit 2 unidentified reactor coolant system (RCS) leakage and quantified the leakage rate at 2.5 gpm. The licensee has since determined that 1.75 gpm of the leakage is associated with the containment spray system and RCP seals. The licensee continues to investigate the source of the remaining 0.75 gpm leak. The resident inspectors maintained coverage throughout the hurricane and continue to monitor licensee corrective actions. Region II has coordinated with FEMA to assess the offsite infrastructure damage.

Crystal River
The licensee issued a Notice of Unusual Event (NOUE) at 5: 10 p. m.(EDT) on September 5 due to the d[illegible]tion of a Hurricane Warning for the west coast of Florida. Hurricane Frances was downgraded to a Tropical Storm and the sustained winds experienced on the site were approximately 35- 40 mph. A reactor trip occurred on September 6 due to a partial Loss of Offsite Power (LOOP) and all necessary safe shutdown systems operated as expected. An emergency diesel generator started to supply electrical power to the " B" train vital bus. On September 7 offsite power was restored to the affected bus and the plant electrical alignment was restored to its normal configuration. Communications with the site have been adequate throughout the storm. The resident inspectors maintained coverage throughout the storm and continue to monitor the licensee's recovery from the plant trip. The licensee exited the NOUE on September 7 at 10: 17 a. m. Region II has coordinated with FEMA to assess the offsite infrastructure damage.
This information presented herein has been discussed with the licensees and the State of Florida, and is current as of 1: 00 p. m., September 7, 2004.
CONTACT: Joel Munday
404- 562- 4560

Organized by:
Office of Nuclear Regulatory Research
United States Nuclear Regulatory Commission
Washington, DC 20555



Nuclear Safety Research Conference

October 25-27, 2004
Marriott at Metro Center
Washington DC USA

The Office of Nuclear Regulatory Research of the
U.S. Nuclear Regulatory Commission is hosting its annual

Nuclear Safety Research Conference (NSRC)

on October 25-27, 2004
at the Marriott at Metro Center
775 12th Street NW
Washington DC

We are planning an exciting program and welcome
your attendance and active participation.

Background and Purpose

The NSRC is an international conference sponsored by the Office of Nuclear Regulatory Research (RES), U.S. Nuclear Regulatory Commission (NRC) that has been held every year since 1973. Its focus on regulatory issues attracts researchers, regulators, and utility representatives from the United States and more than twenty other countries. Participants interact with the NRC's staff and with colleagues to discuss the insights from research programs undertaken to support the NRC's mission.

Conference Format

This year's sessions will cover technical research in materials aging and degradation, new reactors, fuels, PRA infrastructure, and radiation protection, codes and operating experience. This information will center on where research has taken us, its current status, and where it will go in the next 3-5 years. Papers and presentations will cover past and present work, with the future being discussed by panels of experts.

Guest speakers and panelists tentatively include NRC Chairman Nils J. Diaz, and Commissioners Edward McGaffigan, Jr., and Jeffrey S. Merrifield, representatives from organizations, industries, government, the research community and public interest groups in the United States and abroad.

Poster Sessions will highlight additional technical material.

This Conference Agenda is preliminary and subject to change. For the most current information regarding the 2004 NSRC, contact Conference Coordinator Sue Monteleone at (631) 344-7235 or visit the website at www.bnl.gov/nsrc.

Conference Agenda (preliminary)

Monday October 25, 2004

Plenary Session

8:30-8:45 am	Opening Remarks – Carl Paperiello, Director, RES
8:45-9:45 am	Keynote Speaker – Nils J. Diaz, Chairman, NRC
10:00-12:00	Two concurrent sessions

Session 1 – Materials Aging and Degradation Research

Objective: To discuss consideration of failures of passive systems and components in risk assessments, and discuss two risk-informed initiatives involving passive failures. Results will be presented from materials degradation and aging research, emphasizing regulatory and safety insights.

A panel with experts in PRA and materials will debate and challenge issues associated with consideration of passive failures and risk assessments. Issues related to methodology, treatment of uncertainties, appropriate risk measures and defense-in-depth, and their use in decision-making will be discussed.

Session 2 – New Reactors

Objective: To discuss ongoing recent or planned NRC and industry safety research activities affecting new and advanced reactors and to discuss infrastructure to support next-generation reactor technology.

12:00-1:30 pm	Lunch
1:30-5:00 pm	Continuation of the morning's concurrent sessions

Session 1 – Materials Aging and Degradation Research (cont'd.)

Subtopic 1 – Risk-Informed Initiatives involving passive component failures will center on the revision of the PTS rule (10CFR50.61), and the development of LOCA frequencies considering the degradation of passive systems.

Subtopic 2 – Materials Aging Research will describe the NRC's and industry's programs on proactive management of degradation, and on barrier integrity-leakage detection systems.

Session 2 – New Reactors (cont'd.)

A panel of NRC staff, industry, national laboratories, and public interest groups will discuss future research needed to support licensing of new reactors.

Tuesday October 26, 2004

Plenary Session

8:30 - 9:30	Commissioner's Speech
9:45 - 11:45	Two concurrent sessions

Session 3 – Materials Fuel Research (click here: for updated information)

Objective: To discuss the issues and research required to establish the technical basis for LOCA rulemaking. The focus will be on state-of-the-art review of LOCA issues, and the effects of alloy and burn-up on cladding behavior.

Session 4 – Development of a PRA Infrastructure

Objective: This session will explore improving the PRA infrastructure to identify the different types of generic- and application-specific guidance, and prescriptive instructions needed to support quality in PRAs. Presently, the analyst's ability to develop realistic, complete models is limited by the need for approaches to treat uncertainties, including completeness uncertainty, in risk-informed decision making. The session will also consider Human Reliability Assessment and advances in Fire Risk Assessment.

11:45 -1:15 pm	Lunch
1:30 - 5:00 pm	Continuation of the morning's concurrent sessions

Session 3 – Materials Fuel Research (cont'd.) - (click here: for updated information)

Topics will cover the characteristics of cladding under LOCA conditions, and the NRC's rulemaking on 10CFR50.46.

A panel of experts will debate research needs, regulatory impacts, and question whether NRC's focus is optimal, and cooperative arrangements are appropriately used.

Session 4 – Development of a PRA Infrastructure (cont'd.)

This session will consider insights on PRA quality from SPAR and ASP, phased approaches (e.g., Rg1.200), and international activities.

A panel discussion on PRA methods & tools will debate our limitations, priorities, and whether future risk applications might require additional PRA tools, methods, and guidance.

Additionally, we will cover modeling organizational factors, integrating equipment aging into the PRA model, and quantifying the risk of low-power shutdown.

Tuesday, October 26, 2004
9:45-11:45

Session 3 - Spent Fuel Research

Objective: Dry storage and transportation casks impose unique temperature and pressure conditions on spent fuel rods, and the consequences of these conditions are considered in cask licensing. Research results under these conditions and their implications will be presented in this session.

1:30-5:00
Session 3B - High Burnup Under LOCA Conditions

Objective: Risk-informed, performance-based rulemaking is planned to revise 10 CFR 50.46 and Appendix K for LOCA analysis. Results of recent research will be presented, and these results will provide the technical basis for changing the embrittlement criteria in the rule.

Wednesday October 27, 2004

8:30 - 11:30 Two concurrent sessions

Session 5 – Radiation Protection

Objective: To discuss current initiatives in radiation protection including new recommendations from the International Commission on Radiological Protection (ICRP), the challenges of homeland security, and establishing a more robust materials program.

A poster session will demonstrate new HP research tools, MACCS improvements, and an effluent database.

Session 6 – Codes

Objective: To explore issues in developing and applying computer codes to analyze reactor systems. Topics will include the analysis, applications, and maintenance of codes, the status of the TRACE project, the analysis of source terms, and the assessment and modernization of the MELCOR code.

The panel discussion will probe the optimization and updating of codes, and identify areas of the greatest benefits to pursue.

11:30 -1:00 pm Lunch

1:00 - 3:15 pm Two concurrent sessions

Session 5 – Radiation Protection (cont'd.)

A panel of specialists will discuss how NRC should respond to the many challenges of the ICRP's evolving recommendations on radiological protection, waste disposal, and their practical implementation.

Session 7 – Operating Experience

Objective: To present the efforts to enhance the NRC Operating Experience Program utilizing the analysis conducted by RES. Prime concern is the impact on this program from the recommendations of the Davis Besse Lessons Learned Task Force and the August 14, 2003 Northeast Blackout.

The presentation will include generic program-process & screening analysis, the operating experience implementation task force and the international safety margins project. We will look at insights from assessments of the electrical grid, the boric-acid program, and from Mexico's operating experience.

Plenary Session

3:30 Closing remarks

Conference Registration

The NSRC is open to the public, and there is no registration fee. However, registration for the conference is strongly encouraged.

Registration Form (Please Print)

Name: ______________________________

Title: ______________________________

Affiliation: ______________________________

Address: ______________________________

______________________ Country: __________

Telephone: ______________________________

Fax: ______________________________

email: ______________________________

Four easy ways to register:

1. Visit our web site **www.bnl.gov/nsrc**
2. Mail this registration form
3. Fax this registration form
4. e-mail the above registration information

Submit registration information before October 15, 2004 to:

Susan Monteleone, Conference Coordinator
Brookhaven National Laboratory
P.O. Box 5000, Building 130
Upton, New York USA 11973-5000
Tel. 1-631-344-7235
Fax. 1-631-344-3957
e-mail: susanm@bnl.gov

Hotel Reservations

A block of rooms has been reserved at the Marriott at Metro Center at the special conference rate of $209 per night. Please contact the hotel directly to make room reservations and mention the "Nuclear Safety Research Conference". Reservations must be made by September 30, 2004, to receive the special rate.

Marriott at Metro Center
775 12th St., NW
Washington, DC 20005
Phone +1-800-228-9290 or +1-202-737-2200
Fax +1-202-824-6106

Transportation

The Washington D.C. area is serviced by three major airports, Ronald Reagan Washington National Airport (National), Washington Dulles International Airport (Dulles), and Baltimore-Washington International Airport (BWI). The Marriott Hotel, National Airport and other local hotels are conveniently reached via the METRO train. Car rentals are available at all three airports. Public transportation via the METRO is recommended.

Inside NRC

Volume 26 / Number 20 / October 4, 2004

Domenici book touts his role in regulatory turnaround; Jackson begs to differ

By the late 1990s, NRC's licensing process had become unpredictable and regulatory delays were so prevalent that the nuclear industry was losing interest in applying for reactor license extensions and uprates or gearing up for possible new plant construction, according to Sen. Pete Domenici (R-N.M.).

As he saw it, the industry was running up costly bills to address issues not directly tied to safety or to tend to prescriptive requirements to pacify the regulators.

The agency had become bloated and out of touch with its safety mission and needed more discipline, says Domenici, who decided it was time to mete out some "tough love." Then-NRC Chairman Shirley Jackson, however, has a slightly different version of those times.

Domenici, a six-term senator who is chairman of the Senate Energy & Natural Resources and head of the appropriations committee with jurisdiction over NRC's budget, is now a first time author. In a new book he wrote with the help of Blythe Lyons and Julian Steyn, consultants with the Washington, D.C.-based firm Energy Resources International, Domenici portrays the NRC as in dire need of reform after years of congressional neglect.

The crush of problems worsened,

according to Domenici, after whistleblowers
sought out the media to expose
Millstone as out of compliance with
NRC design basis regulations. He says
the agency, under the leadership of then-Chairman Jackson,
reacted by developing more regulations and cracking
down on minor infractions.

Domenici says he began hearing complaints not just
from the industry but also from NRC staffers. "I became
very concerned that many NRC directives were not promoting
safe operations, and certainly were not using riskinformed
criteria," Domenici wrote in his book, "A
Brighter Tomorrow."

"The answer was to move to risk-informed, performance-based regulation," he said. "The NRC needed to
review existing regulations to reform those that were outdated,
paperwork-oriented, or that consumed resources for
compliance but had no positive impact on safety."
Domenici recalled he decided to "get some things
straight" with Jackson at a meeting arranged at her
request. But before the two met, word traveled back from
the NRC staff that it was Jackson who wanted to clear the
air on some issues.

When Jackson arrived, he confronted her with a copy
of the proposed fiscal 1999 budget for NRC, which would
have required the agency to absorb deep spending cuts
and lay off 700 staffers. At the time, Domenici staffers had
signaled the senator was prepared to slash between $100-
and $150-million of the agency's $488.6-million budget
request (INRC, 11 May '98, 1). House appropriators were
equally resolved to force agency changes, directing in
report language that NRC make improvements to its adjudicatory
and licensing processes.

Jackson was "stunned" when she saw the budget proposal,
Domenici said. "The chairwoman wondered aloud,
'You can't be serious?' But as it became perfectly clear that
I not only had the backup but the budget cuts in hand,
she asked for details. I gave them," Domenici said.
Domenici stressed he wasn't bluffing about the budget
cuts, which he promised to push through if action wasn't
taken within a "few months at most."
"Chairman Jackson got up, left, and didn't look back,"
he said.

Historical inaccuracies?

Domenici blames the NRC for hindering the progress of the commercial nuclear sector over the past two decades in an 11-page chapter titled "Regulatory Roadblocks to Nuclear Power."

But Jackson, now president of Rensselaer Polytechnic Institute in Troy, N.Y., says Domenici didn't quite get it right on the regulatory revolution during her tenure. Jackson became a commissioner and assumed the chairmanship in May 1995 and served through June 1999. She said Domenici's description of NRC's regulatory approach while she was chairman was "a gross mischaracterization of the facts" and asserted that NRC began shifting to a new regulatory paradigm years before the meeting described in the book. Jackson says she began speaking publicly as early as September 1996 about the need to refocus on the "safety-significant aspects of reactor operations." She took exception to Domenici taking credit for moving the agency toward a new regulatory philosophy, contending she was the one "who both coined the phraseology 'risk-informed, performance-based regulation,' and articulated its meaning." In a page and a half of comments to Inside NRC, Jackson wrote, "Moreover, this is and was known in nuclear circles around the world." She noted one of her first speeches on the subject was made at a meeting in Madrid.

The former chairman also said she was well aware of the senator's budget cut proposal before her meeting with Domenici. "I reviewed the budget proposal, and the meeting did not prompt any change in direction for the NRC, but a continuation (and some acceleration) of an approach we had already begun," Jackson said. "The agency already had embarked on an aggressive program designed to heighten the security and safety of nuclear power plants, and to reformulate the regulatory approach. Any assertions to the contrary distort the public record and provide a disservice to those who might read Sen. Domenici's book, to me, and to the staff of the NRC,"Jackson said.

Book release this month

Senate Majority Leader Bill Frist (R-Tenn.) is hosting a reception Oct. 6 at the U.S. Capitol to celebrate the publication of Domenici's book, which is slated for release this month. Among those invited are members of the press, industry officials, and current and former congressional colleagues and staffers.

The book, published by Rowman & Littlefield Publishers Inc., has a foreward by former Sen. Sam Nunn (D-Ga.), who praises Domenici for contributing to the public's understanding of the importance of nuclear power. "Citizens of the world will always link the safety of nuclear power with the prospect of terrorists acquiring nuclear materials and exploding a nuclear bomb or a dirty bomb," Nunn said. But he says Domenici emphasizes that nuclear energy can be produced safely, while maintaining tight security of the materials and managing the waste. Nunn called Domenici an influential lawmaker because of his grasp of subjects and persuasive arguments. "Pete is the kind of colleague who makes you want to go back and check your facts if he differs with you on an issue," he said. In his book, Domenici does little to shade or hide his views on everything from the nuclear fuel cycle to radiation standards. The senator criticizes the use of the linear non-threshold model for assessing radiation risks, makes the case that the U.S. should not try to permanently dispose of high-level waste, which he believes should be reprocessed, and scoffs at claims that nuclear plants aren't safe or are vulnerable to terrorist attacks.

He says he became convinced over the past three decades that nuclear power must have a prominent role in meeting future U.S. energy needs after examining the costs and benefits of coal, natural gas, solar, wind, hydro, and particularly oil. He also concludes it is the most efficient and environmentally sound baseload option.

"Our nation is poised to meet not only our needs for electricity with clean, low-cost generation, but we can reach out to other nations and help them," Domenici says.
—Jenny Weil, Washington

Inside NRC

Volume 26 / Number 20 / October 4, 2004

Divided commission to approve new 50.69 rule

An effort that began more than a decade ago under the banner of graded quality assurance is expected to reach a penultimate milestone Oct. 7 when the commissioners affirm their votes to approve a final, voluntary rule—10 CFR 50.69—that will risk-inform NRC's special treatment requirements on procuring components for nuclear plants.

Sources said the vote will be 2 to 1, with Commissioner Edward McGaffigan disagreeing with the decisions of Chairman Nils Diaz and Commissioner Jeffrey Merrifield to significantly modify the staff's proposal on the treatment of so-called risk-informed safety class 3 (RISC-3) structures, systems, and components (SSCs). By using a rigorous categorization process for plant SSCs, utilities hope they can save money by procuring commercial-grade RISC-3 components. Such components, while safety-related, are of low risk significance.

Diaz and Merrifield have apparently voted to streamline language on treatment that the industry found objectionable in the final version of the rule that the staff forwarded at the end of June. But it is unclear whether the commission also will change a key section on categorization that the industry also didn't like.

After the commission votes, the staff will have to conform the final rule package, including the preface, or statement of considerations, to the commission's directives, a process that could take several weeks to months.

Once the rule is published in the Federal Register, NRC will have to wait to see how many utilities move to adopt the new procedures. One barometer of the rule's usefulness could be whether STP Nuclear Operating Co., which asked for and received an exemption from NRC's current special treatment requirements, moves to adopt the rule or continues to use its current exemption.—***Michael Knapik, Washington***

Inside NRC
Volume 26 / Number 20 / October 4, 2004

Industry concerned that staff's SE challenges NEI sump methodology

Issues raised in a draft NRC safety evaluation (SE) could complicate resolution of PWR sump safety issues, an industry source said last week.

The Nuclear Energy Institute (NEI) has developed, and NRC staff is reviewing, a guidance document that provides a methodology for PWR operators to evaluate the ability of their containment sump systems to maintain core cooling despite debris accumulation after a serious reactor accident. In a generic letter dated Sept. 13, NRC requested that operators conduct such evaluations, using either the NEI or an alternative NRC-approved methodology, and upgrade their sumps if necessary (INRC, 20 Sept., 5).

NRC's draft SE issued Sept. 20, concluded that "portions of the proposed guidance were acceptable as is; and other portions were found to need additional justification and/or modification." In the SE, NRC staff "identified conditions, limitations, and required modifications, including alternative guidance to supplement those portions determined by the staff to need additional justification and/or modification in the NEI submittal."

If the issues raised in the SE are adequately addressed, "the resultant combination of the NEI submittal and staff safety evaluation provide an acceptable overall guidance methodology for the plant-specific evaluation" requested by the generic letter, NRC staff concluded in the SE.

The SE was reviewed by the thermal-hydraulic subcommittee of the Advisory Committee on Reactor Safeguards (ACRS) at public meetings Sept. 22-23. The subcommittee "felt that staff's technical justification wasn't up to par in some places," an industry source said last week. The subcommittee raised a number of technical issues, including the possible contribution of chemical effects in the post-accident containment environment to debris generation and accumulation, John Hannon, chief of the plant systems branch in the Office of Nuclear Reactor Regulation (NRR), told Inside NRC last week. Staff will respond to this and other issues raised by the subcommittee when the full ACRS reviews the

SE at its Oct. 7 meeting, Hannon said.

Use of NEI's guidance "does not exempt a plant from concerns not explicitly addressed" in the evaluation methodology, such as chemical effects, staff said in the SE. "Initially, licensees should evaluate whether the current chemical test parameters, which are available in the test plan for the joint NRC/Industry Integrated Chemical Effects Tests, are sufficiently bounding for their plant specific conditions," the SE said.

NRC's Office of Nuclear Regulatory Research is "beginning testing this month with contractors to get additional data that we are going to share with industry by the end of the year," David Solorio of NRR's plant systems branch told Inside NRC. These tests "could show there is no additional concern, or could show a little more debris buildup and head loss [a measure of diminished sump effectiveness]," Solorio said. Therefore, "staff is asking licensees to consider the impact of chemical effects" in their evaluations, Solorio said.

"If chemical effects are observed during these tests, then licensees should evaluate the sump screen head loss consequences of this effect," the SE said. "A licensee who chooses to modify their sump screen before tests are complete should consider potential chemical effects in order to avoid additional screen modification should deleterious chemical effects be observed during testing."

NEI developed "what we still believe to be a very conservative methodology that is going to bias [PWR operators] to modify their sump designs in some way," John Butler, NEI project manager for the sump evaluation guidance, told Inside NRC. "It seems NRC has tried to add additional conservatisms in a number of ways that are going to make this problem a lot more difficult to resolve." Citing one example, Butler said the SE challenged NEI's estimate of the amount of insulation likely to be turned into debris by pipe breaks in a loss-of-coolant accident. NRC's suggested modification to the methodology "significantly increases the zone of influence" —that is, the area of insulation damaged by water from a pipe break—"and amount of debris that would be postulated to occur," Butler said. "NRC's modification increases the zone of influence by a factor of three," he said.

In a letter dated Sept. 27, NRR's Hannon told NEI that industry "may provide clarification on any factual errors or technical misinterpretations contained in the SE provided they are received by October 1." NEI's reply to Hannon's letter was not available by press time. “The final SE will be issued after making any necessary changes and will be made publicly available,” Hannon said in his letter. NRR has previously stated that it plans to complete its review of NEI’s methodology by the end of October.

—Steven Dolley, Washington

Inside NRC
Volume 26 / Number 20 / October 4, 2004

ACRS recommends steam dryers be part of license renewal reviews

License renewal should be approved for Exelon's Dresden and Quad Cities plants but additional issues should be included in NRC's application review, including an evaluation of steam dryers for all future BWR license renewal applicants, the Advisory Committee on Reactor Safeguards told the agency last month.

NRC staff "should require that, prior to entering the period of extended operation, Exelon conduct an evaluation to ensure that operating experience at extended power uprate (EPU) levels is properly addressed by the aging management programs" at Dresden and Quad Cities, ACRS Chairman Mario Bonaca said in a Sept. 16 letter to NRC Chairman Nils Diaz. Also, "steam dryers should be included in the scope of license renewal" for the units, Bonaca said. Additionally, NRC staff should develop guidance to apply these recommendations to "future boiling water reactor license renewal applications," he said.

At a Sept. 9 ACRS meeting, Bonaca took issue with the position of NRC staff and Exelon Nuclear that steam dryers should be considered a "current operating issue" rather than a license renewal issue (INRC, 20 Sept., 6). Because pieces of a Quad Cities-2 steam dryer broke off and "went through safety-related components" after the dryer cracked in 2002, steam dryers should be considered "a component that could have an impact on safety-related components," and therefore in scope, Bonaca said at the meeting. NRC's safety evaluation report on the Dresden-Quad Cities application excluded steam dryers from the license renewal review. NRC approved extended power uprates of 17.8% for Dresden and Quad Cities in December 2001, but Quad Cities has been operating at its previous power rating since the 2002 steam dryer incident. Exelon plans to replace the Quad Cities steam dryers during a spring 2005 refueling outage, Bill Bohlke, Exelon Nuclear vice president, said at the Sept. 9 meeting.

NRC staff is preparing a response to the ACRS letter, said P.T. Kuo, NRC's program director for license renewal and environmental impacts. Kuo did not say whether NRC would accept the committee's recommendations. NRC will reply to ACRS "within the next few weeks, before the Dresden-Quad Cities decision," another NRC staffer told Inside NRC last week. NRC staff expects to issue a decision on the application by Nov. 1.

Kuo said at the Sept. 9 ACRS meeting that "if plans to maintain the integrity of the Dresden and Quad Cities steam dryers during extended power uprate conditions should be unsuccessful, the applicant has committed to include the dryers within the scope of license renewal."

Exelon Nuclear is "crafting an answer" to the ACRS request but hasn't reached a "final conclusion" on whether steam dryers should be considered in scope, said spokeswoman Ann Mary Carley. Regardless of the disposition of the scoping issue, steam dryers are already included in Exelon's "asset management plan," Carley said. Whether steam dryers should be included in license renewal aging management plans, which would entail "a more stringent review process with the NRC," is still under consideration by Exelon, she said. Either way, the steam dryer issue "won't affect the schedule to conclude license renewal review unless there's a showstopper, which we don't anticipate," Carley said.—***Steven Dolley, Washington***

Inside NRC

Volume 26 / Number 19 /September 20, 2004

NEI task force to come up with SDP recommendations

The Nuclear Energy Institute (NEI) is pulling together a task force to come up with recommendations on improving NRC's significance determination process (SDP) for assigning color codes to inspection findings. NRC and the industry agree that disputes over whether an inspection finding is green—of very low risk significance—or white, yellow, or red—with those colors indicating progressively higher risk significance—are consuming too much time and resources.

In particular, licensees are sensitive to a finding being classified as white—low to moderate safety significance—rather than green, an NEI representative said, given the potential impact on bond ratings and stock prices of getting enough white findings to push a nuclear unit into one of the escalated-inspection columns of NRC's action matrix. As an example of the resources that are going into resolving the limited number of "greater-than-green" preliminary findings, eight NRC officials, including Region I Administrator Sam Collins, will meet Sept. 27 with nine Exelon representatives, including the vice president of the company's mid-Atlantic regional operating group. They are slated to discuss an NRC inspection finding that involved a failure of personnel at Oyster Creek to follow procedures while replacing the cooling fan drive belts on an emergency diesel generator cooling system during an overhaul in April. Oyster Creek is already an NRC Column-2 plant—meaning it is in the regulatory response column of the action matrix of NRC's reactor oversight process—because of a white inspection finding in the mitigating systems cornerstone. A second white in this cornerstone would push Oyster Creek into Column 3, or the degraded cornerstone column, meaning there would be more inspections.

Also on Sept. 27, representatives from NRC's Region IV and NRC headquarters will meet with representatives from Nebraska Public Power District to discuss a greater-thangreen preliminary finding involving the utility's failure to restore Cooper's service water system to its normal configuration following maintenance.

The intense resources that are often involved in resolving color findings led the NEI's Nuclear Strategic Issues Advisory Committee, which is composed of utility chief nuclear officers, last month to bless the formation of the task force. The NEI representative said he hopes the task group comes up with recommendations by the end of October so that they can be vetted through a senior NEI working group and forwarded to NRC.

NEI representatives for some time have been suggesting that when there is a dispute over the color of an inspection finding, NRC should just go ahead and conduct a further inspection to see the extent of the condition and what corrective actions the utility has taken. If the extent of the condition is limited and if the utility has taken effective corrective action, then the finding should be listed as a green finding, NEI representatives have said. "Resources should be spent on fixing the problem rather than on arguing over the color of the finding," one NEI source said.

At NRC's Regulatory Information Conference in March, NRC's Stuart Richards said that the SDP timeliness for greater-than-green issues "must improve," and he acknowledged that some SDPs are "very resource intensive."
—Michael Knapik, Washington

Rutland Herald

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Article published Sep 18, 2004

State seeks new opinion on Yankee

The Douglas administration has asked a national nuclear advisory panel to review what the state considers the most controversial — and potentially dangerous — aspect of a plan to increase power production at Vermont Yankee.

In a letter to the Advisory Committee on Reactor Safeguards, state Public Service Commissioner David O'Brien asked the committee to review Entergy's plans, specifically the plan to maintain pressure in the reactor's containment during an emergency.

"The department is questioning whether Entergy should be allowed to count on a certain amount of pressure in the reactor to allow emergency core cooling pumps to run, in the event of an accident," the Public Service Department said in a statement Friday afternoon.

O'Brien noted that the state had already lodged a challenge on the same issue when it requested a formal hearing from the Nuclear Regulatory Commission on the containment-pressure issue three weeks ago. Vermont's congressional delegation has urged the NRC to hold such a hearing.

The NRC, which has never held such a hearing on power increase, is still considering the request, according to a spokesman. O'Brien said that the state's nuclear engineer, William Sherman, had raised the same concerns last year. The NRC's response, which came six months later, failed to answer the state's concerns, he said.
"We're asking for an independent body of experts to look at this issue," O'Brien said Friday. "We want to highlight the overpressure issue." He said the advisory committee was an independent body from the NRC staff.
"I think they have some genuine influence," he said.

NRC spokesman Neil Sheehan said the advisory committee was already in line to review the uprate case as part of the "checks and balances" inherent in the NRC's uprate review process. Sheehan, who hadn't read Vermont's request, said that if the state's request for a hearing is granted, the hearing will be held before the Atomic Safety and Licensing Board, a division of the NRC.

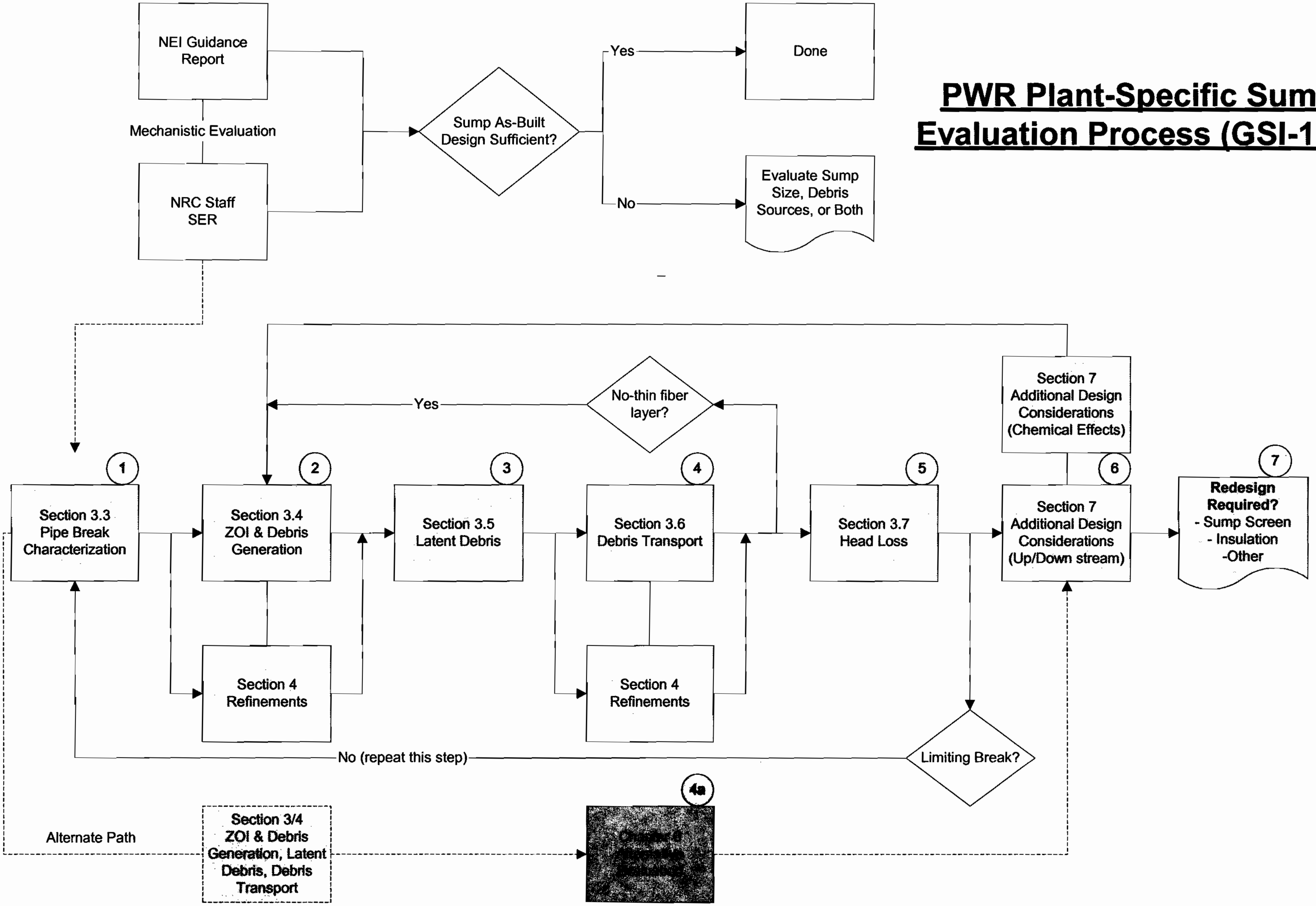
Raymond Shadis, senior technical advisor for the New England Coalition, a nuclear watchdog group, said the Advisory Committee on Reactor Safeguards currently reviews all power uprates, and has never turned one down. The New England Coalition has also asked for a formal hearing, but on different grounds.
"For the time being, they review all extended power uprates," Shadis said. "Back in 1999, they expressed grave reservations of granting uprates of more than 8 percent."
But since then, the committee has been involved in reviewing all the uprates and have given its approval on each one, including those that have included similar containment-pressure plans as Vermont Yankee.

Shadis said to call the group independent was misleading. The committee, made up of experts from around the country, relies on NRC staff for research." To say they are independent is a great deal of hokum," he said.

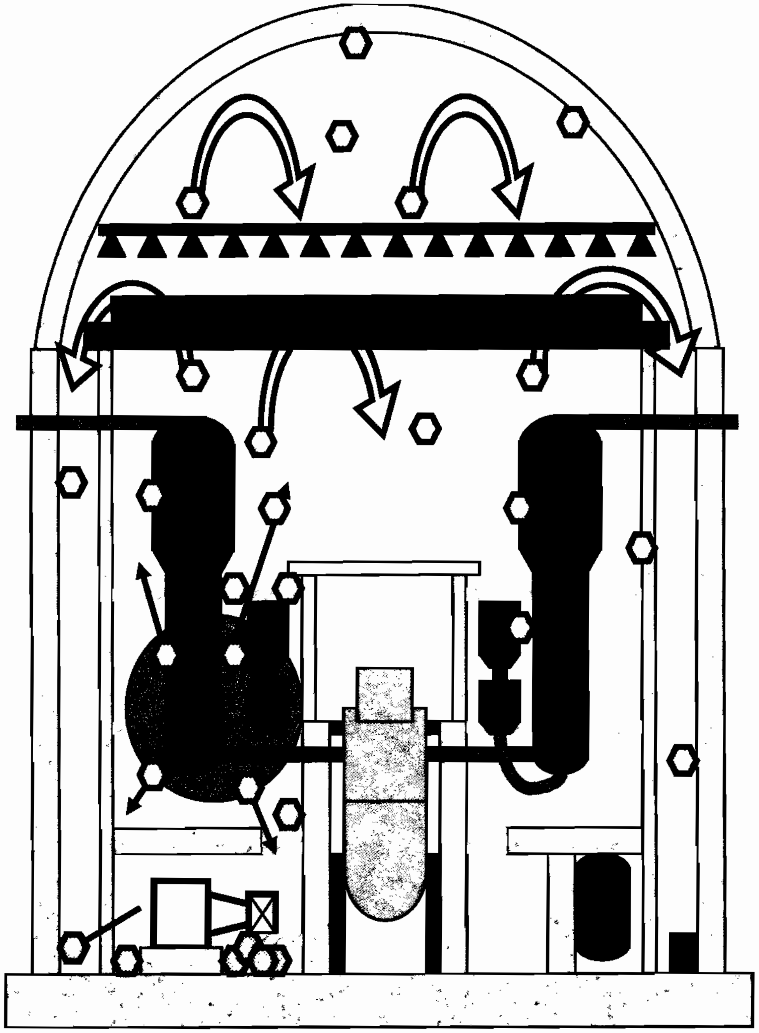
Entergy spokesman Robert Williams said the company has full confidence in its application to the NRC to generate an additional 20 percent, or 100 megawatts, of power. " Our filing for an uprate is well grounded in NRC regulations, that have allowed containment-pressure credits at 25 other plants,"
Williams said.

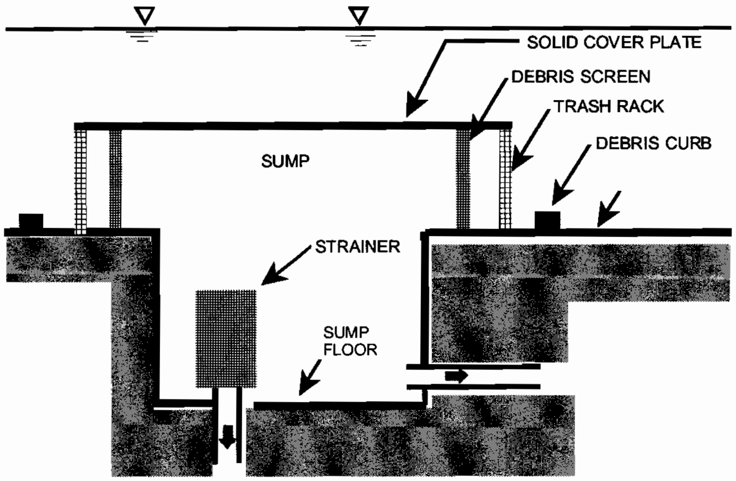
"The ACRS was set up by Congress to give an independent view on safety matters and we welcome the oversight," he said.

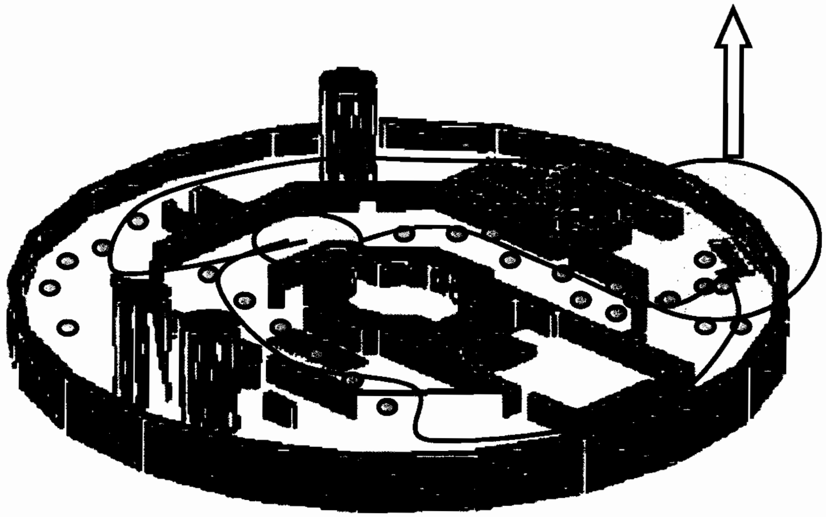
PWR Plant-Specific Sump Evaluation Process (GSI-191)



Accident Progression







Large Break LOCA Progression

NUREG/CR 6770 Table 10: CL DEGB - Large Dry Containment

(MELCOR)

Parameter	Blowdown Phase			Injection Phase			Recirculation Phase		
	0+	**20 s**	**45 s**	**45 s**	**15 min**	**27 min**	**27 min**	**2 h**	**24 h**
RCS pressure at break (psia)	2250	393	55						
RCS temperature at break (°F)	531	291	250	250	173	144	144		
Break flow velocity (ft/s)	296	930	100						
Break flow quality	0	0.25	0.3	0.3	0				
Safety injection (gpm)				11500	11500	11500			
Recirculation flow (gpm)							17500	11800	11800
Spray flow (gpm)				0	5700	5700	5700		
Containment pressure (psig)	0	36	33	33	11.5	7	7	1.5	0
Containment temperature (°F)	110	305	250	250	190	163	163	115	95
Pool depth (ft)					2	3.5	3.5	3.5	3.5
Pool temperature (°F)					212	187	187	125	100

LOCA Debris Estimates

Demonstration Calculations				
For a W-4 Loop with Large Dry Containment - assume 10,000 ft3 of fibrous insulation, latent fiber approx 20 ft3		10,000		
Assume each SG, RV, PZR approx 1300 ft3 (6 big items)	1300	7800		
Remaining miscellaneous insulation		2200		
ZOI fraction of SG	0.9	1170		
ZOI fraction of miscellaneous insulation (compartment)	0.25	550		
Total estimate of debris		1720		
Transport Phase (approximate percentages/values)	Small Fines		Large Pieces	
	factor	ft3	factor	ft3
Debris Profile Fraction	55%	912	45%	808
Fraction transported to Upper Levels by Blowdown	90%	839	65%	509
Fraction transported directly to Pool	10%	73	35%	299
Fraction Washed Down into Pool	70%	595	20%	107
Fraction transported to Inactive Pools	5%	27	5%	57
Fraction in pool transported to Sump Screens	100%	625	75%	264
Fraction of Debris Generated That Accumulates on Sump Screens	70%	629	35%	267
With a 100 ft2 screen, small fines only, yields an approx debris depth of	6	ft		
A debris bed of 6 ft, with a particulate load of 300 #, would yield an estimated head loss of 10-17 ft				
All RMI/Latent Fiber Only				
For Latent Fiber only - 20 ft3 (all small fines, overall transport 70%)	14	ft3 bed		
	0.14	ft = 1.7 inches thick		
Back calculate a 1/8 (0.125) inch bed, results in latent debris volume of	1.04	ft3 bed on 100 ft2 screen		
	1.5	ft3 of latent fiber		
Practical Solutions: double jacket fiber insulation, modify sump screen, refine ZOI model, trash racks/barriers, operator actions, revised setpoints, change insulation types, etc…				

Safety Evaluation Report, GSI-191
PWR ECCS Sump Performance

UNITED STATES NUCLEAR REGULATORY COMMISSION

Presenters
Mark Kowal
Ralph Architzel
Hanry Wagage
Shanlai Lu
Steve Unikewicz

ACRS Full Committee Briefing
Rockville, MD
October 7, 2004

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SER Sections 3.3 and 4.2.1 – Break Selection

- **Guidance Report**
  - Considerations for selecting the limiting break location
  - Limiting break location criterion – Head loss across the sump screen
    - Maximum amount of debris transported
    - Worst combination of debris mixes transported
  - Break size and piping system considerations
  - Consider all phases of the accident scenario
  - Refinement - Application of SRP 3.6.2/BTP MEB 3-1

UNITED STATES NUCLEAR REGULATORY COMMISSION

SER Sections 3.3 and 4.2.1 – Break Selection

- **SER**
  - GR acceptable with two exceptions:
    - No guidance for plants that can substantiate no thin fiber layer (no thin bed effect)
    - Secondary side break locations - staff position is analyses consistent with LOCA piping
  - Staff concludes that it is not appropriate to cite SRP 3.6.2 and BTP MEB 3-1 as methodology for determining break locations for PWR sump analyses
- **ACRS Questions**
    - Added Appendix VIII describing thin bed effect and Calcium Silicate behavior

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SER Section 3.4 Debris Generation

- **Guidance Report**
  - NEI Zone of Influence using ANSI/ANS 58.2-1988 free-jet model, resizing to sphere
  - ZOI Refinements
    Direct Impingement using modeling two freely-expanding jets
    Use of Debris specific destruction zones
    Simplification to a Compartment
  - Debris characteristics provided for transport and head loss input:
    - Destruction pressure, density, size, and distribution

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SER Section 3.4 Debris Generation

- **SER**
  - GR Approach is acceptable with modifications identified by the staff
  - Destruction pressures based on air jet testing should be reduced by 40% to account for two-phase effects; truncation allowed at robust barriers limits impact
  - Two categories of debris:
    - Coatings - Lack of data leads to staff positions for (1) use of data from experimentation to justify values used, or (2) use of conservative alternative guidance
    - All other debris types - Debris-specific data and default values recommended in the baseline and refinements, are generally acceptable

- **ACRS Questions**
  - Destruction pressure definition – Appendix I figures and revisions
  - Paint chip size for no thin bed analyses

SER Section 3.6 – Debris Transport

- **Guidance report**
  - Based on NUREG/CR-6762 logic tree
  - Conservative mass of debris on sump screen
  - Transport only the small fines: blowdown, washdown, pool fill, and pool recirculation
  - Conservatively quantify the logic tree
  - Analytical refinements (Section 4.2.4): nodal network and CFD

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SER Section 3.6 – Debris Transport

- **SER**
  - Staff accepts GR
  - Supplemental guidance: blowdown and washdown (App. VI), pool transport using CFD (App. III), debris transport comparison (App. IV)
  - Limitations: relocation into inactive pools, large debris transport, and uniform debris distribution on the pool floor
- **ACRS questions**
  - Debris moving into upper containment

SER Section 3.7 – Head Loss

ACRS Questions

1. NUREG/CR-6224 testing data range covers temperature 60 - 125 °F. Can the industry use it beyond 125 °F?
2. No concise description of “Thin Bed” effect.

Staff Response: Temp range has been extended to 220+ °F.

Basis: Staff analysis indicates that the most limiting physical phenomenon is the air bubble formation through the bed due to the depressurization.

The air void fraction depends on water temperature, head loss and containment pressure. The criteria is that void fraction <3%.

SER Section 3.7 – Head Loss

Thin Bed Effect And Its Impact

Definition:

A relative thin layer of fibrous bed with particulate causes a high head loss due to the bed porosity approaching the corresponding particulate sludge limit.

Plant application:

A small amount of fiber can challenge the NPSH margin.

SER requirement:

Both the actual bed thickness and a thin bed need to be evaluated for a given screen design and debris types.

UNITED STATES NUCLEAR REGULATORY COMMISSION

SER Section 7.3 – Downstream Effects

- **Guidance Report**
  - Blockage of flow paths
  - Wear and abrasion of surfaces
  - Blockage of flow clearances through fuel assemblies

- **SER**
  - Licensees to determine downstream source term based on Sections 3.3 to 3.6 calculations

SER Section 7.3 – Downstream Effects

- **SER**
  - Licensees to consider conditions of operation, mission times, wear/abrasion, blockage mechanisms, engineering evaluation of ECCS and CS
  - Licensees to determine downstream source term based on Sections 3.3 to 3.6 calculations
  - Licensees to consider conditions of operation, mission times, wear/abrasion, blockage mechanisms, engineering evaluation of ECCS and CS

SER Section 6 - Alternate Evaluation

- **Guidance Report**
  - Realistic and risk-informed elements
  - Comparable to the ongoing 10 CFR 50.46 risk-informed rulemaking effort
  - Define a “debris generation” LOCA break size
    - All auxiliary piping attached to the RCS
    - Break size equivalent to the area of a double ended rupture of a 14 inch diameter pipe (approximately 197 square inches)
  - Region I analyses - RCS breaks up through and including the “debris generation” break size (customary design basis analyses)
  - Region II analyses - RCS break sizes larger than the “debris generation” break size

SER Section 6 - Alternate Evaluation

- Region II analyses (Continued)
  - More realistic analyses and assumptions
  - Safety-related and single failure-proof considerations
  - May require plant-specific exemptions or license amendments
  - Acceptance criteria – NPSH margin to demonstrate adequate core cooling flow and containment cooling
- Risk-informed aspects
  - Associated plant modifications and operator actions
  - Analyses performed consistent with RG 1.174

- **SER**
  - Alternate evaluation approach is acceptable
  - SECY-04-150 - informed the Commission

- **ACRS Questions**
  - Region II acceptance criteria
  - Overall risk reduction

UNITED STATES NUCLEAR REGULATORY COMMISSION

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Date: 10/5

FAX MESSAGE

TO: Dr. Ralph Caruso & Dr. Graham Wallis

FAX: (301) 415-5589

(Transmitting 10 pages including cover sheet)

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Please find attached Prof. Banerjee's Review of Head Loss prediction

REVIEW OF HEAD LOSS PREDICTION ACROSS SUMP SCREENS

1. Introduction

A review has been conducted regarding the basis for the sump-screen head-loss correlation in Appendix B of NUREG/CR 6224, related papers (in particular Kyan et al. (1970), MacDonald et al. (1979), and Ingmanson et al. (1959)), more recent literature, such as the review by Lee and Wang (2000), a Los Alamos Report LA-UR-04-1227 which presents head loss data across debris beds which may form during PWR/LOCA (including the effects of calcium silicate insulation), two memoranda from Dr. Graham Wallis titled, "The NUREG/CR 6224 Head Loss Correlation [9/03/2004]" and "Flow Through a Compressible Porous Mat—Analysis of the Data Presented in Series 6 Tests reported by LANL in LA-UR-04-1227 [9/3/2004]". Some responses to Dr. Wallis's memoranda made by LANL have also been reviewed. Finally, relevant sections in the Draft SER on proposed document number NEI 04-07 was reviewed.

More specifically, the focus of this review was to determine

a) Whether the recommendation in the Draft SER, particularly with regard to "thin-bed" and "mixed-bed" configurations containing calcium silicate presented on page 63, Table 3-5 were defensible.

b) Whether the correlation presented as Eq. B-21 in NUREG/CR 6224, coupled with the bed compression expressions given in B-22 to B-24, can be used to provide "bounding" head-loss predictions for flow through debris beds that accumulate on sump screens during PWR/LOCA.

2. Comments

The main comments based on this review are summarized below, and the basis for each discussed in the following sections.

a) There appears to be insufficient basis for the SER section titled, "Staff Evaluation for Section 3.7.2.3.1.4", pages 62-64, including Table 3-5. The values presented in Table 3-5 are based on what appears to be incomplete data in LA-UR-04-1227—in particular the data and discussion in the LANL report commencing on page 5, Section 5.5, "Head Loss Results and Analysis". As discussed in Section 3 of this review, the data presented in Figures 5.13 (for Test 6H) and Figure 5.14 (Test 6G) of the LANL report, are inconclusive with regard to, and may severely underestimate, the parameter values presented in Table 3-5 of the SER.

b) On the broader subject of the "NUREG/CR 6224 correlation" and associated calculational procedures, there is reason to believe that both its form (equation B-21 in the NUREG) and the associated calculation of actual packing density (equation B-24) are not defensible. The first point is discussed in Section 4 and the second in Section 5.

The literature suggests (see the extensive review by MacDonald et al. (1979) and the paper by Ingmanson et al. (1959), which are cited in NUREG/CR 6224) that an Ergun equation with modified coefficients correlates a wide range of data, and there appears to be no reason why it, together with the (correct) procedure to estimate bed compressibility in Ingmanson et al. (1959), should not be used for the problem at hand.

3. The LANL Data

The recommended values in the Draft SER Table 3-5 come from Section 5.5 of LA-UR-04-1227. To fit the so-called NUREG correlation (discussed in the next section) to the data, the tunable parameters, specific surface area and sludge density are adjusted. It is notable that these parameters then vary significantly between tests. In particular, as the head loss is approximately proportional to the square of the specific surface area, there is a significant effect of tuning this. (Note also that about a 20% increase in sludge density decreases the specific surface area by about 10% in the tuning process).

Be that as it may, the recommended values come from these tunable parameters being adjusted to fit Tests 6G and 6H. These are taken from the LANL report and are shown in Figures 1 and 2 here. Note that the highest point in Test 6G to which the fit was made may or may not be the highest pressure loss. (The x-axis in Figure 1 appears to be mislabeled and should probably be approach velocity). The trend of the increasing velocity lines can be continued to indicate that the pressure loss could perhaps have been higher (double or more). If the velocity had then been decreased, perhaps it would have followed a line from the "NUREG correlation" that was higher—and was based on a higher specific surface area.

To be conclusive, 6G and 6H would have had to be done like Test 6C shown in Figure 3. Here the approach velocity was increased until several points with <u>increasing</u> velocity could be fitted to the "NUREG correlation". Until such a procedure is followed for conditions corresponding to 6G and 6H, the experimental data should not be used to recommend values for the specific surface area and the sludge density even for the questionable NUREG correlation.

4. The NUREG/CR 6224 Correlation

The NUREG/CR 6224 correlation, which for some reason is called "semi-theoretical" in LA-UR-04-1227 (perhaps to give it more legitimacy than it deserves), seems simply to be a sum of what appears to be an early empirical fit to head losses in fibrous beds proposed by Ingmanson et al. (1959) for the viscous regime, and to an incorrectly formulated term which purports to come from the paper by Kyan et al. for the inertial effects. The term is similar to that in the Ergun equation but is incorrectly divided by ε rather than ε^3. As noted in the extensive review by MacDonald et al., which is also cited in NUREG/CR 6224, Kyan et al.'s data for fibrous beds is primarily in the flow regime region where viscous resistance predominates. Therefore, the value of the constant that is used for the inertial term is of little significance [see discussion in MacDonald et al., p. 203, B.2, Kyan Data].

At this point it is worth considering whether lashing together yet another correlation, such as the one in NUREG/CR 6224, is justified in view of the fact that MacDonald et al. note that an Ergun equation with slightly modified coefficients does as well as anything else. In particular, they state that use of porosity numbers, etc., seems an unnecessary complication. Be that as it may, the porosity dependence in the inertial term in NUREG/CR 6224 is in any case incorrect and should be divided by ε^3.

To proceed, let us consider the form of the NUREG/CR 6224 correlation given below

$$\frac{\Delta P}{L_o} = \left[A\,S_v\,a(\varepsilon,S_v)\mu U + Bb(\varepsilon,S_v)\rho U^2\right]\left[L_m / L_o\right] \qquad (1)$$

where L_c, L_m are the uncompressed and compressed bed thicknesses, respectively, U is the superficial velocity, ε is the porosity and S_v is the particle surface area for unit volume. The forms proposed for a and b are:

$$a(\varepsilon,S_v)=S_v^2(1-\varepsilon)^{1.5}\left[1+57(1-\varepsilon)^3\right] \tag{2a}$$

$$b(\varepsilon,S_v)=S_v(1-\varepsilon)/\varepsilon \tag{2b}$$

Also, A and B are constants set at 3.5 and 0.66, respectively. The form is essentially a restricted Ergun equation, with A and B taking slightly different values, and a(ε , S_v) of a slightly different form. As noted previously, the expression for b(ε, S_v) is wrong in (2b). Note that in this restrictive Ergun form $\sqrt{a}(\varepsilon)/b(\varepsilon)$ in this expression is a unique function of ε, as the interfacial area per the unit volume, or particle diameter, cancels out. The NUREG expression, therefore, implicitly assumes that such a unique relationship exists between the coefficients of the two terms and also of each term individually. Is this assumption likely to be correct? Consider the data presented in Tables I to IV in MacDonald et al.'s paper. Here they fitted the best values for the constants in the Ergun equation for each data set and found that they vary by about a factor of 3 depending on the structure of the packed beds.

It is clear that there is no unique relationship between a and b in terms of just the porosity. The relationship will vary depending on the geometry of the interstitial spaces.

At best, one can hope for such a relationship to hold for a type of bed, such as a fibrous mat, and a different expression may hold for particulate beds, etc. In essence, the form of the dependence on porosity varies considerably depending on the type of bed. The NUREG correlation adds nothing and has an incorrect inertial term.

As an aside, it is somewhat surprising that LANL calls this expression "semi-theoretical" in view of the extensive work done in that laboratory on flow in porous media, particularly in regard to oil reservoirs, where extensive use of Lattice-Boltzmann calculations have been done using scanned micrographs of cored samples as input to the calculations.

5. The Bed Compression Relationship

The issue is in regard to Eq. B-24 in NUREG/CR 6224 that predicts bed compression to be proportional to some power of the pressure gradient rather than the pressure itself. This suggests that, for a given pressure loss across the bed, as the thickness increases, the compression will decrease. As Dr. Wallis correctly points out in his memoranda, the stress balance is usually written in terms of solid pressure in the filtration literature (see for example the extensive review by Lee and Wang, Eqs. 30-33). This results in a porosity related to the applied pressure difference, rather than the pressure gradient, as given by Lee and Wang in Eq. 37 of their review. In NUREG/CR 6224 it is stated that the relationship comes from Ingmanson, et al. (1959). However, on reviewing Ingmanson et al.'s paper, it is clear that this is false. Ingmanson et al. (1959) proposed that the actual bed density C be given by

$$C = M\, p_s^N \qquad (3)$$

(Their equation (4)) where p_s is the compacting pressure at a point in the bed given by

$$p_s = P_t - P \tag{4}$$

where P_t is the force per unit area acting on the total upstream (exposed) face of the bed (or mat) and P is the hydrostatic liquid pressure at the point in the bed where p_s is calculated.

In Table I of Ingmanson et al., values of M and N are presented for several materials, including wood fibers. Therefore, the expression in NUREG/CR 6224 does not appear to come from Ingmanson et al. The expression used by Dr. Wallis in his review is therefore consistent with the published literature and the NUREG is not.

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Ingmanson, W.L., Andrews, B.D., Johnson, R.C., 1959, "Internal Pressure Distributions in Compressible Mats Under Fluid Stress", *PAPPI Journal*, 42, 840.

Kyan, C.B., Wasan, D.T., Kintner, R.C., 1970, "Flow of Single-Phase Fluids Through Fibrous Beds", *Ind. Eng. Chem. Fundam.* 9, 596.

Lee, D.J., Wang, C.H., 2000, "Theories of Cake Filtration and Consolidation and Implications to Sludge De-Watering", *Water Research*, 34, 1.

MacDonald, I.F., El-Sayed, M.S., Mow, K., and Dullien, F.A.L., 1979, "Flow Through Porous Media: The Ergun Equation Revisited", *Ind. Eng. Chem. Fundam.*, 18, 199.

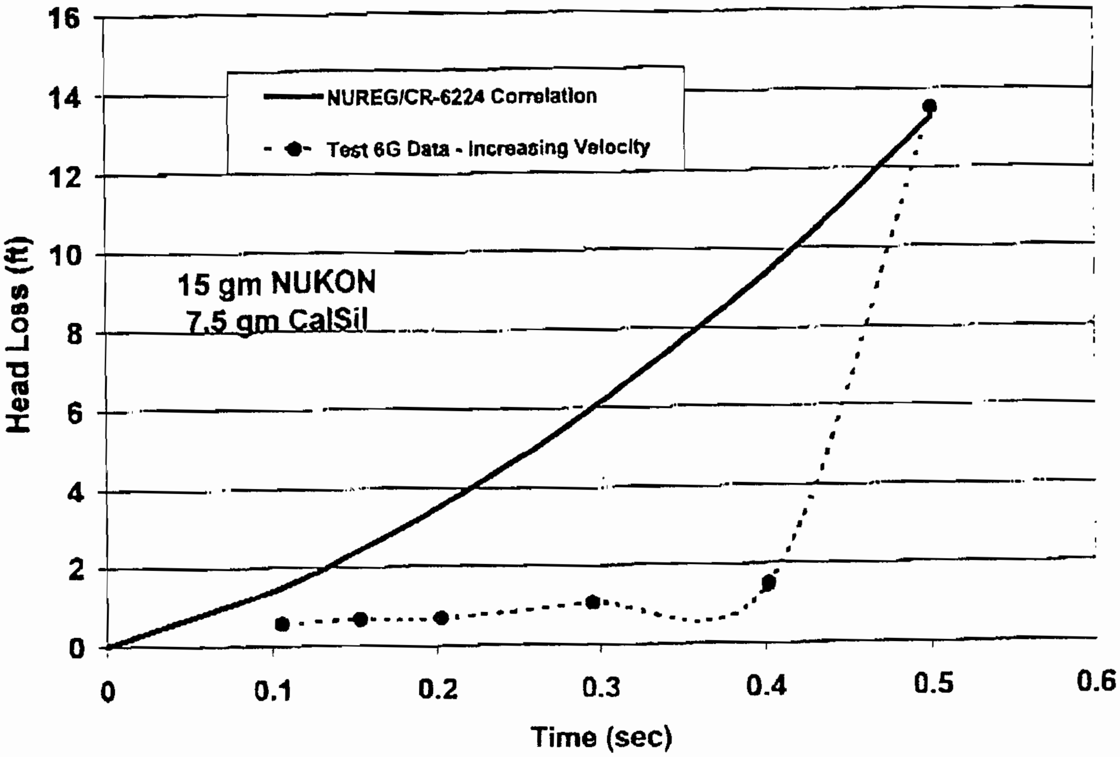

Figure 1. . Comparison of NUREG/CR-6224 correlation with Test 6G head-loss data.

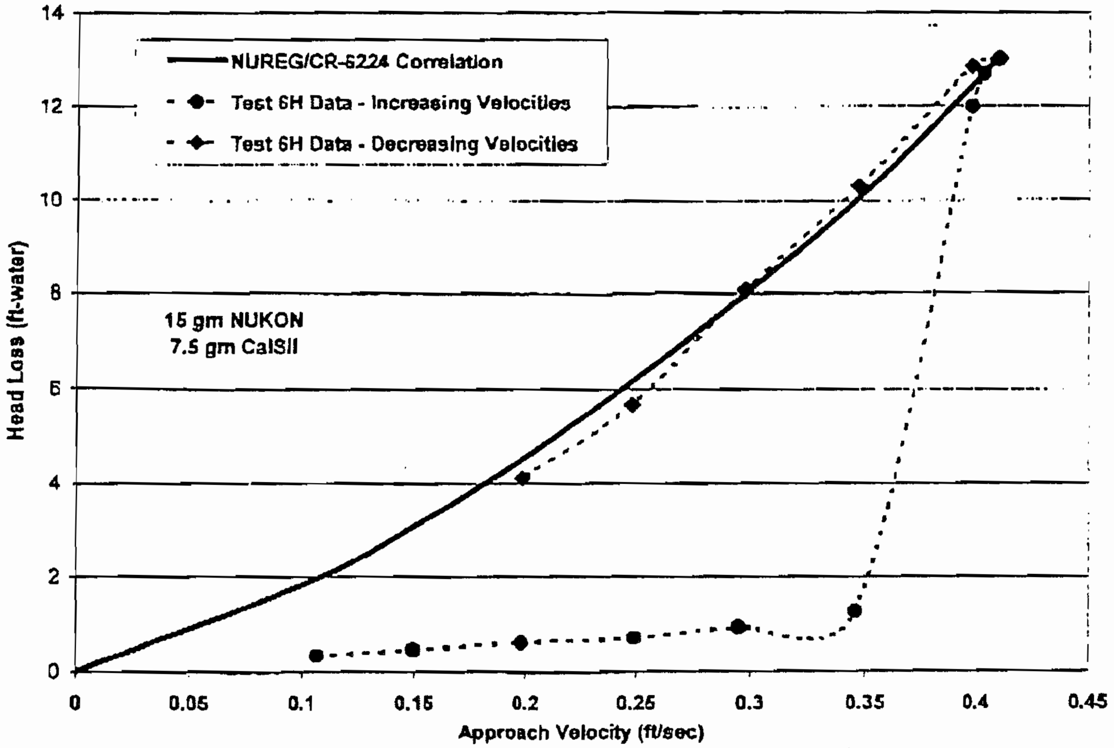

Figure 2. Comparison of NUREG/CR-6224 correlation with Test 6H head-loss data.

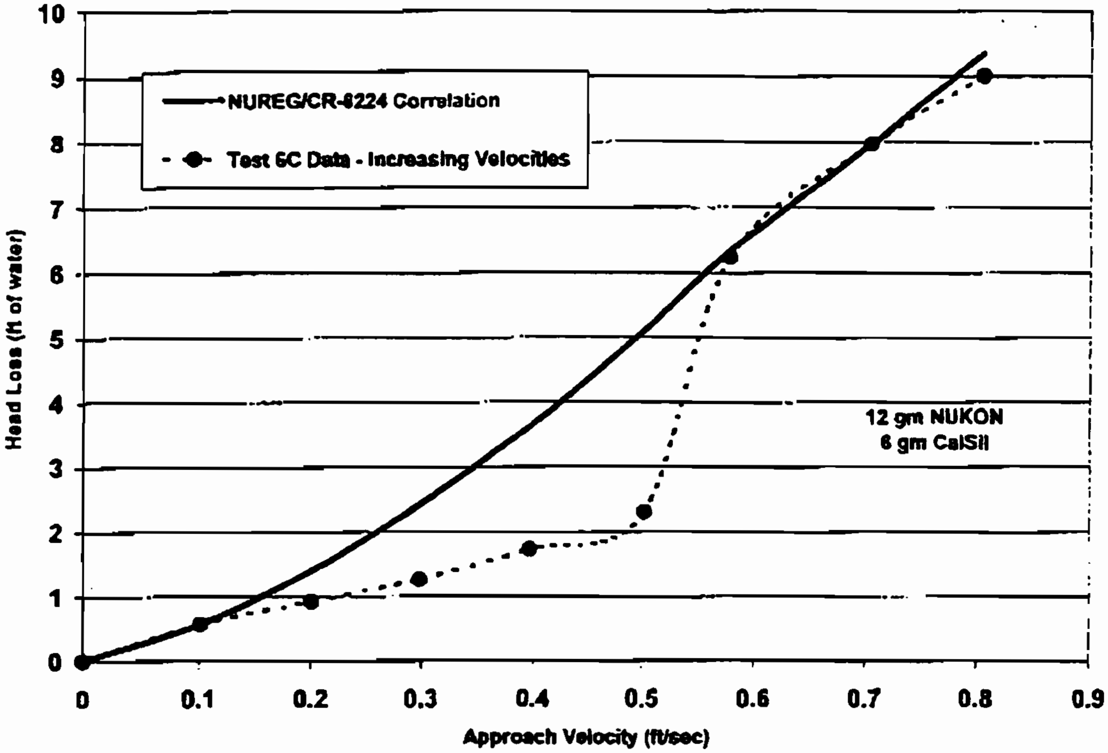

Figure 3. Comparison of NUREG/CR-6224 correlation with Test 6C head-loss data.

(7)

NUCLEAR ENERGY INSTITUTE

Anthony R. Pietrangelo
SENIOR DIRECTOR, RISK REGULATION
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October 1, 2004

Mr. John N. Hannon
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Office of Nuclear Reactor Regulation
Mail Stop O11-A11
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: Safety Evaluation for NEI Guidance Report "Pressurized Water Reactor Containment Sump Evaluation Methodology"

PROJECT NUMBER: 689

Dear Mr. Hannon:

This letter responds to your September 27 letter that provided notification of an opportunity to clarify any factual errors or technical misinterpretations contained in the subject safety evaluation (SE). The enclosure provides a set of detailed comments on the SE.

The industry evaluation guidance was developed to provide a practical and realistically conservative method for resolution of PWR sump performance issues. We believe this objective was accomplished with the document submitted to NRC on May 28, 2004 and supplemented on July 13, 2004.

The industry guidance addresses debris generation, debris size distribution, transport and headloss in a comprehensive fashion that focuses attention and necessary conservatism on the risk-significant event scenarios and phenomena. The SE modifications significantly increase the conservatism of individual aspects of the methodology for all event scenarios without regard to the risk-significance of the affected scenarios and with no apparent recognition of the overall conservatism of the final result. For example, the SE:

- increases the size of the zone affected by break jet impingement by a factor of three for all insulation materials and
- increases the affected zone for qualified coatings by three orders of magnitude.

These changes significantly increase the level of conservatism for the least risk-significant spectrum of breaks.

The July 13, 2004 supplement to Section 6 of the industry guidance provided an alternative evaluation method that addresses the low-risk spectrum of breaks. The main attribute of this alternative method is the selection of a "debris-generation" break size, above which more realistic inputs, methods and criteria may be used. The SE places limitations and restrictions on the use of the alternative evaluation method that significantly limit its value and use. For example, the guidance allows use of nominal parameters in the calculation of NPSH for the low-risk spectrum of breaks. However, the SE removes any allowance for exceeding the "nominal" parameters during normal operation, thereby necessitating the use of bounding values to avoid numerous operability evaluations.

PWR plants have been proactively working to address GSI-191 in advance of the issuance of the final SE. A number of companies with specialized knowledge on containment sump performance have worked with NEI on the development of the evaluation guidance and have already utilized the guidance for several PWRs. We believe it would be instructive for NRC reviewers to become more familiar with the application of industry evaluation guidance. We encourage, and can help facilitate, meetings between NRC and GSI-191 service vendors so that a greater appreciation of the overall conservatism of the evaluation guidance, methods and results can be obtained. This would also serve to provide an engineering perspective on cumulative results of the evaluation. These meetings can be arranged and conducted quickly, with minimal impact on resolution schedules.

An acknowledged and complicating factor in both NRC and industry efforts to resolve GSI-191 is the continuing need to address new concerns and phenomena. Testing to investigate the potential for adverse chemical effects in the containment following a LOCA will begin in the next few weeks with initial results becoming available after the planned issuance of the final SE and after initiation of efforts by industry to implement resolution guidance. While every effort is being made to address chemical effect concerns in a timely manner, the potential for resolution schedules to be impacted must be acknowledged. The NRC recommendation for PWR plants to incorporate margin in their designs in advance of the test results is not supportable in that there is no basis against which to assess the validity or appropriateness of chosen margin factors.

Industry activities and schedule to address the impact of debris passage on systems and equipment downstream of the containment sumps are complicated by ongoing and planned NRC tests. The schedule and scope of these tests is unknown and introduce a significant uncertainty in efforts to address downstream effects. While some downstream effects (e.g., component clearances) can have an impact on containment sump screen design (primarily screen mesh size), these effects are known and are addressed in the industry guidance. Other potential effects,

primarily wear and abrasion concerns, are relatively independent of containment sump screen design in that the screen design acts as a boundary condition for the subsequent treatment and resolution. Given the potential for planned testing to impact the manner in which these downstream effects are addressed, we urge NRC to consider a resolution schedule that is separate from upstream effects and which accommodates the NRC test schedule.

The brief time allotted for review of the DSER has been insufficient to allow the level of review appropriate for this important document. We intend to continue our review, focusing on identification of areas requiring additional clarification, in order to provide the instruction and guidance necessary for proper implementation by PWRs.

Please contact John Butler 202-739-8108, jcb@nei.org, or me if you have any questions on this transmittal or if we can assist in arranging the recommended vendor meetings.

Sincerely,

Anthony R. Pietrangelo

Anthony R. Pietrangelo

c: Mr. Michael Johnson, U.S. Nuclear Regulatory Commission
Mr. David L. Solorio, U. S. Nuclear Regulatory Commission

October 1, 2004

Comments on 9/20/04 Draft Safety Evaluation Report on Guidance Document "Pressurized Water Reactor Sump Performance Evaluation Methodology"

General Comment:
It would be helpful to summarize guidance exceptions in a stand alone section. At present the guidance exceptions are distributed throughout the document including: the executive summary, sections entitled "Staff Evaluation for Section XXX", sections entitled "Staff Conclusions Regarding Section YYY", Table 3-6 entitled "Non-Conservative Assumptions in the Baseline Evaluation Methodology", the chapter entitled "Conditions and Limitation", and the chapter entitled "Conclusion".

Page vi, fourth paragraph
The DSER states:

> **For plants needing to evaluate secondary-side piping such as main steam and feedwater pipe breaks, break locations should be postulated in a manner consistent with the guidance in Section 3.3 of this SER**

This should be modified to state that secondary side break locations should be postulated in a manner consistent with the plant licensing basis. Section 3.3 of the DSER (specifically section 3.3.4) excludes the use of BTP MEB 3-1 on the basis that its application for debris generation purposes does not meet the intent of 10 CFR 50.46. Irrespective of the validity of this basis, it is not applicable to secondary side breaks.

Page ix, ES.4 second paragraph
Guidance should be provided that one alternative for proving that large pieces do not transport is through verification with a detailed CFD model. Using pool velocity data from only one plant, as referenced in the DSER, can be misleading.

Page x, ES.6 Analytical refinements
Computational fluid dynamics (CFD) models were extensively used and were an accepted basis in BWR resolutions in regards to debris transport. The DSER conclusions regarding application of CFD methods are unclear. The Executive Summary states:

> **For debris transport, two methods for computing flow velocities in a sump pool – i.e. network method and the computational fluid dynamics methods – are provided -----. However, the staff finds the guidance offered in either option to be insufficient to provide an acceptable alternative to the baseline approach.**

Page 96, Staff Conclusions Regarding Section 4.2.4 states:

> **Consistent with Regulatory Guide 1.82, Revision 3, the staff accepts (1) the CFD method and (2) the nodal network method as an alternative method to**

> **calculate debris transport onto the sump screens. However, the licensees using the nodal network method should support it using experimental data to ensure that the debris transport estimates are conservative with respect to the quantities and types of debris transported to the sump screen. The GR recommended debris transport model in Section 4.2.4 that assumes using a uniform distribution of debris across the sump floor is not acceptable because the debris entrance into the pool is not uniform. Appendices III and VI provide additional staff guidance on adapting the debris transport methodologies for refined analyses.**

The Executive Summary and/or staff conclusions on page 96 should be revised to be consistent.

Page xii, ES.11 Downstream Effects
The DSER directs licensees to consider the downstream effects of particles larger than the screen openings and axially-oriented particles that pass through the screen.

This requirement is excessively conservative. The differential pressure expected across the screens is very low, and would not be expected to extrude debris. Also, the transportability of larger particulate is comparatively less than smaller particulate. Thus the bypass fraction for large or axially oriented particulate, even for clean screens, would be expected to be much lower than for small particulate. As the fiber bed forms, this bypass fraction quickly reduces to very near zero. This requirement should be deleted.

Page xii, ES12 Chemical Effects
The DSER states the following in several areas:

> **Initially, licensees should evaluate whether the current chemical test parameters, which are available in the test plan for the joint NRC/Industry Integrated Chemical Effects Tests, are sufficiently bounding for their plant specific conditions. If they are not, then licensees should provide technical justification in order to use any of the results from the tests in their plant-specific evaluation.**

This language would appear to be in conflict with the test plan (developed jointly by NRC-research and industry) which was designed to provide data for a "representative" recirculation ECCS recirculation system and containment. One of the objectives stated in the test plan is: "Determine, characterize and quantify the chemical reaction products that may develop in a representative post-LOCA containment sump environment."

It is recognized that the intent of the SER statements may be to restrict licensees from placing excessive amounts of reactive material (such as aluminum scaffolding) which may result in formation of reaction products which may contribute to head loss into containments. However, creation of material restrictions in containment was not intended in the development of the present chemical effects test program nor have the sensitivity of various materials vis-à-vis head loss been determined.

This language should therefore be removed from the SER. Once the final results from these chemical effects and possible necessary future chemical effects tests are available, and if there are issues discovered which affect sump screen head loss, appropriate guidance can be developed.

Page 21, 3rd bullet:

> **The staff position is that licensees should use a coatings ZOI equivalent to 10D or a ZOI determined by plant specific analysis. The specified ZOI of 10D is based upon the previous staff position used for BWR sump analysis.**

The staff position for estimating coatings debris based on a 10D ZOI should be clarified. The 10D ZOI defined in the BWR URG is a cone-shaped volume with double 10 degree subtended angles of expansion. The cone-shaped jet extends 20 feet from a 2 ft diameter pipe break and affects an equivalent coated flat surface of 302 square feet. The BWR URG then assumes that another 302 square feet of coated surface, positioned between the pipe break and the 20 foot outward extension of the jet, are affected by the jet, bringing the total affected coated surface area to 604 square feet.

The DSER implies that a 10 D spherical ZOI be considered. Use of a 10D spherical ZOI would result in an impacted surface area that is orders of magnitude greater than that utilized as part of the BWR URG or industry guidance.

Section 3.4.2.2

The DSER reduces destruction pressures by 40% for debris sources tested with AJIT to account for uncertainties related to two-phase fluid effects. This requirement imposes unreasonable conservatism upon a baseline analysis approach that is already laden with conservatism. The presumption of 40% more damage is not substantiated by testing, and is little more than a "fudge factor" to account for an effect observed in a single test of different material composition. It places an undue emphasis on the maximizing sump screen debris loading in lieu of analyzing for the thin bed, which is the more probable scenario. This modification to the guidance is unwarranted.

Page 31

Staff Evaluation for Section 3.4.2.6

> **The sample calculation is inconsistent with the baseline methodology discussed above because it implies that potentially affected insulation type with the minimum destruction pressure can be selected from within an accounting region in the vicinity of the break rather than from the entire containment inventory as specified in Section 3.4.2.2.**

It was not the intent to construct a sample calculation to define a ZOI in conflict with the criterion of Section 3.4.2.2. The sample calculation should have stated that the two types of insulation inside containment were fiberglass and RMI, then identified the regions affected by the ZOI and resulting debris volumes.

Page 37, "Staff Evaluation for Protective Coatings Quantification:"

Staff Evaluation for Protective Coatings Quantification: The staff finds that the quantity of coating debris that will be generated as a result of a LOCA jet should be based on the following:

- For plants that substantiate a thin bed, use of the basic material constituent (10 μm sphere) to size coating debris is acceptable.
- For those plants that can substantiate no formation of a thin bed at which particulate debris can collect, the staff finds that coating debris should be sized based on plant specific analyses for debris generated from within ZOI and from outside the ZOI. Such an analysis should conservatively assess the coating debris generated with appropriate justification for the assumed particulate size or debris size distribution. Degraded qualified coatings that have not been remediated should be treated as unqualified coatings. Finally, testing regarding jet interaction and coating debris formation could provide insight into coating debris formation and help remove some of the potential conservatism associated with treating coatings debris as highly transportable particulate. If coatings, when tested at corresponding LOCA jet pressures and temperatures, are found to fail by means other than erosion or the erosion is limited, the majority of debris may be larger, less transportable or pose less of a concern for head loss."

The GR identifies use of a 10 μm coating debris size when applicable experimental data is not available. Licensees that have plant specific test data of coating systems should be able to use this data to characterize the debris size for coatings assumed to fail inside and outside the ZOI. Licensees should be able to apply this data for either the thin bed or non-thin bed cases.

Page 38, section 3.4.3.4

> **The staff concludes that the baseline alternatives to plant specific data for the determination of the coatings thickness may not be conservative and are not acceptable without plant specific justification. Rather, the staff concludes that each plant should perform a plant specific evaluation of their respective coatings to determine conservative coating thicknesses. This conclusion was drawn despite the perceived conservatism of the recommendations of assuming all unqualified coatings in containment fail and all coating debris forms a fine 10 micron particulate. It is considered reasonable for each plant to assess their respective coatings thicknesses as well as the soundness of their coatings rather than assume an indefensible default recommendation.**

While the DSER acknowledges the conservatism of the GR treatment of unqualified coatings, we do not agree that it is a reasonable expectation for each plant to fully assess the coatings thicknesses on potentially hundreds of individuals items in containment (pumps, motor operators, electrical cabinets, junction boxes, light fixtures) that possess unqualified coatings.

Page 39, Section 3.4.3.6
We agree that use of plant specific data is desirable with respect to defining insulation characteristics, but in situations when explicit data is not available, a level of flexibility must be afforded to utilize generic insulation information.

Page 46, Section 3.5.2.2.2
Regarding equipment tags, stickers and placards, the staff evaluation states that, if they remain intact and are transported to the sump screen, the sump screen flow area should be reduced by an area equivalent to the original single-sided surface area of the tags. If there is information that the tags will not remain intact, the staff recommends that the equivalent mass of the tags be treated as fibers.

The reduction of the sump screen surface area by the area equivalent to the original single-sided surface area of tags, stickers and placards is very conservative. Past practices in evaluating screen blockage of larger particulates allowed for some overlap of the particulates, typically about a 50% overlap. This accepted practice should be maintained.

The treatment of tags, stickers and placards as fully fibrous is overly conservative. Consideration should be given for treatment of these materials in a more realistic fashion.

Page 55, second paragraph,
It is not clear what is meant by large "flocks" – are these to be considered anything larger than 4 inches?

Page 59, first paragraph
In BWR applications utilization of flow reduction via operator actions was allowed to be credited for reducing pump flow from maximum runout conditions to a throttled flow condition. Can such an approach be utilized for PWR applications? Maximum conditions typically specified are at runout, and do not necessarily represent the plant configuration. Credit for operator action, especially if it is defined in EOPs and has been validated in simulator applications should be allowed to be credited to reduce this over conservative input with respect to debris head loss.

Page 78, 6th paragraph
This paragraph states that it is not conservative to truncate the ZOI whenever it intersects a robust structure implying that the ZOI should be re-sized. This is apparently in contradiction to the second paragraph of page 30.

Page 95
The SER basically finds the uniform debris deposition in the pool at the onset of recirculation unacceptable. This is an initial condition that allows for the development of

the transport fraction. Is there a size of debris that would be uniformly distributed (e.g. fines or small pieces)? For example if the large pieces were retained by debris interceptors to the SG compartment, can it be assumed that all the fines and small pieces are uniformly distributed in the pool from blowdown, washdown and fillup?

Page 96

At the end of 2nd paragraph the DSER instructs the licensee to use the four (4) size categories used in both Appendices III and VI for fibrous debris. The classifications are: 1) fines, 2) small pieces, 3) large pieces and 4) intact blankets (jacketed). Appendix III and VI are essentially the 6772 Separate Effects Characterization, 6773 Integrated Debris Transport Tests and 6369 Drywell Debris Transport Studies and provide a fibrous debris category distribution of 7% fines, 26% small pieces, 32% large pieces, 35% intact pieces. However, in Appendix VI, page VI-8, middle of the page, is a rather large disclaimer "Neither the debris size distributions nor the overall transport fractions in this report are valid for plant specific evaluations." This seems to contradict the DSER direction on pg 96 of the SER to utilize the information contained in the appendices. Transport is predicated on debris size distribution and the DSER has limited application of industry guidance and directed the use of incomplete guidance. Additional information is needed on what NRC would find acceptable, i.e., the DDTS, 6772 and 6773

Page 99, 5.1.5:

> **"However, coating systems that are currently unqualified could be qualified through appropriate testing. Depending upon the rigors of the ASTM standards, some of this testing might be accomplished in place to avoid destructive sample collection from existing surfaces."**

ASTM Committee D-33, Protective Coatings and Linings for Power Generation Facilities, has examined in-situ qualification of coatings in containment on several occasions. Committee D-33 does not endorse nor recommend this approach to upgrading safety-related coatings, and there are no plans to prepare new or revised standards to provide for in-situ DBA testing of coatings. As such, "appropriate testing" based on accepted standards does not currently exist.

Page 117, Section 6.4.7

The GR guidance in Section 6 allows use of nominal parameters as part of the design basis analysis of Region II breaks. Use of nominal parameters implies a recognition that normal operational values will sometimes be higher and sometimes be lower. In recognition of this, the GR includes an accommodation for application of GL 91-18 guidance. We believe the Region II analysis is sufficiently conservative to accommodate operation outside the nominal value for a short period.

The DSER revokes this accommodation for GL 91-18 and will result in the need to use bounding values instead of nominal values. This significantly limits the application of Section 6 guidance and is contrary to the intent of the analysis.

Page 137, 1st full paragraph:

> **"Specifically, this includes the plant-specific consideration of larger sized chips, flakes or other forms of breakdown which is realistically-conservative."**

This concept of the failure morphology of containment coatings is flawed for a number of reasons, primarily because it is based on the spontaneous in-service failure of one particular coating system and subsequent failure by the affected licensee to clean up coating debris in accordance with good housekeeping practices.

Industry has demonstrated that one of the most significant concerns in PWR sump failure analysis is the thin-bed effect, which is exacerbated by small particulate debris. The coatings which may fail and produce transportable debris post-LOCA or post-HELB will all produce debris which, when exposed to the sump pool environment, will conservatively produce small (10µm -50 µm) particles.

No experimental evidence exists to support the staff's premise that containment coatings will fail to produce "larger sized chips, flakes or other forms of breakdown."

OCTOBER 1, 2004

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SAFETY EVALUATION BY
THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO NRC GENERIC LETTER 2004-02,
NUCLEAR ENERGY INSTITUTE
GUIDANCE REPORT
(PROPOSED DOCUMENT NUMBER NEI 04-07),
"PRESSURIZED WATER REACTOR
SUMP PERFORMANCE EVALUATION METHODOLOGY"

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FOREWORD

The objective of this SER on (proposed document number NEI 04-07) "Pressurized Water Reactor Sump Performance Evaluation Methodology" (NEI, 2004a), submitted by the Nuclear Energy Institute (NEI) to the NRC, is to document the staff's review of methodology guidance for licensees of pressurized water reactors (PWRs). This SER relates to recently-issued NRC Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors" (GL-04-02).

In the staff's review of the NEI submittal, it found that portions of the proposed guidance were acceptable as is; and other portions were found to need additional justification and/or modification. Therefore, in an effort to expedite the resolution of Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on PWR Sump Performance," the staff has identified conditions, limitations, and required modifications, including alternative guidance to supplement those portions determined by the staff to need additional justification and/or modification in the NEI submittal. The NEI submittal as approved in accordance with the staff safety evaluation, provides an acceptable overall guidance methodology for the plant-specific evaluation of emergency core cooling system (ECCS) or core spray system (CSS) sump performance following any postulated accident for which ECCS or CSS recirculation is required, with specific attention given to the potential for debris accumulation that could impede or prevent ECCS or CSS from performing its intended safety functions.

Deleted: objective of a safety evaluation report (SER) issued by the U.S. Nuclear Regulatory Commission (NRC, the staff) is typically to determine and describe the acceptability of a submittal from a domestic licensee, vendor, or nuclear industry organization related to a nuclear power plant(s). However, the

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EXECUTIVE SUMMARY

The Nuclear Energy Institute (NEI) submitted (proposed document number NEI 04-07) "Pressurized Water Reactor Sump Performance Evaluation Methodology" (NEI, 2004a, referred to herein as the Guidance Report or GR), for review by the U.S. Nuclear Regulatory Commission (NRC, the staff). NRC approval of this methodology guidance would allow licensees of pressurized water reactors (PWRs) to use the document in their responses to recently-issued NRC Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors" (GL-04-02), as the cited NRC approved methodology for their evaluation of plant-specific sump performance. The Generic Letter identifies inadequacies in previous approaches for modeling sump screen debris blockage and related effects, such that the staff no longer considers many licensing-basis analyses acceptable for confirming compliance with NRC regulations. The NEI submittal offers guidance to all PWR licensees in response to those inadequacies identified during resolution of Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on PWR Sump Performance," documented in the Generic Letter.

The NEI submittal as approved in accordance with the staff safety evaluation, provides an acceptable overall guidance methodology for the plant-specific evaluation of ECCS (or CSS) sump performance following all postulated accidents for which ECCS or CSS recirculation is required, with specific attention given to the potential for debris accumulation that could impede or prevent ECCS or CSS from performing its intended safety functions.

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The GR is divided into two primary sections, the baseline evaluation and the refinements sections. The baseline is intended by NEI to provide a conservative approach for utilities to perform a "baseline evaluation" of their PWR containment sump using a sample calculation for a consistent and simplified first-step in determining susceptibility to head loss. The refinements sections are intended to address, for those plants that do not "pass" the baseline evaluation, options for refinements to the baseline calculation that result in acceptable results, or hardware "fixes" to provide acceptable results. This NEI submittal addresses the following major areas:

- Pipe Break Characterization
- Debris Generation/Zone-of-Influence
- Latent Debris Accumulation within Containment
- Debris Transport to the Sump Screen(s)
- Head Loss as a Result of Debris Accumulation
- Analytical Refinements to Remove Conservatism(s) from the Evaluation
- Physical Refinements to Plant
- Alternate Evaluation (realistic and risk-informed)
- Sump Structural Analysis
- Upstream Effects of Debris Accumulation
- Downstream Effects of Debris Accumulation
- Chemical Precipitation Effects of Debris Accumulation

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The following is a brief summary of each major area of the staff's evaluation.

ES.1 PIPE BREAK CHARACTERIZATION

Analysis of the most challenging postulated accident with regard to sump performance during long-term core cooling, involves selection of the most limiting pipe break size, location, and debris combination within containment. For a PWR, RG 1.82, Rev. 3, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," (RG 1.82-3), Section C, Regulatory Position 1.3.2.3, specifies that a sufficient number of breaks in each high-pressure system that relies on recirculation should be considered to reasonably bound variations in debris generation by size, quantity and type of debris. Regulatory Guide 1.82 stipulates the following set of break locations to be considered as a minimum:

- Breaks in the reactor coolant system (RCS) and, depending on the plant licensing basis, main steam and main feedwater lines with the largest amount of potential debris within the postulated zone-of-influence (ZOI),
- Large breaks with the most variety of debris, within the expected ZOI,
- Breaks in areas with the most direct path to the sump,
- Medium and large breaks with the largest potential particulate debris to insulation ratio by weight, and
- Breaks that generate an amount of fibrous debris that, after its transport to the sump screen, could form a uniform thin bed that could subsequently filter sufficient particulate debris to create a relatively high head loss referred to as the "thin-bed effect." The minimum thickness of fibrous debris needed to form a thin bed has typically been estimated at 1/8 inch thick.

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The GR states the objective of the break selection process is to identify the break size and location which results in debris generation that produces the maximum head loss across the sump screen. All phases of the accident scenario must be considered for each postulated break location, including debris generation, debris transport, and sump screen head loss calculations. The break selection process outlined in the GR identifies limiting break locations as those that result in:

- The maximum amount of debris that is transported to the sump screen.
- The worst combination of debris mixes that are transported to the sump screen.

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The GR also provides the following guidance:

- Break exclusion zones are disregarded for this evaluation (pipe breaks must be postulated in pre-existing break exclusion zones).
- Exclude consideration of NRC Branch Technical Position MEB 3-1, as a basis, since limiting conditions for ECCS sump concerns are not related to the pipe vulnerability issues addressed in MEB 3-1.
- For plants needing to consider main steam and feedwater line breaks, break locations should be consistent with the plant's current licensing basis.
- Consider locations that result in a unique debris source term (i.e., not multiple identical locations).

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- Consider locations with high concentrations of problematic insulation.
- Consider breaks that generate an amount of fibrous debris that could create a thin-bed effect.
- Small breaks less than 2 inches in diameter (for piping attached to the RCS) need not be considered.
- If a significant amount of fibrous debris is not generated, consider breaks that produce the greatest contribution of latent debris sources which may produce the limiting debris loading condition for sump screen blockage concerns.

The staff finds that the GR is consistent with staff positions, with the following exceptions:

1. The GR does not provide guidance for those plants that can substantiate no thin bed effect, which may impact head loss results and limiting break location.
2. For plants needing to evaluate secondary-side piping such as main steam and feedwater pipe breaks, break locations should be postulated in a manner consistent with the guidance in Section 3.3 of this SER.

To address these exceptions, the staff provided enhanced guidance in the appropriate sections of this SER. Additionally, Appendix VIII to this SER provides a description of a thin bed, including its formation and effects. The guidance provided in the GR in accordance with the enhanced guidance offered in the SER, provides an acceptable overall approach.

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ES.2 DEBRIS GENERATION/ZONE-OF-INFLUENCE

With the rupture of piping come shock waves and jets of coolant that project from within the piping via the closed system pressure, until that pressure dissipates. Debris is generated as the shock waves and jets impact surrounding insulation, coatings, surfaces, and other materials within the zone. The volume of space affected by this impact, or zone-of-influence (ZOI), is modeled in order to define and characterize the debris generated.

The ZOI recommended in GR Section 3.4, is a spherical boundary with the center of the sphere located at the break site. The use of a spherical ZOI is intended to encompass the effects of jet expansion resulting from impingement on structures and components, truncating the sphere wherever it intersects any structural boundary or large robust equipment. The GR recommends that ZOI sizing be determined using the ANSI/ANS 58.2-1988 standard for a freely expanding jet. The baseline ZOI is to be based on the insulation type that generates the largest ZOI of all potentially affected insulation types located inside containment—i.e., the insulation type with the lowest destruction pressure. The resulting ZOI will then be applied to all insulation types.

Coating debris generation, however, is treated separately. Coating debris in the GR, are generated from postulated failure (destruction) of both DBA-qualified and unqualified coatings within the ZOI and from postulated failure of all unqualified coatings outside the ZOI. For coatings, the GR recommends a ZOI destruction pressure of 1000 psi, with a corresponding ZOI radius of one pipe diameter. The GR assumes that all coating debris will fail to a particulate size equivalent to the basic material constituent.

Debris characteristics are described in the GR in terms of size distribution, size and shape, and density. The GR identifies two size distributions for material within the ZOI, i.e., small fines and large pieces. Small fines are defined as debris able to pass through the largest openings of the gratings, trash racks, and radiological fences, which are less than a nominal four inches. Debris that cannot pass through these barriers is classified as large pieces.

For debris sizing assumed within the ZOI, most fibrous debris is assumed to degrade to 60% small fines and 40% large pieces. Some fibrous debris is considered to degrade to 100% small fines and no large pieces. Reflective metallic insulation (RMI) is assumed to degrade to 75% small fines and 25% large pieces. And most other debris types are considered to degrade to 100% small fines and no large pieces. Erosion is neglected based on the assumption that the small fines are already reduced to their basic constituents of individual particles and fibers. Jacketed large debris is also assumed not to erode.

Debris material densities and size distributions were tabulated in the GR for select debris types. Properties of materials for which limited data is available are listed as "best available." For those materials for which no data is available, maximum destruction is assumed.

The GR assumes that coatings will fail as particulate. The amount of particulate is a function of coating properties including the thickness and area. The GR indicates that where plant-specific data does not exist regarding the thickness of unqualified coatings, an equivalent thickness of 3 mils of inorganic zinc (IOZ) be used.

The staff has reviewed the use of a spherical model sized in accordance with the ANSI/ANS standard, and finds this approach acceptable. The spherical geometry proposed encompasses a zone which considers multiple jet reflections at targets, offset between broken ends of a guillotine break, and pipe whip. The confirmatory analysis performed by the staff (Appendix I) verifies the applicability of the ANSI/ANS standard for determining the size of this zone. Use of a ZOI model is identified as an acceptable approach for analyzing debris generation per RG 1.82, Rev. 3. (This approach was also used and approved by the staff in the BWR sump performance SER.) The GR recommendation to truncate the spherical ZOI when a robust barrier or large piece of equipment is encountered is acceptable to the staff. The refinement offered in the GR to apply spherical ZOIs that correspond to material-specific destruction pressures for each material that may be affected in the vicinity of a break, is also acceptable.

A LWR LOCA jet is a two-phase steam/water jet. The destruction pressures cited in the GR are referenced from the BWROG URG which were determined using an air jet. Based on staff study of this difference and due to limited experimental evidence from two-phase jets, the BWROG destruction pressures could be too high and thus could underestimate debris quantities. The staff position in this Safety Evaluation is to lower the debris destruction pressure by 40% in order to account for two-phase jet effects (see Section 3.4.2.2).

With regard to coatings, the staff agrees with the approach taken; however the staff considers there to be insufficient technical justification to support a value of 1000 psi as a destruction pressure, with corresponding ZOI of one pipe diameter. The staff position is that the licensees should use a coatings ZOI spherical equivalent to 10D or a ZOI determined by plant specific analysis, based on experimental data that correlate to plant materials over the range of temperatures and pressures of concern. Note that an equivalent to ten pipe diameters was used for coatings characterization and was approved by the staff in the BWR sump performance SER.

The staff concurs with the characterization of debris in GR Section 3.4.3. Confirmatory analyses provided in Appendix II, verifies the acceptability of the size distributions recommended in the GR. However, the staff position is that licensees apply insulation-specific debris size information where available.

For the characterization of coatings in Section 3.4.3.4, the staff position is that the alternative offered to use of plant-specific data for the determination of coatings thicknesses include plant-specific justification. The equivalent inorganic zinc (IOZ) thickness of 3 mils recommended may be nonconservative and unsubstantiated because, although the assumption that all unqualified coatings outside the ZOI fail is consistent with the position provided in NUREG-0800, Section 6.1.2, "Protective Coatings Systems", the staff is aware of numerous cases in which containment coatings, qualified and unqualified, are much thicker than the recommended 3 mil IOZ equivalent thickness.

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Also, for those plants that substantiate no formation of a fibrous thin bed, the assumptions and guidance provided in the GR for coatings may be nonconservative in that the particulate-sized debris assumed would simply pass through the screens, thereby not causing a head loss concern. Therefore, for any such plant, the staff position is that assumptions related to coatings characterization, be realistically-conservative based upon the plant-specific susceptibilities and data identified by the licensee, or that a default area equivalent to the area of the sump screen openings be used for coatings size.

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ES.3 LATENT DEBRIS

Section 3.5 of the GR provides guidance for estimating the amount of latent debris as a source for contribution to head loss across the ECCS sump screen. Generally, miscellaneous fiber, dust, and dirt are primary sources of this debris type. It is noted that for all-RMI plants, the primary contribution of fibrous debris toward formation of a thin fibrous bed may come from latent debris sources.

The staff has reviewed the guidance provided for estimating the impact of latent debris and agrees that it is necessary to determine the types, quantities and locations of latent debris sources. The staff also agrees that it is not appropriate for licensees to assume that their existing foreign material exclusion (FME) programs have entirely eliminated miscellaneous debris. Results from plant specific walkdowns should be used to determine a realistic amount of latent debris in containment and to monitor cleanliness programs for consistency with committed estimates.

The guidance provided in the GR for consideration of effects of latent debris is considered acceptable for: (1) general considerations for latent debris; (2) estimates of some surface areas for evaluation of latent debris; and (3) some attributes associated with evaluation of debris buildup, quantity of miscellaneous debris, and defining debris characteristics. Alternate guidance is provided in Section 3.5 of the SER for sampling techniques and analysis to allow licensees to more accurately determine the impact of latent debris on sump screen performance. This revised approach is based on generic characterization of actual PWR debris samples. If desired, a licensee could pursue plant-specific characterization as a refinement.

ES.4 DEBRIS TRANSPORT

Debris transport is described in Section 3.6, and is separately specified for each of three containment types—highly-compartmentalized, mostly uncompartmentalized, and ice condenser containments. Transport of the two size distributions identified in ES.2, above, and discussed in Section 3.4.3 (i.e., small fines and large pieces) are considered in the staff's review of debris transport.

The staff finds that the transport guidance for small fines of debris is acceptable. However, the guidance for the large pieces of debris is unacceptable because of the unrealistic assumption that large pieces of debris will not transport. Specifically, plants with configurations conducive to fast pool velocities will realistically transport some large pieces, therefore the staff position is that consideration for transport of large pieces of debris is necessary.

The staff also finds that the method recommended for determining the quantity of fine debris trapped in inactive pools based on the volume ratio of inactive pools to the total pools is unrealistic for plants with large inactive pools. Therefore the staff position is that licensees should limit the maximum fraction of fine debris being trapped in inactive pools to 15% to avoid nonconservative results.

ES.5 HEAD LOSS

Computation of head loss in the GR involves input of design characteristics and reflection of thermal-hydraulic conditions into a head loss correlation (NUREG/CR-6224). The approach is acceptable to the staff, with specific areas of additional guidance offered in Section 3.7.2.2 and 3.7.2.3 of this SER. The licensees should ensure the validity of the NUREG/CR-6224 correlation for their applications of type of insulations and the range of parameters using the guidance provided in Appendix V of this report.

The following additional guidance on fibrous thin bed formation should be considered:

- Use of the appropriate density in the determination of the quantity of debris needed to form a thin bed—i.e., the as-manufactured density.
- Careful evaluation of the limiting porosity for the particular particulate or mixture of particulates in the debris bed.
- Consideration of uncertainties in specifying a one-eighth-inch bed thickness criteria—e.g., the indication that calcium silicate can form a debris bed without supporting fibers.
- Consideration of other uncertainties—e.g., uncertainties associated with mixing of constituents, or uncertainties associated with latent debris data collection.

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Before using the NUREG/CR-6224 correlation that is recommended in the GR or any other head loss correlation, the licensees should ensure that it is applicable for the type of insulation and the range of parameters. If the correlation has been validated for the type of insulations and the range of parameters, the licensees may use it without further validation. If the correlation has not been validated for the type of insulations and the range of parameters, the licensees should validate it using head loss data from tests performed for the particular type of insulations.

ES.6 ANALYTICAL REFINEMENTS

Three analytical topics are identified in this section—i.e., debris generation, debris transport, and head loss. A fourth, break selection, is addressed in Section 4.2.1.

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For debris generation, the GR recommends two refinements for insulation materials. First, the GR proposes use of debris-specific ZOI's versus use of the most conservative debris type applied to all. Second, the GR proposes use of two freely-expanding jets emanating from each broken pipe section versus use of spherical ZOI. The staff finds both debris generation refinements to be acceptable.

For debris transport, two methods for computing flow velocities in a sump pool—i.e., the network method and the computational fluid dynamics method—are provided in the Analytical Refinements section of the GR. However, the staff finds the guidance offered in either option to be insufficient to provide an acceptable alternative to the baseline approach.

For head loss, only refinements offered in GR Section 3.7.2.3.2.3, "Thin Fibrous Beds," are offered. This section addresses the need for consideration of fibrous thin bed formation, and the alternative consideration of latent debris as the primary contributor to this thin bed for all-RMI plants.

ES.7 PHYSICAL REFINEMENTS TO PLANT

GR Section 5.0 provides guidance for refinements in the areas of debris source term, debris transport obstructions, and screen modifications.

The five following areas for refinement are offered for debris source term:

- Housekeeping and foreign material exclusion (FME) programs
- Change-out of insulation
- Modify existing insulation
- Modify other equipment or systems
- Modify or improve coatings program

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The staff has reviewed these refinements and finds them to be acceptable. However, with regard to insulation change-out or modification, the staff emphasizes that minimum loadings required to form a thin-bed be considered. Also, related to coatings, the statement that DBA-qualified coatings have very high destruction pressures has not been demonstrated (see Sections 3.4.2, 3.4.2, and 4.2.2.2.3).

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This section of the GR also discusses the potential use of floor obstructions to provide barriers to prevent debris transport to the sump. It mentions that barriers can be used either near the sump or closer to the debris source. Key considerations regarding the use of floor obstructions and barriers are that the barrier be located where flow velocities and turbulence are insufficient to lift debris over the barriers, and the barrier should cover the entire cross-section of flow.

To credit debris transport obstructions for trapping debris, plant specific documentation should be available on site to demonstrate an appropriate correlation to the test results in terms of debris type and velocity limits.

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With regard to screen modifications, those discussed in the GR are found to be acceptable; however, licensees are not limited to those identified.

ES.8 ALTERNATE EVALUATION

NEI has proposed an alternative evaluation approach which incorporates realistic and risk-informed elements to the PWR sump analysis. The following steps are proposed for this alternative approach, or "Option B":

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- Define a "debris generation" LOCA break size to distinguish between customary and more realistic design basis PWR sump analyses.
- Perform customary design basis analyses for break sizes up through the debris generation break size identified above (i.e., Region I analyses).
- Perform analyses demonstrating long-term cooling and mitigative capability for break sizes larger than the debris generation break size up through the double-ended rupture of the largest RCS piping (i.e., Region II analyses).

The GR proposes realistic treatment of Region II break sizes based on the low probability of these larger breaks. Models, assumptions, and equipment availability for mitigation used for this analysis are proposed to be realistic and demonstrated as functionally reliable, and may not necessarily be safety-related or single failure-proof. Risk evaluations would be performed as a basis for plant modifications and credit taken for operator actions. Such analyses may require plant-specific exemption and/or license amendment requests.

In considering risk-informing aspects of the resolution of GSI-191, the staff recognized that there is the potential that the containment sump may clog if the mitigation capability credited in the Region II analysis does not function properly. Based on the industry proposed approach in the Region II analysis, which also uses the conservative NUREG-1150 LBLOCA frequency to calculate the target reliability of the mitigation capability, and using the related generic study information, the largest LBLOCA CDF would be 1.4E-5/year. This indicates that at a minimum the risk associated with LBLOCAs will be reduced from the current condition by nearly an order of magnitude.

- The staff concludes that GR Section 6.0 provides an acceptable approach for evaluating PWR sump performance. Application of more realistic and risk-

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informed elements is technically justified based on the low likelihood of such breaks occurring.

ES.9 SUMP STRUCTURAL ANALYSIS

The staff provides information in this section to show that structural loads on a sump screen should be computed using the total pressure drop across the screen. The limiting conditions correspond to the break location and debris source term that induce the maximum total head loss at the sump screen after full consideration of transport and degradation mechanisms. This represents the minimum required performance criterion for judging recirculation-sump operability. In other words, the recirculation sump must be able to accommodate both the clean-screen head loss and the debris-induced head loss associated with the limiting break while providing adequate flow through both the ECCS injection pumps and the CSS pumps if needed. For some licensees, the minimum structural design criterion for the sump screen can depend on the plant NPSH margin. Revised plant-specific licensing bases may dictate the structural capacity of the sump screen for supporting water flow through a debris bed under recirculation velocities depending on screen geometry (fully-submerged versus partially-submerged designs).

ES.10 UPSTREAM EFFECTS

The GR states that certain hold-up or choke points exist which could reduce flow to and possibly cause blockage upstream of the sump. Such areas within containment are: (1) narrowing of hallways or passages; (2) gates or screens that restrict access to areas of containment such as behind the bioshield or crane wall; and (3) refueling canal drain.

The staff finds the guidance with respect to upstream blockage to be acceptable.

ES.11 DOWNSTREAM EFFECTS

This section provides guidance on the evaluation of entrained debris downstream of the sump causing downstream blockage. The three areas of concern identified are: (1) blockage of flow paths in equipment such as containment spray nozzles and tight-clearance valves, (2) wear and abrasion of surfaces such as pump running surfaces, and heat exchanger tubes and orifices, and (3) blockage of flow clearances through fuel assemblies.

The staff finds this section to need clarification and additional considerations and provides the following alternative guidance with regard to downstream blockage:

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- Licensees should consider the potential for particles larger than the flow openings in a sump screen to deform and flow through or orient axially and flow through, and determine what percentage of debris would likely pass through their sump screen and be available for blockage of piping, core spray nozzles, and instrument tubing at downstream locations.
- Licensees should consider term of operating line-up (short or long), conditions of operation, and mission times.
- Licensees should consider wear and abrasion of pumps and rotating equipment, piping, spray nozzles, instrumentation tubing, and HPSI throttle valves. The

potential for wear to alter system flow distribution and/or form plating of slurry materials (in heat exchangers) should be included.

- An overall ECCS or CS system evaluation should be performed considering the potential for reduced pump/system capacity due to internal bypass leakage or through external leakage.
- Licensees should consider flow blockage associated with core grid supports, mixing vanes, and debris filter, and their effects on fuel rod temperature.

ES.12 CHEMICAL EFFECTS

GR Section 7.4 addresses how reaction products formed in a post-LOCA environment can contribute to blockage of the sump screens and increase the associated head loss across the screens. The GR also defers guidance for dealing with these effects until current testing is completed and the data has been appropriately evaluated.

The staff has considered NEI's response and finds that chemical effects should be addressed on a plant-specific basis. Initially, licensees should evaluate whether the current chemical test parameters are sufficiently bounding for their plant specific conditions. If they are not, then licensees should justify the use of test results in their plant-specific evaluation. If chemical effects are observed during these tests, then licensees should evaluate the sump screen head loss consequences of this effect. A licensee who chooses to modify their sump screen before tests are complete should consider potential chemical effects in order to avoid additional screen modification should deleterious chemical effects be observed during testing.

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GUIDANCE DEVELOPMENT BACKGROUND

The staff began working with NEI on the resolution of GSI-191 in 1997, with the establishment of the PWR Industry Sump Performance Task Force. The staff also conducted a study on the susceptibility of PWRs to ECCS sump blockage following a LOCA. This study was entitled, "GSI-191: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance" (Rao, 2001), and was performed by Los Alamos National Laboratory (LANL) in support of the NRC's GSI-191 technical assessment to determine if sump failure is a plausible concern for PWRs.

On July 26 and 27, 2001, the NRC held a public meeting with the industry and other stakeholders including NEI, the Westinghouse Owners Group, the Babcock and Wilcox Owners Group, and the Combustion Engineering Owners Group, on the preliminary findings of that study. This meeting was documented in a meeting summary dated August 14, 2001 (Mtg, 2001). The preliminary results of the study indicated that significant quantities of fibrous and particulate debris will be generated during various size LOCAs, and that a sufficient fraction of this debris may be transported to the sump screen and cause sump screen blockage. However, before determining what regulatory action was needed, the staff presented the results to the industry and interested stakeholders, to discuss the assumptions and calculations in the report. Since that time, the parametric report was approved and issued (NUREG/CR-6762), and the staff concluded that GSI-191 is a credible concern for the population of domestic PWRs and that detailed plant-specific evaluations are needed to determine the susceptibility of each U.S.-licensed PWR to ECCS sump blockage.

The staff has worked closely with NEI, providing feedback into the development of an acceptable approach to resolution of GSI-191, through a series of public meetings held between July 2001 and October 2003, until the submittal of NEI's October 31, 2003, "PWR Containment Sump Evaluation Methodology" (NEI, 2003b). Following the public meeting on July 26 and 27, 2001, described above, which involved discussions of risk considerations as well as the parametric evaluation results, a public meeting was held on March 28, 2002, described in a meeting summary dated April 16, 2002 (Mtg, 2002a). The staff presented its approach toward resolution of GSI-191, as did the industry, making references to the revision of Regulatory Guide (RG) 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident (DG-1107)," issuance of a generic letter, Standard Review Plan update, chemical testing, data collection guidance, and evaluation guidance. Industry also committed to take the lead for issue resolution.

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By the next meeting on May 30, 2002, NEI had issued NEI 02-01, "Condition Assessment Guidelines: Debris Sources Inside PWR Containments," dated April 19, 2002 (NEI, 2002a). The staff's comments in response to NEI 02-01 identified minor concerns with a lack of firm direction in some areas of data collection although the staff considered that NEI 02-01 provided reasonable overall guidance. The staff's conclusions are included in Attachment 3 to the meeting summary dated June 6, 2002, as are status presentations from the staff, NEI, and the industry (Mtg, 2002b).

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In the next two public meetings on July 2, 2002, and August 29, 2002, the staff raised discussions on the schedule for the draft generic letter, the development of temporary instructions for NRC inspectors regarding GSI-191, concerns surrounding downstream effects such as high pressure safety injection (HPSI) throttle valve blockage, and presented fault tree modeling for ECCS injection. NEI's discussion focused on Interim Plant Assessment templates and guidance on related compensatory measures, as well as their response to the staff's comments on NEI 02-01. The discussions in both meetings are documented in meeting summaries dated July 31, 2002 (Mtg, 2002c), and September 5, 2002 (Mtg, 2002d), respectively.

The following two public meetings on October 24, 2002, and December 12, 2002, revolved around NEI's proposed ground rules for the sump evaluation guidance, discussion of head loss behavior and leak-before-break (LBB) considerations for break selection, as well as the HPSI issue. The staff objected to the use of LBB as applied to break selection assumptions. "NEI Draft Evaluation Methodology Ground Rules" was issued on December 12, 2002 (NEI, 2002b). The material presented during both meetings is included in the meeting summaries dated October 31, 2002 (Mtg, 2002e), and December 31, 2002 (Mtg, 2002f), respectively.

The staff, NEI, LANL, and interested stakeholders participated in discussions of GSI-191 issues and toured the University of New Mexico (UNM) experimental facilities on March 5, 2003. NRC presented the schedule for generic letter issuance, chemical testing status and expectations, response to NEI's "ground rules" for sump evaluation guidance, and supporting data and research by LANL including debris accumulation, ECCS vulnerability, and pool flow analysis. NEI presented material on the use of LBB for break selection, the use of a Nodal Network Method as an alternative to Computational Fluid Dynamics computer modeling for debris transport analysis, and the use of fracture mechanics for debris generation. A meeting summary was generated, and several individual presentations were documented (Mtg, 2003a).

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NEI requested a meeting on April 29, 2003, summarized in a meeting summary dated May 15, 2003 (Mtg, 2003b), where the technical basis for using LBB arguments for break selection was discussed at length. The staff recommended that NEI provide for staff consideration an official submittal on their proposed approach to break selection. The staff presented the proposed Bulletin in the meeting, which was titled "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors."

On June 30, 2003, the staff held a public meeting with NEI and interested stakeholders on the issuance of NRC Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Recirculation during Design-Basis Accidents at Pressurized-Water Reactors," dated June 9, 2003 (NRCB, 2003). NEI had forwarded 73 industry questions and comments on the bulletin, to which the staff responded in a handout distributed at this meeting. The effect of the bulletin on the overall GSI-191 resolution schedule was also raised by the public. All meeting material was attached to the meeting summary dated August 12, 2003 (Mtg, 2003c).

On July 1, 2003, a separate public meeting was held between the staff, NEI, and industry representatives. Sections of the draft methodology guidance were presented to the staff. The staff discussed progress in four major regulatory areas: RG 1.82, Revision 3 (issuance), head loss task report, debris characterization project, and chemical effects testing. The credibility of metal corrosion, precipitation of low solubility lead, and significant head loss effects from fiber debris beds, were also addressed. The public raised the question of ranking the plants' susceptibility to sump blockage; to which the staff replied that no ranking was intended beyond the parametric study results for 69

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"cases" already issued. The associated meeting summary is dated August 11, 2003 (Mtg, 2003d).

The NRC participated in a public workshop on Debris Impact on ECCS Recirculation held in Baltimore, MD, on July 30 and 31, 2003, where NRC and LANL presented material on sump evaluation methodology and the use of computer codes and volunteer plant studies in sump evaluation analyses. The NRC presentations were documented (Wkshp, 2003).

A public meeting between the staff, NEI, and industry representatives was held on September 10, 2003, the results of which were documented in a meeting summary dated October 16, 2003 (Mtg, 2003e). The NRC staff expressed concern over chemical effects on sump screen blockage based on testing. NEI and the industry also presented material on chemical effects. Considerable discussion centered on the formation of gelatinous material due to chemical effects.

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On October 31, 2003, NEI submitted to the staff the "PWR Containment Sump Evaluation Methodology" (NEI, 2003b). The staff provided to NEI a preliminary review of the October 31, 2003, submittal, by letter dated February 9, 2004 (NRC, 2004a). The staff transmitted two requests for additional information (RAIs) by electronic mail to NEI on March 10, 2004, and June 28, 2004, respectively. The staff met with NEI and stakeholders in a public meeting on March 23 and 24, 2004, to discuss the draft submittal and the March 10, 2004, RAIs. The results of this meeting are described in a meeting summary dated April 22, 2004 (Mtg, 2004a). NEI responded to the staff's RAIs by letters dated June 10, 2004 (NEI, 2004c), and July 8, 2004 (NEI, 2004d), respectively.

On April 19, 2004, NEI submitted to the staff a preliminary version of a Baseline Evaluation Method (NEI, 2004b), or Section 3.0 of the proposed GR. On May 28, 2004, NEI submitted the final version of the "PWR Containment Sump Evaluation Methodology" (NEI, 2004a), including a revised Section 3.0, and including a draft version of Section 6.0. On July 7, 2004, NEI provided the staff with a "Table of Refinements," via electronic mail, clarifying what refinements were being offered in the GR. On July 13, 2004, NEI submitted a final version of the Risk-Informed Section, or Section 6.0 (NEI, 2004e) of the GR.

NEI submitted a total of three draft versions of the GR, which were reviewed by the staff. They are: a draft of key sections of the evaluation guidance submitted July 1, 2003 (NEI, 2003a); a first draft of the "PWR Containment Sump Evaluation Methodology," submitted October 31, 2003 (NEI, 2003b); and a preliminary version of the current Baseline Evaluation Method, or Section 3.0 of the proposed GR, submitted April 19, 2004 (NEI, 2004b). The final GR was submitted to the NRC staff for review on May 28, 2004 (except Section 6.0, which was submitted to the staff on July 13, 2004), and is the subject of this safety evaluation. The final GR provides "baseline" guidance to utilities for evaluating plant-specific issues of pipe break selection, debris generation, latent debris, debris transport, sump screen head loss, and ECCS pump NPSH. In addition the GR provides "supplemental" guidance that can be used by licensees to refine their analysis and evaluations. The GR baseline guidance does not provide detailed guidance for several important related issues, including long-term chemical effects and head-loss correlations for particular insulation materials (e.g., calcium silicate), nor does it provide guidance for evaluating the impacts of debris passing through the screens and being ingested into the ECCS (downstream effects). The GR does note that licensees must consider these additional elements in the overall performance evaluation in their plant-specific analysis.

The process used between the industry and the staff involved (1) direct discussions between the industry and the staff on key issues, (2) the NRC staff's independent research in support of the GSI-191 resolution effort, and (3) the submittal by NEI of three separate versions of the GR, which significantly contributed to the development of the technical basis for an acceptable methodology which is described in this SER.

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Acronym List

ACRS	Advisory Committee on Reactor Safeguards
AJIT	air jet impact test
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
BWR	boiling water reactor
BWROG	Boiling Water Reactor Owners' Group
B&W	Babcock and Wilcox
CalSil	calcium silicate
CDF	core damage frequency
CFD	computational fluid dynamics
CP	corrosion products
CS	containment spray
CSS	containment spray system
DBA	design basis accident
DGBS	"debris generation" break size
DDTS	Drywell Debris Transport Study
DEGB	double-ended guillotine break
DPSC	Diamond Power Specialty Co.
ECC	emergency core cooling
ECCS	emergency core cooling system
GDC	General Design Criteria
GR	NEI PWR Sump Performance Evaluation Methodology guidance report
GSI	Generic Safety Issue
HELB	high-energy line break
HPSI	high-pressure safety injection
IEF	initiating event frequency
IOZ	inorganic zinc
LANL	Los Alamos National Laboratory
LBB	leak before break
LBLOCA	large break loss of coolant accident
LDFG	low density fiberglass
LOCA	loss-of-coolant accident

NEI	Nuclear Energy Institute
NIST	National Institute for Standards and Technology
NPSH	net positive suction head
NRC	Nuclear Regulatory Commission
PE	Parametric Evaluation
PWR	pressurized water reactor
RAI	Request for Additional Information
RCS	Reactor Coolant System
RG	Regulatory Guide
RMI	reflective metal insulation
SEM	scanning electron microscope
SER	Safety Evaluation Report
SMC:FP	sump mitigation capability failure probability
SRP	Standard Review Plan
SS	stainless steel
TMI	Three Mile Island
TPI	Transco Products, Inc.
TR	target reliability
UNM	University of New Mexico
ZOI	zone of influence

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO NRC GENERIC LETTER 2004-02,

NUCLEAR ENERGY INSTITUTE

GUIDANCE REPORT (PROPOSED DOCUMENT NUMBER NEI 04-07)

"PRESSURIZED WATER REACTOR SUMP PERFORMANCE EVALUATION METHODOLOGY"

1.0 INTRODUCTION

By letter dated May 28, 2004, the Nuclear Energy Institute (NEI) submitted for review by the U.S. Nuclear Regulatory Commission (NRC, the staff) a document entitled (proposed document number NEI 04-07,) "Pressurized Water Reactor Sump Performance Evaluation Methodology" (NEI, 2004a), herein referred to as the guidance report (GR). NRC approval of the GR would allow licensees of pressurized water reactors (PWRs) to use the GR in their response to NRC Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors" (GL-04-xx), as the cited "NRC approved methodology" for their evaluation of plant-specific sump performance. The Generic Letter identifies inadequacies in many of the current PWR licensing-basis analyses for modeling sump screen debris blockage and related effects, such that the staff no longer considers those analyses acceptable for confirming compliance with NRC regulations. The NEI GR offers guidance to all PWR licensees in response to those inadequacies raised during resolution of Generic Safety Issue (GSI) 191, "Assessment of Debris Accumulation on PWR Sump Performance," which are documented in the Generic Letter.

The staff has completed its review of the GR and associated documentation, and the conclusions are documented in this safety evaluation report (SER). In general, the staff found that portions of the GR are acceptable for use in conducting plant-specific analyses of emergency core cooling system (ECCS) sump screen blockage and resultant ECCS and/or core spray system (CSS) loss of net positive suction head (NPSH) for pumps required following a loss-of-coolant-accident (LOCA). However, the staff found that several portions of the GR are not acceptable because the proposed methods lack sufficient guidance, supporting data, or analysis to justify their technical basis. For each of these areas, the staff has provided a recommendation and/or alternative guidance to that offered in the GR. This SER addresses each section of the GR, discusses the staff's evaluation of the proposed methodologies, and documents the basis for the staff's conclusions.

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This SER addresses each part of a plant-specific analysis of sump performance, and is organized so that its discussions parallel the guidance discussions presented in the GR. The SER includes sections on each of the following topics:

- Pipe Break Characterization (Section 3.3)

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- Debris Generation/Zone of Influence (Section 3.4)
- Latent Debris (Section 3.5)
- Debris Transport (Section 3.6)
- Head Loss (Section 3.7)
- Analytical Refinements (Section 4.0)
- Design and Administrative Control Refinements (Section 5.0)
- Debris Source Term Refinements (Section 5.1)
- Refinements by use of Debris Transport Obstructions (Section 5.2)
- Refinements via Sump Screen Modifications (Section 5.3)
- Risk-Informed Evaluation (Section 6.0)
- Sump Structural Analysis (Section 7.1)
- Upstream Effects (Section 7.2)
- Downstream Effects (Section 7.3)
- Chemical Effects (Section 7.4)

1.1 BACKGROUND

In 1979, Unresolved Safety Issue (USI) A-43, "Containment Emergency Sump Performance," was established as a result of evolving staff concerns related to the adequacy of PWR recirculation sump designs. After extensive research, the staff found that the design assumption of 50 percent sump blockage used by licensees was nonconservative under certain conditions, and published the technical findings in NUREG-0897, "Containment Emergency Sump Performance," dated October 1985. Although the staff's regulatory analysis concerning USI A-43 did not support imposing new sump performance requirements, the staff issued GL 85-22, "Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage," dated December 3, 1985. GL 85-22 documented the resolution of USI A-43, recommending that all reactor licensees replace the 50 percent blockage assumption with a comprehensive mechanistic assessment of plant-specific debris blockage potential for future modifications related to sump performance, such as thermal insulation changeouts. The staff also updated the NRC's regulatory guidance, including Section 6.2.2 of the Standard Review Plan (NUREG-0800) and Regulatory Guide (RG) 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident" (RG 1.82), to reflect the USI A-43 technical findings documented in NUREG-0897.

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Following the resolution of USI A-43 in 1985, several events challenged the staff's conclusion that no new requirements were necessary to prevent the clogging of ECCS strainers at operating BWRs:

- On July 28, 1992, at Barseback Unit 2, a Swedish BWR, the spurious opening of a pilot-operated relief valve led to the plugging of two containment vessel spray

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system suction strainers with mineral wool and required operators to shut down the spray pumps and backflush the strainers.

- In 1993, at Perry Unit 1, ECCS strainers twice became plugged with debris. On January 16, ECCS strainers were plugged with suppression pool particulate matter, and on April 14, an ECCS strainer was plugged with glass fiber from ventilation filters that had fallen into the suppression pool. On both occasions, the affected ECCS strainers were deformed by excessive differential pressure created by the debris plugging.
- On September 11, 1995, at Limerick Unit 1, following a manual scram due to a stuck-open safety/relief valve, operators observed fluctuating flow and pump motor current on the "A' loop of suppression pool cooling. The licensee later attributed these indications to a thin mat of fiber and sludge which had accumulated on the suction strainer.

In response to these ECCS suction strainer plugging events, the NRC issued several generic communications, including Bulletin 93-02, Supplement 1, "Debris Plugging of Emergency Core Cooling Suction Strainers," dated February 18, 1994; Bulletin 95-02, "Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode," dated October 17, 1995; and Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors," dated May 6, 1996. Through these bulletins the staff requested that BWR licensees implement appropriate procedural measures, maintenance practices, and plant modifications to minimize the potential for the clogging of ECCS suction strainers by debris accumulation following a loss-of-coolant accident (LOCA). Bulletin 96-03, in particular, noted the experience-based finding that clogging by fibrous debris is not limited to fibrous insulation as a debris source. These bulletins were adequately addressed by all BWR licensees.

However, findings from research to resolve the BWR strainer clogging issue in the 1990s raised questions concerning the adequacy of PWR sump designs by confirming what the aforementioned BWR strainer clogging events had earlier indicated: (1) that the amount of debris generated by a HELB could be greater than estimated by the USI A-43 research program, (2) that the debris could be finer (and, thus, more easily transportable), and (3) that certain combinations of debris (e.g., fibrous material plus particulate material) could result in a substantially greater head loss than an equivalent amount of either type of debris alone. Therefore, in 1996 the staff identified GSI-191, to ensure that post-accident debris blockage would not impede or prevent the operation of the ECCS and CSS in the recirculation mode at PWRs in the event of a LOCA or other HELB accidents for which sump recirculation is required. The staff began evaluating the potential vulnerability of PWRs and contracted LANL to evaluate the potential for debris to cause degraded PWR recirculating sump performance. In July 2001, preliminary parametric calculations were completed on PWR sump performance, which confirmed the potential for debris accumulation in a representative number of operating PWRs. A number of studies (e.g., NUREG/CR-6771, LA-UR-02-7562) have been performed to evaluate the potential for sump clogging and the concerns associated with generic safety

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issue GSI-191. Designing the containment sump so that it is not susceptible to clogging has been generically estimated in the above studies to reduce the risk associated with large break LOCAs by a factor of 45. Using the conservative NUREG-1150 LBLOCA frequency (5E-4/year) in the generic calculation results in a risk reduction from 1.6E-4/year to 3.6E-6/year. Using a current (more realistic) LBLOCA frequency (4E-6/year) would result in a risk reduction from 1.2E-6/year to 2.6E-8/year.

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On June 9, 2003, having completed its technical assessment of GSI-191 (summarized below in the Overview section), the NRC issued Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at Pressurized-Water Reactors," requesting an expedited response from PWR licensees regarding the status of their compliance with regulatory requirements concerning the ECCS and CSS recirculation functions. PWR licensees unable to assure regulatory compliance pending further analysis were asked to describe any interim compensatory measures that have been implemented or will be implemented to reduce risk until the analysis could be completed. All PWR licensees have since responded to Bulletin 2003-01.

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In developing Bulletin 2003-01, the NRC staff recognized that it may be necessary for PWR licensees to undertake complex evaluations to determine whether regulatory compliance exists in light of the concerns identified in the bulletin and that the methodology to perform such evaluations was not currently available. As a result, that information was not requested in the bulletin but PWR licensees were informed that the staff was preparing a generic letter that would request this information. On September 13, 2004, the staff issued GL 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors."

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1.2 OVERVIEW

In the event of a HELB inside the containment of a PWR, energetic pressure waves and fluid jets would impinge upon materials in the vicinity of the break, such as thermal insulation, coatings, and concrete, causing them to become damaged and dislodged. Debris could also be generated through secondary mechanisms, such as severe post-accident temperature and humidity conditions, flooding of the lower containment, and the impact of containment spray droplets. In addition to debris generated by jet forces from the pipe rupture, debris can be created by the chemical reaction between the chemically reactive spray solutions used following a LOCA and the materials in containment. These reactions may result in additional debris such as disbonded coatings and chemical precipitants being generated. Through transport methods such as entrainment in the steam/water flows issuing from the break and containment spray washdown, a fraction of the generated debris and foreign material in the containment would be transported to the pool of water formed on the containment floor. Subsequently, if the ECCS or CSS pumps were to take suction from the recirculation sump, the debris suspended in the containment pool would begin to accumulate on the sump screen or be transported through the associated system. The accumulation of this suspended debris on the sump screen could create a roughly uniform covering on the screen, referred to as a debris bed, which would tend to increase the head loss across the screen through a filtering action. If a sufficient amount of debris were to accumulate, the debris bed would reach a critical thickness at which the head loss across the debris bed would exceed the net positive suction head (NPSH) margin required to ensure the successful operation of the ECCS and CSS pumps in recirculation mode. A loss of NPSH margin for the ECCS or CSS pumps as a result of the accumulation of debris on the recirculation sump screen, referred to as sump clogging, could result in degraded pump performance and eventual pump failure. Debris could also plug or wear close tolerance components within the ECCS or CSS systems. The effect of this plugging or wear may cause a component to degrade to the point where it may be unable to perform its designated function (e.g. pump fluid, maintain system pressure, or pass and control system flow).

Assessing the likelihood of the ECCS and CSS pumps at domestic PWRs experiencing a debris-induced loss of NPSH margin during sump recirculation was the primary objective of the NRC's technical assessment of GSI-191. The NRC's technical assessment culminated in a parametric study that mechanistically treated phenomena associated with debris blockage using analytical models of domestic PWRs generated with a combination of generic and plant-specific data. As documented in Volume 1 of NUREG/CR-6762, "GSI-191 Technical Assessment: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance," dated August 2002 (NUREG/CR-6762-1), the GSI-191 parametric study concludes that recirculation sump clogging is a credible concern for domestic PWRs. As a result of limitations with respect to plant-specific data and other modeling uncertainties, however, the parametric study does not definitively identify whether or not particular PWR plants are vulnerable to sump clogging when phenomena associated with debris blockage are modeled mechanistically.

The methodology employed by the GSI-191 parametric study is based upon the substantial body of test data and analyses that are documented in technical reports generated during the NRC's GSI-191 research program and earlier technical reports generated by the NRC and the industry during the resolution of the BWR strainer clogging issue and USI A-43. These pertinent technical reports, which cover debris generation, transport, accumulation, and head loss, are referenced in the GSI-191 parametric study:

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- NUREG/CR-6770, "GSI-191: Thermal-Hydraulic Response of PWR Reactor Coolant System and Containments to Selected Accident Sequences," dated August 2002.
- NUREG/CR-6762, Vol. 3, "GSI-191 Technical Assessment: Development of Debris Generation Quantities in Support of the Parametric Evaluation," dated August 2002.
- NUREG/CR-6762, Vol. 4, "GSI-191 Technical Assessment: Development of Debris Transport Fractions in Support of the Parametric Evaluation," dated August 2002.
- NUREG/CR-6224, "Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris," dated October 1995.

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In light of the new information identified during the efforts to resolve GSI-191, the NRC staff determined that the previous guidance used to develop current licensing-basis analyses did not adequately and completely model sump screen debris blockage and related effects. As a result, due to the deficiencies in the previous guidance, an

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analytical error could be introduced which results in ECCS and CSS performance that does not conform with the existing applicable regulatory requirements outlined in GL 04-02. Therefore, the staff revised its guidance for determining the susceptibility of PWR recirculation sump screens to the adverse effects of debris blockage during design basis accidents requiring recirculation operation of the ECCS or CSS (RG 1.82-3). The NRC staff determined that it was appropriate to request that addressees perform new, more realistic analyses and submit information to confirm their plant-specific compliance with NRC regulations and other existing regulatory requirements listed in GL-04-02 pertaining to post-accident debris blockage.

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In addition to demonstrating the potential for debris to clog containment recirculation sumps, operational experience and the NRC's technical assessment of GSI 191 have also identified three integrally related modes by which post-accident debris blockage could adversely affect the sump screen's design function of intercepting debris that could impede or prevent the operation of the ECCS and CSS in recirculation mode.

First, as a result of the 50-percent blockage assumption, most PWR sump screens were designed assuming that relatively small structural loadings would result from the differential pressure associated with debris blockage. Consequently, PWR sump screens may not be capable of accommodating the increased structural loadings that would occur due to mechanistically determined debris beds that cover essentially the entire screen surface. Inadequate structural reinforcement of a sump screen may result in its deformation, damage, or failure, which could allow large quantities of debris to be ingested into the ECCS and CSS piping, pumps, and other components, potentially leading to their clogging or failure. The ECCS strainer plugging and deformation events that occurred at Perry Unit 1 (further described in Information Notice (IN) 93-34, "Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post LOCA Debris in Containment," dated April 26, 1993, and LER 50 440/93-011, "Excessive Strainer Differential Pressure Across the RHR Suction Strainer Could Have Compromised Long Term Cooling During Post LOCA Operation," submitted May 19, 1993), demonstrate the credibility of this concern for screens and strainers that have not been designed with adequate reinforcement.

Second, in some PWR containments, the flowpaths by which containment spray or break flows return to the recirculation sump may include "choke-points," where the flowpath becomes so constricted that it could become blocked with debris following a HELB. Examples of potential choke-points are drains for pools, cavities, isolated containment compartments, and constricted drainage paths between physically separated containment elevations. Debris blockage at certain choke-points could hold up substantial amounts of water required for adequate recirculation or cause the water to be diverted into containment volumes that do not drain to the recirculation sump. The holdup or diversion of water assumed to be available to support sump recirculation could result in an available NPSH for ECCS and CSS pumps that is lower than the analyzed value, thereby reducing assurance that recirculation would successfully function. A reduced available NPSH directly concerns sump screen design because the NPSH margin of the ECCS and CSS pumps must be conservatively calculated to determine correctly the required surface area of passive sump screens when mechanistically determined debris loadings are considered. Although the parametric study (NUREG/CR-6762, Volume 1) did not analyze in detail the potential for the holdup or diversion of recirculation sump inventory, the NRC's GSI 191 research identified this phenomenon as an important and potentially credible concern. A number of LERs associated with this concern have also been generated, which further confirms its credibility and potential significance:

- LER 50-369/90-012, “Loose Material Was Located in Upper Containment During Unit Operation Because of an Inappropriate Action,” McGuire Unit 1, submitted August 30, 1990.
- LER 50-266/97-006, “Potential Refueling Cavity Drain Failure Could Affect Accident Mitigation,” Point Beach Unit 1, submitted February 19, 1997.
- LER 50-455/97-001, “Unit 2 Containment Drain System Clogged Due to Debris,” Byron Unit 2, submitted April 17, 1997.
- LER 50-269/97-010, “Inadequate Analysis of ECCS Sump Inventory Due to Inadequate Design Analysis,” Oconee Unit 1, submitted January 8, 1998.
- LER 50-315/98-017, “Debris Recovered from Ice Condenser Represents Unanalyzed Condition,” D.C. Cook Unit 1, submitted July 1, 1998.

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Third, debris blockage at flow restrictions within the ECCS recirculation flowpath downstream of the sump screen is a potential concern for PWRs. Debris that is capable of passing through the recirculation sump screen may have the potential to become lodged at a downstream flow restriction, such as a high-pressure safety injection (HPSI) throttle valve or fuel assembly inlet debris screen. Debris blockage at such flow restrictions in the ECCS flowpath could impede or prevent the recirculation of coolant to the reactor core, thereby leading to inadequate core cooling. Similarly, debris blockage at flow restrictions in the CSS flowpath, such as a containment spray nozzle, could impede or prevent CSS recirculation, thereby leading to inadequate containment heat removal. Debris may also accumulate in close tolerance sub-components of pumps and valves. The effect may either be to plug the sub-component thereby rendering the component unable to perform its function or to wear critical close tolerance sub-components to the point at which component or system operation is degraded and unable to fully perform its function. Considering the recirculation sump screen’s design function of intercepting potentially harmful debris, it is essential that the screen openings are adequately sized and that the sump screen’s current configuration is free of gaps or breaches which could compromise the ECCS and CSS recirculation functions. It is also essential that system components are designed and evaluated to be able to operate with debris laden fluid as necessary post-LOCA.

To assist in determining, on a plant-specific basis, whether compliance exists with 10 CFR 50.46(b)(5), licensees may use the guidance contained in RG 1.82, Revision 3, “Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident,” dated November 2003. Revision 3 enhanced the debris blockage evaluation guidance for PWRs provided in Revision 1 of the RG to better model sump screen debris blockage and related effects. The NRC staff determined after the issuance of Revision 2, that research for PWRs indicated that the guidance in that revision was not comprehensive enough to ensure adequate evaluation of a PWR plant’s susceptibility to the detrimental effects caused by debris accumulation on debris interceptors (e.g., trash racks and sump screens). Revision 2 altered the debris blockage evaluation guidance found in Revision 1 following the evaluation of blockage events, such as the Barsebäck Unit 2 event mentioned above, but for BWRs only. Revision 1 replaced the 50-percent blockage assumption in Revision 0 with a comprehensive, mechanistic assessment of plant-specific debris blockage potential for future modifications related to sump performance, such as thermal insulation changeouts. This was in response to the findings of USI A-43.

The NEI GR expands on RG 1.82, Rev. 3 (requirements for long-term cooling), using portions of NUREG/CR-6808 (knowledge-base report) and other NRC and industry related documents. The NEI research contributions are (1) in the area of alternate break size, including options for risk-informing the analysis as it relates to the initial postulated break size, and (2) on the behavior of protective coatings (a potential debris type) under high-pressure, two-phase jet impact.

In support of the GSI-191 resolution effort, the staff also conducted research which was not completed, for a plant-specific sump performance analysis based on sample plant data. Although the work was not published, some of the work was completed and simply not documented. Therefore, the staff has provided results from specific areas of this research, to supplement areas in the GR that lack supporting data and experimentation, as a basis for alternative guidance and has provided details in such cases, in Appendices III and VI to this SER.

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2.0 REGULATORY EVALUATION

This section details the regulatory requirements, associated guidance, and precedent upon which the staff based its review of the GR submitted by NEI to be used for the evaluation of PWR sump recirculation performance.

In accordance with Title 10 of the Code of Federal Regulations (10 CFR) Part 50.46, Sub-section (b) (5), licensees of domestic nuclear power plants are required to provide long-term cooling of the reactor core. Specifically, that regulation provides that "after any calculated successful initial operation of the ECCS", the "calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core." For this evaluation of PWR recirculation performance, the staff has considered this extended time to be thirty days, and requires cooling by recirculation of coolant via the ECCS sump, where coolant is accumulated for this purpose. However, if debris collects and clogs the sump screen or other components or pathways that prevent adequate suction for ECCS or CSS pumps, then compliance with this regulation may be in question.

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Guidance for determining compliance with 10 CFR 50.46(b) (5), is contained in RG 1.82, Revision 3. The staff review guidance for evaluating licensee compliance with 10 CFR 50.46(b) (5), is contained in NUREG-0800, "Standard Review Plan" (SRP) 6.2.2, "Containment Heat Removal Systems." Additionally, SRP 6.1.1, "Engineered Safety Features Materials," provides the review process for thermal insulation and coating systems, which impact long-term cooling evaluation; SRP 9.2.5, "Ultimate Heat Sink," provides review guidance from which the extended time for recirculation performance is derived; and SRP 6.1.2, "Protective Coating Systems (Paints)," provides review guidance for coating systems, which is a debris type generated and evaluated in the sump analysis.

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For PWRs licensed to General Design Criteria (GDC) in Appendix A to 10 CFR Part 50, GDC 35 specifies additional ECCS requirements, GDC 38 specifies heat removal systems requirements, and GDC 41 provides requirements for containment atmosphere cleanup. Many PWR licensees credit a CSS, at least in part, with performing the safety functions to satisfy these requirements, and PWRs that are not licensed to the GDC may credit a CSS to satisfy similar plant-specific licensing basis requirements. In addition, PWR licensees may credit a CSS with reducing the accident source term to meet the limits of 10 CFR Part 100 or 10 CFR 50.67.

Technical specifications pertain to the ECCS and CSS insofar as they require the operability of these systems for the mitigation of certain design basis accidents. Other plant-specific licensing commitments concerning the ECCS and CSS are also documented in the Final Safety Analysis Report.

The staff considered the NRC's August 28, 1998, SER on the Utility Resolution Guidance (URG) for ECCS Suction Strainer Blockage (NEDO-32686-A), (URG SER) used for resolution of the related strainer blockage issue for BWRs in its evaluation of the GR. This approach helped to assure consistency and efficiency. In some areas, departures from the GR and the URG SER were warranted due to differences in the design features of BWRs and PWRs, as well as later information obtained through regulatory research.

The Commission's staff requirements memorandum from A. L. Vietti-Cook to L. A. Reyes, SECY-04-0037, "Issues Related to Proposed Rulemaking to Risk-Inform Requirements Related to Large Break Loss-of-Coolant-Accident (LOCA) Break Size and Plans for Rulemaking on LOCA with Coincident Loss-of-Offsite-Power," dated July 1, 2004 (SECY-04-0037) and RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated July 1998 (RG 1.174), were considered in the review of industry-proposed alternatives, and in the realistic and risk-informed options with regard to break size selection and mitigative equipment requirements.

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3.0 BASELINE EVALUATION

Section 3 of the GR provides an evaluation methodology referred to as a baseline set of methods that help identify the dominant design factors for a given plant. The baseline evaluation methodology is intended to provide an approach which includes sufficient conservatism such that simpler analytical methods can be used.

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3.1 INTRODUCTION

Section 3.1 of the GR describes the purpose of the baseline, and it presents background information regarding general accident scenarios of concern and accident phenomena. This section also notes the limitations of the evaluation method. It makes reference to supplemental guidance for refinements, and data collection to support base evaluations.

Key introductory points include the following:

1. This section states: "If a plant uses this method and guidance to determine that sufficient head loss margin exists for proper long-term Emergency Core Cooling (ECC) and Containment Spray (CS) function, no additional evaluation for head loss is required."
2. The baseline evaluation method only addresses the phenomena and issues up to and including head loss across the sump screen. Insufficient information presently exists to evaluate the effects of chemical reaction products on head loss across a sump screen and the associated debris bed. Also, the Baseline Methodology does not include the evaluation of holdup of flow by debris upstream of the sump screen, the structural integrity of the sump screen, or the effects resulting from debris passing through the sump screen and being ingested into the ECC or CS systems.
3. The baseline evaluation guidance provides a conservative approach for evaluating the generation and transport of debris, and the resulting head loss across the sump screen. If a plant determines that the results of the baseline approach are not acceptable, or additional design margin is desirable, the refinement guidance provided in subsequent sections may be used to further evaluate the post-accident performance of the ECC sump.

Staff Evaluation for Section 3.1: The baseline guidance acknowledges that the chemical reaction product effects on head loss, downstream effects, and upstream effects are not fully considered in the baseline evaluation methodology. However, the guidance does not make it explicitly clear that the plant must still address these issues even if the plant successfully applied the baseline method to their plant. Therefore, the staff position is that licensees address these effects in accordance with the staff positions specified in Section 7.0 of this SER.

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The staff questions the GR statement that the baseline provides a conservative approach. Aspects of the baseline guidance have been identified that are clearly not conservative while other aspects are conservative. The subject aspects are identified at the appropriate locations in this SER. Acceptance of the baseline evaluation requires that the baseline approach results in an evaluation that, overall, is realistically

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conservative. The staff has sponsored research to confirm whether or not specific aspects of the guidance are truly conservative as stated by the guidance. Results of this research are included in Appendices I, II, IV, and V to this SER, and are referenced appropriately in the pertinent section of this document. Section 3.8 documents the staff evaluation of assumptions for which conservatism is in question, and provides alternative guidance toward ensuring an overall realistic conservatism for the baseline.

3.2 METHOD OVERVIEW

Section 3.2 presents the five major areas of the baseline guidance as break selection, debris generation, latent debris, debris transport, and head loss.

3.3 BREAK SELECTION

This section of the GR presents considerations and guidance for selecting an appropriate postulated break size and location for use in the baseline analysis. The stated objective of the selection process is to identify the break conditions that present the greatest challenge to post-accident sump performance.

The staff review resulted in two exceptions to the proposed GR guidance for break selection. These two exceptions involved the treatment of secondary side breaks and the guidance for plants that can substantiate that no thin bed develops. A discussion on the evaluation of secondary side breaks is included in this section of the SER. Discussions regarding guidance for thin bed considerations are included in other sections of this SER, including those on debris generation, latent debris, transport and head loss. Additionally, Appendix VIII of this SER provides a description of a thin bed, including its formation and effects.

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3.3.1 Introduction

Break selection is described in the GR as a two-step process involving selection of (1) the size of the break and (2) the location of the break.

Staff Evaluation of Section 3.3.1: The staff notes that DEGB breaks need to be assumed for the baseline analysis of primary system piping (GR section 3.3.3), so the size of the break is then determined by the diameter of the pipe. Other break-size criteria may be adopted for postulated breaks in secondary piping depending on assumptions in the plant licensing basis.

The GR states that the objective of the break selection process is to identify the break size and location that results in debris generation that is determined to produce the maximum head loss across the sump screen. The staff finds this objective to be acceptable. Because the assessment will address several complex phenomena for each break location, the location of the most challenging break cannot be identified with confidence until a number of postulated-break locations have been evaluated.

3.3.2 Discussion

As stated in the GR, the criterion used to define the most challenging break conditions is the estimated head loss across the sump screen. The break location that maximizes estimated head loss is referred to in the GR as the "limiting break location." All phases of the accident scenario must be considered for each postulated break location including debris generation, debris transport, and sump-screen head loss calculations. The outcome of head loss predictions from each candidate break location should be performed systematically, and should be self-contained.

Two attributes of break selection which are emphasized in the GR that can contribute to head loss are (1) the maximum amount of debris transported to the screen and (2) the worst combination of debris mixes that are transported to the screen. The proper metric for comparison, head-loss effect upon arrival at the screen, has been emphasized. The GR requires that break locations be surveyed to provide for both items 1 and 2 because under given circumstances, either could represent the limiting break. For example, relatively small quantities of fiber in combination with LOCA-generated or latent-debris particulate can induce head losses that exceed the effects of much larger debris beds. Regulatory Guide 1.82, Rev. 3 [RG 1.82-3] itemizes additional features of a break that may dominate effects on the screen, but these two criteria stated in the GR encompass quantity, type, transport and mixed composition as key issues.

3.3.3 Postulated Break Size

Staff Evaluation of Section 3.3.3: The NRC agrees that double-ended guillotine breaks (DEGB) with full piping separation and offset should be used for baseline evaluation of LOCA debris generation for breaks assumed to occur in primary system piping (RCS main loop piping and attached auxiliary piping). For plants that require recirculation to maintain long-term cooling after secondary-system pipe ruptures, either DEGB conditions may be assumed or conditions consistent with the plant's licensing basis for those breaks may be used for size characterization (typically, a spectrum of break sizes is evaluated, up through a double ended rupture). The staff finds the GR guidance with respect to break size is acceptable because this approach provides for large volumes of debris and worst combinations of debris.

3.3.4 Identifying Break Locations

Staff Evaluation of Section 3.3.4: In accordance with 10 CFR 50.46, the NRC agrees that all reactor coolant system (RCS) piping, and connected piping, must be considered in the evaluation of locations to identify the limiting break. As stated in the GR, some plant designs require eventual coolant recirculation from the sump for pipe ruptures other than a LOCA. If recirculation is required under the plant licensing basis to mitigate these events, then breaks must be examined in this piping as well. Any actuation of the recirculation pumps implies an initiating event that should be examined for potential debris generation regardless of whether the recirculation supplies containment spray or safety injection systems.

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3.3.4.1 General Guidance

The staff position is provided here for each of the seven principles of break selection guidance offered in the GR.

1. The GR states that break exclusion zones must be disregarded for this evaluation. The staff finds this to be acceptable because all piping locations should be considered. The GR also states that for main steam and feedwater line breaks, licensees should evaluate the licensing basis and include potential break locations in the evaluation, if necessary. The staff finds this to be acceptable. However, the staff position is that if secondary breaches such as main steam line and feedwater line breaks rely on sump recirculation, as described in the plant licensing basis, breaks should be postulated in these systems at locations chosen in a manner consistent with the remaining guidance in this section.

2. The GR states that application of NRC Branch Technical Position MEB 3-1 is not appropriate for determining potential LOCA break locations. The staff finds this to be acceptable (see section 4.2.1 of this SER for a more detailed discussion of the staff position).

3. The GR states that for plants for which secondary-system breaks such as main steam line and feedwater line breaks rely on sump recirculation as described in the licensing basis, postulated break locations should be consistent with the plant's current licensing basis. The staff finds this position to be unacceptable. The staff position is that secondary side break locations should be postulated in a manner consistent with the remaining guidance in this section. The reason supporting this position is that inclusion of secondary-break scenarios in the licensing basis acknowledges the possible need for recirculation, but the break locations evaluated in the licensing basis may not have been defined specific to sump performance and could not have anticipated the range of concerns identified in the course of resolving GSI-191. Although secondary side breaks are not analyzed in accordance with the requirements of 10 CFR 50.46 or to demonstrate compliance with 10 CFR 50.46, the staff's position is that licensees relying on the ECCS sumps to mitigate the consequences of secondary side breaks, e.g., for EEQ purposes, should identify and evaluate the limiting break locations to ensure acceptable sump performance.

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4. The GR recommends that pipe breaks be postulated at locations that result in unique debris source terms to avoid multiple locations with identical composition and quantity of debris. The staff considers that in order to assess the potential head loss on the sump screen, the break location must also be judged based on the degree of transport that is expected. Licensees may analyze the first few break locations in full detail, quantifying all phases of the accident sequence. Additional breaks may then be evaluated by comparing debris composition, debris quantity and debris transport potential. This approach will avoid some duplication of effort and will permit a systematic survey of break locations.

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5. The GR states that pipe breaks shall be postulated that affect locations containing high concentrations of problematic insulation (microporous insulation, calcium-silicate, fire barrier material, etc.). The staff finds this position to be acceptable. Additionally, in keeping with the objective of identifying limiting break conditions, zones of problematic insulation might be affected by smaller breaks in their vicinity or by larger breaks that encompass

them. Both possibilities should be considered because the overall composition of the debris arriving at the screen may be different.

6. As discussed above, the initial quantity and composition of the debris source are important attributes of break selection, but potential transport must be considered also. The GR states that "Pipe breaks shall be postulated with the goal of creating the largest quantity of debris and/or the worst-cast combination of debris types at the sump screen." The staff agrees that these conditions should be evaluated. The GR correctly notes that the largest quantity at the screen may not produce the highest head loss. Additional discussion of screen head loss analysis found in Section 3.7 of the SER may help guide the selection of break locations that may create adverse conditions at the sump screen.

7. The GR proposes that piping less than 2 inches in diameter need not be considered in order to identify the limiting break conditions. The staff finds this to be acceptable. While it may be possible for a 2-inch break to challenge net-positive-suction-head (NPSH) margins for some existing screens, larger breaks postulated with minimal transport would pose an identical challenge. Larger breaks with higher transport potential will certainly bound the maximum on-screen debris permitted by a 2-inch break. Eliminating 2-inch diameter breaks from the baseline greatly simplifies the systematic survey.

3.3.4.2 Piping Runs to Consider

The staff agrees that breaks, ruptures and leaks other than a LOCA will be considered in this analysis if these scenarios eventually require recirculation for any purpose and if they are part of the plant licensing basis.

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The staff's position is that all broken lines, regardless of piping system, that meet the following criteria should be considered: (1) incorporated in the licensing basis; (2) capable of generating debris; and (3) lead to a recirculation demand on the sumps. This position is not meant to imply that breaks must be fully analyzed in every length of every system. Many postulated locations will be eliminated by comparison with other collocated break possibilities of their respective debris volume, composition, and transport potential. Note that all piping in containment should be considered regardless of its location within containment because breaks in secondary systems may also be of interest if the above criteria for consideration are satisfied (e.g., main steam and feedwater piping).

The level of detail pursued in the application of breaks in alternative piping systems depends largely on assumptions made in other steps of the accident analysis. For example, if assumptions made in the transport and head loss analyses both require the assessment of thin-bed formation, then break selection can focus on (1) particulate sources that may contribute to the thin-bed, and (2) maximum debris quantities that may dominate the debris bed. An example of a case where detailed examination of an alternative system might be required is a high energy line with debris generation potential that is either insulated with or that might affect problematic or diverse insulation types in locations outside the range of larger pipe breaks. Locations of this type might be found in upper containment near component cooling lines near the pressurizer, for example. Scenarios of this type could be conservatively analyzed using bounding jet

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parameters relevant to the primary system piping or a new jet calculation could be performed specific to the conditions of the line in question. The actuation of spray for breaks postulated in alternative systems is also a key consideration in their assessment as potentially limiting conditions, because containment spray will enhance transport to the recirculation pool and to the sump screen. This discussion is intended to recognize that there may be candidate break locations outside of the larger break ZOI. Conversely, if such locations are already considered within larger postulated breaks with large ZOI, then detailed examination may not be required.

Note that the explicit assumption of thin-bed formation regardless of break size or location offers a significant simplification for break selection, because more focus can be placed on the larger piping systems that envelope more spatial volume. Breaks outside of the crane wall may require more detailed examination for pipe size, pipe pressure, nearby insulation types, and transport potential.

3.3.4.3 Other Considerations for Selecting Break Locations

Three additional considerations for selecting break locations are presented in the GR. The staff position regarding each respective consideration is discussed here.

1. The staff finds that the GR correctly emphasizes proper consideration of relative locations between the postulated break location and the affected containment material targets. Additionally, the staff notes that a good understanding of spatial volume obtained from the ZOI discussion in Section 3.4.2 of this SER and related calculations will assist in determining the level of detail needed for the break location survey.

2. The second consideration focuses on the potential for the formation of a thin fiber layer on the screen that filters particulates very efficiently, the so-called "thin-bed" effect. In general, state-of-the art debris transport methods are not sufficiently advanced to preclude the formation of a thin bed when fibrous insulation is damaged within any ZOI. The degree of vulnerability to this effect is specific to the sump screen in question. This GR consideration for break selection sets a marginal value for debris generation that might already be bounded by larger breaks with minimal transport. The staff agrees that the "thin-bed" effect should be evaluated. Additionally, the staff's position is that smaller breaks affecting unique combinations of insulation not encompassed by larger break should still be examined for potential thin-bed formation. When computing the volume of fibrous debris needed to form a 1/8-inch thick uniform layer on a given sump screen, the dry-bed or "as-manufactured" density should be used, and only the wetted screen area relevant to the break in question should be credited.

3. The GR offers an additional consideration that recognizes the importance of latent debris inventory as a potentially limiting debris source for plants with little or no fibrous insulation. The staff agrees with this consideration, and refers to Section 3.5 of this SER for a more complete discussion of latent debris characterization. The staff notes that the use of an appropriate dry-bed density for latent fiber and a wetted screen area can be used by plants with non-fiber insulation to establish a plant cleanliness criterion for their FME programs.

3.3.4.4 Selecting the Initial Break Locations

The staff finds that the guidance offered in the GR for initial break location selection is acceptable and notes that spatial perspectives gained from implementation of the ZOI models will be helpful at directing the break-location survey further. In general, the survey should first consider larger breaks with more complex debris composition and proceed down to smaller breaks with more unique debris compositions that have not yet been captured in the survey. The degree of transport, which can be affected by the use of containment spray, should be considered during the comparison of potential break locations. Starting with this initial break location and moving to other large breaks that envelope any previously identified debris-source concerns will quickly build a set of comparative source-term and transport factors that can be used to judge other locations and classes of postulated breaks without as much detailed quantification. Comparative rationale that disqualifies a candidate location from designation as a limiting break condition should be documented to illustrate the systematic and comprehensive scope of the break-selection survey.

3.3.5 Evaluation of Break Consequences

<u>Staff Evaluation of Section 3.3.5:</u> The staff finds that the proper metric of comparison between break locations has been emphasized in the GR, i.e., head loss across the sump screen as a result of generation, transport, and accumulation of debris on the sump screen. Break locations cannot be eliminated from consideration based on any single attribute alone. The staff agrees that all breaks should be evaluated in the context of the complete accident sequence and the potential effect on sump-screen head loss. Nevertheless, many comparisons will be found that are useful. For example, all large break locations within a compartment may be found to have similar transport characteristics and spatial volume, so only one or two locations within the compartment are needed to bound the variation in debris composition.

3.3.5.1 Purpose of Break Consequence Evaluation

Once the limiting break condition(s) have been identified, the corresponding head loss will be compared to the required NPSH either as a measure of vulnerability to sump blockage or as a design criterion for sump-screen modifications. The staff finds that the GR provides an acceptable and concise summary in this section of the steps involved with evaluating each candidate break location against the criterion of maximum sump-screen head loss.

3.3.5.2 Selection of Intervals for Additional Break Locations

This section of the GR describes a systematic approach to break selection along individual piping runs that starts at an initial location along a pipe, generally a terminal end, and steps along in equal increments (3 foot increments) placing breaks at each sequential location. The staff position is that break intervals can be relaxed to 5-ft increments along the pipe in question and notes that the concept of equal increments is only a reminder to be systematic and thorough. Earlier work reported by NRC contractors using automated analysis tools to evaluate higher spatial resolution (1 to 3 ft increments) was motivated by a risk assessment approach that required an accurate sampling of piping lengths and break sizes to represent their proportional contribution to the overall frequency of sump screen failure. For the purpose of identifying limiting break conditions, a more discrete approach driven by the comparison of debris source term and transport potential can be effective at placing postulated breaks. The key difference between many breaks (especially large breaks) will not be the exact location along the pipe, but rather the envelope of containment material targets that is affected.

The staff agrees that as the plant-specific analysis develops, many break locations along a pipe will be determined by inspection of potential debris inventory, similarity of transport paths, and piping physical characteristics compared to a smaller number of fully quantified break scenarios.

As discussed previously, the staff does not accept the GR position regarding the treatment of secondary break locations. The staff position is that if secondary break scenarios involve a recirculation-sump demand and if these scenarios are part of the plant licensing basis, the same considerations for break location must be applied as discussed in this section for LOCA events in primary piping. The reason supporting this position is that inclusion of secondary-break scenarios in the licensing basis acknowledges the possible need for recirculation, but the break locations evaluated in the licensing basis may not have been defined specific to sump performance and could not have anticipated the range of concerns identified in the course of resolving GSI-191. Although secondary side breaks are not analyzed in accordance with the requirements of 10 CFR 50.46 or to demonstrate compliance with 10 CFR 50.46, the staff's position is that licensees relying on the ECCS sumps to mitigate the consequences of secondary side breaks, e.g., for EEQ purposes, should identify and evaluate the limiting break locations to ensure acceptable sump performance.

The staff accepts the GR-stated position regarding breaks in attached piping beyond isolation points so long as there is no possible need for recirculation should a break occur in these sections. The decision whether to include piping segments beyond the isolation points should consider possible failure of the isolation valves in a manner consistent with the licensing basis.

3.4 DEBRIS GENERATION

3.4.1 Introduction

This section of the GR discusses the process of determining, for each postulated pipe break location, the zone within which the break jet forces will be sufficient to damage materials and create debris, the amount of debris generated by the break jet forces and the need to determine the characteristics of the debris.

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3.4.2 Zone of Influence (ZOI)

The GR in Section 3.4.2 recommends a spherical boundary for the ZOI with the center of the sphere located at the break site. The ZOI is defined as the volume about the break in which the fluid escaping from the break has sufficient energy to generate debris from insulation, coatings, and other materials within the zone. The use of a spherical ZOI is intended to encompass the effects of jet expansion resulting from impingement on structures and components.

Staff Evaluation for Section 3.4.2: The recommended spherical ZOI is a key feature to the baseline evaluation and any alternatives other than spherical or alternatives specifically reviewed and approved by the staff for use within the baseline as described in Section 6 of this safety evaluation report. The staff evaluation of refinements to the spherical ZOI are addressed in Section 4.2.2 of this SE.

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The spherical zone is a practical convenience that accounts for multiple jet reflections and mutual interference of jets from opposing sides of a guillotine break as well as pipe whip. It is important to note that when the spherical volume is computed using an acceptable approximation for unimpeded free-jet expansion, the actual energy loss involved in multiple reflections is conservatively neglected to maximize the size of the ZOI. The staff concurs with the use of spherical ZOI as a practical approximation for jet-impingement damage zones.

3.4.2.1 Recommended Size of Zone of Influence

The GR recommends using the ANSI/ANS 58.2-1988 standards to determine the radius of the spherical ZOI that represents the effects of the jet originating from a postulated pipe break. Appendices B, C, and D of the ANSI/ANS standard provide guidance necessary to determine the geometry of a freely expanding jet for jets originating from a variety of reservoir conditions, including subcooled conditions. This section of the GR reviews the key steps used in the ANSI/ANS 58.2-1988 procedure that determine the size of the ZOI.

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Section 3.4.2.1 of the GR also specifically addresses the break jet pressures that will result in coating debris generation within the ZOI.

Table 3-1 presented in this section of the GR contains the recommended destruction pressures for typical protective coatings and for several types of insulation.

Staff Evaluation for Section 3.4.2.1: The staff agrees that ANSI/ANS 58.2-1988 Standard (cited as reference 3 in the GR) provides a suitable basis for computing spatial volumes inside a damage zone defined by a jet impingement pressure isobar. Appendices in the standard do provide a set of equations that can be evaluated for this purpose, but the presentation is somewhat confusing, and the physical limitations of the model are not discussed thoroughly. For these reasons, Appendix I has been provided in this document to add guidance on the proper evaluation and interpretation of results from the ANSI model.

Six steps are outlined in the GR for performing ZOI calculations using the ANSI jet model:

1. The mass flux from the postulated break was determined using the Henry-Fauske model, as recommended in Appendix B of the Standard, for subcooled water blowdown through nozzles, based on a homogeneous non-equilibrium flow process. No irreversible losses were considered.

2. The initial and steady-state thrust forces were calculated based on the guidance in Appendix B of the Standard, with reservoir conditions postulated.

3. The jet outer boundary and regions were mapped using the guidance in Appendix C, Section 1.1 of the Standard for a circumferential break with full separation.

4. A spectrum of isobars was mapped using the guidance in Appendix D of the Standard.

5. The volume encompassed by the various isobars was calculated using a trapezoidal approximation to the integral with results doubled to represent a DEGB.

6. The radius of an equivalent sphere was calculated to encompass the same volume as twice the volume of a freely expanding jet.

These steps are acceptable for generic implementation of the model and conversion of isobar volumes to a volume-equivalent spherical radius. However, the following observations are provided in this SE which concern details of implementation of this method that should be considered when using the model. These details are further explained in Appendix I:

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1. Plots of metrics related to the Henry-Fauske mass flux presented in the standard do not extend to the desired state point, so it is not clear exactly how the mass flux was evaluated in the GR. Licensees using this technique should refer to confirmatory Appendix I for guidance.

2. It should be noted that neglect of irreversible losses refers to internal pipe and pipe-component friction losses between the upstream reservoir and the location of the break.

3. Only the steady-state thrust coefficient should be used in this calculation as a conservative bound.

4. Insulation damage pressures such as the 10 psi cited for Nukon fiberglass can only be interpreted with a full understanding of the test conditions under which they were experimentally measured. The computed jet conditions will not match the experimental test conditions; therefore care should be taken to assure that equivalent damage effects are considered. Finally, it should be noted that the GR exercised the model for a spectrum of pressure isobar values because different materials have different resistances to damage from jet impingement.

Regarding the three conditions offered for jet expansion calculations, the staff agrees that DEGB break configurations with circular geometries, and full separation and offset between the broken ends provides the maximum debris generation volume. However, as further discussed in Appendix I, the choice of fluid reservoir conditions is not justified as bounding for the baseline evaluation and the reported thermodynamic properties do not match the stated conditions. Using automated NIST/ASME steam tables [NIS96], the stagnation enthalpy and degree of subcooling for the stated conditions of 2250 psia and 540 ºF are 534.9 Btu/lbm and 112.7 ºF, respectively. Appendix I confirms that these conditions bound nominal conditions for a hot-leg break, and some guidance is offered

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there for licensees to estimate the effects of minor system-pressure increases without the need for reevaluating the model.

The staff agrees with the GR choice of ambient containment pressure, versus crediting containment backpressure. The staff considers this choice since zone-of-influence volumes are strongly driven by the system stagnation pressure, which is highest when the containment is at ambient conditions. The maximum debris generation would occur instantaneously within this ZOI. Furthermore, the use of atmospheric pressure may not be non-conservative for subatmospheric containment designs that would permit the discharge of a slightly higher mass flux across a break. However, the effect is judged to be small and is compensated by jet pressure equations in the standard that do neglect ambient pressure in containment. See Appendix I for a discussion of mass flux calculations and the dependence of ANSI correlations for thrust coefficient on the choice of psia.

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The staff finds that the citation of 10-diameter limits for jet damage recommended in NUREG/CR-2913 [WEI83] for structural loadings on equipment and components is not applicable to the present concern regarding insulation damage. The criteria for onset of damage and the implications of structural damage vs. debris generation are not directly related. Furthermore, any comparison of conservatism between methods should consider the range of damage pressures for various insulation types. Therefore, the 10-diameter limit for jet damage may only be used for structural loading and for coatings as described below.

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Protective Coatings Destruction

The potential debris term generated by failed coatings can be a significant contributor to the total containment sump debris term for some plants. The GR assumes the following LOCA effects on coatings:

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- that all coatings in the ZOI will fail;
- that all qualified (DBA-qualified or Acceptable) coatings outside the ZOI remain intact; and
- that all unqualified coatings will fail.

The GR also assumes that coating failure will generate debris in the form of fine particulate which is equivalent in size to the basic material constituents. This is descriptive of the size of the average zinc particle in inorganic zinc (IOZ) coatings or the pigment used in epoxy coatings, which is approximately a 10μm (in diameter) spherical particle in both cases. The GR states that because there is a lack of experimental data regarding coating debris size values, a debris size distribution of 100% small fines (10μm IOZ equivalent) is adopted for all coatings inside the ZOI. For coatings outside the ZOI, the GR states that all indeterminate and DBA-unqualified and unacceptable coatings should be treated as a single category of coating which produces debris of the same characteristic independent of the type of coating. As such, the coating debris size within the ZOI is applicable to all unqualified, indeterminate and unacceptable coatings that fail outside the ZOI as well.

Outside the ZOI, the GR assumes that all qualified coatings remain intact and do not contribute to the debris term. Although the GR assumes that all unqualified coatings will fail and break down into 10μm particles, it also indicates that plant specific data should be used to estimate the area and thickness of the unqualified coating in order to determine the amount of debris generated.

The GR indicates that "the ZOI for DBA-qualified coatings or coatings determined to be 'Acceptable,' applied to PWR containment surfaces, which results from fluid impingement from the break jet, has not been clearly defined." However, two key pieces of evidence are offered in the GR to support the argument that DBA qualified and acceptable coatings are resistant to direct jet impingement: (1) DBA qualification tests subject samples to elevated temperatures with no apparent loss of structural integrity or performance degradation; and (2) water-jet pressures in excess 2250 psia are commonly required to efficiently remove coatings in industrial applications.

This GR-assumed destruction pressure is tied to experience for removing coatings by the commercial water blast industry and industry waterjet testing detailed in Appendix A of the GR. This testing was performed using a 3500 psig positive displacement pump, hose and nozzle attachment (high pressure washer) at two temperatures, i.e., approximately 80°F and 150°F, to investigate coating degradation under jet impingement conditions. The test apparatus was used at various distances from substrates coated with qualified coatings. The testing indicated that coating debris generated in the ZOI would fail as the result of erosion and would generate debris sized roughly equivalent to the coating pigment size. Both IOZ and epoxy were tested. The testing also indicated that coating degradation was influenced by temperature.

Staff Evaluation for Protective Coatings Destruction: The staff finds the spherical modeling of the ZOI to be consistent with the approach approved in Section 3.4.2 of this SER, and therefore an acceptable approach for application to coatings. The staff finds that the following assumptions should be applied with regard to coating debris destruction subjected to a LOCA jet:

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- Qualified coatings outside the ZOI are assumed to remain intact and will not contribute to the sump debris load during a postulated event.
- All unqualified coatings outside the ZOI are assumed to fail and act as a potential contributor to the debris load during a postulated event.
- All coatings, regardless of qualification are assumed to fail within the LOCA jet ZOI. The baseline guidance does not provide sufficient technical justification to support use of a 1000 psig coating destruction pressure and corresponding ZOI equivalent to 1 pipe diameter. The staff position is that licensees should use a coatings ZOI equivalent to 10D or a ZOI determined by plant specific analysis. The specified ZOI of 10D is based upon the previous staff position used for BWR sump analysis. Any plant specific analysis should incorporate at a minimum the temperature and pressure effects of the jet on plant coating systems in the ZOI. Such an analysis should be based on experimental data over the range of pressures and temperatures of concern using coating samples correlatable to plant materials. The analysis should also seek to accurately estimate the amount of coating on a plant specific basis within the ZOI. If a realistically conservative approach is taken, the basis and justification for why the method is realistically conservative should be provided.

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The staff agrees that it is conservative to treat coating debris as highly transportable particulates in the range of 10 to 50 microns in diameter, based on plant susceptibility to thin bed formation at the sump screen. However, for those plants that can substantiate no formation of a thin bed at the sump, this assumption may be non-conservative with regard to sump blockage since fine particulates would pass through the sump screen and generate no blockage concerns. Therefore, for those plants that are susceptible to thin bed formation at the sump screen, use of the basic material constituent (10 μm sphere) to size coating debris is acceptable. However, for those plants that can substantiate no formation of a thin bed at which particulate debris can collect, the staff finds that coating debris should be sized based on plant specific analyses for debris generated from within ZOI and from outside the ZOI. Such an analysis should conservatively assess the coating debris generated with appropriate justification for the assumed particulate size or debris size distribution. Degraded qualified coatings that have not been remediated should be treated as unqualified coatings. Finally, testing regarding jet interaction and coating debris formation could provide insight into coating debris formation and help remove some of the potential conservatism associated with treating coatings debris as highly transportable particulate. If coatings, when tested at corresponding LOCA jet pressures and temperatures, are found to fail by means other than erosion or the erosion is limited, the majority of debris may be larger, less transportable or pose less of a concern for head loss.

The staff agrees with the assumption that qualified coatings outside the ZOI remain intact during a postulated event and will not contribute to the ECCS sump debris load, because it is based on qualified coatings meeting established quality criteria and acceptance testing and is consistent with the position outlined in NUREG 0800, Section 6.1.2 Protective Coating Systems. The assumption is also based on the coatings being in good condition at the initiation of the postulated LOCA. However, operating experience indicates qualified coatings require periodic maintenance throughout the coating service life and operating experience has identified cases were qualified coatings have exhibited significant degradation during the coatings normal service life. Therefore, the staff position is that a periodic coating condition assessment be identified, described and implemented during routing outages, to assure that qualified coatings remain capable of performing in a manner consistent with assumptions used to evaluate sump debris loads. Further the staff has concluded that qualified coatings which have degraded, but which have not yet been remediated should be considered to fail during a postulated accident and will potentially contribute to the debris load. The staff finds that the estimated quantity of debris from degraded qualified coatings (if any) should be based on plant specific data and should follow the guidance for debris resulting from unqualified coatings.

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The staff agrees with the assumption that all unqualified coatings outside the ZOI fail, based on the position outlined in NUREG-0800, “Standard Review Plan,” Section 6.1.2 “Protective Coating Systems.”

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The staff agrees with the assumption that all coatings, regardless of type and qualification will fail within the ZOI because it conservatively addresses the LOCA jet interaction with all coatings (unqualified coatings are assumed to fail regardless of location) in this zone; however, the staff believes there is insufficient technical justification for the assumption of a 1000 psig destruction pressure and corresponding spherical ZOI with a radius equivalent to one pipe diameter.

Although Appendix A of the GR provides useful test data illustrating the erosion effects of high pressure water jets on coating systems, no test data are offered that combine both the effects of mechanical insult and elevated temperature in the same test, and no data appear to be available on the effects of very rapid thermal transients on coating performance. Specifically, the initial conditions of the LOCA jet established in the baseline methodology are 540°F and 2250 psig, while industry testing referenced in Appendix A of the GR was performed at approximately 3500 psig and 150°F. Although the initial LOCA jet pressure is expected to be lower than the industry test pressure used (~3500 psig) and waterjet pressure data, the initial LOCA jet temperature expected is significantly higher than the industry test temperatures used (150°F). No correlation or extrapolation was provided in the NEI baseline methodology illustrating how the elevated test pressure accounts for the reduced test temperature to produce a similar damage mechanism and degree of damage as the combined temperature and pressure from a LOCA jet and thus can be used to adequately establish the coating ZOI. Therefore, the staff finds the results of the waterjet testing to be inconclusive in this regard.

Additional information offered in Appendix I of this report presents spatial contours of estimated jet impingement temperature for a reference cold-leg break condition. Temperature zones exceeding 300 °F are observed to extend out to 10 pipe diameters from the break, and exceed 220 °F for most of the jet envelope. Given the small thickness of the paint and the differences in heat conduction between the layer and the substrate, it is presumed that the coating would reach the impingement temperature almost instantly when directly hit by the break jet. Thermal shock may affect bonding with the substrate, induce expansion cracking in the coating layer, and change its tensile properties. All of these potential effects increase the vulnerability of paint to jet impingement. The occurrence of very rapid thermal transients in combination with the mechanical insult of water-laden jet impact is a unique environment that should be subject to experimental study.

The NRC staff acknowledges that the five reasons given to defend the selection of 1000 psi as a destruction pressure for DBA-qualified or "Acceptable" coatings are factual, while the GR arguments do not address important phenomenology of the accident environment. It is premature to accept the proposed value of 1000 psi as either appropriate or conservative. Individual licensees should provide data to support the robustness of their DBA-qualified and "Acceptable" coatings system for use in the baseline analysis. Spatial contours of jet-impingement temperature such as that offered in Appendix I may be useful in judging the cost-benefit of alternative test conditions.

Because (1) the temperature effect may be influenced by the coating system, i.e., IOZ alone, IOZ topcoated with epoxy or multiple coats of epoxy, (2) epoxy and IOZ each would be expected to have a different temperature response, and (3) no testing replicating the effects of LOCA jet pressures and temperatures on coatings (epoxy, IOZ, qualified or unqualified coatings) have been performed or referenced; the staff position is that either a coating spherical ZOI of 10D be used, or be determined by plant specific analysis. If an analysis is performed, it should incorporate, at a minimum, the combined temperature and pressure effects of the jet on potential coating systems in the ZOI. Such an analysis should be based on experimental data over the range of pressures and temperatures of concern using coating samples correlatable to plant materials. The analysis should also seek to accurately estimate the amount of coating on a plant specific basis within the ZOI. If a bounding approach is taken, it is the staff position that the basis and justification why the method is conservatively bounding must be provided.

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The staff believes that a comprehensive test program investigating the effects of direct impingement of a LOCA jet (accounting for jet pressure and temperature) on coating degradation should be performed in order to have a sound basis for the destruction pressure and size of the coating ZOI.

3.4.2.2 Selecting a Zone of Influence

Section 3.4.2.2 recommends that for the baseline calculation, the ZOI for a break is selected based on the potentially affected insulation inside containment with the minimum destruction pressure. This ZOI is then applied to all insulation types.

Staff Evaluation for Section 3.4.2.2: The baseline approach of selecting ZOI size based on the potentially affected insulation type in containment with the lowest destruction pressure is acceptable to the staff provided that 1) there are no other materials in containment more fragile than insulation that might pose a debris generation potential, and that 2) defensible damage pressures are available or can be ascertained conservatively with engineering judgment, for all insulation types, coatings and other materials of concern. The implication of this assumption that the presence of a single vulnerable material means that all candidate debris materials should be presumed damaged to the same level. Credit for the individual response of well-characterized insulation types can be given under the refinement offered in Chapter 4 of the GR.

Table 3-1 is offered in the GR as a mechanism for matching experimentally determined damage pressures with "calculated" values of volume-equivalent spherical ZOI radii. Presumably, the calculations were performed in the manner described in Appendix D of the GR, but no cross reference or explanation is offered. Appendix D cites an evaluation of the ANSI/ANS 58.2-1988 that was used to generate spatial jet pressure contours, but no insights are offered in the GR for how to interpret the resulting pressures with respect to material damage.

In order to confirm that the ANSI jet model was implemented properly the model was independently programmed and the results compared with the isobar map tabulated in Table D-1 of the GR. This comparison is shown in Fig. 3-1 where the blue contour lines represent the GR evaluation of a break at 2250 psia and 540 ºF and the black contour lines represent a reference cold-leg break at 2250 psia and 530 ºF. See Appendix I of this SER for an explanation of the independent calculation and additional guidance on interpreting results of the ANSI jet model.

Good agreement is seen between the calculations for downrange behavior (Zone 3), but discrepancies exist in Zones 1 and 2. It appears that contour termination points on the centerline are not accurate and that the quadratic behavior of the Zone 2 isobar equations is not implemented correctly. These differences will have a negligible effect on volume integrals for jet pressures less than 20 psig, but may become more of a concern for higher pressures near the break. To quantify the magnitude of the difference, Table 3-1 below presents a comparison of ZOI radii computed from both methods. In particular, the GR approach may not have preserved the system stagnation pressure throughout the volume of the liquid core region as specified by the standard. However, the GR recommended values essentially bound both sets of calculated values.

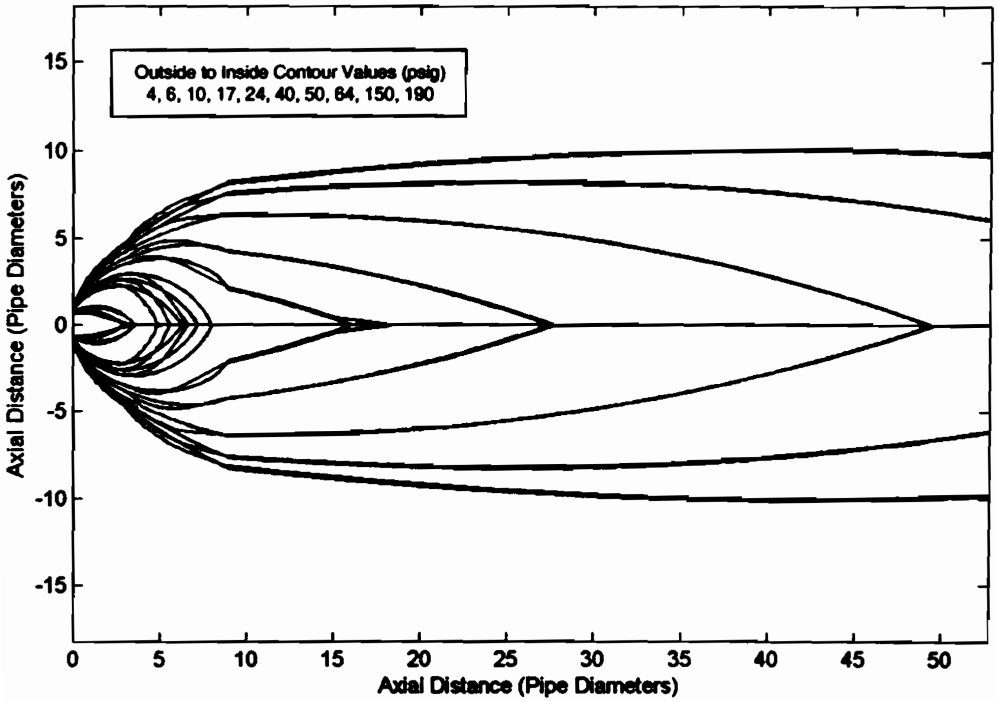

Figure 3-1. Comparison of GR Isobar Map with Isobars from Independently Evaluated ANSI Jet Model

Table 3-1. Comparison of Computed Spherical ZOI Radii from Independent Evaluations of the ANSI Jet Model

Impingement Pressure (psig)	ZOI Radius/Break Diameter		
	Guidance Report Recommendation	Calculated Value	SER Appendix I
1000	1.0	0.24	0.89[a]
333	1.0	0.55	0.90
190	1.3	1.11	1.05
150	1.6	1.51	1.46
40	3.8	3.73	4.00
24	5.5	5.45	5.40
17	7.8	7.72	7.49
10	12.1	12.07	11.92
6	17	16.97	16.95
4	21.6	21.53	21.60

[a] The core volume at stagnation pressure P0 gives a minimum possible ZOI radius of 0.88 diameters.

The larger question of what damage pressure to recommend for each material type requires an understanding of both the limits of the jet model and the knowledge base of existing experimental data.

First, as discussed in Appendix I, the jet model predicts impingement pressures in the longitudinal (downstream) direction only and may underestimate the radial extent of isobars in Zones 1 and 2 when considering the impingement pressure that would develop on the face of a target perpendicular to the local flow velocity.

Second, the ANSI model appears to be unbounded in the downstream direction. This means that for very small impingement pressures the isobar volume will grow unrealistically large. These two limitations compensate to some extent when volume-equivalent spherical radii are computed, and because the jet envelope provides a rigid constraint to radial growth of the contours, unbounded downstream growth will eventually dominate.

Unreasonable growth of low-pressure isobars can be illustrated by comparing the spherical radius plot in Figure I-13 (Appendix I) to Figure 3-3 in the parametric evaluation (PE) supplement [NUREG/CR-6762-3]. The PE study plots a function of spherical ZOI radii that was determined by the BWROG using the NPARC computational fluid dynamics (CFD) model for BWR blowdown conditions. Despite the differences in thermodynamic state point, the differences in qualitative behavior for target pressures less than 20 psig is evident; the ANSI trend appears to be diverging while the BWROG correlation appears to approach a finite maximum at zero pressure. The NRC reviewed the BWROG calculations and found the NPARC code to be a more capable method of modeling steam jets than the ANSI model.

The staff notes that a comparison using a CFD model for PWR break conditions was not performed for either the GR or this safety evaluation. Caution should be used in the comparison of calculated and experimentally determined pressures to ensure that the computed parameter of the field matches the measured parameter as closely as possible. For example, while it is trivial to fractionate a computed pressure into static and dynamic components over any incident angle, it may be difficult to obtain high-fidelity measurements under equivalent conditions and diagnostic orientations.

Third, the correlation between any prediction of jet pressure and an experimental observation of "damage pressure" depends on how the measurements were taken, how the debris was characterized, and what the thermodynamic conditions of the test actually were. Data from the references cited in Table 3-1 of the GR are dominated by tests conducted for resolving the strainer blockage issue for BWRs using high-pressure air as a working fluid. Therefore, much of the test data is not directly applicable to PWR or BWR blowdown conditions where jets consist of steam and water mixtures. Without directly applicable data and/or high-fidelity predictive models, this surrogate information can only be applied with appropriate caution. The NRC was concerned about potential differences in debris generation between air surrogates and two-phase jets, and therefore initiated a joint test program with Ontario Power Generation (OPG). Testing of low-density fiberglass ended prematurely after only one test and the concerns were not fully resolved, but the available results are documented in Vol. 3 of the PE report [NUREG/CR-6762-3] and in Reference 7 of the GR. These data were cited but not discussed in GR Table 3-1 in reference to damage pressures for calcium silicate. Therefore, there is a very limited set of data to evaluate the effects of two-phase jets on low-density fiberglass.

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Destruction pressure is the threshold pressure for the onset of damage. This is normally determined by experimentally measuring the differential pressure on the face of a target. One recurring problem with definitions of damage pressure is inconsistency in the degree of damage that is correlated to the pressure value. Two obvious choices exist. The first option is to define the minimum pressure (threshold) at which jacketing is breached in any way. Issues regarding contribution to potential screen blockage are then handled with a complete description of the debris size distribution from fines to partially intact cassettes and blankets. The second option is to presume a debris size that is suspected to contribute to the blockage potential and to report the damage pressure as the point where significant quantities of this debris size are generated. The second option will have higher values of damage pressure than the first, and the debris size distribution will be skewed towards smaller, and therefore more transportable, pieces if the two options are to give equivalent results in a vulnerability assessment. The second method also requires more a priori subjective judgment. Damage-pressure values reported in Table 3-1 of the GR are based on the second approach. The single fiberglass test performed by OPG resulted in conversion of approximately 50% of the insulation volume into debris of sufficiently small size to be a concern. It is assumed that this test meets by a significant margin the criteria for significant quantity implicit in the second damage-pressure definition.

The OPG test for fiberglass was conducted at a distance of 10D on the centerline downstream of a heated vessel of water at 1450 psia. Comparisons with more extensive OPG data for calcium-silicate suggested [NUREG/CR-6762-3] that the lower threshold for fiberglass damage in two-phase jets might be as low as 4 psig. The actual range can only be determined by bracketing with two tests at differing distance the transition from significant damage to negligible damage. While it is true that the insulation products tested by OPG were not identical to those tested in the BWROG air-jet tests, substantially different debris characteristics were observed.

In the absence of more complete test data, it is prudent to attribute the observed effects to the differences in the jet medium, i.e. the difference between air used in the BWROG tests and the two-phase steam/water mixture used by OPG. Several plausible physical mechanisms may contribute to enhanced debris generation in two-phase jets including penetration and erosion from impingement of entrained droplets, increased shear forces within the jet caused by radial velocity components of the expanding fluid, and higher local velocities because of the lower density of water vapor compared to air. To judge the potential contributions of these effects without more extensive data would be speculative, as would be any counter arguments offered to refute their importance. The potential for material degradation by erosion has already been acknowledged in the GR in relation to coatings damage. Although offered there as an ostensible conservatism, the same phenomenon should be considered for all material types.

Based on the OPG test results, an argument could be made for reducing damage pressures determined through air-jet testing by a factor of 2 or more. That approach was recommended, in fact, in the PE study by reducing the damage pressure for fiberglass from 10 psig to 4 psig. A corresponding spherical ZOI radius was then recommended based, not on the ANSI model for PWR break conditions, but rather, on the BWROG correlation for BWR break conditions that were similar to the OPG test. The corresponding radius was reported to be 12-D for an incident pressure of 4 psig while the ANSI model predicts a 21.6-D radius for nominal PWR break conditions at the same impingement pressure. Hence, there appears to be an inconsistency in the PE report because no compensation was made for increased ZOI volume induced by the higher initial pressure of a PWR break.

Given the uncertainties discussed above regarding: (1) interpretations and applicability of the ANSI jet model and its performance compared to CFD correlations for very low impingement pressures; (2) the dissimilarity of insulation types, jacketing and target orientation used in the OPG test compared to U.S. PWRs; and (3) the practical definition of damage pressure and its empirical correlation to the degree of insult, it would be speculative to assess the full damage-pressure reduction derived in the PE report. Therefore, based on the 50% destruction of fiberglass observed in the only publicly accessible two-phase debris generation test for this insulation type, comparison with OPG data on greater than 40% reduction in damage pressure for calcium-silicate insulation, and on the similarity of this degree of damage to the definitions used in Table 3-1 of the GR, the NRC staff position is that damage pressures for all material types characterized with air jet testing should be reduced by 40% to account for potentially enhanced debris generation in a two-phase PWR jet.

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Of course, specific materials may respond differently (if at all) to the effects of a two-phase jet, but this reduction in damage pressure provides adequate recognition of the issue and could focus some attention on the remediation or mitigation of high-debris volume accident scenarios. When available, the reduced damage pressure thresholds should be replaced with material-specific test data, so the GR recommendation of 24 psig for the damage pressure of calcium silicate is appropriate based on the findings of the OPG study. Table 3-2 lists the revised destruction pressures and the corresponding ZOI diameters computed as described in Appendix I for the reference cold-leg break.

Table 3-2. Revised Damage Pressures and Corresponding Volume-Equivalent Spherical ZOI Radii

[illegible]	[illegible]tion Pr[illegible] (psig)	[illegible]
Protective Coatings (epoxy and epoxy-phenolic paints)	TBD[1]	NA[2]
Protective Coatings (untopcoated inorganic zinc)	TBD[1]	NA[2]
Transco RMI Darchem DARMET	114	2.0
Jacketed Nukon with Sure-Hold® bands Mirror® with Sure-Hold® bands	90	2.4
K-wool	24	5.4
Cal-Sil (Al. cladding, SS bands)	24	5.45
Temp-Mat with stainless steel wire retainer	10.2	11.7
Unjacketed Nukon, Jacketed Nukon with standard bands Knaupf	6	17.0
Koolphen-K	3.6	22.9
Min-K Mirror® with standard bands	2.4	28.6

[1] To be determined by experiment.

[2] Not available for evaluation at this time.

Formal debris generation studies have confirmed that insulation products having outer casings, jackets, or other similar mechanical barriers resistant to jet impingement yield smaller quantities of debris than do less robust materials. Various studies have also demonstrated dependence between the orientation of the jacketing seam relative to the jet and the amount of debris generation. This suggests that the integrity of the jacket during impingement is an important feature for minimizing debris generation. Reference [Russell, J., "Jet Impact tests - Preliminary Results and Their Applications," Ontario Power Generation, N-REP-34320-10000, Rev R00, (April 2001).] reports, for example, that double jacketing an insulation product with a second overcladding of stainless steel having a rotated, opposing seam was very effective at minimizing the distance from the jet to the onset of damage. As mentioned in Appendix I, any improvement in the mechanical resistance of the insulation product will help to avoid inflated ZOI volumes predicted by the ANSI jet model for very low damage pressures.

As noted above, the ANSI/ANS jet model has been proposed in the GR and found acceptable by the staff for the purpose of estimating potential damage volumes associated with empirically measured damage pressures. Various attributes and interpretations of the ANSI jet model are presented in Appendix I. Among those observations is the explanation of potentially exaggerated conservatism for very low damage pressures. While this is conservative, it may be detrimental for the identification and design of practical mitigation strategies. The staff notes that the use of robust insulation materials is one possible approach for avoiding excess conservatism. Another approach, which can be accomplished concurrently with the testing of specific insulation products, is to properly instrument jet tests for the purpose of refining the ANSI model for the specific application of debris generation. Particular emphasis should be placed on the measurement of impingement pressures on small targets placed both perpendicular to the jet centerline and at radial locations parallel to the jet centerline. A test program such as this would be most effective when combined with concurrent insights gained from models including ANSI-58.2-1988 and CFD.

3.4.2.3 The ZOI and Robust Barriers

Section 3.4.2.3 recommends truncating the spherical ZOI whenever the ZOI intersects a robust barrier such as walls and components such as supports, pressurizer, steam generator, reactor coolant pump or jet shields. Such barriers will terminate further expansion of the ZOI. The area in the shadow of the component or structure will be free from damage. The baseline assumes there is sufficient conservatism in drawing the sphere that it is not reasonable that a jet reflected off of a wall or structure would extend further than the unrestrained sphere.

Staff Evaluation for Section 3.4.2.3: Conceptually, the volume integral under a computed jet expansion isobar represents the potential for material degradation at pressures equal to the isobar value and higher. Multiple reflections and deflections of a LOCA jet within a confined space would dissipate energy, so conservation of the jet volume under an impingement pressure isobar provides an upper bound on the integral volume of the spatial damage zone, regardless of the shape it is mapped into either by the local geometry of obstacles or by convention for the purpose of analysis. Spherical zones were originally conceived as an adequate approximation for opposing jets from each side of a guillotine break in the congested piping environment of a BWR containment structure. Spherical zones also provide significant convenience for mapping onto piping layouts.

The only conservatism inherent to the ZOI mapping within containment is the conservation of damage potential computed as the volume under a relevant damage-pressure isobar. The degree of conservatism depends on the piping and equipment congestion in the vicinity of the break. More deflections and redirections lead to greater local deposition of energy, and hence, to greater conservatism in the preservation of damage volume, which maximizes the size of the ZOI by assuming no interference with jet development. It is difficult to quantify the degree of conservatism introduced by ignoring jet reflections, but for BWR break conditions, CFD calculations were performed in a spatial domain with contrived obstacles and flow paths to demonstrate rapid dissipation of the potential damage volume. Similar examples have not been offered in the GR to quantify the conservatism that would rationalize the truncation of spherical ZOI. Relevant attributes of this calculation would include representative spatial complexity and scale relative to the damage volume for PWR break conditions.

PWR containment structures often have structural paths that are designed to direct the principal expansion flow. These features include the ice columns in ice-condenser plants and steam generator compartments in large-dry plants that are vented to upper containment domes with spray deluge systems. Given the potentially large damage volumes that may be predicted from the previous section, it seems reasonable that these spherical ZOI will be redirected along the designed flow paths for many break scenarios.

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The potential benefits of shadowing by equipment and components are also difficult to quantify. Undoubtedly, shadowing is a relevant effect for impingement on a large steam generator from one side in a relatively unconfined location, but within a doghouse enclosure, flows may accelerate completely around the generator causing damage on all sides. Shadowing effects cannot be approximated by strict geometric obstruction angles. Limited guidance is provided by the GR on the practical implementation of proposed method.

For the baseline analysis, the NRC staff position is that licensees should center the spherical ZOI at the location of the break. Where the sphere extends beyond robust barriers such as walls or encompasses large components such as tanks and steam generators, the extended volume can be truncated. This truncation should be conservatively determined with a goal of+0-25% accuracy, and only "large" obstructions should be considered. The shadow surfaces of components should be included in this analysis and not truncated, as debris generation tests clearly demonstrate damage to shadowed surfaces of components.

3.4.2.4 Simplifying the Determination of the ZOI

Section 3.4.2.4 offers a conservative simplification for the determination of the ZOI. Given the complexity of the analysis as a whole, it may be desired to make conservative assumptions with the goal of simplifying the analysis. For example, for some breaks it may be only slightly more conservative and much simpler to assume that an entire subcompartment (but not outside the subcompartment) becomes the ZOI.

Staff Evaluation for Section 3.4.2.4: The staff concurs that simplifications may be desirable. As a point of practical guidance, it may be useful to precalculate the free volume of subcompartments and rooms that may host a break location or be affected by an adjacent break location. This will facilitate cumulative volume estimates for the total affected zone.

The staff finds the example simplification acceptable; provided the simplification procedure properly justifies that significant jet destruction cannot occur beyond the assumed boundaries of the affected compartments.

3.4.2.5 Evaluating Debris Generation within the ZOI

Section 3.4.2.5 provides a general statement regarding the assessments of debris within the ZOI and refers to the following section (Section 3.4.3). It notes that plant-specific information on the type, location, and amount of debris sources within containment is needed. This information is obtained from plant drawings and the results of condition assessment walkdowns.

Staff Evaluation for Section 3.4.2.5: The general statement in GR Section 3.4.2.5 is acceptable. As a point of clarification the staff suggests that once the spatial region of the ZOI has been determined, the next step is to calculate the volume of insulation, the surface area of coatings both qualified and unqualified, and the amounts of any other potentially frangible debris sources within that ZOI. Guidance provided in other sections determines how this insulation is distributed by size and character into debris.

3.4.2.6 Sample Calculation

A sample calculation is provided in Section 3.4.2.6 of the GR. The sample postulates the break of a 10-inch diameter pipe attached to the RCS. The break occurs at the base of a steam generator. Two types of insulation materials are specified (Nukon and RMI), and the quantities of each in the affected zone are given. A ZOI radius is determined based on the pertinent ZOI/break diameter values given in Table 3-1 of the GR. All of the insulation material within the affected zone is assumed to be damaged and becomes debris. The sample also calculates the surface area of coatings estimated to be destructed by the break jet forces.

Staff Evaluation for Section 3.4.2.6: Separation of the containment into inventory zones appears to be a very effective aide in moving through the break selection and ZOI mapping processes in a systematic way. Alternative segmentation schemes (or useful subdivisions) other than the uniform grid shown in Fig. 3-1 of the GR might be based on structural barriers or groupings of diverse but collocated insulation types. In Step 4, the volume of the evaluation zone and the estimated surface area of coatings (both qualified and unqualified) are not provided even though this step should represent all available information about the potential impacts of a break in the postulated location.

The sample calculation is inconsistent with the baseline methodology discussed above because it implies that the potentially affected insulation type with the minimum destruction pressure can be selected from within an accounting region in the vicinity of the break rather than from the entire containment inventory as specified in Section 3.4.2.2. For example, if Min-K were present in an adjacent evaluation zone (or anywhere else in containment), the ZOI radius would have to be larger to account for the lower damage pressure of that insulation type. ZOI may easily overlap several evaluation zones for large breaks.

If Nukon is the most fragile insulation in containment, then the example is consistent through step 5 except that, using the revised damage pressures presented in Table 2 above for two-phase jet impingement, the ZOI radius would be 17 pipe diameters, the ZOI radius would be 14.1 ft, and the ZOI spherical volume would be 11,742 ft^3. All potential debris generation materials within this zone should be included in the debris inventory.

Step 6 appears to invoke the simplification of assuming 100% inventory within the zone. The decision to make this simplification might be assisted by comparing the ratio of the ZOI volume to the volume of the evaluation zone. It is further reinforced by considering the relative volume of the ZOI obstructed by the steam generator and major piping. When this additional volume is added back to account for flow divergence, the ZOI occupies an even larger proportion of the evaluation zone.

For strict compliance with the baseline methodology, step 6 should also include all of the coatings within the evaluation zone as debris, both qualified and unqualified. Instead, step 7 illustrates an example of a proposed refinement presented in Chapter 4 of the GR where a ZOI specific to a material type is computed to account for the possible higher resistance of coatings to jet impact. Under this refinement, a separate ZOI radius can be computed for each potentially affected debris source. It is likely that many licensees will choose this refinement rather than accept the conservatism of applying at all break locations damage zones defined by the most vulnerable material in containment.

Because acceptable damage pressures for coatings have not been developed, the staff does not agree with the step 7 calculation. However, once a ZOI has been established, the total area (or equivalent mass) of qualified paint within the spherical zone should be added to the initial debris inventory. There is no basis for the assumption of a coating area equal to the surface area of the ZOI except to satisfy the intent of conservatism for very small damage zones. This assumption of a minimum coating contribution is not necessary if there is no paint present within the potential ZOI that is eventually defined by a coatings damage pressure.

3.4.3 Quantification of Debris Characteristics

3.4.3.1 Definition

Section 3.4.3.1 defines debris characteristics as post-accident size distribution of material, material size and shape, and material densities. The input information needed to determine debris characteristics is also noted.

3.4.3.2 Discussion

Section 3.4.3.2 provides a discussion of the debris size distributions that have been used in various studies and specifies the distribution recommended for the baseline evaluation. The GR adopts a two-size distribution for material inside the ZOI of a postulated break. These two size groups are small fines and large pieces. Small fines were defined as any material that could transport through gratings, trash racks, or radiological protection fences by blowdown, containment sprays, or post-accident pool flows. Furthermore, the small fines are assumed to be the basic constituent of the material for fibrous blankets and coatings (i.e., individual fibers and pigments, respectively). The GR assumes the largest openings of the gratings, trash racks, or radiological protection fences to be less than a nominal 4 inches (less than 20 square inches total open area). The remaining material that cannot pass through gratings, trash racks, and radiological fences is classified as large pieces.

The erosion and potential disintegration of some debris materials by post-DBA environment water flows are also discussed in Section 3.4.3.2. Because the small fines were already classified as reduced down to the basic constituent, further erosion of the small fines does not apply (e.g., for fibrous and coating debris). For fibrous insulation material, the large pieces are assumed to be jacketed or canvassed. According to NUREG/CR-6369, jacketed pieces are not subjected to further erosion. Also, for material outside the ZOI, all insulation material that is jacketed is assumed not to undergo erosion or disintegration by containment spray or break flow.

The discussion noted the NUKON™ debris size distribution from the test as the insulation that had the most data points and that produced the smallest fines and adapted this point as the bounding value of fines production for unjacketed fibrous blankets. The GR references the OPG testing (OPG, 2001) for a low-density fiberglass, which indicated that 52% of the debris was in the category defined as small fines.

The GR assumes that if a material has a higher destruction pressure than NUKON™ then it signifies that the material has a higher resistance to damage, hence the size distribution would be larger than a more fragile material indicated by a lower destruction pressure. Therefore, it is conservative to adopt the NUKON™ blanket size distribution for material with a higher destruction pressure.

Staff Evaluation for Section 3.4.3.2: The categories in any size distribution must correlate to the transport model assumptions. The recommended two category size distribution (i.e., small fines and larger pieces) adapted by the NEI baseline for material inside the ZOI of a postulated break is suitable to the baseline transport assumptions, which are based on the transport of either the basic constituent (e.g. individual fibers) or large pieces. The division between the two categories of a nominal 4-in size is adequate in that it agrees well with debris generation testing data. The two-category size distribution, however, is likely to become highly problematic for debris transport refinements that more realistically treat the transport processes. For example, a transport model designed to treat small fibrous debris that transport along the pool floor rather than as suspended fibers will require the small fines in the NEI baseline to be further subdivided into suspended fines and small pieces. The staff finds the two-category size distribution suitable to the baseline but the use of this size distribution should be reevaluated when debris transport refinements are proposed, such as the refinements proposed in Section 4 of the GR.

The baseline approach contains the assumption that all large pieces of fibrous insulation material would be jacketed or canvassed and therefore would not be subject to further erosion due to water flows. Although this assumption is inconsistent with debris generation data acquired through NRC-sponsored tests, the staff position is that the overall impact of this nonconservatism on the results of this analysis is relatively minor in regards to the acceptance of the baseline guidance, and therefore acceptable. This is based on GR assumptions which include a large fraction of small debris (60%), all of which is assumed to be small fines. These are unrealistically conservative assumptions which substantiate the minor importance of addressing degradation of large debris. Further, it is agreed that for material outside the ZOI, all insulation material that is jacketed will not undergo significant erosion or disintegration by containment spray or break flow.

The NEI baseline guidance for determining a conservative fraction for the small fines based on one insulation type, i.e. NUKON™, is not realistic even though the 60% determination is adequate. The GR indicates that the debris generation test with the most destruction for their determination is the low-density fiberglass test conducted by OPG and documented in NUREG/CR-6808, which indicated 52% of the debris was in the category defined as small fines, which is in close agreement with the GR assumption of 60%. During the debris generation for the drywell debris transport tests (DDTS) documented in NUREG/CR-6369, Transco™ fiberglass blankets (similar to NUKON™ blankets) were located at a distance in front of the air jet nozzle so that the blankets were routinely completely or nearly completely destroyed (so noted on Page 3-20 in NUREG/CR-6369). Therefore, it must be concluded that fiberglass blankets will be essentially totally destructed into small fines given sufficient jet pressures (approximately 17 psi for Transco™). However, because this testing was based on a small distance between the nozzle and the insulation target, a realistic determination of the fraction of the insulation in a spherical ZOI that would be destructed to small fines requires integration over the sphere based on damage versus pressure and a mapping of the test jets into the spherical ZOI. Analyses documented in Appendix II confirmed the adequacy of the recommendation of 60% for the fraction of small fines debris generation for NUKON™ fiberglass insulation. Further, this analysis confirmed the 60% number for

Transco and Knauf insulations, which are similar to NUKON™ (all low density fiberglass insulations). The Appendix II analyses also illustrate the correct process to determine the debris size recommendation.

The baseline guidance assumes it is conservative to adopt the NUKON™ blanket size distribution for other materials with a higher destruction pressure than NUKON™. This assumption has been supported, but not conclusively assured, by debris generation confirmatory analyses documented in Appendix II. This assumption should only be applied if insulation-specific debris size information is not available.

In addition, although the GR provides damage pressures for a number of insulation products, this list reflects only those products that have received some type of prior testing. The list is not comprehensive either in trade name or by mechanical insulation type. Acceptable default assumptions regarding material damage have been discussed, but product-specific test data can be performed in order to avoid unnecessary conservatism. Test data should be used to quantify the performance of mitigation strategies such as double cladding, double banding, or other redesigned insulation-application methods.

3.4.3.3 Size Distribution

Section 3.4.3.3 provides the recommended size distributions (i.e., percentages that are small fines versus large pieces) for fibrous materials in a ZOI, reflective metallic insulation (RMI) in a ZOI, other material in ZOI, and material outside the ZOI. These recommendations are summarized in Table 3-3.

Table 3-3. NEI Recommended Debris Size Distributions

Material	Percentage Small Fines	Percentage Large Pieces
Fibrous Materials in a ZOI		
NUKON Fiber Blankets	60	40
Transco Fiber Blankets	60	40
Knaupf	60	40
Temp-Mat	60	40
K-Wool	60	40
Min-K	100	0
Generic Low-Density Fiberglass	100	0
Generic High-Density Fiberglass	100	0
Generic Mineral Wool	100	0
Reflective Metallic Insulation in a ZOI		
All Types	75	25
Other Material in ZOI		
Calcium Silicate	100	0
Microtherm	100	0
Koolphen	100	0
Fire Barrier	100	0
Lead Wool	100	0
Coatings	100	0
Material Outside the ZOI		
Covered Undamaged Insulation	0	0
Fire Barrier (Covered)	0	0
Fire Barrier (Uncovered)	100	0
Lead Wool (Covered)	0	0
Unjacketed Insulation	100	0
Qualified Coatings	0	0
Unqualified Coatings	100	0

Staff Evaluation for Section 3.4.3.3: The baseline recommendations can be grouped as follows:

- Materials for which adequate debris generation data exists to evaluate the debris size distribution, i.e., NUKON™ fiberglass and DPSC Mirror™ RMI insulations.
- Materials deemed to have a size distribution no finer than the materials for which debris generation data is available.
- Materials for which the debris generation is not known well enough to conservatively estimate debris size distributions, therefore maximum destruction is assumed.
- Materials outside the ZOI that are not expected to form debris due to qualification of or lack of protective coverings.

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The size distribution for materials located within the ZOI should be specified in conjunction with the specification of the spherical ZOI radius. Insulation damage progresses from total destruction at the location of the break to substantially less damage near the outer boundary of the ZOI. The specification of the outer boundary is based on a judgment of the conditions under which the damage becomes relatively insignificant. For example, with NUKON™ insulation where the destruction pressure of 10 psi has been accepted even though significant damage was seen at 6 psi in a debris generation test [BWROG AJIT Test 6-2 documented in the BWROG URG], the general test results indicate that a larger portion of the 10 psi ZOI would be in small fines than if the ZOI was based on 6 psi. Therefore, within the ZOI, the size distribution used should be based on the radius of the spherical ZOI determined.

For section 3.4.3.3 of the GR, the staff finds the following:

1. Analyses documented in Appendix II confirmed the adequacy of the recommendation of 60% for the fraction of small fines debris generation for NUKON™ fiberglass insulation. Further, this analysis confirmed the 60% number for Transco and Knauf insulations, which are similar to NUKON™. The small fine generation fraction of 60% is a realistic value that is only slightly conservative.

2. The GR assumes it is conservative to adopt the NUKON™ blanket size distribution for other materials with a higher destruction pressure than NUKON™. This NEI assumption has been supported but not conclusively assured by debris generation confirmatory analyses documented in Appendix II. This assumption should only be applied if insulation-specific debris size information is not available.

3. The staff agrees with the assumption of 100% of the materials becoming small fines for materials for which the debris generation is not known well enough to conservatively estimate debris size distributions.

   However, for those plants that can substantiate no formation of a thin bed at the sump, this assumption would be nonconservative with regard to sump blockage since fine particulates would pass through the sump screen and generate no blockage concerns. Therefore, for those plants that can substantiate no formation of a thin bed at the sump at which particulate debris can collect, the staff finds that debris generated should be assumed to be sized with realistic conservatism based on the plant-specific environment and susceptibilities identified for that facility, with appropriate justification for the sizing used.

4. The staff agrees that covered insulations and fire barrier material outside the ZOI will not form significant debris provided the covering is substantial enough to remain intact and to stop significant water from passing through the insulating materials. For example, an exception would be a vinyl covering of fibrous or particulate material that might melt at post-LOCA containment temperatures, and thus would not protect the materials inside from the effects of water erosion.

3.4.3.4 Calculate Quantities of Each Size Distribution

Section 3.4.3.4 provides guidance for estimating the quantities of debris for each material and each size distribution category. For materials located within the ZOI, other than coatings, the volumes of materials are simply multiplied by the respective size distribution fractions for either small fines debris or large piece debris to obtain the debris volumes of small fines and large pieces, respectively.

Staff Evaluation for Section 3.4.3.4: The staff agrees that for materials other than coatings, it is appropriate to multiply the volumes of the ZOI by the appropriate debris size distribution fractions to determine the volumes of debris.

Protective Coatings Quantification

The ZOI for protective coatings is based on the coating destruction pressure assumed in the GR. The same approach used to map the ZOI for other debris types (described in Section 3.4.2) is also used to map the ZOI for coatings, that is, modeling the ZOI as a spherical volume resulting from the freely expanding LOCA jet that will be exposed to pressures greater than or equal to the assumed destruction pressure. Depending on the break location, coated components may or may not exist within this sphere. Where plant specific data does not exist regarding the amount of coating within the ZOI, the GR assumes that coated components equivalent to the surface area of the sphere will exist within this volume and will fail, generating fine particulate debris. The amount of coating debris is a function of the coating thickness as well as the surface area. If plant specific coating thicknesses are not available, then the GR provides guidance on assuming a coating thickness in the ZOI that consists of 3 mils of IOZ primer plus 6 mils of epoxy topcoat.

Staff Evaluation for Protective Coatings Quantification: The staff finds that the quantity of coating debris that will be generated as a result of a LOCA jet should be based on the following:

- For plants that substantiate a thin bed, use of the basic material constituent (10 μm sphere) to size coating debris is acceptable.
- For those plants that can substantiate no formation of a thin bed at which particulate debris can collect, the staff finds that coating debris should be sized based on plant specific analyses for debris generated from within ZOI and from outside the ZOI, or that a default area equivalent to the area of the sump screen openings be used for coatings size. If analyzed, then such an analysis should conservatively assess the coating debris generated with appropriate justification for the assumed particulate size or debris size distribution. Degraded qualified coatings that have not been remediated should be treated as unqualified coatings. Finally, testing regarding jet interaction and coating debris formation could provide insight into coating debris formation and help remove some of the potential conservatism associated with treating coatings debris as highly transportable particulate. If coatings, when tested at corresponding LOCA jet pressures and temperatures, fail by means other than erosion or the erosion is limited, the majority of debris may be larger, less transportable or pose less of a concern for head loss.

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The GR stipulates that all unqualified coatings outside the ZOI are assumed to fail. This assumption is consistent with the position provided in NUREG-0800, "Standard Review Plan," Section 6.1.2, "Protective Coating Systems." The amount of debris will be a function of the area of unqualified coating and the coating thickness as described in the GR, but the staff recommends that plant-specific values regarding the unqualified coating properties and thickness should be used. The GR recommendation to use 3 mils of IOZ as a default thickness for unqualified coatings outside of the ZOI was based on the fact that 3 mils of IOZ, being 4.5 to 5 times more dense than epoxy, epoxy phenolic or alkyd coatings, would yield approximately the same mass as 13.5 to 15 mils of epoxy coating film. This concept of an "IOZ equivalent" coatings quantity can lead to inaccurate results in the calculation of the amount of debris generated because the GR does no clearly explain that the mass of coatings debris estimated in this way must then be combined with the actual coating density (not the density of IOZ) in order to accurately determine the amount of particulate that may impact sump screen head loss.

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Further, the staff is aware of numerous instances where containment coatings, qualified and unqualified, are much thicker than the assumed equivalent thickness of 13.5 to 15 mils, so the assumed equivalent thickness may not be conservative. The staff concludes that the GR alternative is not acceptable without plant specific justification and recommends that plant-specific evaluation of the plant's unqualified coatings be performed to determine conservative coating properties and thicknesses. The staff recognizes that the amount of unqualified coating in a plant may change due to changes in plant equipment and modifications which could affect the sump debris load. Therefore, the staff recommends that licensees periodically assess the amount of unqualified coating identified and used in the sump analysis to ensure the quantity remains bounding and if non-conservative changes in the amount of unqualified coating occur, that the impact of this change be evaluated.

<u>Staff Conclusions Regarding Section 3.4.3.4</u>: The staff concludes that the baseline alternatives to plant-specific data for the determination of the coatings thickness may not be conservative and are not acceptable without plant specific justification. Rather, the staff concludes that each plant should perform a plant specific evaluation of their respective coatings to determine realistically-conservative coating thicknesses. This conclusion was drawn despite the perceived conservatism of the recommendations of assuming all the unqualified coatings in containment fail and all coating debris forms a fine 10 micron particulate. It is considered reasonable for each plant to assess their respective coating thicknesses as well as the soundness of their coatings rather than assume a default recommendation that may not be conservative.

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3.4.3.5 Sample Calculation

Section 3.4.3.5 provides a sample calculation for estimating the quantities of debris from the ZOI by size category and for the DBA-unqualified coatings outside the ZOI.

<u>Staff Evaluation for Section 3.4.3.5:</u> The staff found the sample calculation presented in this section of the GR to be adequate in concept and practice, but numerically inconsistent with revised guidance explained in this SER, particularly in its treatment of coatings debris. First, the size distribution of fine and large pieces for both fiberglass and RMI insulation should be reviewed for consistency with SER recommendations in Section 3.4.3. Second, the estimate of coating debris from within the ZOI should be based on plant-specific characterization of coating thickness and a defensible ZOI radius. Finally, the estimate of coating debris from outside the ZOI should also be based on a plant-specific characterization of unqualified coating thickness and total inventory, not the suggested default thickness.

3.4.3.6 Debris Characteristics for Use in Debris Transport and Head Loss

Section 3.4.3.6 provides Tables 3-2 and 3-3 that present selected debris characteristics for a variety of materials, specifically material densities and characteristic sizes. The baseline guidance declared the characteristic sizes to be the most conservative values that can be associated with debris transport and head loss. The tables include data for fibrous, cellular, RMI, and particulate (granular) insulation materials. It is noted that the manufacturer should be contacted to obtain information for materials not listed.

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Staff Evaluation for Section 3.4.3.6: The staff notes the following concerns regarding the use of the data in Tables 3-2 and 3-3:

1. The range of variation for several data entries is substantial, e.g., the as-fabricated density for Kaowool ranges from 3 to 12 lb/ft^3. The reason for such wide variation was not provided but is likely due to the variability in the manufacture of that insulation. Further, the specification of such a wide range is not specific enough for head loss predictions because using 3 versus 12 lb/ft^3 for an as-manufactured density could easily make a drastic difference in the prediction. For example, it would take four times the volume of insulation to form a uniform 1/8-in thick layer if the density was 12 rather than 3. It is important that each plant locate data specific to their installed insulation.

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2. An inconsistency exists in the guidance regarding the particulate size for coatings debris outside the ZOI. The characteristic size for epoxy and epoxy phenolic coating chips (outside the ZOI) in Table 3-3 is listed as 25 microns. But the discussion on Page 3-25 appears to recommend a 10 micron particulate size for all unqualified coatings. It is the staff's understanding that the intent of the baseline guidance was to recommend the 10 micron size for the coating particulate; therefore acceptance of the baseline is based on the 10 micron recommendation.

3. In Table 3-2, a range of as-fabricated densities is recommended for Microtherm of 5 to 12 lb/ft^3. However, the reference provided (i.e., GR Reference 13), provided several ranges, i.e., 8 to 25 lb/ft^3, 12.5 to 22 lb/ft^3, and 15 to 22 lb/ft^3; none of which match the range recommended in the GR. Therefore, the value used for Microtherm should be confirmed by the licensee before its application.

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4. The data tables provide a characteristic size to represent the material in head loss calculations rather than the specific surface area required when using a correlation such as the NUREG/CR-6224 head loss correlation. In head loss discussions in Section 3.7 of the GR, the characteristic size is used to estimate the specific surface area from simple geometric formulas. The staff is concerned with the method of converting characteristic dimensions into specific surface area because it has been demonstrated that the method shown in Section 3.7 is not reliable. This concern is particularly important when estimating a specific surface area for a particulate with a distribution of particle

sizes where the tendency of using the mean of the size distribution is incorrect and leads to an underestimate of the specific surface area that, in turn, can lead to an underestimate of the head loss. This issue is further discussed in the staff evaluation of Section 3.7 of this SER. Confirmatory research presented in Appendix V was performed that illustrates the application of simple geometric equations (e.g., 4/d for fibers and 6/d for particles).

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Staff Conclusions Regarding Section 3.4.3.6: The staff concludes that acceptance of this section depends upon each plant-specific evaluation properly determining that the parameters selected for the analysis adequately reflect the insulation types actually used in that containment, and that the specific surface area used in the head loss calculation is properly determined.

The staff did not independently verify all the data located in GR Tables 3-2 and 3-3, however, the values presented agree with analyst perceptions for these materials.

Failed Coatings

The GR assumes that all failed coatings generate debris sizes equivalent to the coatings' basic constituent or pigment sizes which the methodology identifies as 10μm. The GR chose this value because experimental evidence was lacking regarding coating debris size generation during a postulated event. The industry pressure wash testing detailed in Appendix A of the GR provided some insight that coatings within the ZOI will likely fail by erosion resulting in debris sized in the range of 10μm - 50μm spheres. The testing also provided insight that the qualified epoxy and qualified IOZ coating that were tested would not fail as chips or sheets during simulated jet impingement testing. Coatings outside the ZOI that fail are also assumed to generate debris in sizes equivalent to their basic constituents or pigment sizes. This debris on the order of 10μm spheres.

Staff Conclusions Regarding Failed Coatings: For plants that substantiate a thin bed, use of the basic material constituent (10 μm sphere) to size coating debris is acceptable.

For those plants that can substantiate no formation of a thin bed at which particulate debris can collect, the staff finds that coating debris should be sized based on plant specific analyses for debris generated from within ZOI and from outside the ZOI, or that a default area equivalent to the area of the sump screen openings be used. If analyzed, then such an analysis should conservatively assess the coating debris generated with appropriate justification for the assumed particulate size or debris size distribution. Degraded qualified coatings that have not been remediated should be treated as unqualified coatings.

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Finally, testing regarding jet interaction and coating debris formation could provide insight into coating debris formation and help remove some of the potential conservatism associated with treating coatings debris as highly transportable particulate. If coatings, when tested at corresponding LOCA jet pressures and temperatures, are found to fail by means other than erosion or the erosion is limited, the majority of debris may be larger, less transportable or pose less of a concern for head loss.

3.5 LATENT DEBRIS

3.5.1 Discussion

Section 3.5.1 of the GR provides a discussion of general considerations for latent debris regarding its potential impact on sump screen blockage and some variables that should be addressed on a plant specific basis. The five generic activities outlined in the GR needed to quantify and characterize latent debris inside containment (1. Estimate horizontal and vertical surface area, 2. Evaluate resident debris buildup, 3. Define debris characteristics, 4. Determine fractional surface area susceptible to debris buildup, and 5. Calculate total quantity and composition of debris) provide a working outline of the process.

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Staff Evaluation for Section 3.5.1: The staff finds the GR guidance with respect to general considerations for latent debris to be acceptable. The staff agrees with the position in the GR that latent debris present in containment during operation may contribute to head loss across the emergency core-cooling sump-screens, and that it is necessary to determine the types, quantities and locations of latent debris. The staff also agrees that it is not appropriate for licensees to claim that their existing Foreign Materials Exclusion (FME) programs have entirely eliminated miscellaneous latent debris. Results from plant-specific walkdowns should be used to determine a realistic amount of dust and dirt in containment and to monitor cleanliness metrics that may be deemed necessary following the overall sump screen blockage vulnerability assessment.

For more detailed analysis, the staff believes that when characterizing the resident debris buildup it would be useful to partition the inventory not only by vertical and horizontal location but also by relationship to spray impingement and washing by containment-spray drainage.

3.5.2 Baseline Approach

The introduction provided in this section of the GR provides practical insights into the level of importance that latent debris may take in the overall vulnerability assessment and helps licensees to judge the level of effort needed to characterize their plants. In this section, NEI acknowledges that latent debris should be considered as an input to sump screen head loss, and recommends the use of conservative strategies rather than evaluating the effects of latent debris to a high level of detail.

Staff Evaluation for Section 3.5.1: The staff finds the GR guidance with respect to the introduction of the baseline approach for consideration of latent debris to be acceptable. For plants that expect to have fibrous insulation debris generated in the ZOI, the additional contribution to head loss from the latent fiber component may be small by comparison and reasonable approximations of inventory will suffice. However, for predominantly RMI plants, the latent fiber component represents the dominant potential for thin-bed formation across the screen. In any case, accurate fiber inventories can provide valuable insight for critical decisions regarding sump-screen vulnerability.

3.5.2.1 Estimate Horizontal and Vertical Surface Area Inside Containment

This section of the GR provides a general outline of steps required to estimate the horizontal and vertical surface areas in containment. The bulleted list of items that should be included in the surface area calculation (floor area, walls, cable trays, major ductwork, control rod drive mechanism coolers, tops of reactor coolant pumps, and equipment such as valve operators, air handlers, etc.) provides a starting point for licensees to consider for major inputs. The five steps provided for surface-area calculations (flat surface considerations, round surface area considerations, vertical surface area considerations, thorough calculation of surface areas in containment, and use of estimated dimensions when exact dimensions are unavailable) are informative.

Staff Evaluation for Section 3.5.2.1: The staff finds the GR guidance for estimating surface areas within containment to be acceptable with provisions outlined below for specific sections/attributes.

The staff agrees that the quantity of ambient dust and dirt collected on vertical surfaces by settling from the air is small compared to that collected on horizontal surfaces in the absence of factors that promote adhesion to those vertical surfaces. Any special factors that might promote adhesion to vertical surface should be noted and examined more carefully for dust accumulation. A list of potential adhesive factors includes oil leaks, moisture or condensate laden surfaces, residue from previously sprayed oils or solutions, and detergent films. Dust that accumulates on vertical surfaces is very small and should be assumed 100% transportable if affected by water during a LOCA.

Other surfaces that should be considered for inclusion in plant-specific inventory estimates include steam generators, pressurizers and pressurizer relief tanks, cooling fans, other large equipment, structural supports like I-beams and seismic restraint collars, access gratings and steps, and piping. In general, the area inventory refers to external surfaces that can be affected by spray wash down. Internal compartments and cabinets with known loadings of dust and debris that are not typical of most surface conditions after containment close out should be examined carefully for water infiltration and potential flushing. Areas of this type include inlet-air filter housings and confined crawl spaces that are accessed infrequently.

The guidance provided in the GR for surface-area calculations treats the contribution of vertical surfaces in an inconsistent manner. In general, the staff agrees that practical simplifications can be made to simplify estimates of surface area, and the 10% factor proposed for general vertical surfaces is an acceptable estimation based on engineering judgment. However, vertical surfaces that are subject to enhanced dust and debris accumulation should be added to the latent debris load estimation separately as part of the resident debris buildup evaluation in Section 3.5.2.2. Additional guidance for considerations to be included in containment surveys for latent debris loading is provided under that section.

The staff agrees that the containment dome does not need to be considered from the point of view of dust accumulation. However, the dome may be a contributor of degraded coatings that are dislodged during vapor expansion and should be addressed as such in the determination of the coatings debris source term.

In its present form, the baseline guidance requires detailed calculations of both horizontal and vertical surface areas and physical surveys of dust accumulation on horizontal surfaces (Section 3.5.2.2.1). To improve consistency in the treatment of vertical surfaces, the staff provides the following two acceptable alternative options for baseline analysis based on the best available information documented by the industry:

Option 1: Adopt a default vertical-surface inventory of 30 lbs to be characterized by the smallest size fraction found in the horizontal-surface inventory and document a simplified but realistic calculation of vertical surface area. Consideration should still be given to the unique deposition areas discussed above and the results should be added to the default vertical inventory. This value is approximately 5 times (established by using a 2 standard deviation expansion from the mean of the reported sample data set to achieve a 95% coverage of the expected data curve, then doubling the result for conservatism) higher than the vertical inventory reported in Appendix B for concrete walls and the containment liner and should be sufficiently high to bound variations in surface area, plant cleanliness and the additional vertical areas represented by piping and equipment.

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Option 2: Conduct swipes for three categories (a, b, c) of vertical surfaces in the manner illustrated in Appendix B of the GR. It should be noted that repeated wiping with a lint-free cloth (Maslin) under manual pressure or HEPA-filtered vacuuming with mild brush agitation of the surface are both effective methods for collecting the full spectrum of particle sizes found on surfaces, and both methods provide collection media that can be weighed before and after collection to determine the mass of debris in the sample (see Appendix VII). Concrete walls (a), the liner (b), and vertical piping/equipment (c) should each be sampled at a minimum of three locations selected and documented by simple rationale to represent typical variations in expected dust loadings within containment. For example, walls near the equipment hatch might represent maxima and the upper containment liner might represent minima. Document a simplified but realistic calculation of vertical surface area for each category of surface that is sampled and use the average of the three (or more) measurements to determine the mass present on vertical surfaces of each surface category. Add the three subtotals to the inventory estimate obtained from any unique deposition areas. If recently cleaned surfaces are used to establish the minima for a surface category, a documented cleanliness plan should be referenced that describes the frequency of this cleaning treatment. This option represents a minimal increase in effort over that required in the GR, namely the collection of vertical-surface swipes, and yet allows maximum credit for individual variations in plant cleanliness.

The staff agrees with Step 2 in the GR regarding the treatment of round surfaces, but notes that piping surfaces should be considered. Steps 4 and 5 also provide some practical recommendations that are acceptable.

3.5.2.2 Evaluate Resident Debris Buildup

Section 3.5.2.2 of the GR provides a high level discussion of general practices needed to evaluate latent debris buildup in containment. The GR cites recent sampling of surfaces inside of containment at a number of plants, and recommends surveys of the containment be performed with the objective of determining the quantity of latent debris. This information is not available in the public domain to allow confirmation of consistency in sampling methods and reporting practices, so any statement of expected maximum dust inventory should be considered speculative. The GR references NEI 02-01 to provide guidance for conduct of these containment surveys and evaluation of the presence of foreign material found. The GR also suggests that the degree of rigor for containment survey and surface swiping be applied in inverse proportion to the attention given to foreign material exclusion under normal operations.

Staff Evaluation for Section 3.5.2.2: The staff finds the GR guidance with respect to the practices for overall evaluation of latent debris to be acceptable provided the provisions outlined below are incorporated into the site-specific surveys for latent debris in containment. These surveys will produce opportunities to maximize credit for plant cleanliness, and identify areas of higher than expected debris loadings.

To ensure a comprehensive evaluation of containment debris, the following items should be considered as part of the containment survey: Phenomena that can enhance dust collection on both vertical and horizontal surfaces include temperature gradients (thermophoresis) and static electrical charge (electrophoresis). The vertical surfaces of cooling fins, heat exchangers and warm electrical panels may attract higher concentrations of dust than painted concrete structures. Hanging lamp shades inside containment are a common location for enhanced dust collection caused by the thermal gradient. Static charge may be accumulated on any surface exposed regularly to air flow. Dielectric materials such as plastics and exposed cable jackets may be principle candidates for inspection. For some plants, these effects and locations may be minor contributors to the total dust inventory, and can be dismissed with proper examination. However, these issues should be considered and their disposition documented.

For the purposes of latent debris characterization, surveys taken after every second outage should be sufficient. Exceptions to this schedule warrant surveys after any invasive or extended maintenance like steam-generator replacement.

3.5.2.2.1 Evaluate the Resident Debris Buildup on Surfaces

This section of the GR focuses on the measurement of dust and dirt found on horizontal surfaces of containment. The four steps presented in the GR (1. Divide the containment into areas based on robust barriers, 2. Determine representative surfaces for each section of containment, 3. Survey the representative surfaces in each section to measure debris quantity, and 4. Calculate the thickness of the debris layer) describe the process. Of these, steps 1 and 2 offer practical and thorough guidance for performing a systematic survey. The primary method for determining latent debris inventory suggested in items 3 and 4 of the GR is direct measurement of debris thickness.

Staff Evaluation for Section 3.5.2.2.1: The staff finds the GR guidance with respect to division of containment areas (step 1) and determination of representative surfaces (Step 2) to be acceptable, however, the methods identified for measuring and evaluating the buildup of debris on surfaces to be unacceptable. The recommendation in the GR for direct measurement of dust thickness is considered impractical. A revised approach for the assessment is offered here that is based on generic characterization of actual PWR debris samples. The revised approach also addresses the question of particulate to fiber ratio as it relates to thin bed effect. If desired, a limited plant-specific characterization can also be pursued as a refinement using this guidance.

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Attempting to directly measure latent debris thickness is not recommended for the following reasons: (1) masses can be measured much more accurately than thickness, (2) comparison of dirt layers to reference thickness standards is subjective and prone to error because of heterogeneous small objects that may reside on the surface and because of nonuniform dust thickness across a surface like piping, (3) in situ estimates of thickness do not characterize size distributions, particulate-to-fiber mass ratios or densities that are needed to define hydraulic head-loss properties. These problems can

Deleted: satisfy the requirements to be avoided by measuring total masses within a known surface area and then partitioning the fiber and particulate mass fractions either by physical measurement or by generic assumptions described in the next section.

Statistical sample mass collection is an acceptable method for quantifying latent debris inventories. This approach will not pose an undue burden if planned in advance and incorporated with other survey activities. A list of unique debris sample locations should be developed starting with the previous discussion in Section 3.5.2.1 that can be checked for each evaluation zone that is defined in containment. For convenient cross reference these evaluation zones should be defined to coincide with the break zones discussed in Section 3.4. For later input in debris transport assessment, the potential for exposure to water from either direct containment spray, containment-spray drainage, or recirculation-pool immersion, should be noted for the surfaces in each evaluation zone. Other areas that should be included in the survey include annular compartments outside of the bioshield and the reactor cavity if the area participates in circulatory flow with the sump pool during recirculation. Using the practical guidance offered in GR Section 3.4.2.2.1 item 2 for selecting typically loaded surfaces within each inventory evaluation zone, several classes of horizontal surfaces should be defined to represent places where latent debris are found. For example, high and low traffic floor areas, tops of equipment, floor near curbing, cable trays, etc. At least three samples should be taken from each category as they appear throughout containment and the results should be treated in the same manner described for vertical surfaces.

The goal of defining debris characteristics is satisfied by collecting swipe or vacuum-filter samples that can be weighed before and after collection to determine the total mass of debris within a measured area. It is important that the collection method adequately capture the full range of particulate sizes from very small (< 10 μm) up to the large miscellaneous chips and pieces, and all fibers in the sample region. Both HEPA-filtered vacuuming with light brush agitation of the surface and repeated swiping under manual pressure with a Maslin cloth were found to be effective collection methods for fine particulates and fiber. Vacuuming is considered more efficient for collecting larger grains and miscellaneous objects. Scraping with a metal blade or sweeping with a bristle-type brush will not adequately collect the full range of debris [DIN04].

3.5.2.2.2 Evaluate the Quantity of Other Miscellaneous Debris

Section 3.5.2.2.2 of the GR provides general guidance for considerations to be used for identifying and evaluating potential sources of miscellaneous debris in containment. The GR refers to and endorses the use of NEI 02-01 to provide guidance for performance of containment surveys. A list of three bulletized items, Equipment tags, Tape, and Stickers or placards affixed by adhesives; is used to provide guidance for these specific sources of latent debris.

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Staff Evaluation for Section 3.5.2.2.2: The staff finds the GR guidance with respect to the methods to identify and evaluate miscellaneous debris acceptable provided the guidance is supplemental with the additional direction identified below. The staff agrees that surveys of containment for the presence of miscellaneous debris should be performed and that miscellaneous debris types should be assessed for potential contributions to sump-screen head loss. In addition to the three categories of miscellaneous debris discussed in the GR; the quantity, characteristics and location of any failed qualified coatings should also be noted in the survey. This issue may be addressed elsewhere in the GR, but it warrants emphasis in this section as well.

- Without specific data to cite regarding the behavior of miscellaneous debris types, the phrases "available for transport" and "transportable debris" should be interpreted as "complete transport to the screen" for fines and particulate debris under the conditions of interaction with water. Larger miscellaneous debris types must be evaluated on a case by case basis for susceptibility to transport as outlined in section 3.6. If data on disintegration and transport become available, they should be documented and used as an acceptable refinement to quantify an assumption of partial degradation or partial transport. If applicable, refinements should include a plausible timeline or necessary operating condition for failure. For example, if adhesives are shown to fail after hours in containment, large or heavy stickers and signs may become detached, but still may not transport in low-velocity recirculation conditions. Or, delayed failure of adhesives on upper levels of containment may not lead to transport if containment sprays are no longer operating. Proper consideration should be given to the location of these items and the logic of the rationale that is used. For example, slow softening of adhesive in a high-humidity environment is much different than erosion by spray-water cascade or break-jet impingement. The following additional guidance is offered on the evaluation of the GR bulletized categories of latent debris.

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- Equipment tags: The GR guidance provided on the post-LOCA status of paper tags is ambiguous. There is an implied assumption that complete tags arriving at the screen will induce more head loss than shredded or dissolved paper fiber contributing to a mixed debris bed. Regardless of their physical condition, tags can only contribute to head loss if they are transportable. Robust lanyards and attachment methods should prevent most equipment tags that exist outside the ZOI from becoming detached (equipment tags within the ZOI shall be assumed to become detached). The size and weight of detached equipment tags and broken lanyards should be evaluated against criteria in Section 3.6 to determine if they should be considered transportable debris. For all equipment tags that are found to be potentially transportable, it is necessary to determine the number and location of tags by type for contribution to screen head loss. If transportability or the capability of tags to remain intact cannot be determined, to preserve conservatism it should be assumed that they remain intact and are transported to the sump screen. In this case, the wetted sump-screen flow area should be reduced by an area equivalent to the original single-sided surface area of the tags. If there is information that indicates the tags will not remain intact, the staff recommends that the equivalent mass of the tags be treated as latent fiber.

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- Tape: The GR mentions some specific applications of tape and recommends that all tape be assumed to fail as transportable debris. The staff agrees that the size, weight, and composition of tape that would interact with water should be evaluated for transportability per Section 3.6 to determine the realistic amount that would arrive on the sump screen. As stated in the GR for equipment tags, all failed tape that is determined to be transportable should be assumed to arrive on the screen intact and obstruct an area equivalent to its original single-sided surface area unless there is evidence that the tapes will not remain intact. If there is evidence that the tapes will not remain intact, for example prior in-service disintegration, then the equivalent mass of the tape should be assumed to be transported to the screen in the form of latent fiber.

- Stickers or placards affixed by adhesives: The staff agrees with the position in the GR that adhesives may fail in post-accident conditions. Under the present guidance offered in the GR, all items attached by adhesives should be assumed to fail and be evaluated for transport to the sump screen as outlined in Section 3.6. The staff considers this an acceptable position. Where evidence is available that these items will degrade, the equivalent mass of the items in question should be assumed to be transported to the sump screen in the form of latent fiber. Otherwise, the wetted flow area of the sump screen should be reduced by the original single-sided area of the items in question.

3.5.2.3 Define Debris Characteristics

This section of the GR notes that two generic methods can be applied for defining debris characteristics: Method 1 - analysis of samples, or Method 2 - assume composition and properties based on conservative values. NEI indicates that the second option (assume conservative values for debris composition properties) is preferable, and provides parameter values for fiber density, particle density, and particle diameter. The GR notes that for this option to be used, an appropriate fiber/particulate mix for the plant being evaluated should be used. The GR goes on to describe some of the difficulties and challenges associated with Method 1 – analysis of samples.

Staff Evaluation for Section 3.5.2.3: The staff finds the GR guidance with respect to defining debris characteristics to be acceptable provided the method used is supplemented with the additional details outlined below.

It should be noted that conservatism with respect to head-loss potential includes both the aspects of transportability and the hydraulic properties of the material in a mixed debris bed. The four GR bullets for evaluating debris characteristics will be addressed in a parallel format that discusses the Method 1 and Method 2 approaches to each topic concurrently. Both methods first require that adequate surface samples be taken to characterize variability in the plant and that total masses in containment be estimated by multiplying the empirically determined concentration for each type of collection area (g/ft^2) by the corresponding surface areas before summing to obtain the total inventory. Since the GR indicates that Method 2 – assume composition based on conservative values, is the preferred choice, it will be addressed first for each bullet provided.

First GR Bullet – use an appropriate fiber/particulate mix for the plant being evaluated.

> Method 2 – Assume that fiber contributes 15% of the mass of the total estimated inventory. If abnormal qualified coating conditions indicate a dominant presence of paint chips compared to normal dust and dirt at a particular sampling location, that location should be characterized by measurement under Method 1. (see Appendix VII on Latent Debris for more specific information)
>
> Method 1 – Characterize the fiber-to-particulate mass ratio in the plant by wet rinsing and manual separation of the fibers from the particulates followed by drying and weighing to obtain mass ratios for samples taken. If this option is chosen, HEPA filtration is recommended as the preferred collection method because of easier separation of the debris from the filter.

Second GR Bullet – Fiber density

- It is conservative to assume that all fiber exposed to water transports to the screen (unless special circumstances are noted as discussed earlier), but material buoyancy is not the primary contributing factor and the density should not be assigned equal to that of water.

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Method 2 – Assume that latent fiber material has a mean density of 1.5 g/cm^3.

Method 1 – Immerse dry fiber samples of known mass in a graduated cylinder with a known quantity of water. Cover with plastic film to prevent evaporation and let stand for several days or heat gently to remove trapped air. Measure new volume of contents and determine fiber material density by displacement.

Third GR Bullet – Particle density

- It is appropriate to assume that latent particulates are primarily geophysical in origin being composed of soil, sand and dust, i.e., "dirt."

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Method 2- Assume latent particulate material has a nominal density of 2.7 g/cm^3.

Method 1 – Measure the particulate density by water displacement as described above for fiber.

Fourth GR Bullet – Particle Diameter

- The principal use of particle diameter is in the estimation of hydraulic properties of the debris like the specific surface area. This information can also affect judgments regarding transportability and retention in a fibrous debris bed.

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Method 2 – The GR provides the guidance to assume all particulate mass is composed of 10-μm diameter grains. The staff considers this assumption to be acceptable, but this approach is very conservative; especially when much of the mass may be composed of small paint chips, hardware, and visible sand grains. However, this assumption offers the convenience of consistency with baseline assumptions applied to failed coatings as mentioned in the GR. A more refined set of assumptions that would also be considered acceptable are as follows:

- Assume that typical mixtures of latent particulate debris have a specific surface area of 106,000 ft^{-1} as defined for use in the NUREG/CR-6224 head-loss correlation.
- Assume that 22% of the particulate mass determined from the raw samples that is above the recirculation-pool flood level is nontransportable.
- Under conditions of low sump-screen flow <0.2 ft/s and estimated particle-to-fiber mass ratios <3, assume that 7.5% of the latent particulate debris penetrates the sump screen and is not permanently deposited in the bed to contribute to head loss.

Method 1 – Dry sieve particulates into size fractions down to 75-μm and characterize the mass distribution as a function of diameter. Assume that the fraction > 2mm is not transportable. Assume that 25% of the 75-μm diameter mass fraction can penetrate the debris bed. Use scanning electron microscopy (SEM) on subsamples of the 75-μm fraction to determine statistically the fraction of particles below 10-μm diameter. Compare measured size distributions to literature reported determinations of latent debris size distribution and adjust the Method 2 specific surface area by ratios of estimated masses in each size bin.

- Two additional factors needed that are not mentioned in the GR:

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- The dry-bed accumulated density of latent fibers is needed for head-loss calculations. For fiberglass, this density is typically reported as the "as manufactured" density but there is no equivalent definition for latent fiber.

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  Method 2 – Assume the dry-bed bulk density for latent fiber is equal to that of fiberglass insulation (2.4 lbm/ft^3 = 38.4 kg/m^3).

  Method 1 – Using the dry fiber component obtained from the Method-1 measurement of fiber-to-particulate mass ratios, separate fibers and small flocks from a sample of known mass and drop them successively through several inches of air into a graduated container. Measure the volume after a bed has been formed by random settling and compute the bulk density of this configuration.

1. The fiber specific surface area is also needed for head-loss calculations to compute the contributions to head loss of latent fiber in a mixed debris bed.

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   Method 2 – Assume the head loss properties of latent fiber are the same as reported in NUREG/CR-6224 for commercial fiberglass. Latent fiber will either be dominated by fiberglass present from the break location or it will form the substrate of a "thin-bed" particulate filter and be dominated by the particulate bed forming on top of the fiber. In either case, the exact properties of the latent fiber are dominated by another debris type, so the error associated with the assumption should be small.

   Method 1 – Measure the hydraulic properties of latent fiber by inference using iterative comparisons of head-loss data and model predictions using the NUREG/CR-6224 head-loss correlation.

The staff agrees with all of the cautionary notes provided in the GR regarding the difficulties of debris characterization except for the presumptive judgment of extreme expense and little benefit. While cost/benefit is an important practical consideration, the NRC never discourages well-documented testing to obtain site-specific information. For some of the more simple steps of the analysis, it may be an immediate benefit to characterize plant conditions more completely than the default assumptions permit. Improved particulate-to-fiber mass ratios, for example, may offer an immediate potential benefit because of the key role latent fiber plays in the assessment of vulnerability for thin-bed formation in a predominantly RMI-insulated plant.

3.5.2.4 Determine Fraction of Surface Area Susceptible to Debris Accumulation

The guidance in this section of the GR is again offered in the form of a baseline approach. The GR offers the two following options for guidance: 1. Assume that 100 percent of the surface area is susceptible to debris accumulation; and 2. Perform an evaluation that consists of estimating fractional surface areas susceptible to debris accumulation on a case-by-case basis. The intent of the guidance in this section is to offer credit for cleanliness programs exercised in certain parts of containment. The GR provides a basic approach for reducing the area considered susceptible to debris accumulation through three bulleted items: 1. calculate the total surface area; 2. calculate the surface area considered to be clean using conservative assumptions; and 3. calculate the ratio of potentially dirty area to total area.

Staff Evaluation for Section 3.5.2.4: The staff finds the GR guidance with respect to fractional surface area susceptible to debris accumulation acceptable with the provisions outlined below:

To implement the baseline approach, the GR intended for a measurement to be made of the thickness of dust on a representative surface within each inventory evaluation zone and that this thickness would be multiplied by the total relevant area in the zone to obtain the volume of debris. This approach is not considered reliable due to the difficulty and subjectivity of measuring a debris thickness as discussed in 3.5.2.2.1

Either approach presented in this section of the GR for establishing a fractional surface area for debris accumulation is acceptable to the staff with the following caveat: If areas are excluded from the surface inventory, documented cleaning procedures should be in place that are exercised before each restart. If periodic cleaning occurs less frequently, the sampling method outlined earlier in this SER is recommended to determine the minimum dust loading in those areas of a surface type that have been previously cleaned.

An issue similar to accumulation susceptibility that may lead to a credit for reduced latent inventory is transport susceptibility. As recommended earlier in this SER, potential exposure to water should be assessed for each inventory evaluation zone. It is expected that most surfaces will be exposed to either direct spray, spray accumulation flow, or immersion in the recirculation pool, but some isolated areas may exist for which little or no water transport can occur (interior cabinets, elevated crawl spaces, locked rooms, etc). For these types of areas where latent debris is known or expected to exist, justification for exemptions ~~from~~ considering for the total latent-debris inventory can be documented on a case-by-case basis.

3.5.2.5 Calculate Total Quantity and Composition of Debris

The GR provides four basis steps for calculation of the total quantity of latent debris: 1. Perform calculations as previously outlined on an area-by-area basis; 2. Compute the total quantity of debris using the area/debris thickness method outlined in the GR; 3. Include other types of debris from containment survey data as outlined previously in the GR; and 4. Categorize and catalog the results for consideration in debris transport evaluation.

Staff evaluation for Section 3.5.2.5: The staff finds the general steps identified with respect to the total process acceptable provided that methods outlined previously in this SER are used in place of those specific items previously identified for computation of quantity of debris and debris density.

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The process for integrating survey findings over all surface types has been alluded to several times in this SER review. Given the revised approach to measurement of debris build up recommended by the staff, the total quantity of debris for each inventory evaluation zone and each surface type will be found by multiplying debris concentration (lbm/ft^2) by the respective areas to obtain the total number of pounds in containment. Proper evaluation of debris for transportability has been discussed previously in other sections of this SER pertaining to evaluation of debris types. Most importantly, the calculation must separate the fiber and particulate components of the debris aggregate.

These fractions behave differently during transport, contribute separately to head loss, and introduce separate considerations regarding sump-screen vulnerability.

3.5.3 Sample Calculation

The sample calculation presented in this section of the GR illustrates the concept and systematic process involved with defining categories of surfaces that reside within a given inventory evaluation zone, calculating areas, and summing debris inventories. Minor points of clarification are offered in the following sections.

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3.5.3.1 Calculate Horizontal Surface Area

This section of the GR illustrates the appropriate level of simplification for computing structural surface areas in containment.

Staff Evaluation for Section 3.5.3.1: The staff finds the sample calculations provided to be acceptable for implementing concepts for calculation of horizontal surface areas in containment. The following clarifications are added for licensees to consider when performing these calculations:

Step 4 of the calculation discusses the calculation of additional horizontal surface areas contributed by equipment, piping, cable trays, etc. Where these items are large and obstruct floor areas computed in previous steps, the projected area of the item is effectively included twice. The duplicate area can either be subtracted from the inventory or cited as a conservatism to account for the complexity of the object in question, whichever is most appropriate.

The treatment given to the recirculation-sump cover as a projected area accounted for in the floor-area calculation is appropriate.

3.5.3.2 Calculate Quantity of Debris

The example calculation in the GR is consistent with guidance given in previous sections assuming that a debris-layer thickness can be measured and that in situ densities can be determined; total latent-debris mass is then computed accordingly.

Staff evaluation for Section 3.5.3.2: The staff finds the GR guidance with respect to the total calculation of quantity debris to be unacceptable. The problems associated with direct measurement of debris thickness have been explained. If inventory analysis options involving sampling are pursued, it might be practical to conduct calculations like the one illustrated in this example.

3.6 DEBRIS TRANSPORT

3.6.1 Definition

Section 3.6 provides guidance for estimating debris that is transported from debris sources to the sump screen. The four major transport modes considered in the GR are blowdown, spray washdown, pool fill-up, and pool recirculation flow.

3.6.2 Discussion

Section 3.6.2 presents a generic transport logic tree used subsequently in the transport recommendations. In addition, three containment type categorizations are also defined. These categories are:

1. Highly compartmentalized containments defined as those containments that have distinct robust structures and compartments totally surrounding the major components of the RCS. For a main steam line break in a highly compartmentalized containment, the mostly uncompartmentalized containment values should be used.

2. Mostly uncompartmentalized containments defined as those containments that have partial robust structures surrounding the steam generators.

3. Ice condenser containments defined as all seven ice condenser plants, which lack lower containment compartmentalization.

Staff Evaluation for Section 3.6.2: The simple generic debris transport chart shown in GR Figure 3-2 is acceptable for a schematic representation of the GR baseline debris transport evaluation methodology. However, the distinction between the highly compartmentalized and mostly uncompartmentalized containments has not been clearly defined. Therefore, if the containment category in a plant specific analysis is not certain then the evaluation should assume the category that predicts the greater debris accumulation on the sump screens. The acceptance of the baseline guidance as a package is the subject of Section 3.8.

3.6.3 Debris Transport

The introduction to Section 3.6.3 introduces the NEI baseline concept for estimating debris trapped in inactive pool volumes defined as volumes located below the containment bottom floor (e.g., the cavity under the reactor vessel) that are not affected by drains from the upper part of the containment that may cause them to participate in the active volumes. All volumes at the containment bottom floor elevation are assumed to participate in the recirculation flow path from the containment sprays and break flow to the sump. The baseline model assumed no preferential direction for water to flow to the sump. Further, the baseline guidance assumes that all debris in the containment bottom floor is uniformly distributed throughout the entire volume of water in containment. This guidance then assumes that the debris transported to the inactive sumps is strictly based on the ratio of the volume of the inactive sumps to the total water volume in containment at the start of recirculation. The baseline guidance states that this assumption is conservative since it ignores the preferential sweeping of the debris on the containment floor to the inactive sumps by the thin sheets of high-velocity water. It was further noted that all small fine debris in active pools on the containment floor is transported to the sump during recirculation.

Subsections 3.6.3.1, 3.6.3.2, and 3.6.3.3 which address the highly compartmentalized, the mostly uncompartmentalized, and the ice condenser containments, respectively, primarily contain compartmental specific debris transport assumptions. These assumptions are summarized in Table 3-4 for the small fine debris generated within the ZOI. The baseline guidance recommends that all debris generated outside the ZOI be treated as small fine debris that subsequently transports to the sump screens (i.e., 100%

washdown transport, 100% sump pool recirculation transport, and no transport into the inactive pools). The baseline guidance recommends the assumption that all of the large piece debris deposits onto the containment bottom floor where it stays.

Table 3-4. Summary of Debris Transport Assumptions for Small Fines Debris from ZOI

Transport Assumption	Fibrous Debris	RMI Debris	Other Debris
Highly Compartmentalized Containments			
Fraction of Debris Generated	0.6	0.75	1
Fraction of Debris Generated that Transports into Upward Levels by Blowdown	0.25	0.25	0.25
Fraction of Debris Generated that Transports Directly to Sump Pool Floor by Blowdown	0.75	0.75	0.75
Fraction of Debris Generated that Blows into Upper Levels and Washes Down into Sump Pool	1	0	1
Fraction of Debris Generated that Enters into Inactive Sump Pools	Volume Ratio	Volume Ratio	Volume Ratio
Fraction of Debris that Enters Sump Pool that Transports to Sump Screens	1	1	1
Mostly Uncompartmentalized Containments			
Fraction of Debris Generated	0.6	0.75	1
Fraction of Debris Generated that Transports into Upward Levels by Blowdown	0*	0	0
Fraction of Debris Generated that Transports Directly to Sump Pool Floor by Blowdown	1*	1	1
Fraction of Debris Generated that Blows into Upper Levels and Washes Down into Sump Pool	1	0	1
Fraction of Debris Generated that Enters into Inactive Sump Pools	Volume Ratio	Volume Ratio	Volume Ratio
Fraction of Debris that Enters Sump Pool that Transports to Sump Screens	1	1	1
Ice Condenser Containments			
Fraction of Debris Generated	0.6	0.75	1
Fraction of Debris Generated that Transports into Upward Levels by Blowdown	0.1**	0.1**	0.1
Fraction of Debris Generated that Transports Directly to Sump	0.9	0.9	0.9

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Deleted: Washdown Transport Fraction

Deleted: Transport into Inactive Pools

Deleted: Sump Pool Recirculation Transport

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Transport Assumption	[illegible]ous [illegible]bris	RMI [illegible]	[illegible]ther [illegible]
Pool Floor by Blowdown			
Fraction of Debris Generated that Blows into Upper Levels and Washes Down into Sump Pool	1	0	1
Fraction of Debris Generated that Enters into Inactive Sump Pools	Volume Ratio	Volume Ratio	Volume Ratio
Fraction of Debris that Enters Sump Pool that Transports to Sump Screens	1	1	1

*Because this value was not actually specified in the baseline guidance (Section 3.6.3.2, fibrous blowdown transport), the table value was assumed to be the same as the stated RMI value.

** Guidance assumes 100% ejected upwards of which 90% returns via ice melt to containment floor.

Staff Evaluation for Section 3.6.3: The staff's evaluation of this section was based on confirmatory research documented in Appendices IV and VI and the base of debris transport knowledge documented in NUREG/CR-6808.

Table 3-5 shows that the baseline recommendations for the fractions of the debris generated that transports into upward levels by blowdown were 0.25, 0, and 0.1 for highly compartmentalized, mostly uncompartmentalized, and ice condenser containments. These fractions are conservative. The detailed analysis performed for the volunteer plant, which was assumed to have a highly compartmentalized containment, the fractions were 0.92 and 0.44 for small fines fibrous and small RMI debris compared to 0.25 recommended for the baseline analysis (Appendix VI to this report). For mostly compartmentalized containments, the GR recommended no debris transporting to the upper containment. For ice condenser containments, the GR recommended value of 0.1 is conservative because these containments are designed to divert a significantly higher fraction of blowdown flow towards the ice condensers.

The inactive pool debris entrapment model does not represent the realities of debris transport. In the detailed volunteer plant debris blowdown/washdown transport analysis (Appendix VI to this report), a majority of the small fine debris was determined to transport upwards in the containment where it deposited onto any number of surfaces. Only a few percent of the small fines would likely deposit directly onto the containment bottom floor where the debris would be subjected to pool formation flows into the inactive volumes. Note that in the volunteer plant, the openings into the bottom sump level floor consisted of two personnel access doorways, which are small compared to the large area that opens directly to the containment dome. The large opening was designed for pressure relief from HELB events in the steam generator compartments housing most of the RCS. A significant time delay would most certainly exist between the blowdown period and the time when major portions of the small fines would be transported down to the sump pool by the containment spray drainage. Therefore, the inactive pools would most likely fill (first few minutes) before a large portion of the debris could wash to the sump pool, hence the assumed volume ratio is non-conservative.

The baseline guidance assumes that the debris transported to the inactive sumps is strictly based on the ratio of the volume of the inactive sumps to the total water volume

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in containment at the start of recirculation. The baseline guidance states that this assumption is conservative because the debris transport methodology ignored the preferential sweeping of the debris on the containment floor to the inactive sumps by the thin sheets of high-velocity water. This basis does not reflect realistic debris transport. Observations made during the integrated tank tests [NUREG/CR-6773] show debris being directionally driven by sheeting flow wave front. Such transport could drive debris across the tank bottom (either away from or to the sump) unless the debris became otherwise trapped along the transport path. With this type of sheeting flow transport of fine debris, a sharp direction change, such as at an entrance into a hallway leading to the reactor cavity, could easily result in the debris being swept past such an entrance because the debris was unable to alter direction with flow into the doorway. Since it is difficult to determine how sheeting flow would actually transport debris, the amount of conservatism achieved by ignoring the preferential transport of debris to the inactive volumes is difficult to quantify.

The baseline assumption that all debris in the containment bottom floor is uniformly distributed throughout the entire volume of water in containment also does not reflect reality, certainly not in the general sense of all PWRs. The volunteer plant detailed analysis of a line break within a steam generator compartment indicated that more of the blowdown-deposited debris on the bottom floor was likely retained within the affected steam generator compartment than being transported outside the compartment. Hence a substantial concentration of debris would initially be located in the affected steam generator compartment. Although the washdown debris would enter the sump pool at multiple locations with the containment spray drainage, the entry points would place the debris directly into the sump pool flow stream rather than into inactive pools or inactive or quieter portions of the sump pool.

The inactive pool debris entrapment model can predict unrealistically high fraction of debris moving into inactive pools for some plants. Therefore, the licensees should limit the fraction of debris moving into inactive pools to a maximum of 15% of the source unless shown otherwise by analysis as described in Appendix IV.

Table 3-4 shows that the only distinguishing feature among the highly compartmentalized, mostly uncompartmentalized, and ice condenser containments relative to the debris transport assumptions is the fraction of the debris assumed to deposit directly onto the containment bottom floor as a result of blowdown debris transport. For fibrous debris transport, however, this fraction becomes irrelevant because all the debris transported upwards is conservatively assumed to wash back down to the sump pool where the washdown debris is treated in the same manner as the blowdown floor deposited debris. In summary, for small fine fibrous debris transport (all three containment categories), the overall transport fraction to the sump screens is one (1.0) minus the fraction assumed to enter the inactive pools (based on a water volume ratio). The 100% washdown assumption for fibrous (and other) debris is conservative.

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For small fine RMI debris transport, the fraction assumed ejected upwards (25%) is subsequently assumed to remain in the upper containment areas. In reality, some portion of the small fine RMI debris deposited in the upper reaches of the containment during blowdown would wash back down to the sump pool; therefore this baseline assumption is non-conservative in isolation. However, based on confirmatory debris transport research in Appendices IV and VI, this non-conservative transport assumption, in conjunction with the relatively high fractions of small fine blowdown assumed to be deposited on the bottom floor (0.75, 1.0, or 0.9), represents a conservative estimate of small fine RMI debris placed in the sump pool.

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The baseline assumption that the recirculation phase pool transport is 100% for small fines is conservative, and removes a need to address the effects of the variety of pool geometries and flow velocities associated with the differences among the PWR containments. However, the baseline assumption of zero sump pool transport of the large piece debris is non-conservative for the plants with relatively fast pool velocities that are capable of moving large debris. The implication of this assumption is that absolutely no large piece debris would accumulate on the sump screens. Based on experimental results from testing performed at the University of New Mexico, the volunteer plant pool model demonstrated that large pieces will degrade and fibers will come out of the large flocks and be transported to the screen (NUREG/CR-6773). As stated in Appendix IV, the characteristic transport velocities must be compared to typical debris transport velocities to determine whether or not the baseline method should be modified to include the transport of large debris. Characteristic transport velocities can be sufficiently estimated using recirculation flow rates and nominal sump dimensions to determine if a potential exists that substantial portions of the large debris will transport. If substantial transport of large debris is reasonably possible and if such transport can alter the outcome of the NPSH margin evaluation, then analytical refinements are needed that evaluate large debris transport.

A conservative assumption recommended in the baseline guidance is that all debris generated outside the ZOI will be of small fine debris that subsequently transports to the sump screens (i.e., 100% washdown transport, 100% sump pool recirculation transport, and no transport into the inactive pools). This assumption removes a need to address the variability and uncertainties due to lack of data on the generation and transport of debris outside the ZOI, especially when considering the differences among the PWR containments.

__Staff Conclusions Regarding Section 3.6.3:__ The staff concludes that two of the transport assumptions given in the baseline guidance are non-conservative. These assumptions are: (1) the assumption that the quantity of fine debris trapped in inactive pools, especially debris washed down from the upper levels of the containment, can be estimated simply by the ratio of the inactive pool volume to the total water volume, and (2) the large piece debris will not transport in the sump pool. In order to avoid predicting unrealistic results when using these assumptions the licensees should (1) limit the fraction of debris moving into inactive pools to a maximum of 15% of the source unless shown otherwise by analysis and (2) evaluate large debris transport if characteristic transport velocities show that substantial transport of large debris is possible.

The baseline assumption that all debris in the containment bottom floor is uniformly distributed throughout the entire volume of water in containment is also not conservative. This assumption was made in the baseline guidance as justification for the inactive pool volume ratio but otherwise does not directly affect the acceptance of the baseline guidance due to the 100% recirculation pool transport assumption. However, should a plant subsequently perform a pool transport refinement, then this assumption would not apply and at that point alternative approaches such as those detailed in Appendix III should be considered.

3.6.4 Calculate Transport Factors

A sample transport calculation is provided in Section 3.6.4 of the GR. For the sample calculation, it was assumed that the containment was highly compartmentalized with an inactive pool fraction of 30%, and that the ZOI insulation debris included NUKON™ and RMI debris. The unquantified logic chart shown in GR Figure 3-2 was applied to both the NUKON™ and RMI debris per the guidance outlined in GR Section 3.6.3. GR Figures 3-3 and 3-4 show quantified transport logic trees for NUKON™ and RMI debris.

Applying the chart to NUKON™ debris, the size distribution is 60% small fines and 40% large pieces that were assumed not to transport. Two transport pathways delivered small fines debris to the sump: (1) 75% of the debris was assumed directly deposited to the sump pool floor, and (2) the remaining 25% of the debris deposited in the upper containment but subsequently washed down to the sump pool after 30% of each case being sequestered in inactive pools. Therefore, 42% of the total NUKON™ debris was assumed to reach the sump with the remaining 58% assumed either trapped in the inactive pools (18%) or as large pieces (40%). Applying the chart to RMI debris, the size distribution is 75% small pieces and 25% large pieces that were assumed not to transport. Only one transport pathway delivered debris to the sump in which 75% of the debris assumed directly deposited to the sump pool floor. The 18.75% of the RMI assumed deposited in the upper containment was assumed to remain there and 30% of the small pieces assumed to reach the lower containment (56% was assumed trapped in the inactive pools. Therefore, 39% of the total (or 53 % of the small pieces) RMI debris was assumed to reach the sump. No large debris transport to the sump. The sample calculation acknowledges 100% transport of coatings debris, from both within and outside the ZOI; and all debris material outside the ZOI including latent debris. A list of all debris by type and size is provided and available for the subsequent sample head loss calculations.

Staff Evaluation for Section 3.6.4: The sample problem is consistent with the baseline methodology discussed above and the specified transport assumptions.

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3.7 HEAD LOSS

3.7.1 Introduction and Scope

Section 3.7.1 consists of an introduction to the head loss guidance.

3.7.2 Inputs for Head Loss Evaluation

3.7.2.1 Sump Screen Design

Section 3.7.2.1 briefly describes several aspects of sump screen design pertinent to estimating the head loss across the sump screen. The aspects described include screen construction, screen orientation, screen mesh size, applicable screen area, flat screen versus alternate geometries such as stacked-disc strainers (circumscribed area versus actual screen area), and clean strainer head loss estimation.

Staff Evaluation for Section 3.7.2.1: The general guidance in this section is acceptable because it is consistent with general engineering practice.

3.7.2.2 Thermal-Hydraulic Conditions

3.7.2.2.1 Recirculation Pool Water Level

Section 3.7.2.2.1 recommends using the minimum water level of the recirculation pool in estimating the head loss across the debris bed accumulated on the screen. The minimum water level will yield the smallest surface area for the water flow through the screens that are partially submerged, as well as the lowest available NPSH to the ECCS pumps.

Staff Evaluation for Section 3.7.2.2.1: The staff determined that the recommendation of using the minimum water level in the pool is appropriate. For partially submerged sump screens, the water level affects the wetted screen area, which affects the water approach velocity used in the calculation of the head loss due to the debris accumulation on the sump screen. A lower water level in the pool would result in a lower wetted screen area giving a higher approach velocity, which would conservatively give a higher head loss across the debris bed. For completely submerged screens, the static water level adds to the NPSH margin. The staff further notes that the determination of the minimum level should consider potential water hold up in the upper levels of the containment including water holdup due to potential debris blockage at water passages such as drains (e.g., refueling pool drains). The minimum level is not merely a conservative assumption but is consistent with ensuring adequate NPSH margin when the pool is actually operated at that level.

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3.7.2.2.2 ECCS Flow Rate

Section 3.7.2.2.2 recommends using the highest ECCS flow rate in calculating the head loss across a screen, i.e., the maximum pump flows as identified in current NPSH calculations. For multiple sump screens, the flow rate for the head loss calculation is the flow through each of the screens.

Staff Evaluation for Section 3.7.2.2.2: The staff concludes that the recommendation of using the maximum pump flows in the head loss calculations is the appropriate assumption although under certain conditions those pumps might be throttled back to a lesser flow rate. This maximum pump flow assumption removes the uncertainty that a lesser flow rate will be exceeded. The rate of flow through the screen along with the screen area is used to determine the velocity of flow through the screen, which is a primary input to the head loss calculation.

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3.7.2.2.3 Temperature

Section 3.7.2.2.3 makes three recommendations for specifying the water temperature to be used in the head loss calculations.

1. The temperature at which the head loss is evaluated should be consistent with the temperature used for the NPSH evaluation.

2. The head loss is to be evaluated at multiple times when different temperatures and flows exist during an accident.

3. The maximum expected temperature may be used for the NPSH analysis, whereas the lowest expected temperature during ECCS operation may be taken for the head loss analysis.

Staff Evaluation for Section 3.7.2.2.3: The water temperature determines the viscosity of the water, which affects head loss. A head loss correlation typically either includes the viscosity or is only valid for a distinct range of temperatures. A higher water temperature lowers the viscosity and therefore conservatively gives a higher frictional head loss across the debris bed on the sump screen. Therefore, Recommendation 3 is acceptable for specifying the water temperature. The licensees should calculate the NPSH margin according to their licensing bases (Regulatory Guide 1.82, Rev. 3).

The estimation of the minimum water temperature may require a different calculation than the typical plant estimation of the maximum water temperature for the design basis. It is conservative in the calculation of the maximum sump pool water temperature to neglect heat transfer processes or systems (e.g., a non-safety related heat removal systems) either to simplify the calculation or because a system cannot be relied upon to limit the temperature. But in a minimum water temperature calculation, all heat removal systems and processes should be included.

Recommendation 2 allows the time-dependency of the temperature to be evaluated, i.e., the evaluation of multiple times, temperatures, and flows during an accident. Staff concerns with the approach include:

1. Recommendation 2 appears to also recommend that the pump flow can vary with time as well, which is in direct conflict with Section 3.7.2.2.2, which states that the maximum pump flow should be used.

2. The debris in the time-dependent calculation must be assumed as the worst case debris accumulation because the debris transport evaluation capability is not sufficient to predict time-dependent accumulation.

3. If one calculation is used to estimate the pool temperature, it should be sufficiently realistic to capture all important heat transport processes. The systems specified in the accident scenario and the specification of the accident scenario must address whether or not systems such as non-safety related heat removal systems are operating.

Recommendation 1 is unacceptable because it does not in any way specify a minimum temperature for the head loss calculation. The licensees should calculate the NPSH margin according to their licensing bases (Regulatory Guide 1.82, Rev. 3).

Staff Conclusions Regarding Section 3.7.2.2.3: The staff concludes that Recommendation 3 for determining the pool temperatures is conservative and adequate if the minimum and maximum temperatures are properly estimated. Recommendation 2 is also a valid approach if properly evaluated with the provisions that the flow should remain that of the maximum pump flow, the debris bed should be the worst case debris accumulation throughout the time-dependent temperature transient, and the pool temperature is properly determined. Recommendation 1 is incomplete and unacceptable by itself.

3.7.2.2.4 Debris Types, Quantities, and Characteristics

Section 3.7.2.2.4 provides a general discussion regarding the parameters needed to specify an accumulation of debris on the sump screen.

<u>Staff Evaluation for Section 3.7.2.2.4:</u> The staff notes that the list of important head loss parameters is incomplete. In addition to quantities specified as volumes or masses, the bulk and fiber densities are needed for fibrous debris; the particle density and limiting porosity are needed for the particulate; and the specific surface areas are needed for each debris bed component. Appendix V gives guidance on determining the specific surface areas.

3.7.2.3 Head Loss Methodology

3.7.2.3.1 General Theoretical/Empirical Formulas

3.7.2.3.1.1 Fibrous Debris Beds with Particulate

Section 3.7.2.3.1.1 describes the NUREG/CR-6224 head loss correlation by providing the basic correlation equation and the supporting constituent equations for solidity (one minus the porosity). This section also discusses fibrous debris bed compression due to the pressure gradient across the sump screen as well as compression limiting factors.

The baseline guidance offers the following options for dealing with debris materials or combinations of materials for which the empirical head loss data do not exist:

1. Characterizing the material with scanning electron microscopy (SEM) analysis and the establishment of a size distribution.
2. Choosing an alternate material that conservatively represents the material in question, via similitude arguments.
3. Testing head loss of the particular material to establish a correlation or else validate an existing correlation for that material.
4. Using other data that may exist to establish head loss data for the material in question.

The section contains a discussion for estimating the specific surface area, S_v, from the constituent characteristic dimension (e.g., particle or fiber diameter). A formula is provided for determining S_v for a mixture of debris constituents that is based on volume averaging the squares of the constituent S_v. The baseline guidance states: "it is best to err on the low side for conservative values of S_v." In addition, the guidance describes obtaining the aggregate density for both particulate and fibrous debris using a simple volume averaging procedure. Finally, a computational procedure is described for solving the correlation equations to obtain the head loss.

<u>Staff Evaluation for Section 3.7.2.3.1.1:</u> The GR options for obtaining head loss parameters for materials that have not been previously characterized are all valid methods of learning more about that material. Performing head loss testing (Option 3) that can be subsequently analyzed to determine appropriate head loss parameters is the best option since it provides results with the least uncertainty. The other three options will improve knowledge but can leave substantial uncertainty in the resultant head loss parameters that must be countered through the use of conservative safety factors.

Confirmatory research presented in Appendix V and head loss testing reports LA-UR-04-1227 and LA-UR-04-3970 illustrate the application of the NUREG/CR-6224 correlation to head loss data to determine applicable input parameters for the correlation.

The baseline adequately presents the concept of compression limiting whereby the compaction of the fiber and particulate effectively prevents further compression of the debris bed, i.e., limiting of the solidity of the debris bed. However, the computational procedure described in the GR for solving the NUREG/CR-6224 correlation equations to obtain the head loss does not include steps for the determination of whether or not the limiting solidity would occur and how to proceed with the calculation should the limiting solidity condition occur within the iterative solution. The reader is left with the impression that the limiting solidity is approximately 0.2 (i.e., limiting porosity of 0.8), which is correct for BWR iron oxide corrosion products. This impression is reinforced in the sample problem (Page 3-71) where the mixed bed solidity is set to 0.20 for a particulate that consists of latent and coating debris. Common sand, a likely component of latent debris has an approximate solidity of 0.60 (data available in common soil handbooks), which is greater than the GR implied limit of 0.2. The surrogate latent debris head loss testing documented in LA-UR-04-3970 tested common sand and verified the handbook values for sand solidity. When applying the NUREG/CR-6224 correlation, the correct value for the limiting solidity should be used for the postulated particulate because the limiting solidity governs the head loss prediction whenever the correlation predicts compression limiting has occurred, as is the case with thin-bed debris accumulations.

The determination of the specific surface area for the debris bed is an important aspect for predicting the head loss. The head loss from the NUREG/CR-6224 correlation is directly dependent on S_v; in fact, the leading laminar term uses the Sv^2. For example, at lower flow velocities, if the S_v were under-predicted by a factor of 2, then the head loss could be under predicted by a factor of 4. The baseline guidance statement "it is best to err on the low side for conservative valves of S_v," should be clarified to indicate that it is the debris size that should be selected on the low side, not the value S_v. It is conservative to estimate S_v high, not low.

The baseline guidance for estimating S_v from the constituent characteristic size dimension (e.g., fiber or particle diameter) has been demonstrated to be unreliable particularly when a particulate is defined by a size distribution. The use of six divided by the diameter is reasonable when specifying S_v for the conservative all-one-size particulate (10 micron) postulated for coatings debris. However, it is unreasonable when a particulate distribution covers a wide range of sizes (e.g., iron oxide corrosion products ranges from 1 to 300 microns) typically described by 3 or 4 subgroups. The value of S_v calculated is sensitive to the value of the diameter which is used to represent the size group in the 6/diameter formula. The natural tendency is to select the mean of the size group but the mean significantly under estimates the specific surface area because all particles in the group less than the mean make a substantially greater contribution to S_v than do the particles larger than the mean value. Selecting an appropriate value within the range is problematic because it depends upon the size distribution within the size group. A conservative solution to this problem is to use the minimum size of each size group. However, this approach can lead to large estimates of S_v, especially when the particles become very small. For example, assume the size group has a uniform distribution ranging from 5 to 100 microns. Using the 5 micron size results in a S_v of 366,000/ft which is conservative (but too large), whereas using the mean of 52.5 micron results in a S_v of only 34,800/ft which is much too small. Smaller particles in a debris bed cause greater head loss than do the larger particles. Confirmatory research presented in Appendix V show significant error in S_v calculated using simple geometric equations (e.g., 4/d for fibers and 6/d for particles) compared to the one deduced using head loss data. Where the particulate for a specific material is defined by a size distribution, the licensees should use applicable head loss data to determine S_v.

The formula provided in the baseline for determining S_v for a mixture of debris constituents that is based on volume averaging the squares of S_v is adequate and conservative relative to the formula actually provided in the cited reference, NUREG/CR-6371.

Before using the NUREG/CR-6224 correlation that is recommended in the GR or any other head loss correlation, the licensees should ensure that it is applicable for the type of insulations and the range of parameters. Appendix V of this report gives the procedures for applying the correlation and the ranges of parameters used to validate it and are publicly available. If the correlation has been validated for the type of insulations and the range of parameters, the licensees may use it without further validation. If the correlation has not been validated for the type of insulations and the range or parameters, the licensees should validate it using head loss data from tests performed for the particular type of insulations.

Staff Conclusions Regarding Section 3.7.2.3.1.1: The staff agrees with the baseline that the NUREG/CR-6224 correlation is an appropriate method for estimating the head loss associated with a debris bed consisting of fibers and particulates. The licensees should ensure the validity of this correlation for their applications of type of insulations and the range of parameters using the guidance provided in Appendix V of this report.

3.7.2.3.1.2 RMI Debris Beds

Section 3.7.2.3.1.2 provides a head loss correlation for estimating the head loss across a bed of RMI debris. This correlation and the values for the constant known as the interfoil gap thickness were extracted directly from NUREG/CR-6808.

Staff Evaluation for Section 3.7.2.3.1.2: The staff agrees with the baseline that the NUREG/CR-6808 is an appropriate method for estimating the head loss associated with a debris bed consisting of RMI as documented in NUREG/CR-6808.

3.7.2.3.1.3 Mixed Debris Beds (RMI, Fiber, and Particulates)

Section 3.7.2.3.1.3 provides guidance for mixed debris beds that include RMI, fibrous, and particulate debris. The baseline guidance recommends that the head loss for the fibrous/particulate debris and the RMI debris be estimated separately and then added together to obtain the head loss for the mixed debris bed (i.e., superposition of individual head losses).

Staff Evaluation for Section 3.7.2.3.1.3: NRC sponsored research found the test data for head loss for mixed debris beds to be bounded by the sum of the head loss of the individual constituents. However, it was noted that the mixed bed tests were not comprehensive in regards to all of the types and combinations of debris that may be possible. NUREG/CR-6808 concluded that the head loss associated with a mixed RMI and fiber debris bed should preferably be based on head loss measurements but can alternately be calculated as an algebraic sum of the fiber and RMI components after accurately accounting for the strainer geometry. The potential for forming a fiber/particulate thin-bed should be evaluated even when mixed debris beds are possible because there is insufficient data to substantiate the conclusion that the presence of RMI debris can prevent the formation of a thin bed.

3.7.2.3.1.4 Calcium Silicate Insulation

GR Section 3.7.2.3.1.4 discusses the calculation of head loss for debris beds containing calcium silicate insulation debris. It states: "Based on current information, the NUREG/CR-6224 correlation can be used according to the methods for fibrous debris beds with particulate if the application is limited to particulate mixtures containing up to about 20 percent calcium silicate by mass." The calcium silicate is treated as the particulate in the fiber/particulate debris bed. The guidance referenced the NRC sponsored calcium silicate test report (issuance pending), which is now available as LA-UR-04-1227.

Staff Evaluation for Section 3.7.2.3.1.4: The staff concludes that the baseline guidance regarding the estimation of head loss for debris beds containing calcium silicate debris is not adequate. The staff recognizes that LA-UR-04-1227 was not available in time for it to be reviewed by industry and its results included in the baseline guidance. Therefore, the recommendations from LA-UR-04-1227 are summarized herein.

The staff recommended parameters for applying the NUREG/CR-6224 correlation to debris beds consisting of fibrous and calcium silicate debris are shown in Table 3-5. Note that the recommendations depend upon whether or not the thin-bed debris configuration is a potential concern. If the potential for a thin-bed debris configuration exists, then the application of the correlation must consider the higher specific surface area deduced from the tests where the high thin-bed head losses were encountered.

The reproducible thin-bed CalSil tests demonstrated that the potential thin-bed accumulation is realistic. Only a small quantity of fibers (or perhaps none) and fine CalSil particulate, which tends to remain in suspension, is needed to form a uniform debris bed. The recommended specific surface area of 880,000 ft^{-1} is 10% higher than the experimentally deduced area, to prudently incorporate a 10% to 20% safety factor to account for (1) experimental uncertainties, such as instrumentation error; (2) an incomplete examination of the experimental test parameter space; and (3) the variance in the manufacture of calcium silicate insulation.

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Table 3-5. Recommended Conservative Calcium Silicate NUREG/CR-6224 Correlation Parameters

Correlation Parameter	Recommended Head Loss Parameters

	[illegible]	[illegible]
Particle Density	115 lbm/ft^3	115 lbm/ft^3
Particulate Sludge Density	22 lbm/ft^3	22 lbm/ft^3
Particulate Specific Surface Area	880,000 ft^{-1}	600,000 ft^{-1}

The sump screen conditions that cannot form a thin-bed configuration include the following: (1) the advanced strainer designs, where test data has indicated that thin-bed configurations would not uniformly form because of complex surface design; and (2) flow conditions insufficient for the required debris bed formation, which can be substantiated by applicable test data. Examples of advanced strainer designs include the stacked-disk strainers, where it has been generally accepted, based on testing of prototypical strainers, that a uniform thin-bed configuration will not form under potential debris loadings. An example of insufficient flow conditions include a maximum screen/strainer approach velocity of less than 0.1 ft/s and particulate-to-fiber mass ratios of less than 0.5—conditions for which a thin bed was not achieved in the calcium silicate head-loss tests because the filtration efficiency apparently was not sufficient to remove enough of the fine calcium silicate from the flow to form a granular debris bed. Beyond these conditions, a thin bed was actually formed during the tests or the tests did not cover that part of the parameter space; thus, it is not known if a thin bed can form.

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The specific surface area for calcium silicate is not a fixed value as it is for hardened particulates such as BWR corrosion products. It was demonstrated that calcium silicate particles are somewhat "spongy" with interior voids so that when compressed, the particulate deforms to fill interparticle spaces. A working theory that fits the experimental results is that the compression forces water through smaller and smaller interior voids and increases the effective specific surface area of the calcium silicate particles.

The three parameters recommended in Table 3-5 (i.e., particle density, particulate sludge density, and particulate specific surface area) are a parameter set and should be applied as a set. The experimental determination of the specific surface areas depended upon the specification of the debris densities. It is also important to note that the calcium silicate tested was obtained from only one manufacturer, and that these recommendations do not necessarily apply to all types of calcium silicate insulation debris.

Whether or not there is sufficient fiber to form a thin-bed has been generally based on the NUREG/CR-6224 recommendation that the quantity of fibrous debris available must be sufficient to form an accumulation 1/8-in thick on the screen. Tests conducted using only calcium silicate fragments have demonstrated that calcium silicate debris can accumulate without the aid of fibrous debris. However, tests conducted using only calcium silicate were not definitive enough to accurately determine the conditions under which a thin-bed can form without the presence of fibrous debris other than the fibers contained in the calcium silicate insulation.

Staff Conclusions Regarding Section 3.7.2.3.1.4: The staff concludes that the recommendations shown in Table -3-5 of this report should be followed for debris beds containing calcium silicate debris unless other data becomes available that is more

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applicable to plant specific conditions. If it can be demonstrated that a thin-bed configuration cannot be formed with calcium silicate debris, then the mixed bed configuration recommendations can be followed. Otherwise, the thin-bed configuration should be assumed. In determining whether or not enough fibrous debris is available, the determination that it may be possible to form a bed of calcium silicate debris without other supporting fiber should be factored into the analysis.

3.7.2.3.1.5 Microporous Insulation

Section 3.7.2.3.1.5 acknowledges that microporous insulation (e.g., MinK and Microtherm) is a granular insulation that is in use in PWRs. For guidance, the GR refers to insights gained in a limited series of head loss experiments for which additional background is provided in the supplemental guidance (GR Section 4.2.5.2.2).

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Staff Evaluation for Section 3.7.2.3.1.5: The staff finds that GR did not provide adequate guidance to predict head loss for microporous-insulation debris beds because it did not recommend any methodology. The licensees should develop correlations or use test data for predicting head loss for microporous-insulation debris beds.

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3.7.2.3.1.6 Microporous and Fiber Debris

Section 3.7.2.3.1.6 provides limited guidance regarding the application of the NUREG/CR-6224 correlation to light loadings of microporous insulation debris on a sump screen for a particulate to fiber mass ratio less than 0.2.

For ratios larger than 0.2, the baseline guidance recommends the following options:

1. Removal of microporous or calcium silicate insulation until the particulate to fiber mass ratios drops below 0.2.
2. Seek an alternative head loss correlation to the NUREG/CR-6224 correlation.
3. Perform head loss experiments using plant-specific debris mixtures, sump screen configuration, and thermal-hydraulic conditions.

The baseline guidance in this section also discusses concerns for microporous or calcium silicate debris only (i.e., no additional fibers other than those integral to the microporous or calcium silicate debris). This guidance recommends the same three alternatives noted above for situations where a debris bed can be accumulated with these insulations without significant other fiber.

The baseline guidance addresses mixtures of granular insulation and RMI debris beds by referring to the superposition guidance presented in Section 3.7.2.3.1.3.

Staff Evaluation for Section 3.7.2.3.1.6: The staff concludes the following regarding the guidance presented in this section.

1. The baseline guidance is adequate for particulate-to-fiber mass ratios less than 0.2.

2. The alternatives for particulate-to-fiber mass ratios greater than 0.2 are adequate with the caveat relative to alternative 2 that the adequacy of the alternate correlation should be verified using applicable test data.

3. Since a debris bed formed of microporous debris without additional fibrous debris would be similar to a fibrous/microporous debris bed with a high particulate-to-fiber mass ratio, the adequacy of the alternatives is the same as for a debris bed with fibers and a particulate-to-fiber mass ratio greater than 0.2.

4. The acceptance of the baseline guidance for thin-beds containing microporous insulation types is also subject to the acceptance of the three alternatives.

5. The superposition guidance for mixtures of granular insulation and RMI debris is acceptable.

3.7.2.3.2 Methodology Application Considerations

3.7.2.3.2.1 Total Sump Screen Head Loss

Section 3.7.2.3.2.1 recommends adding the clean strainer head loss to the debris bed head loss to get the total head loss across the screen.

Staff Evaluation for Section 3.7.2.3.2.1: The staff concludes this guidance is acceptable because it is consistent with general engineering practice. Regulatory Guide 1.82, Rev. 3, recommends a different approach, which is based on NPSH margin. Either approach is acceptable.

3.7.2.3.2.2 Evaluation of Breaks with Different Combinations of Debris

Section 3.7.2.3.2.2 recommends that analysts evaluate a spectrum of breaks with different combinations of debris types to ensure the identification of the break with the mixture of debris on the screen that causes the highest head loss. The guidance notes that the limiting break is not necessarily the break that generates the largest total quantity of debris.

Staff Evaluation for Section 3.7.2.3.2.2: The staff concludes this guidance is acceptable because the break size recommended in the GR gives conservatively higher head loss across the debris bed on the sump screen..

3.7.2.3.2.3 Thin Fibrous Beds

GR Section 3.7.2.3.2.3 recommends that the head loss associated with a thin-bed be calculated as a sensitivity analysis. To analyze a thin fiber bed, a fiber quantity sufficient to form a one-eighth-inch thick debris bed should be determined to be available and, if present, could be deposited on the sump screen. The head loss calculations are the same as described for fiber and particulate beds using the full value of particulate matter transported to the sump screen. The particulate matter includes the latent debris such as dirt, concrete dust, rust, inorganic zinc, epoxy fines, etc. The particulate layer is characterized by a high sludge-to-fiber ratio; hence a limiting value for the compression

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is used. If under these conditions, the thin-bed head loss should exceed the NPSH margin, then the allowable particulate loading can be evaluated by reducing the particulate quantity until the calculated head loss is within the NPSH margin.

Staff Evaluation for Section 3.7.2.3.2.3: The staff agrees that the potential for developing a thin-bed head loss must be evaluated regardless of the composition of the potential containment debris. However, the staff gives the following supplement to the thin-bed guidance provided in the GR to ensure conservatism.

1. The appropriate density to apply to the fibrous debris in the determination of the quantity of debris needed to form a one-eighth-inch bed is the as-manufactured density. The one-eighth-inch minimum thickness has been based on the NUREG/CR-6224 (Appendix B, Page B-60) finding: *"The head loss model is applicable only to fiber bed thicknesses where uniform bed formation is expected. Typically, this is valid for fiber bed thicknesses larger than 0.125" (0.318 cm). Below this value, it appears the bed does not have the required structure to bridge the strainer holes and filter the sludge particles."* The NUREG/CR-6224 analysis used the as-manufactured density to specify the 'theoretical bed thickness', which is used to specify whether or not a one-eighth-inch thick bed exists. For NUKON™ debris, the accepted as-manufactured density has been 2.4 lb/ft^3. For latent debris, the as-manufactured density is not applicable because latent fibers can come from any number of sources. However, after examining the latent fibers collected from volunteer plants LA-UR-04-3970 conservatively recommended a density of 2.4 lb/ft^3 which is equal to that of NUKON™.

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2. For a thin-bed debris accumulation, the limiting bed compression specified as either the limiting porosity or limiting solidity becomes a controlling parameter in the NUREG/CR-6224 correlation, i.e., the bed solidity essentially approaches that of the granular materials. It is important that the limiting solidity is correctly evaluated for the particular particulate or mixture of particulates in the debris bed. For example, the limiting solidity for BWR iron oxide corrosion products is about 0.2 (NUREG/CR-6224) but for common sand, it varies between 0.57 to 0.60 (standard handbook data). This issue was discussed in Section 3.7.2.3.1.1.

3. Because a number of uncertainties are associated with specifying the one-eighth-inch bed thickness criteria, the parameter values that go into the bed thickness determination need to be sufficiently conservative to compensate for uncertainties to ensure adequate NPSH margin. One consideration is the fineness of the fibrous debris accumulating on the screen. Tests have been conducted since the NUREG/CR-6224 study was completed where thin-beds have been formed that were somewhat thinner than one-eighth-inch (e.g., one-tenth-inch), principally because the bed was formed from suspended individual fibers rather than the shredded fiber debris used in the NUREG/CR-6224 testing. Another consideration is the fact that the one-eighth-inch criteria was based on NUKON™ debris and has not been actually determined for other type of fibrous debris. Another consideration is the indication that calcium silicate can form a debris bed without supporting fibers (other than the fibers integrated into the calcium silicate).

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4. In determining the mass of allowable particulate on the sump screen that is needed to overcome the NPSH margin, the uncertainties associated with predicting this value should be noted. Specifically, the determination of the limiting solidity has a significant uncertainty due to inaccurate specifications of the densities of the particulate components or perhaps the mixing of constituents, and due to the involvement of fibers interlaced with the particulate.
5. To compensate for these noted uncertainties, sufficient conservatism should be used in estimating the quantities of fibrous debris available to form a thin bed. This point is particularly important for plants that do not have significant fibrous insulation (e.g., an all RMI plant) so that the main contribution to the fiber quantities on the sump screen comes from latent debris. In such cases the estimate of the latent fiber becomes a determining factor but substantial uncertainty is also associated with that estimate.

3.7.2.3.2.4 Sump Screen Submergence

Section 3.7.2.3.2.4 described the applicable characterization for partially versus completely submerged sump screens. The limiting criterion for submerged screens occurs when the combined clean sump and debris bed head loss exceeds the NPSH margin. The limiting criterion for a partially submerged screen is when the debris bed accumulation on the screen reduces the flow to less than the flow requirements for the sump. An effective head loss across the debris, which is approximately equal to one-half of the pool height, is sufficient to prevent adequate water flow. The head loss estimate is applied to the submerged portion of the sump screen area.

Staff Evaluation for Section 3.7.2.3.2.4: The staff concludes that the baseline guidance in this section regarding partially and completely submerged sump screens is acceptable because it is consistent with Regulatory Guide 1.82, Rev. 3.

3.7.2.3.2.5 Buoyant Debris

Section 3.7.2.3.2.5 addresses the conditions where buoyant debris could become a problem for strainer head loss. For fully submerged screens, buoyant debris is not considered a problem since it would not reach the sump screens. For partially submerged screens where buoyant debris is determined to reach the screen, the baseline guidance recommends that the effective area be reduced by the thickness of the buoyant debris layer times the length of the covered perimeter, to the extent that it fully envelopes the screen.

Staff Evaluation for Section 3.7.2.3.2.5: The staff agrees with the necessity of considering the potential for buoyant debris affecting sump screen head loss. The baseline guidance is acceptable with the exception that shallowly fully submerged sump screens could still draw buoyant debris down to the submerged screen. An analysis should be performed to determine the submerged depth needed to ensure buoyant debris cannot be drawn down onto the sump screen.

3.7.2.3.3 Methodology Limitations and Other Considerations

3.7.2.3.3.1 Flat Screen Assumption

Section 3.7.2.3.3.1 makes the point that head loss data obtained using a vertical pipe test section of a closed loop test apparatus with a horizontally mounted flat screen yielded conservative data for the development of the NUREG/CR-6224 correlation because all debris was forced onto a very small screen. Further, it states that in the alternative design screens, the direct application of the NUREG/CR-6224 correlation may yield overly conservative results and that for these alternate geometry screens, independent head loss correlations should be developed based on actual design configurations, debris loads, and test data to reduce conservatism.

Staff Evaluation for Section 3.7.2.3.3.1: The staff finds that the guidance in this section need the following clarification. The development and application of the NUREG/CR-6224 correlation is based on uniform and homogeneous debris beds. Applicable test data must therefore be measured on test debris beds that match these correlation assumptions. The vertical pipe closed-loop test apparatus generally meets these conditions provided the debris is introduced in such a manner that it settled uniformly on the test screen. The baseline statement that "all debris was forced onto a very small screen" does not reflect testing realities. The debris is allowed to settle uniformly but the important point is that the correlation is based on the bed thickness and composition as tested.

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A uniform debris bed is a realistic and a likely form of debris accumulation when debris accumulation is accomplished by filtering out suspended fibers. For example, during the conduct of the integrated tank tests [NUREG/CR-6773], the typical accumulation of fibrous debris was due primarily to suspended debris transport and resulted in uniform debris buildup on both horizontally and vertically oriented screens. Also consider the operational incidents at Perry [NUREG/CR-6808] where a coating of fine dirt covered most of the surface of the strainers and at Limerick where a thin mat of material covered the strainer. The flat screen assumption is reality based and is not merely a conservative assumption, nor is it overly conservative.

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While it is adequate to develop independent head loss correlations based on actual design configurations, debris loads, and test data for alternative screen designs, it should also be noted that the NUREG/CR-6224 correlation has been successfully applied to these designs without over conservatism. The application of the NUREG/CR-6224 correlation involves the selection of the appropriate screen area versus debris loading (i.e. total screen area, circumscribed area, or some area in between based on test data) but then so will any other successful correlation that models an alternate design from a clean screen to its fully loaded condition. The NUREG/CR-6224 correlation has been and can be applied to prototype alternate geometry screens/strainers to determine effective screen areas for specific debris loadings that can be subsequently used in plant specific evaluations.

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3.7.2.3.3.2 Non-Uniform Deposition on Sump Screen Surfaces

Section 3.7.2.3.3.2 discusses the conservatism of the assumption that the debris is uniformly distributed on the screen relative to potential non-conservative accumulation associated with vertical and inclined screens.

Staff Evaluation for Section 3.7.2.3.3.2: The staff agrees that it is conservative to assume uniform debris accumulation on all types and orientations of screens.

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3.7.2.3.3.3 Very Thin Fiber Beds

Section 3.7.2.3.3.2 discusses aspects where the fiber loading is less than that required to form a thin bed. It states that experiments have shown that very thin fibrous beds (with a thickness of less than one-eighth-inch) are characterized by large scale non-uniformities on the screen and negligible head losses. The baseline guidance recommends assuming a negligible head loss whenever the debris bed thickness is less one-eighth-inch.

<u>Staff Evaluation for Section 3.7.2.3.3.3:</u> The staff concludes that it is adequate to neglect the head loss associated with low density fiberglass insulation debris beds of less than one-eighth-inch provided the concerns expressed in the staff's response to Section 3.7.2.3.2.3 regarding the determination of the thin bed thickness are adequately addressed. These concerns included using the appropriate density to determine the thickness for a given quantity of debris and the uncertainties associated with the original specification of one-eighth-inch as the threshold thickness. The uncertainties include the relative fineness of the insulation debris used to make the threshold thickness determination and the fact that the thickness determination was made only for NUKON™ debris and has not been directly determined for other types of insulation debris. An example where it is not appropriate to neglect the head loss for a debris bed less than one-eighth-inch thick is when there is substantial calcium silicate debris in the bed because there has been experimental indications that calcium silicate can form a debris bed without supporting fibers.

3.7.2.3.4 Sample Calculations

Sample head loss calculations are provided in Section 3.7.2.3.4 of the GR. In the sample calculations, flat-plate strainer geometry, steady-state ECC flow conditions, and the final debris loadings are assumed. The debris sources were developed in the sample problem sections for debris generation (GR Section 3.4.3), latent debris (GR Section 3-5-3), and debris transport (GR Section 3.4). Sample head loss calculations were presented for a fiber/particulate debris bed, an RMI debris bed, a mixed RMI, fiber/particulate debris bed, and a thin-bed debris condition.

<u>Staff Evaluation for Section 3.7.2.3.4:</u> The sample problems are consistent with the baseline methodology discussed above and with the specified head loss calculational assumptions, with the exception that the sample problem used a fiber density of 175 lbs/ft^3 rather than the 159 lbs/ft^3 recommended in GR Table 3-2. However, the sample problems fail to clarify the differing volumes and densities associated with each constituent. For example in the fiber/particulate calculation, two volumes are provided for NUKON™ fibers without distinguishing the type of volume quoted: (1) 129 ft^3 for the bulk volume, and (2) 1.77 ft^3 for the material (solid) volume. The reader must take care to ensure the proper volumes and densities are used in the appropriate calculational steps.

In Section 3.7.2.3.1.1, the GR discusses maximum solidity for particulates as a material dependent property but then also leaves the reader with the impression that 20% is a reasonable limiting value for general use. The staff comments to this section pointed out that many particulates have maximum solidities much higher than 20%, e.g., common sand has an approximate solidity of 60%. Therefore, the general use of 20% is not appropriate. Rather, the maximum solidity should be determined for each particulate constituent and then the particulate constituent effective average must be determined. It should also be noted that the maximum solidity also depends upon the particulate size distribution. The sample head loss calculations, specifically the thin-bed calculation where the limit is applied, failed to treat material-specific maximum solidities. The failure to correctly treat the maximum solidities can lead to erroneous and non-conservative head loss predictions for pack limited debris beds.

3.8 ACCEPTANCE OF NEI BASELINE GUIDANCE

The purpose of the baseline evaluation methodology is to provide U.S. PWR licensees with a common and consistent approach for evaluating the susceptibility of containment sumps to blockage resulting from the effects of postulated LOCA events. The baseline evaluation methodology is the application of a conservative set of methods that help identify the dominant design factors for a given plant (GR Section 3) that could be subsequently followed by separate guidance on possible analytical refinements to the baseline approach (GR Section 4) and potential design/operational refinements (GR Section 5).

The baseline, however, goes beyond the scoping intent with the statement that "If a plant uses this method and guidance to determine that sufficient head loss margin exists for proper long-term Emergency Core Cooling (ECC) and Containment Spray (CS) function, no additional evaluation for head loss is required." Rather, the baseline methodology becomes an acceptance methodology for plant specific evaluations. Therefore, the NRC staff acceptance of the baseline evaluation methodology is based on whether or not any and all PWRs that determine an adequate head loss margin by applying the baseline evaluation methodology will actually have adequate sump performance capabilities to support long-term cooling functions.

The NRC staff acceptance depends upon providing adequate assurance that the baseline assumptions taken as whole and applied generically to any PWR will not result in a plant operating without adequate ECC or CS head loss margin. In addition, the staff's acceptance considers how follow up analytical refinements will affect the baseline methodology retained in the final evaluation. Specifically, the acceptance of the baseline evaluation methodology as a package must balance conservative assumptions against non-conservative assumptions; therefore an analytical refinement that decreases the degree of conservatism on a particular assumption has the potential to alter the package balance such that the degree of conservatism is reduced, or even reversed to nonconservatism.

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The primary difficulties with assessing whether the assumptions used in the baseline guidance result in the baseline guidance as a package being conservative with respect to estimating NPSH margin is that each assumption is variable with respect to the plant evaluated and the conservatism for each assumption cannot be quantified without actually performing a detailed evaluation. Without quantification for at least the more influential assumptions, it is difficult to judge the baseline package conservatism. For example, assuming that all unqualified coatings fail into 10 micron particles could be overly conservative for containments with large quantities of unqualified coatings. However, for plants with little unqualified coatings this assumption does not provide any extra conservatism to counter the non-conservative assumptions in the baseline guidance. The more influential assumptions with potential notable conservatism are

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summarized in Table 3-6 and the more influential assumptions that are clearly not conservative are summarized in Table 3-7.

Table 3-6. Conservative Assumptions in the Baseline Evaluation Methodology

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Debris Generation Assumptions			
1	All unqualified coatings in containment are assumed fail.	Compensate for lack of data, i.e., no basis for estimating failure of unqualified coatings.	Variable depending upon plant conditions, therefore the associated conservatism to the baseline package could range from essentially none to excessive.
2	All coatings debris (qualified and unqualified) assumed to become 10 micron particulate. The implication of the small particulate size is complete transport to sump screen and complete filtration.	Compensate for lack of data, i.e., no basis for estimating coatings debris size distributions.	Variable depending upon plant conditions, therefore the associated conservatism to the baseline package could range from minimal to excessive.
3	100% destruction of materials for which suitable debris generation data is not available including all such materials inside the ZOI and unprotected materials outside the ZOI.	Compensate for lack of data, i.e., the fraction of the materials that becomes small fine debris cannot be ascertained without material-specific debris generation data.	Variable depending upon the types and quantities of such materials. Additionally, it depends upon the relative quantities of such materials compared to dominant insulation with known destruction characteristics. The associated conservatism to the baseline package could range from a minor correction to substantial.
Debris Transport Assumptions			
4	Washdown transport to the sump pool is 100% for fibrous debris and a large fraction of the blowdown transported debris is directed to the sump with the end result that all small fibrous debris fines transport to the sump pool.	Avoidance of complex analyses.	Variable depending upon containment design. Some containment designs could result in high washdown transport, e.g., the volunteer plant study (Appendix VI), while other may retain debris in the upper levels of the containment.

No.	Baseline Guidance Assumption	Rationale for Assumption	Perceived Level of Conservatism
5	100% of small fines ZOI debris, not allocated to an inactive pool is transported to the sump screens.	Avoidance of complex analyses.	Variable depending upon the transport characteristics of the pool. Given a fast flowing pool, the transport could be high therefore this assumption would not necessarily be conservative. But for a slow pool, a substantial portion of the small fines debris could sink to the floor and not transport to the screen, i.e., substantial conservatism with this assumption.
6	All debris generated outside ZOI assumed to transport to sump screen.	Avoidance of complex analyses and compensation for lack of data.	Variable depending upon the types and quantities of such materials. The associated conservatism to the baseline package could range from a minor correction to substantial.

Table 3-7. Non-Conservative Assumptions in the Baseline Evaluation Methodology

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		Debris Generation Assumptions	
7	The adaptation of the BWROG URG destruction pressures to PWR LOCA jets.	Lack of BWR or PWR specific data. Similar application suggests the BWR data appropriate to PWRs.	Because a LWR LOCA jet is two-phase steam/water jet and the destruction pressures cited in the URG were determined using an air jet and due to limited experimental evidence from the OPG two-phase jets, the BWROG destruction pressures could be too high. The baseline methodology could underestimate debris quantities. Therefore, based on the study of this issue and testing, the staff position is to lower the debris destruction pressure by 40% in order to account for two-phase jet effects (see Section 3.4.2.2).
8	A spherical ZOI is truncated whenever the ZOI intersects a robust structure. The radius of the remaining ZOI is not increased to compensate for jet reflection effects.	Assumption that jet reflections off the robust structure would not extend further than the unrestrained sphere. This approach was used for resolving the BWR strainer issue.	Jet reflections off the robust structures would reinforce other components of the LOCA jet. A major portion of the energy of the jet may be preserved.

No.	Baseline Guidance Assumption	Rationale for Assumption	Perceived Level of Non-Conservatism
9	The destruction pressures for coatings within the ZOI were based on high pressure water jet data rather than two-phase jets typical of a PWR LOCA.	Lack of applicable data.	The water jet data may not properly address thermal shock effects that spalled concrete in the HDR tests (NUREG-0897, Page C-2 and Figure C-5). The ZOI coatings debris quantities may be underestimated. Therefore, the staff position is that either destruction pressures and spherical ZOI sizing for coatings be determined on a plant-specific basis, based on experimental data as described in Sections 3.4.2 and 3.4.3, or a spherical ZOI of 10D be used.
10	Default worse case paint thickness of 3 mil thickness for unqualified coatings outside ZOI	Default alternative when plant specific coating thickness data is not available.	Not worse case and the assumption was not been properly justified. Therefore, the staff recommends plant-specific justification of this thickness, or plant-specific evaluations to determine unqualified coating properties and thicknesses as described in Section 3.4.3.
Debris Transport Assumptions			
11	Debris transport into inactive pools based on the ratio of the inactive pool water volume to the total water volume in the sump pool. Implies a uniform distribution of debris throughout the water pools formed following the LOCA.	Assumptions of uniformly-distributed (as opposed to preferential) sweeping of debris on the containment floor into inactive pools by thin sheets of high-velocity water, and of 100% transport of small fines to the sump during recirculation.	Baseline assumption that debris entrapment in inactive pools (e.g., reactor cavity) based on ratio of water volumes is not realistic. Debris will not be uniformly distributed in the sump water and washdown transported debris likely to arrive in sump after inactive pools filled.

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No.	Baseline Guidance/Approach	Rationale/Assumption	[illegible]
			Potentially large non-conservatism that depends upon inactive pool volume relative to total water volume. In addition, the same sheeting flow mechanism credited by the GR has the nonconservative result of sweeping debris preferentially to the screens. Therefore, the staff position is that licensees limit the ratio of debris transported to the inactive pools to 15% unless a higher fraction is adequately supported by analyses or experimental data (see Section 8.0).
12	Large piece debris (> 4 in.) is assumed to not transport in sump pool, hence large piece debris accumulation on sump screen completely neglected.	Avoidance of complex analyses.	The impact of neglecting all large debris on the baseline conservatism depends upon pool transport characteristics and sump screen geometry. Little impact for a slowly flowing pool where detailed analyses would predict little large debris transport, but potentially a large impact for a fast flowing pool where substantial large debris could accumulate on the screen, or for geometries such as sump screens protected by gratings at floor level.
	Head Loss Assumptions		
13	The baseline recommends using simple geometric formulas to use characteristic diameters for fibers and particles to determine specific	Lack of experimentally determined specific surface areas.	Confirmatory research has demonstrated that this approach is not reliable in that it has the potential to result in large underestimates of debris bed head loss.

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No.	Baseline Guidance Assumption	Rationale for Assumption	Perceived Level of Non-Conservatism
	surface areas needed for the NUREG/CR-6224 head loss correlation.		Therefore, the staff provides additional guidance in Appendix V to deduce the specific surface areas from applicable head loss data through the application of the correlation.

The baseline methodology assumptions were apparently made for a variety of reasons. Worst case conditions were assumed in certain situations where there is nearly a complete lack of data required to support a more realistic evaluation. These assumptions primarily include the generation of debris such as the treatment of unqualified coatings where all unqualified coatings are assumed to fail and then form fine particulate debris that would readily transport and accumulate in a fibrous bed of debris. In reality, much if not most of this coatings debris would either remain attached to the surfaces or would form chip debris that may not transport so readily. In addition to the unqualified coatings, other materials both within and outside the ZOI were assumed to fail into 100% small fines debris. The difficulty with judging the impact of these assumptions is that a particular containment may not have much of these materials; therefore the relative conservatism associated with these types of assumptions cannot be quantified for PWR containments in general.

Other baseline assumptions were made so that complex debris transport analyses could be avoided. The baseline methodology does not recommend debris transport methods but does credit debris entrapment in inactive pools. Also, it does not consider washdown transport of RMI debris and does not consider the transport of large pieces of debris. Again, the conservatism and non-conservatism of these assumptions cannot be judged for PWR containments in general but only by plant specific analyses. Assuming all fine fibrous and particulate debris washes back down to the sump pool is conservative for all plants. However, neglecting the transport of large piece debris is not conservative for all plants. Judging whether or not a conservative assumption can compensate for a non-conservative assumption requires the consideration of plant specific features. The assumption that debris entrapment within inactive pools could be made on a simple water volume ratio is not realistic because it does not consider the timing of debris washdown relative to the fill up of the inactive pools, which would occur early in the sequence. The volunteer plant study estimated that a majority of the small fine debris was blown upwards in the containment where it subsequently would be subject to washdown processes. That study estimated a majority portion of the small fine debris returning to the sump pool but the analytical capabilities cannot determine the timing of the debris entrance into the pool. If the inactive pools filled before the small fine debris washed back to the sump, then only relatively minor quantities might become so trapped. Therefore, the inactive pool entrapment assumption is probably non-conservative.

As an illustration of the variability of these assumptions as applied to the fleet of PWR plants, consider the following hypothetical situations. Assume that the application of the baseline guidance to both Plants A and B results in the prediction of adequate NPSH margin. The importance of the key assumptions is summarized in Table 3-8. The containment of Plant A is characterized as having relatively large quantities of debris with unknown debris generation characteristics and debris transport characteristics and the containment has debris transport characteristics that tend to entrap debris thereby preventing transport to the sump screens. The variability of the baseline assumptions would tend to over-predict debris generation and over-predict debris transport by substantial amounts. Therefore, if Plant A has sufficient NPSH margin evaluated using the baseline guidance, Plant A should then have an adequate NPSH margin with reasonable certainty. Plant B, however, would be characterized as having limited quantities of debris other than the ZOI insulation with reasonably well known destruction properties. Realistic debris transport fractions to the sump screen would be relatively high. Substantial larger debris transport would be expected with relatively minor quantities trapped in inactive pools. With hypothetical Plant B, there is a concern that the baseline evaluation could predict an adequate NPSH margin whereas an adequate margin may not actually exist if the collective uncertainties resulted in a non-conservative fashion.

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Table 3-8. Baseline Guidance Application to Divergent Hypothetical Plants A and B

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Unqualified Coatings (#1 and #2)	Large quantities of unqualified coatings	Little if any unqualified coatings
100% destruction of ZOI materials with unknown destruction pressures and unprotected materials outside ZOI (#3) and complete transport of the outside ZOI material (#6)	Large quantities of such materials.	Small quantities of such materials.
100% washdown transport for fibrous and particulate small fines debris (#4)	Containment design would likely retain substantial debris at the upper levels	Most debris would likely washdown to the sump pool.
100% pool transport for small fines debris not entrapped in inactive pools (#5)	Relative slow sump pool flow velocities results in significant small fines debris entrapment on sump pool floor.	Relative fast sump pool flow velocities results in little small fines debris entrapment on sump pool floor.
Debris entrapment in inactive pools (#11)	Inactive pool volumes are relatively small therefore debris entrapment in the inactive pools become minor consideration.	Inactive pool volumes are relatively large therefore debris entrapment in the inactive pools become substantial consideration.
Neglect large piece debris (#12)	Relative slow sump pool flow velocities results in little actual large piece debris transport.	Relative fast sump pool flow velocities results in substantial actual large piece debris transport.

It cannot be conclusively demonstrated with rigor that the application of the baseline evaluation methodology, as a package, to PWR plants in general can be relied upon to guarantee that any PWR predicting an adequate NPSH margin using the baseline will truly have an adequate NSPH margin. However, a reasonable assessment of the methodology is that sufficient overall realistic conservatism exists in the baseline to accept its application with the use of acceptance qualifications or alternative guidance for specific outlier situations such as the one described below.

For example, consider a hypothetical plant that has extensive unqualified coatings but insufficient fibrous debris to form a fibrous debris accumulation sufficient to filter particles. Under the baseline methodology, all the coating debris would be in the form of 10 micron particles, which would be assumed to simply pass through the screens thereby not causing a significant head loss. But in a potential LOCA, the coating debris could fail in large quantities and possibly transport as chips that could accumulate on the screen without the aid of fibrous debris, and thus result in significant head loss.

This example raises two major concerns. First, the baseline guidance excludes transport and blockage of large piece debris. The staff position is that the sump screen blockage evaluation should address whether outlier scenarios such as these exist and evaluate any that are identified. If a plants sump pool flow is relatively fast, then neglecting large piece debris could lead to substantially under-estimated debris effects. Second, for debris characterization, a caution is needed regarding the determination of whether or not there is sufficient fiber to form a thin-bed. If this determination is a close call then all aspects of that determination become critical. Licensees will need to examine inputs to ensure that each of those aspects are realistic, with appropriate conservatism added before reaching the final conclusion that there is not sufficient fibrous material in containment to form a thin bed debris accumulation.

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The results of supporting confirmatory research and information available in the knowledge base [NUREG/CR-6808] cause concern in several aspects of the baseline guidance acceptability. These concerns include:

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- Concerns regarding two phase jet effects relative to data collected from air jet testing indicate a potential need to reduce the NEI recommended destruction pressures (which are based on air jet testing) unless over conservatism can be demonstrated in the analytical estimates for debris quantities.
- The baseline evaluation recommendation of truncating a ZOI whenever it intersects a robust structure without resizing the remaining ZOI to maintain jet volumes is not conservative. Jet reflections from the robust structure may affect the remaining ZOI.
- The default coating thickness recommended by the baseline evaluation guidance are not worst case thickness. Only plant specific coating thickness evaluations can adequately assess not only the coating debris volumes but also the appropriate parameters for the head loss correlation, e.g., the particle densities.
- Because conservative estimates for the debris specific surface areas used in the NUREG/CR-6224 head loss correlation are critical to ensuring conservative estimates for the NPSH margins, the staff is concerned that the baseline

evaluation methodology recommendations for estimating the areas using only the characteristic diameters will lead to non-conservative head loss predictions. Confirmatory research recommendations should be addressed.

- The baseline methodology neglects potential erosion of large piece debris by water flows by assuming all large piece debris remains in protective coverings, which debris generation data clearly shows is not realistic. Even though such erosion is not expected to result in large quantities of additional fine debris, it should still be considered in the baseline evaluation if large portions of the large piece debris are physically located directly below large flows of fallings water.

In summary, the baseline evaluation coupled with the methodology enhancements provided in this SER is acceptable. The baseline evaluation methodology by itself cannot be given a blanket acceptance because: (1) non-conservative assumptions are recommended in the baseline guidance; (2) it is not possible to quantify the degree of conservatism or non-conservatism of each important assumption without performing detailed analyses for comparison; especially considering the diversity in the containment and RCS designs; and (3) confirmatory research has resulted in concerns associated with key aspects of the guidance. Therefore, the baseline evaluation methodology as modified in accordance with staff positions established in the preceding sections, is acceptable. If the baseline evaluation is based on planned design/operational changes, as opposed to current plant configuration, then acceptance of the evaluation is also based on the implementation of planned changes. The baseline evaluation guidance does not resolve concerns not explicitly addressed by the baseline, e.g., chemical effects and downstream effects.

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Subsequent analytical refinements to the baseline evaluation must reconsider the non-conservative assumptions of the baseline evaluation; not merely reduce identified over-conservatisms. Supplemental NEI analytical refinements include recommendations for reducing the sump pool transport fractions by means of evaluating pool flow velocities and comparing those velocities with test data for threshold velocities for moving debris along the pool floor. If such analyzes are performed on small piece debris then those analyses need to also treat large piece debris transport.

The sample problem developed in the baseline evaluation methodology may serve to illustrate the evaluation process but is not detailed enough to serve as a template for plant evaluations.

4.0 ANALYTICAL REFINEMENTS

Some acceptable analytical refinements are provided in the GR. Some sections contain additional information to support the development of refinements. Some of this information is already in the baseline. For clarity, the NEI has presented the following table (Table 4-1) that lists the refinements offered in Sections 4.0 and 5.0 of the GR.

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For the purpose of this review, the staff provides its position on each of those analytical refinements recognized in this section of the GR for use by the industry. Any analytical refinement(s) proposed by a licensee in its plant-specific analysis of sump performance which is not addressed by the staff in this section of the SER, should be presented to the staff for approval.

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4.1 INTRODUCTION

Section 4.1 defines four main analytical topics where analytical refinements to the baseline evaluation are offered in the GR. They are (1) break selection, (2) debris generation, (3) debris transport, and (4) head loss.

4.2 METHOD DESCRIPTION

Section 4.2 identifies three main analytical topics where refinements to the baseline evaluation are offered in Section 4.0, of the GR. They are (1) debris generation, (2) debris transport, and (3) head loss. It is stated that discussions on the other two topics, i.e., break selection and latent debris, are included for completeness.

Table 4-1. Pressurized Water Reactor Sump Performance Evaluation Methodology Refinements Table

No.	Section	Page	Topic	Description
1	4.2.1	4-1	Break Selection	This section identifies that plants may use Generic Letter 87-11, "Relaxation in Arbitrary Intermediate Pipe Rupture Requirements," consistent with their licensing basis, to select break locations for evaluating post-accident sump operability.
2	4.2.2.1	4-2	Debris Generation	This section identifies that plants may refine the Zone of Influence (ZOI) definition from a single all-encompassing region based on the material with the minimum destruction pressure by assigning multiple ZOIs to each break site. Each ZOI would correspond to the destruction pressure of one insulation species located near the break site.
3	4.2.2.1	4-3	Debris Generation	This section identifies that plants may refine the Zone of Influence (ZOI) definition by modeling two freely-expanding jets, each originating at one end of a postulated DEGB. The ZOI for a specific material would be evaluated as the region enclosed within the calculated isobar corresponding to a given destruction pressure of an insulation species located within the jet.
4	4.2.2.2	4-5	Debris Characteristics	This section provides additional refinements with respect to the characteristics of debris that might be generated from a postulated break. Specifically, the use of plant-specific or publicly available vendor-specific information, where applicable, is identified as source for refining debris sizes considered in the transport and blockage evaluation.
5	4.2.3	4-14	Latent Debris	This section identifies that plant-specific conditions (for example, cleanliness programs) may be used to support improvements to the latent debris source term
6	4.2.4	4-14	Debris Transport	This section identifies two refinements to evaluate debris transport. • The first refinement is the use of an open channel nodal network to evaluate bulk fluid movement about the containment. • The second refinement is the use of a Computational Fluid Dynamics (CFD) model to calculate a detailed flow field within the containment sump and assess debris transport.

Table 4-1. Pressurized Water Reactor Sump Performance Evaluation Methodology Refinements Table (Continued)

No.	Section	Page	Topic	Description
7	4.2.4.1	4-14	Debris Transport	This section provides guidance on the development of an open channel network model Guidance is given on: • Use of the physical configuration of the containment geometry to define the model, • Development of boundary conditions based on sources and sinks of cooling water, • Defining hydraulic channels • Calculation of hydraulic losses in the channels, and, • Refinements to the channel pattern. A sample calculation is included for demonstration purposes.
8	4.2.4.2	4-23	Debris Transport	This section provides guidance on the development of detailed flow patterns in the containment pool using state-of-the-art 3D computational fluid dynamics (CFD) codes. Guidance is given on: • Selection of CFD software, • Building a CAD model of the containment to be used as input to the CFD model • Building the CFD model, including mesh generation and selection of material properties and boundary conditions, • Solution convergence considerations, and, • Use of computed results for evaluating debris transport. A sample calculation is included for demonstration purposes.
9	Table 4-2	4-29	Debris Transport	This table provides additional transport data for debris generated from common insulation materials. This information may be used in conjunction with either the Open Channel Nodal Network or CFD models to evaluate debris transport in the sump pool during operation of the ECCS in the recirculation mode.
10	4.2.5.1	4-35	Head Loss	This section identifies that no refinements for evaluating thin bed effects are offered beyond those already given in Section 3.7.2.3.2.3.

Table 4-1. Pressurized Water Reactor Sump Performance Evaluation Methodology Refinements Table (Continued)

No.	Section	Page	Topic	Description
11	4.2.5.2	4-35	Head Loss	This section presents information that may be helpful in refining the head loss analysis as a whole including a brief background discussion on head loss correlation development. This section identifies the parameters to be considered when developing a head loss correlation. This discussion is given to identify the considerations to be accounted for when developing a design-specific head loss correlation for a sump screen.
12	4.2.5.2.1	4-37	Head Loss	This section presents a summary of early sump screen head loss testing. Included in the discussion is the method of test, a summary of the nature of the tests and the data obtained, and how the data were correlated. This is provided to facilitate understanding of the nature and complexity of head loss testing. Add statement regarding plant-specific basis.
13	4.2.5.2.2	4-39	Head Loss	Several special head loss correlations are presented and discussed. Specifically: • An empirical correlation for fiber-only beds, • The US NRC NUREG/CR-6224 head loss model, • The US BWROG combined debris head loss correlation, and, • Correlations for head loss due to flow through reflective metallic insulation (RMI). The basis for, and considerations to be accounted for, in applying the RMI head loss equations are also listed.
14	4.2.5.2.3	4-50	Head Loss	This section presents information that may be useful in the development of correlations for alternate strainer designs. Two potential improvements identified for head loss modeling for alternate strainer designs are identified: • Accounting for geometry of the screen, if it varies significantly from a flat plate, and, • Non-uniform deposition of debris on the strainer, if appropriate and justifiable.

Table 4-1. Pressurized Water Reactor Sump Performance Evaluation Methodology Refinements Table (Continued)

No.	Section	Page	Topic	Description
15	5.1	5-1	Debris Source Term	This section identifies possible design and operational activities that may be undertaken to reduce the debris source term, such as: • Improved housekeeping and foreign materials exclusion (FME) programs • Insulation change-out, • Insulation modifications, • System and equipment modifications, and, • Modifications to protective coatings programs.
16	5.2	5-4	Debris Transport	This section identifies information that might be used for debris barriers that might mitigate debris transport about the containment. These barriers include: • Floor obstructions, and, • Debris racks.
17	5.3	5-6	Screen Modifications	This sections identifies options for sump screen modifications, including: • Passive strainer designs, • Backwash strainer designs, and, • Active strainer designs. In addition to the sump screen modification options, a list of considerations for each of the options is identified.

4.2.1 Break Selection

Section 4.2.1 of the GR discusses an analytical refinement involving pipe break locations to be considered when performing PWR sump analyses. The proposed guidance suggests application of NRC Generic Letter 87-11, "Relaxation in Arbitrary Intermediate Pipe Rupture Requirements," (GL-87-11) to preclude arbitrary intermediate pipe break locations from consideration in PWR sump analyses. The refinement suggests consideration of only those break locations which are consistent with Branch Technical Position MEB 3-1, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," of NUREG-0800, "Standard Review Plan (SRP)," (SRP), Section 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping." Application of Branch Technical Position MEB 3-1 for PWR sump analyses is intended to focus attention on high stress and fatigue break locations, such as at the terminal ends of a piping system and intermediate pipe ruptures at locations of high stress.

Staff Evaluation for Section 4.2.1: The staff's evaluation of this section considered the proposed GR guidance in conjunction with existing, corresponding guidance on this subject. The staff's review considered the requirements of 10 CFR 50.46, the staff's evaluation and conclusions for a similar proposal from the boiling water reactor owners group (URG SER), the guidance provided in Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," (RG 1.82-3), and the Commission's staff requirements memorandum (SRM) regarding a proposed rulemaking to risk-inform requirements related to large break LOCA break size (SECY-04-0037).

GSI-191 and the concern of PWR sump blockage is directly associated with the long-term cooling acceptance criteria listed in 10 CFR 50.46 (b)(5). To ensure acceptable ECCS cooling capability, 10 CFR 50.46 requires that, "ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated." The staff notes that the worst breaks with respect to peak clad temperature and the other acceptance criteria of 10 CFR 50.46 may not necessarily be the limiting breaks for debris generation and sump head loss. When evaluating ECCS performance for compliance with 10 CFR 50.46, SRP Sections 6.3, "Emergency Core Cooling System," and 15.6.5, "Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary," are the appropriate SRP sections to consider. SRP Section 15.6.5 states that reviewers "evaluate whether the entire break spectrum (break size and location) has been addressed." The proposed GR guidance to consider only break locations which are consistent with Branch Technical Position MEB 3-1 is not consistent with the requirements of 10 CFR 50.46 because MEB 3-1 may not provide assurance that the most severe postulated LOCA's are calculated.

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NRC Regulatory Guide 1.82, Revision 3 (RG 1.82-3) provides NRC staff guidance regarding an appropriate spectrum of breaks to be considered when evaluating PWR sump performance. Specifically, regulatory position 1.3.2.3 of Regulatory Guide 1.82 states that a "sufficient number of breaks in each high-pressure system that relies on recirculation should be considered to reasonably bound variations in debris generation by the size, quantity, and type of debris." As a minimum, the staff position is that the following postulated break locations should be considered: (a) Breaks in the hot leg, cold leg, intermediate leg, and, depending on the plant licensing basis, main steam and main feedwater lines with the largest amount of potential debris within the postulated zone of influence, (b) Large breaks with two or more different types of debris, including the breaks with the most variety of debris, within the expected zone of influence, (c) Breaks in areas with the most direct path to the sump, (d) Medium and large breaks with the largest potential particulate debris to insulation ratio by weight, and (e) Breaks that generate an amount of fibrous debris that, after its transport to the sump screen, creates a minimum uniform thin bed (1/8-inch layer of fiber) to filter particulate debris. The staff considers that Regulatory Guide 1.82 provides the complete scope of breaks which should be evaluated to ensure that 10 CFR 50.46 is satisfied. The proposed GR guidance to consider only break locations which are consistent with Branch Technical Position MEB 3-1 does not provide an adequate alternative to the guidance provided in Regulatory Guide 1.82, Revision 3 to demonstrate compliance with 10 CFR 50.46 because the complete scope of break locations may not be evaluated.

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The staff previously reviewed a similar request to apply SRP Section 3.6.2 and Branch Technical Position MEB 3-1 for identifying break locations to be considered when evaluating ECCS strainer concerns in BWRs. As documented in the staff's safety evaluation report for the BWR's (URG SER), the staff rejected the BWROG proposal for two reasons. The first reason is that SRP Section 3.6.2 and Branch Technical Position MEB 3-1 do not provide guidance or acceptance criteria for demonstrating compliance with the requirements of 10 CFR 50.46. The staff noted that the only acceptance criterion specified in SRP Section 3.6.2 is compliance with General Design Criteria (GDC) 4. GDC 4 requires that licensees must protect structures, systems and components important to safety from the dynamic effects (e.g., pipe whip, direct steam jet impingement, etc.) and environmental effects (e.g., temperature, pressure, radiological effects) of postulated pipe ruptures. The staff communicated through Generic Letter 87-11, which transmitted the revised SRP Section 3.6.2 and Branch Technical Position MEB 3-1, that licensees could still provide an adequate and practical level of protection for compliance with GDC 4 by reducing the number of postulated pipe breaks and by physically protecting equipment important to safety from the postulated pipe breaks that have a relatively higher potential for failure (e.g., postulated failures at high-stress and fatigue locations). As a result, when demonstrating compliance with GDC 4, licensees may analyze pipe breaks through the use of pipe stress analysis methodologies similar to that provided in SRP Section 3.6.2 and Branch Technical Position MEB 3-1. The staff considers SRP Section 3.6.2 and Branch Technical Position MEB 3-1 to be inappropriate for postulating break locations for the purpose of determining the extent of debris generated in order to comply with 10 CFR 50.46 because these are applied to demonstrate compliance with GDC 4, not 10 CFR 50.46. The second reason given by the staff in rejecting the BWROG proposal was that the BWROG had not demonstrated that break locations selected consistent with SRP Section 3.6.2 and Branch Technical Position MEB 3-1 would bound the worst-case debris generation scenarios and, therefore, meet the intent of 10 CFR 50.46. The staff finds that this discussion also applies to the PWR's and the GR proposal.

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Finally, in evaluating the GR proposal, the staff considered the current effort involving a proposed rulemaking to risk-inform requirements related to large break LOCA break size. For a risk-informed 10 CFR 50.46, the staff is revising the design basis LOCA break size, but does not plan on changing its current position regarding break locations which need to be considered for purposes of meeting the requirements of 10 CFR 50.46. The staff's intention is to ensure that GSI-191 resolution methodology be consistent with the 50.46 rulemaking effort.

Based on the above discussions, the staff concludes that it is inappropriate to cite SRP 3.6.2 and Branch Technical Position MEB 3-1 as methodology to be applied for determining break locations to be considered for PWR sump analyses because these may not identify the limiting break location. The staff concludes that the guidance regarding break locations, as described in GR Section 3.3 (and as amended in Section 3.3 of the staff's safety evaluation report) should be followed when performing PWR sump analyses. The staff's conclusion applies for the entire spectrum of pipe break sizes which are considered. When performing analyses described in Section 6 of the GR, "Alternate Evaluation," this conclusion applies for both Region I and Region II analyses.

4.2.2 Debris Generation

4.2.2.1 Zone of Influence

This section reiterates that, for the baseline calculation, the GR recommends the use of a spherical ZOI to encompass the effects of jet expansion resulting from impingement on structures and components. It notes that two refinements are to be presented for insulation materials, but none are offered relative to coatings.

Staff Evaluation for Section 4.2.2.1: The spherical zone is a practical convenience that accounts for multiple jet reflections and mutual interference of jets from opposing sides of a guillotine break. It is important to note that when the spherical volume is computed using an acceptable approximation for unimpeded free-jet expansion, the actual energy loss involved in multiple reflections is conservatively neglected to maximize the size of the ZOI. The staff concurs with the use of spherical ZOI as a practical approximation for jet-impingement damage zones.

4.2.2.1.1 Method 1: Debris-Specific Spherical ZOIs

Method 1 refines the evaluation of ZOI by recommending that multiple ZOIs be assigned to each break site, with each corresponding to the destruction pressure of one insulation species located near the break site. Pressure isobars used to define the equivalent volume spherical ZOI pertinent to a particular insulation type are determined using the methodology of the ANSI/ANS 58.2-1988 standard. Destruction pressures for several insulation types were presented in Table 3-1 of the GR. That table provided the ratio of the ZOI radius to the break diameter for each insulation type listed. The Method 1 discussion notes that no changes to insulation destruction pressures are to be made to account for differences between dry and saturated steam jets. Robust barriers and the effects on the ZOI are to be treated as discussed in Section 3.4 of the GR.

Once the ZOI for each insulation type has been determined, the debris generated within each ZOI is calculated and the individual contributions are summed to arrive at a total debris source term.

Staff Evaluation for Section 4.2.2.1.1: The NRC agrees that the definition of multiple spherical ZOI at each break location that correspond to the damage pressures of potentially affected materials is an appropriate refinement for debris generation calculations. Furthermore, it is also appropriate to apply this refinement in a selective manner. For example, a separate well-characterized ZOI can be applied for coatings and all insulation types can be treated according to the baseline assumption of damage equivalent to the most vulnerable material in containment. This approach was illustrated in the Sample Calculation presented in Section 3.4.2.6. Target material inventories within their respective ZOI should be calculated as in accordance with the staff evaluation in Section 3.4 of this SER, including the treatment of robust barriers.

Definition of Spherical ZOI

Application of the ANSI/ANS 58.2-1988 jet model was reviewed in Section 3.4 of the GR and in Appendix I of this report and was found to be an acceptable approach for computing volume-equivalent spherical ZOI. However, material-specific damage pressures that were experimentally determined using high-pressure air as a surrogate working fluid should be treated in a manner similar to that presented in Section 3.4.2.2 to account for potential differences between dry and flashing two phase water jets. The listing of damage pressure provided in Table 3-1 of the GR implicitly acknowledges the potential for enhanced destruction by citing two-phase destruction tests for calcium silicate. The staff position to reduce destruction pressure by 40% for materials not tested under two-phase conditions is substantial; however, it is less than the decrease measured for calcium silicate.

Three additional refinements related to the application of the ANSI jet model can be developed on a case-by-case basis for selected breaks if it is advantageous to do so:

1. First, the application of worst-case thermal hydraulic conditions to every break location can be relaxed if there is supporting evidence to demonstrate that a particular break location or class of break locations exhibits substantially different conditions that can be conservatively calculated or measured. Maximum damage volumes are generally driven by increased pressure, but these volumes can exhibit unexpected changes related to the degree of subcooling. (See Appendix I).
2. Second, the assumption of equivalent maximum mass flux from both ends of a guillotine break can be relaxed if there are supporting calculations to conservatively substantiate important differences between the thermal hydraulic conditions upstream in either direction. Damage volumes from each side would be calculated independently and then added similar to the way that damage volumes are doubled for the baseline analysis.
3. Third, some credit can be taken via conservative approximation for friction losses in lines leading to the break location if adequate documentation of roughness coefficients, and flow losses in piping components can be provided. This refinement will have the effect of reducing the effective total pressure at the exit plane below the stagnation pressure of the upstream system reservoir. The system stagnation enthalpy should be assumed constant.

It is expected (but not necessary) that these refinements would be pursued on a selective basis for break locations that are found to drive key decision points. For example, limiting breaks identified under the baseline assumptions might be found that impact vulnerable insulation types that are located in high-radiation areas. While replacement of vulnerable insulations with more robust material might be the desired mitigation option, these refinements might be effective in demonstrating that the material should be left in place. If these refinements are applied as described for the purpose of exempting specific targets, the corresponding assumed break locations should be located such that the flow-path distance between break and target is minimized. These refinements can be applied selectively in any combination, and they apply as well to the Method 2 refinement for direct jet impingement.

The ZOI and Robust Barriers

Target material inventories within their respective ZOI or generic ZOI should be calculated as discussed in Section 3.4 of this SER, including the treatment of robust barriers. Section 3.4 does not allow simple truncation for robust barriers as proposed in the GR.

Evaluating Debris Generation Within the ZOI

The NRC agrees that the contributions of each material type to the total debris inventory should be added to determine the debris source term available for transport as described in other sections of the GR and is an acceptable approach.

4.2.2.1.2 Method 2: Direct jet Impingement Model

This section of the GR offers the refinement of defining the ZOI by modeling two freely-expanding jets emanating from each broken pipe section as opposed to using the spherical ZOI approach presented in Section 3.4. The ANSI standard ANSI/ANS 58.2-1988 is recommended for determining the jet geometry. The specific procedures to be followed for determining jet geometry are summarized, and an example calculation is discussed. The results of the isobar mapping calculations and an example of a plotted isobar are presented in Appendix D of the GR. The treatment of robust barriers and the determination of overall debris generation are the same as for Method 1.

Staff Evaluation for Section 4.2.2.1.2: The NRC staff has reviewed this refinement and finds it acceptable. This refinement retains some spatial information inherent to the direction of the severed pipe. It implicitly assumes that the ends of the pipe are fully separated and fully offset, but yet, remain basically aligned in the original direction. The staff notes that there is no specific analysis of pipe-whip potential if this method is used. However, the spherical ZOI approximation carries similar inherent assumptions (basic alignment of pipe segments to create a spherical ZOI from opposing and interfering jets). Although not explicitly stated, the perceived advantage of this method under strict implementation of the GR would follow from truncation of a jet segment that impinges directly on a barrier like a wall or floor, as well as the economy associated with use of ZOI calculations that have already been performed for local dynamic effects (GDC4 analyses). The practice of ZOI truncation was reviewed in Section 3.4 and was judged to be nonconservative compared to the concept of ZOI volume conservation. Licensees electing direct impingement model refinement should retain the volume for conservatism. In fact, the mapping of an independent directional jet segment within containment would be necessary for postulated sidewall ruptures if they are considered for analysis. Analysis of sidewall ruptures would carry the additional burden of investigating

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alternative jet directions. In lieu of mapping directional jet segments for sidewall ruptures, Section 6 of this Safety Evaluation reviews the use of directional (worst debris generation) hemispherical break geometry as an acceptable alternative to assuming a sphere for partial breaks in RCS main loop piping (non-DEGB).

The information provided in this section on ANSI jet modeling is identical to that provided in Section 3.4.2.1 and was reviewed previously. However, the staff would like to emphasize the GR statement that this refinement relies upon a high degree of rigor in determining what stagnation pressure each insulation type is subjected to. The first task is to model unimpeded jet expansion using the ANSI standard and Appendix I of this SER for guidance, and the second task is to map relative spatial geometries of targets and the jet in the vicinity of the break location. It is also true, as stated in the GR, that isobar contours like those presented in Appendix D of the GR and Appendix I of the SER have rotational symmetry and can be rotated about the longitudinal axis to define the three-dimensional surface of equivalent damage potential, i.e. impingement pressure.

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As a point of nomenclature consistency, there is a conceptual difference between the classical definition of stagnation pressure in a moving fluid as approximated by Bernoulli's Law and the pressures predicted by the ANSI model. The predicted pressures are referred to throughout the SER as impingement pressures because they represent nonisentropic stoppage of the fluid on the face of a target that should be slightly higher than the theoretical stagnation pressure at a freestream point in the flow field. Other limitations to this interpretation of the predicted jet pressures also apply as discussed in Appendix I.

It should be noted that the additional optional refinements discussed above as Method 1 refinements for debris-specific ZOI also apply to this Method 2. The choice of using an approximate spherical geometry or the more realistic geometry of a directed jet is largely independent of the thermal hydraulic assumptions used to compute a jet contour.

The ZOI and Robust Barriers

Target material inventories within their respective ZOI or generic ZOI should be calculated as discussed in Section 3.4 of this SER. The isobar volume of interest should be mapped and conserved independently for the jet on each side of the break. The total damage volume of the two jets should be preserved in a contiguous region rather than crediting overlapping reflections.

Evaluating Debris Generation Within the ZOI

The guidance offered in this section is identical to that presented in Section 3.4.2.5 and has been reviewed previously. Additionally, the contributions of debris from both independently evaluated jets are added to represent the total debris source term.

4.2.2.2 Debris Characteristics

Section 4.2.2.2 provides additional information regarding the characteristics of debris following a postulated break. The section recommends using plant-specific or publicly available vendor-specific information, where applicable, for refining debris sizes considered in the transport and blockage evaluations. The section includes Table 4-1 that contains recommendations for destruction pressures, fabrication and material densities, and debris characteristic sizes. In addition to replicating data presented in baseline Tables 3-1 and 3-2, Table 4-1 includes recommendations for other materials as well.

__Staff Evaluation for Section 4.2.2.2:__ The staff has the following concerns regarding the guidance provided in Section 4.2.2.2.

1. In Section 4.2.2.2.1 "Fibrous Insulation", the guidance states "Not all generated fibrous debris needs to be assumed to be of a transportable size." The reality is all debris not specifically attached to a structure can transport given a sufficient driving force. For example, an entire intact blanket of fibrous debris will move in a pool of water if the flow velocities are sufficiently fast. Sheeting flows during testing has shown the capability of moving intact RMI cassettes under certain conditions. The point is that all debris should be considered transportable until plant-specific analyses determine otherwise.

2. GR Reference 27 was cited in Section 4.2.2.2.2 "Reflective Metallic Insulation (RMI)" as a source of information for the debris size distribution for RMI debris. However, Reference 27 is a report on the testing of NUKON™ insulation and does not contain RMI information. Therefore, an appropriate debris size distribution for RMI debris is not available in the GR. Reference 27 is also inappropriately cited for evaluating coatings in Section 4.2.2.2.3, "Coatings".

3. In Section 4.2.2.2.3.1, "Coatings within the ZOI", the GR recommends using the properties of a multiple coating system that produces the post-accident debris with the most detrimental effects to the containment sump. However, the GR does not provide guidance regarding which types of properties (e.g., a light or heavy coating density) would produce the most detrimental effects. The most detrimental properties for debris transport may differ from those most detrimental to head loss. The staff is concerned that such ambiguity in the guidance could lead to improperly determined properties from a conservative standpoint and recommends that each component in a multiple coating system be evaluated separately with its applicable properties. Effective properties for multiple types of debris can then be determined. In a similar statement in Section 4.2.2.2.3.2 "Coatings outside the ZOI", assuming properties for unidentified non-DBA-qualified coatings systems used outside the ZOI should assume the most detrimental properties needs more supporting guidance regarding which types of properties are most detrimental.

4. In Section 4.2.2.2.4, the GR recommends assuming that all tape and stickers located in the ZOI are destroyed into small pieces and fibers. The positive aspect of the assumption is the subsequent transport to the sump screens would than be 100% of this debris. However, it is not a forgone conclusion that assuming the debris is destroyed into small pieces and fibers would cause a higher head loss than if this debris arrived at the screens intact, which is one of the potential realities, at least for non-soluble tapes, stickers, and tags. As intact debris, this debris could effectively interdict flow through covered portions of the screen thereby effectively reducing the size of the screen. Hence, the GR statement that it is conservative to assume that all debris created from tape and stickers is reduced into fine or small pieces or individual fibers is not supported. It is recommended that the head loss evaluation estimate the head

loss by assuming each condition of the debris, and then use the higher head loss in the NPSH margin determination.

5. In Section 4.2.2.2.5 "Fire Barrier Materials", fire barriers consist of many types of insulation and other materials including board materials, blanket materials, and foam materials. With a few exceptions, debris generation data does not exist for fire barrier materials that differ from the piping insulations tested. The GR recommends "For materials that are unique to fire barrier applications and do not have supporting test data, a destruction pressure equal to that of low-density fiberglass may be assumed." While this guidance seems reasonable for fire barrier materials consisting of a low-density fiberglass or even a high-density fiberglass, it is not acceptable to apply data for low-density fiberglass to the variety of fire barrier materials, e.g., board and foam materials.

The staff did not independently verify all the data contained in GR Table 4-1 and has the following concerns:

1. Table 4-1 provides four seam orientation calcium silicate destruction pressures (i.e., 0°, 45°, 180°, and generic orientation) without additional guidance and the zero degree reference was not stated. Application of seam oriented destruction pressures requires orientation specific jet destruction models. As discussed in Appendix II, because substantial insulation damage occurred at a jet pressure of 24 psi in the OPG tests (45° orientation), the lowest pressure tested; the threshold pressure for destruction is actually less than 24 psi. The staff recommends using the recommendation in NUREG/CR-6808 of 20 psi for calcium silicate.

2. The destruction pressure recommended in Table 4-1 is 2.5 psi for blanketed and unjacketed Min-K whereas in the baseline Table 3-1 the GR recommendation is 4 psi. Hence, these two recommendations are in conflict. The staff recommends using a destruction pressure of 2.5 psi for blanketed unjacketed Min-K in the baseline as well as in the refinements. The GR recommended destruction pressure of 6 psi for blanketed jacketed Min-K with SS bands and latch and strike locks does not specify the jacket construction. Unless a specific jacket construction can be correlated to test data whereby it can be shown that a pressure of 6 psi or greater is needed to compromise that specific jacket, then the lower destruction pressure of 2.5 psi should be used.

3. It is noted that several data are missing from Table 4-1 that will be required by the analyst. For example, the material density for Min-K is specified as NA but will be required when applying the GR recommended NUREG/CR-6224 head loss correlation.

4. The destruction pressure for Microtherm was apparently set equal to that of Min-K in Table 4-1, without justifying remarks. Some rationale should have been presented for this justification.

5. For Knaupf with an as-fabricated density of 2.4 lb/ft^3, a destruction pressure of 10 psi is recommended in Table 4-1; and for Knaupf with an as-fabricated density of 4.0 or (blank), the GR does not recommend a destruction pressure. However, in Table 3-1 of the GR, one entry exists for Knaupf which

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recommends a destruction pressure of 10 psi, alone. Because of the inconsistency, application of this guidance for Knaupf should be based on its as-fabricated density, as appropriate.

6. The destruction pressure recommended in Table 4-1 for Kaowool was made without justifying remarks or reference. Some rationale should have been presented for this assumption.

7. The as-fabricated density of Kaowool is specified as 9.4 lbs/ft^3 in Table 4-1 but given as range of 3 to 12 lbs/ft^3 in baseline Table 3-2. If this density is a manufacturing variable, then the plant-specific as-applied density should be used. As illustrated in Appendix V, the head loss evaluation is very dependent upon this number.

8. The reference number provided for the material density of Kaowool is given as "xx," which is not listed in the GR References Section (i.e., Section 9), and should be corrected and/or provided.

9. The destruction pressure for Mirror® with Sure-Hold® bands is recommended in Table 3-1 of the GR, as 150 psi; however, this item is missing from GR Table 4-1. The staff evaluation of this value is provided in Section 3.4.2 of this SER. The acceptable value provided in Table 3-2 of this SER should also be used if applied as a refinement.

10. The destruction pressure recommended in Table 4-1 for Silicone foam was made without justifying remarks or reference. Some rationale should have been presented for this assumption.

11. The destruction pressure recommended in Table 4-1 for Gypsum board was made without justifying remarks or reference. Some rationale should have been presented for this assumption.

12.

Staff Conclusions Regarding Section 4.2.2.2: The staff finds that use of debris-specific characteristics as a refinement to the baseline is acceptable. However, the cautions listed above should be considered in the use of this refinement and debris-specific data should be sought.

4.2.3 Latent Debris

Although the GR does not identify any generic analytical refinements for quantifying latent debris in this section, other methods identified as acceptable alternatives by the staff in Section 3.5 for sampling plans could be viewed as refinements to a conservatively assumed baseline inventory.

4.2.4 Debris Transport

Section 4.2.4 recommends two methods of analytical refinements for determining the flow characteristics of the sump pool for the purpose of predicting the transport of debris in the sump pool to the recirculation sump screens. These methods included the open channel flow network method (Section 4.2.4.1) and the three-dimensional computational

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<#>Some data were assumed without justifying remarks, e.g., the destruction pressure for Microthe was apparently set equal to that Min-K. Some rationale should hav been presented for this and other justifications.¶
The as-fabricated density of Kaowool is specified as 9.4 lbs/ft^3 in Table 4-1 but given as range of 3 to 12 lbs/ft^3 in baseline Table 3-2. If this density is a manufacturing variable, then the plant-specific as-applied density should be used. As illustrated in Appendix V, the head loss evaluation is very dependent upon this number.

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fluid dynamics (CFD) method (Section 4.2.4.2). Aspects of the network method discussed included the following: the analytical approach, model input development, and the network solution. An example network model was superimposed onto a corresponding CFD result. No discussion was provided regarding the use of network predicted results to estimate debris transport within the sump pool. Aspects of the CFD method discussed included the following: the selection of software, the building of a computer aided design model that could be used to generate the computational mesh, the CFD analysis, and the prediction of debris transport using the CFD results.

The debris transport discussion associated with the CFD modeling included a discussion of plotting velocity magnitude contours for the minimum bulk transport velocity at selected levels within the containment pool. After the area within this transport velocity contour is determined, the debris within this area is assumed to transport to the sump screen.

The GR also includes Table 4-2, "Debris Transport Reference Table," that provides transport data such as the minimum velocities needed to transport debris.

Staff Evaluation for Section 4.2.4: Of the two methods of analytical refinements for transport of debris in the sump pool the staff identified the following challenges in using the open channel network method:

1. The implementation of the network method requires the adaptation of multiple correlations for estimating form loss coefficients and friction factors (correlations typical of piping pressure loss calculations). At each network node junction, a form loss coefficient is required that simulates flow for the connecting nodes. The complexity of the sump pool channel will require the analyst to make engineering judgment adaptations for the application of generic correlations and the complexity of the model input development can severely limit the detail of the model resulting in a rather coarse nodalization.

   The coarseness of the network method, as illustrated by the example nodalization in GR Figure 4-4, limits the simulation of important aspects of the sump pool such as the complexity of the flow channel, obstacles to flow, and the complex distribution of containment spray drainage entering the pool. The example nodalization has ignored portions of the sump pool without providing a rationale for determining which portions of the pool do not need to be modeled.

2. The model coarseness forces the analyst to rely on predicted bulk velocities between coarse nodes and therefore cannot predict localized flow conditions that are capable of moving debris even if the bulk flow velocities indicate no movement of debris. An example of localized flow is vortices that could be completely internal to a network node. Testing [NUREG/CR-6773] has shown that vortices affect debris transport.

3. The network method is not capable of predicting sump pool turbulence or its effects on debris transport. Sump pool turbulence has been shown to affect debris suspension within the pool (e.g., water flows falling into the sump pool can suspend debris that would normally settle in calm water) and the rates of erosion (Section III.3.3.3) for certain types of debris (e.g., fiberglass insulation debris).

4. The network method is not capable of predicting pool characteristic during pool formation that affect the transport of debris during this period such as the initial spreading of water across the floor or the filling of inactive portions of the sump (e.g. reactor cavity).

5. The large number of input parameters associated with specifying a network nodalization model (e.g., inputs to form loss correlations) could make the performance of a quality sensitivity evaluation for those input values difficult.

Appendix C compares the results of the open channel network method to the results of CFD method. The staff concluded that the results do not agree in contrast to the assertion in the GR that the network and CFD results compare favorably. The difference in flow rates of less than 10% were calculated by dividing by the total recirculation flow. For example, the GR quoted error for Channel 156 is 7.7% (Table C-1) but the flow for the network method is in the opposite direction to that of the CFD analyses. If the difference for Channel 156 were calculated as the difference between the network and the CFD predicted flow rates divided by the CFD the result would have been 56% instead of 7.7%. In addition, the flows of the network and CFD methods are in the opposite direction.

The GR recommends adding 10% to the calculated channel flow rates but the staff recommends that the safety factor applied to the network calculated results be based on benchmark analyses of the network methodology against experimental debris transport results and/or superior analytical methods. In addition, a method is still needed to perform the needed analysis that is well beyond the capabilities of the network method.

Regulatory Position 1.3.3.4 of Regulatory Guide 1.82, Revision 3, states the following:

> An acceptable analytical approach to predict debris transport within the sump pool is to use computational fluid dynamics (CFD) simulations in combination with the experimental debris transport data. Examples of this approach are provided in NUREG/CR-6772 and NUREG/CR-6773. Alternative methods for debris transport analyses are also acceptable, provided they are supported by adequate validation of analytical techniques using experimental data to ensure that the debris transport estimates are conservative with respect to the quantities and types of debris transported to the sump screen.

Consistent with the above regulatory position the staff accepts the nodal network method as an alternative method to calculate debris transport onto the sump screens. However, the licensees should support this method using experimental data to ensure that the debris transport estimates are conservative with respect to the quantities and types of debris transported to the sump screen.

The staff finds that the GR presentation regarding the CFD method and analysis is thorough. Specific staff comments include:

1. The GR suggests using turbulent turbine kinetic energy (TKE) profiles in the pool as a pool characteristic but fails to prescribe how this information would be useful in the debris transport analysis. The staff recommends a potential adaptation of a CFD method employed in the BWR drywell debris transport study [NUREG/CR-6369, Vol. 3] where the CFD code is also used to simulate

applicable tests where debris settling was correlated to the CFD predicted turbulence indicators.

2. The GR discussions regarding the level of detail or analytical fineness to model does not adequately address potential plant features that can significantly affect sump pool hydraulics. For example, the GR statement that "Obstructions less than 6 inches in diameter or the equivalent may be omitted," is too general a statement. If there is a single 6 inch obstacle, it might be argued that it can be neglected but if there is a series or array of 6 inch objects, then the array may need to be modeled.

3. Other model development aspects, including the following, should be properly assessed before selecting modeling options: the type and size of calculational mesh, boundary conditions inflow and outflow options, and convergence criteria. Many of the modeling options depend upon the CFD code selected and the model development should properly select the best options for the plant-specific sump pool evaluation.

The GR recommends using a uniform distribution of debris on the sump floor, i.e., the sump pool debris transport fraction is equal to the floor area fraction where the velocity is greater than the minimum transport velocity (GR Section 4.2.4.2.5). This recommendation is not acceptable because the debris entrance into the pool is not uniform. The staff provided supplemental guidance in Appendices III and VI addressing sump pool debris transport and blowdown/washdown transport, respectively, in the volunteer plant. Appendix III demonstrated that the GR floor area transport model would under-predict the sump pool debris transport in the volunteer plant by a wide margin. Debris initially deposited onto the sump floor in the volunteer plant was preferentially deposited within or near the break compartment due to the partial confinement of debris in the break compartment and debris initially deposited in the upper levels of the containment would washdown with the drainage of the containment sprays entering the sump pool at discrete locations, typically in the faster areas of the pool. The licensees should use the debris transport methodologies presented in Appendices III and IV for refined analyses.

In the GR baseline, a two group size distribution was recommended where the small fines would completely transport to the sump screens and the large debris would not transport at all. Therefore, the sump pool debris transport refinement cannot be applied to small fines because at least a portion of this group must be treated as suspended fines with complete transport. A refinement can be applied to the large size group but in the baseline guidance this group is assumed to not transport. In order to proceed with a sump pool analytical refinement, a better defined size distribution that addresses the key aspects of debris transport should be used. In addition, if the analytical refinement is applied to the small debris, it should also be applied to the large debris that is neglected in the baseline methodology. The licensee should use the four size categories used in both Appendices III and VI for fibrous debris. This size distribution has: (1) fines that remain suspended, (2) small piece debris that transport along the pool floor, (3) large piece debris with the insulation exposed to potential erosion, and (4) large debris where the insulation is still protected by a covering thereby preventing further erosion.

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GR Table 4-2, "Debris Transport Reference Table," provides useful data and references NRC published documents as the source of the data. However, one column in the table provides selected values for TKE energies required to suspend debris that are not in the referenced NRC published documents. The staff has not assessed or accepted the TKE numbers presented in GR Table 4-2.

Staff Conclusions Regarding Section 4.2.4: Section 4.2.4 recommends the open channel flow network method and the three-dimensional computational fluid dynamics method for refining the analysis for transport of debris in the sump pool to the recirculation sump screens. Consistent with Regulatory Guide 1.82, Revision 3, the staff accepts (1) the CFD method and (2) the nodal network method as an alternative method to calculate debris transport onto the sump screens. However, the licensees using the nodal network method should support it using experimental data to ensure that the debris transport estimates are conservative with respect to the quantities and types of debris transported to the sump screen. The GR recommended debris transport model in Section 4.2.4 that assumes using a uniform distribution of debris across the sump floor is not acceptable because the debris entrance into the pool is not uniform. Appendices III and VI provide additional staff guidance on adapting the debris transport methodologies for refined analyses.

4.2.5 Head Loss

The GR states that no head loss refinements are offered other than those given in Section 3.7.2.3.2.3. (See SER Section 3.7.2.3.2.3, "Thin-Fibrous Beds," for the staff evaluation of that section.) The supporting Appendix E repeats the text found in Section 4.2.5, and provides tables that summarize available domestic and international head loss testing and results.

Staff Evaluation for Section 4.2.5: The staff did not identify any specific analytical refinement(s) offered in Section 4.2.5 or Appendix E. Therefore, no evaluation is provided for analytical refinement(s) to the head loss analysis.

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5.0 DESIGN AND ADMINISTRATIVE CONTROL REFINEMENTS

Industry representatives including the Nuclear Energy Institute (NEI), the Westinghouse Owners Group, and various participants from individual utilities have followed the development, research, and resolution process of GSI-191 for several years. Over this time, practical insights have been gained by the participants regarding the relative importance of each stage of the accident sequence to the overall assessment of recirculation-sump vulnerability. This section addresses the phenomenology associated with debris generation, debris transport, debris accumulation and head loss across beds of mixed composition. As the knowledge base of research data and plant survey information has improved, and as analytic methods have developed to address each aspect of the complex accident sequence, so too has the awareness of potential vulnerabilities grown. Recognition and understanding of the principle contributors to sump-screen vulnerabilities has initiated a discussion about possible mitigation strategies that seek to interdict the accident progression at one or more of the aforementioned stages.

Self assessment of recirculation sump vulnerability and the identification of site-specific contributing factors is a responsibility of each licensee, but this chapter attempts to share the broader industry perspective on possible improvements that a licensee can make to improve their sump-performance posture, regardless of their current plant condition.

Based on the findings of individual licensees, the range of mitigative actions pursued across the industry may range from status quo operation to sump-screen replacement. In many cases, though, new awareness of the issues involved with ensuring sump-screen performance will lead to at least procedural changes that help avoid unnecessary exposure to the risk of sump-screen blockage. With improved understanding of a problem comes a new perspective of common-sense regarding the simple things that can be done to improve safety as well as the detailed knowledge required to affect engineered solutions to a specific technical problem. This chapter provides insights at both levels. The discussion presented here may be sufficient for a given licensee to address any identified problems. For others it may motivate progress towards a site-specific solution of their own devising. It should also be recognized that successful management of sump-screen vulnerability may require a combination of the approaches presented in this chapter.

Given the diversity of possible responses to this issue and the variety of site-specific solutions that will be developed at varying degrees of complexity, at this time the NRC cannot endorse any one mitigation strategy that is offered here. Assessments of relative effectiveness expressed in the GR are the opinion of the industry representatives. The staff believes that this information improves the practicality of the GR because licensees are immediately motivated to find workable solutions to any problems that are identified during their vulnerability assessments. Any necessary changes to plant configuration, technical specifications, operating procedures or other licensing basis changes should still consider the need for NRC staff review and approval. Licensees should consider existing regulatory processes, and if necessary, submit any required information for staff review. An important aspect of the existing review process is the need for applicable testing and analysis of any new equipment or materials that are incorporated into the ultimate resolution strategy. In this manner, the NRC can judge the effectiveness of the approaches chosen by each licensee. For these reasons, the staff review of Chapter 5

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is limited. The staff found the technical descriptions in this chapter to be acceptable as an introduction to the topic of mitigating sump-screen vulnerabilities.

5.1 DEBRIS SOURCE TERM

Five categories for design and operational refinements are examined in this section. Staff comments on each category are itemized below.

1. Housekeeping and FME Programs: The GR recommends that if housekeeping or FME programs are implemented or revised to reduce the latent/miscellaneous debris burden then appropriate procedures should be designed to ensure a high level of performance. The staff wishes to emphasize that such procedures and performance metrics, based on swipe sample analyses, for example, should be used if vulnerability assessments rely on periodic cleaning activities to maintain debris loadings below some minimum level of concern.

2. Change-Out of Insulation: Two additional comments are offered by the staff in addition to those itemized in the GR. First, it should be noted that while change out of problematic insulation types may address the issue of maximum debris loadings on the screen, it might not address the issue of minimum loadings required to form a thin filtration bed. To satisfy both concerns, a combination of strategies in addition to change-out might be needed. Second, the large-scale removal of some insulation types may inadvertently increase the latent debris loading of residual insulation materials unless removal is performed carefully to minimize the spread of fine materials or effective plant cleaning routines are implemented after insulation removal to recover dispersed material.

3. Modify Existing Insulation: This may effectively address the issue of maximum debris loads on the screen without changing the minimum loadings required to form a thin filtration bed. To satisfy both concerns, a combination of strategies in addition to modification of existing insulation may be necessary.

4. Modify Other Equipment or Systems: The staff agrees that changes to noninsulation items should be considered in the context of the entire sump performance evaluation. Another example of beneficial change to equipment was suggested by the discussion of latent debris surveys that identify unique collections of particulate or fibrous material like filter housings that are vulnerable to water infiltration. If such sources can be sealed or protected from containment spray, then the internal inventory will not be released to the sump pool.

5. Modify or Improve Coatings Program: Under the conservative assumption that 100% of unqualified coatings will fail, the staff agrees that conversion to DBA qualified systems would reduce the source term contributed by failed coatings. Although no current approved staff guidance exists, it is possible that coatings systems that are currently unqualified could be qualified through appropriate testing. Depending on the rigors of the ASTM testing standards, some of this testing might be accomplished in place to avoid destructive sample collection from existing surfaces and equipment. Additionally, the staff does not agree with the statement that DBA-qualified coatings have very high destruction

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pressures. This statement has not been proven for the simultaneous combination of high-temperature and high-pressure jet impingement. See Sections 3.4.4 and 4.2.2.2.3 for more discussion on acceptable coatings destruction pressures.

5.2 DEBRIS TRANSPORT OBSTRUCTIONS

This section examines various options for redirecting or retarding the movement of debris towards the sump screen. The objective of these approaches is to trap or sequester debris so that it cannot reach the sump screen during recirculation. Transport velocities are highest during pool fill up when sheeting velocities can move large pieces of debris that are initially impacted on the floor near the break or washed to the floor by the break effluent. During this timeframe, flow direction is not preferentially towards the sump. As the containment pool fills, sheeting velocities decrease. With the onset of recirculation flow, debris transport with a preferential direction aligned towards the sump screen is established. Design of obstructions to provide a barrier to debris transport to the sump screens should consider all phases of pool fill and establishment of recirculation flow.

5.2.1 Floor Obstruction Design Considerations

Careful thought must be given to the stability of the holding location with respect to turbulence introduced by cascading containment spray water. For example, if diversion baffles successfully collect debris during fill up in a drainage zone that is highly agitated by falling water, the net result may be to increase the fraction of individual fibers and fine material available for transport to the screen under low recirculation velocities. During initial fill up, curbs may be subjected to significant flow velocities, so heights would need to be designed accordingly in order to be effective. Removable structures like debris rakes and baffles may also experience significant hydrodynamic force loadings during fill up. The test data cited from GR reference 54 for the effectiveness of curbs is very rudimentary. Significant opportunity exists for optimizing curb designs to accomplish the complimentary objectives of debris capture and/or debris diversion.

5.2.1.1 Test Results

During pool fill up, flow directions are dictated by the location of the break and the containment geometry. During recirculation, there is a directed flow path towards the sump screens, but perhaps at lower bulk velocities. None of the data apply to turbulence induced from direct water splashing near the curbing. It is noted that curbs could be an especially important strategy for protecting horizontal sump screens from debris build up while the sump cavity is filling. To effectively design curbing a reasonable detailed understanding of water velocity and direction is needed during the phase of transport for which the curbs are intended to be effective. The staff also notes that while curbing may be effective at impeding the migration of larger debris along the floor, curbs do not address the problem of suspended fines. Thus, the overall effectiveness of curbing and debris racks (next section) will depend on the site-specific debris types that they were designed to mitigate.

5.2.2 Debris Obstruction Rack Design Considerations

There is ample room for optimization of rack designs for trapping debris before it reaches the sump screen. One conceptual design that has been discussed involves two or more parallel racks placed across the flow path to act as weirs over which the water must flow while depositing larger debris in the spaces between racks. For this to be effective, the mesh size and height of the baffles would need to be optimized for the size of the debris and the depth of the pool in order to prevent obstruction of water flow. This design concept of interstitial capture between vertical risers might also be incorporated directly into a multilayered suction strainer where the outer layers serve initially to attract and capture debris leaving the inner layers clear to provide adequate water flow.

5.2.2.1 Test Results

The test results cited from GR reference 55 focus on tumbling and sliding of debris along the floor. During pool fill, water velocities could be much higher than the incipient velocities listed in GR Table 5-1. The use of racks may effectively manage larger debris items moving along the floor, but would not stop the migration of individual suspended fines.

5.2.2.2 Debris Rack Grating Size

In this section, the GR emphasizes several of the design considerations mentioned above in Section 5.2.2 of the SER.

5.3 SCREEN MODIFICATION

Staff Evaluation of Section 5.3: This section of the GR provides guidance regarding potential sump screen designs and features.

The relative effectiveness of curbs and debris racks depends on the characteristics of the debris that challenge the sump screen. While these design features may be effective at preventing the migration of large volumes of debris along the floor, they may not be effective at preventing transport of suspended fines. Therefore, depending on the dominant debris types at a site, licensees may determine that it may be more cost effective to modify screen configurations to manage the entire range of debris size. The GR considers the attributes of three generic design approaches that licensees might pursue. These include passive strainers, backwash strainers, and active strainers.

The staff emphasizes two performance objectives that should be addressed by a sump-screen design. First, the design should accommodate the maximum volume of debris that is predicted to arrive at the screen given full consideration of debris generation, containment transport and auxiliary mitigation systems like curbing that may be in place. Second, the design should address the possibility of thin-bed formation. When fibrous debris are expected as part of the limiting break condition, the screen should accommodate a large fraction of the expected fines (both from the ZOI and from potential pool degradation) as individual fibers with the potential to form a uniform layer. The difference between these objectives relates to the degree of uncertainty in debris transport methodology that the screen design should accommodate. While it is difficult to argue that debris will not transport (first objective), it is equally difficult to demonstrate that it will transport (second objective). Thus, both extremes should be satisfied by the screen design.

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5.3.1 Considerations for Passive Strainer Designs

The large appeal of passive strainers relates to their simplicity of maintenance and high reliability for an adequately tested design, both important considerations for safety-related equipment. While the GR accurately presents the general attributes of existing passive designs, the presentation is focused on applications of one-dimensional head-loss correlations that have traditionally lead to large strainer designs. Water velocity through the debris bed is an important factor in predicting head loss, so larger surface areas imply lower velocity for a given recirculation flow, and hence, lower head loss. The challenge with this approach is to achieve a large surface-to-volume ratio by using a convoluted screen geometry that traps debris while providing adequate recirculation flow and not taking up too much space in containment.

Given the requirement in some plants to address thin-bed formation for potentially large amounts of fine fibrous debris, large compact surface areas alone may not be sufficient. Two alternative design concepts may be effective, perhaps in combination with compact geometries that achieve large surface-to-volume ratios. Generically, these design concepts may be described as disrupting the formation of a uniform fiber layer by (a) using a complex porous filter structure to capture fiber, or (b) designing hydraulic flow paths that amplify velocity gradients across the flat surfaces of the strainer where fiber first approaches.

The first design concept can be imagined as a prefilter, made perhaps of crumpled wire cloth (~1-inch mesh) or similar material that creates a very porous volumetric filter on the face of a standard sump screen for the purpose of capturing fibers with minimal head loss. Porosity and thickness of the prefilter section would require design optimization to accommodate a specific quantity and size of suspended fiber debris. The second concept utilizes small friction losses internal to the body of a convoluted filter structure that has many fins, fingers, plates or other protuberances on which to capture debris. Small internal friction losses can be enhanced and designed to create velocity gradients across the external surfaces of the filter. If properly designed, this feature might be effective at directing the build up of fiber in a controlled way that avoids uniform simultaneous coverage of the strainer face. This might be used to efficiently pack material on an essentially sacrificial surface while leaving other flow areas unobstructed. These concepts, and other innovations, share a common need for adequate design testing, but they may offer effective solutions to the drawbacks of large passive strainers presented in the GR.

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5.3.2 Considerations for a Backwash Strainer Design

In addition to the practical considerations for a backwash strainer design offered in the GR, the NRC staff identifies the following observations. The staff agrees that backwash systems may need to undergo design testing and possible surveillance testing to demonstrate that they will work as intended.

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1. Any design that attempts to clear an existing debris blockage should give careful consideration to the problem of resuspension and redeposition of that debris. If the working fluid is applied too violently, a cloud of debris may temporarily disperse and then reform a bed on the screen. Testing may show that this is acceptable behavior that reduces the screen loading enough to be effective regardless of bed reformation.

2. It is stated implicitly in the GR that normal recirculation flow will be stopped during backflushing. This may raise concerns about restart reliability of the ECCS system. Some backflush designs might be able to operate effectively without interrupting ECCS flow. For example, a continuous water-jet curtain directed across the face of the screen might be effective at preventing debris buildup to unacceptable levels. This water flow might be provided as a side stream from the main ECCS system so that no additional pumps, actuators or valves need be qualified.

3. Debris beds, especially fiber-based mats, are effective filters of suspended particulate. If the entire debris mat is disturbed very quickly, the local concentration of material that can pass through the screen is suddenly very high. This may represent a unique challenge to downstream components that is not present during normal recirculation flow.

4. Most debris beds studied to date are held to the screen only by the pressure of the water flowing through them. They form no particular adhesive or mechanical attachment to the screen. Fibrous beds have been observed to slump or sluff off of the screen in contiguous mats. For designs where ECCS flow is interrupted, this behavior presents an opportunity for collecting or trapping the debris that loosens from the screen without dispersing it greatly. Debris racks, or bins might be designed to sequester the debris mats and minimize redeposition. Minimum-flow backflush systems in combination with inclined screens that provide gravity assist for the detachment might benefit the most from this behavior.

5. Item 5 in the GR suggests automated control systems to actuate the backwash cycle based on measurement of pressure drop or flow. For backwash systems that function intermittently upon actuation, some degree of information feedback and/or intervention might be given to operators to increase the flexibility and utility of the backflush system as a recovery alternative for potential sump blockage.

5.3.3 Considerations for an Active Strainer Design

Active strainer concepts offer much greater design flexibility for addressing the challenges of debris accumulation in PWR recirculation pools. Therefore, they offer some unique advantages over the other two generic screen designs. Several such advantages are presented as favorable technical considerations in the GR. One contradiction that the staff would point out relates to favorable technical consideration number 3, which offers the opinion that self-cleaning strainers may avoid uncertainties related to various debris generation and transport phenomenology. However, the same active strainer features that indicate success for some phenomena might also exacerbate problems for other phenomena. As an example, adhesive chemical corrosion byproducts might be smeared into a semi-impervious layer across the sump-screen mesh by a scraping device whereas the same debris might be dislodged by an optimized backflush system.

Active designs can carry a greater burden of proof for effectiveness and operability depending on their complexity, and the staff agrees with additional consideration number

1 that experimental studies would be needed to demonstrate the effectiveness of proposed active strainer designs. In general, many of the considerations for an active strainer design like power supply, control system reliability and functional reliability are similar to those presented in the GR for backwash systems.[1] Many of the staff observations are also similar. For example, active-strainers may be most effective when combined with mechanisms for debris collection and sequestration that over time reduce the local suspended debris concentration that poses a challenge to the strainer surface.

To maintain the generality of this discussion, the NRC prefers the terminology "active strainer" over the description of "self-cleaning." The GR accurately defines an active strainer as a design that incorporates active components to maintain flow to the sump, but there the generality of the presentation ends and discussions of self-cleaning mechanisms begin. Because there are no active strainer applications for either BWR suppression pools or PWR sumps, there should be no preconceptions imposed regarding typical active designs. Similarly, while continuous cleaning of the strainer surface area might be one desirable performance metric of an active design, it is not the only method of maintaining flow to the sump.

Another class of design solutions exists that periodically clean the strainer surface rather than continuously cleaning the surface. Consider, for example, a set of flat, parallel, inclined sump screens that are latched at the top corners and hinged at the bottom corners. When the outer face is loaded with debris, the latches are released and the screen swings to the floor, exposing a fresh screen for debris collection and trapping its debris inventory from further transport. Other methods may be developed using gravity assisted debris detachment on downward inclined screen surfaces. Internal flows could be alternately switched between separate chambers of the strainer to permit detachment on one side while drawing flow from the other side. Flow baffles might be switched with actuation mechanisms and control logic systems or by simple rotation of a spindle based on hydraulic flow imbalance between the chambers. The success or failure of any innovative design concept depends on how completely it can satisfy the additional considerations presented in the GR, but once the commitment has been made to facing these design challenges, no restrictions should be placed on the options available for a successful plant-specific solution.

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5.3.4 Summary

In combination with staff comments provided in this SER, the NRC finds this chapter of the GR to be a useful and acceptable introduction to the variations in sump-screen design that may be pursued for sump modification by an individual licensee. The exact definitions of the generic categories and the particular label given to an innovative design are not as important as the generic attributes that have been defined in the GR. These attributes serve as a basis for comparing the technical challenges and benefits, and the potential programmatic costs of alternative design solutions. Any consideration of screen modifications should be made in the context of the comprehensive site-specific vulnerability assessment. Alternative combinations of source mitigation, design changes, and administrative control should be weighed against existing debris types, containment geometry constraints, and NPSH margins.

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[1] In fact, after correcting a typographical error near the end, item 6 should read, "Margin must be available to initiate *active strainer mode* before sump blockage affects either ECC or CS operation."

6.0 ALTERNATE EVALUATION

6.1 BACKGROUND AND OVERVIEW

Section 6 of the GR describes an alternate evaluation methodology for demonstrating acceptable containment sump performance. The alternate evaluation methodology described in this section is shown as Option B in Figure 2-1 of the GR.

For the last several years, the NRC has recognized that probabilistic risk assessment (PRA) has evolved to the point that it can be used increasingly as a tool in regulatory decision making. Through its policy statement on PRA (ADAMS Accession number ML021980535), the Commission expressed its expectation that enhanced use of PRAs will improve the regulatory process in three ways: through safety decision making enhanced by the use of PRA insights; through more efficient use of agency resources; and through a reduction in unnecessary burdens on the licensees.

The NRC staff has considered the development of risk-informed approaches to the technical requirements specified in 10 CFR 50.46, and these considerations are documented in numerous communications between the Commission and the staff (SECY and Staff Requirements Memorandums (SRM)). The NRC Commissioners, in their March 31, 2003 SRM, directed the staff to undertake several rulemakings, one of which would develop a proposed rule to allow, as a voluntary alternative, a redefinition of the design basis LOCA break size. In a March 4, 2004 letter to NEI (SB, 2004), the staff stated that it would discuss, in public meetings, the use of current or planned work to risk-inform 10 CFR 50.46 as a suitable technical basis for defining a spectrum of break sizes for debris generation and containment sump strainer performance.

Specific to GSI-191, the Commission recently requested the staff to, "implement an aggressive, realistic plan to achieve resolution and implementation of actions related to PWR ECCS sump concerns." One such resolution path involves the LOCA break size used in PWR sump analyses. For example, it is well understood that the amount of debris generation to be expected following a LOCA is dependent on the break size, and generally that less debris would be generated with a smaller LOCA break size (although less debris generation may be worse in certain situations when considering debris type and break location). The staff is already working to risk-inform 10 CFR 50.46 to redefine the design-basis large break LOCA break size based on expected LOCA frequencies. A comparable approach for use in GSI-191 resolution would identify a "debris generation" break size which would be used to distinguish between customary and realistic design basis analyses. However, it is very important to note that an alternative approach for resolving GSI-191 would not redefine the design basis LOCA break size in advance of the 10 CFR 50.46 rulemaking effort. In developing an alternate approach for resolving GSI-191, the staff intends to remain at least as conservative as, and consistent with any forthcoming revision to 10 CFR 50.46.

On May 25, June 17 and June 29, 2004, the staff met with NEI, industry representatives and stakeholders, in category 2 meetings, to discuss alternate, realistic and risk-informed approaches for resolution of the PWR sump issue. Throughout these meetings, both NRC and NEI staff presented proposals and positions regarding technical and regulatory elements of alternative approaches.

These interactions between the staff, NEI, industry representatives and stakeholders yielded an alternative approach which includes both realistic and risk-informed elements. For such an approach, licensees would continue to perform design basis long-term cooling evaluations and satisfy design basis criteria for all LOCA break sizes up to a new "debris generation" break size that would be smaller than a double ended guillotine break (DEGB) of the largest pipe in the reactor coolant system (RCS). This analysis space is referred to as Region I in the GR. Long-term cooling must be assured for breaks between the new "debris generation" break size and the double-ended rupture of the largest pipe in the RCS, but the evaluation may be more realistic than a customary design basis evaluation, consistent with the small likelihood of the break occurring. For breaks larger than the "debris generation" break size, licensees could apply more realistic models and assumptions. This analysis space is referred to as Region II in the GR. Additionally, any physical modifications to plant equipment, or operator actions credited to demonstrate mitigative capability for these larger breaks (Region II) would not necessarily need to be safety-related or single-failure-proof. Changes to the existing facility designs, and credit for operator actions would include risk evaluations consistent with Regulatory Guide 1.174. Licensees should ensure that the changes to the facility design would have sufficient reliability to provide reasonable assurance that they will perform their intended function.

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While not a component of the 10 CFR 50.46 ECCS evaluation model, the calculation of sump performance is necessary to determine if the sump and the residual heat removal system are configured properly to provide enough flow to ensure long-term cooling, which is an acceptance criterion of 10 CFR 50.46. Therefore, the staff considers the modeling of sump performance as the validation of assumptions made in the ECCS evaluation model. Since the modeling of sump performance is a boundary calculation for the ECCS evaluation model, and acceptable sump performance is necessary for demonstrating long-term core cooling capability (10 CFR 50.46 (b)(5)), the requirements of 10 CFR 50.46 are applicable. Based on this, such an alternative approach might require plant-specific license amendment requests or exemption requests from the regulations, depending on each licensee's chosen resolution approach. Licensees could request, on a plant-specific basis, exemptions from the requirements associated with demonstrating long-term core cooling capability (10 CFR 50.46 (b)(5)). For example, exemptions from the requirements of 10 CFR 50.46 (d) may be required if a licensee chose to classify new equipment as non-safety related or non-single failure proof. For purposes of GSI-191 resolution, exemption requests would not be applicable to the other acceptance criteria of 10 CFR 50.46 (peak cladding temperature, maximum cladding oxidation, maximum hydrogen generation and coolable geometry), and would be submitted in accordance with existing NRC regulations (10 CFR 50.12). Additionally, license amendment requests may be needed for changes in analytical methodology or assumptions. Licensees would assess the need for license amendment requests in accordance with the requirements of 10 CFR 50.59.

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NRC staff review and acceptance of such plant-specific license amendment or exemption requests would consider the following elements:

- Application of the principles of Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis," (e.g., defense-in-depth, safety margins, delta Core Damage Frequency, delta Large Early Release Fraction).

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- Consistency with NUREG-0800 (Standard Review Plan), Section 19, "Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance."
- Design-Basis, deterministic analyses necessary to verify compliance with 10 CFR 50.46 (b)(5) for break sizes up through "debris generation" break size.
- Acceptable mitigative capability up through the DEGB of the largest pipe in the RCS. The equipment needed for mitigative capability would have some functional reliability requirements, but would not necessarily need to be safety-related or single failure proof.

One key element of Regulatory Guide 1.174 involves assurance that defense-in-depth is maintained. Although a "debris generation" break size is selected to distinguish between customary and more realistic design basis analyses, the staff would require that licensees demonstrate acceptable mitigative capability for LOCA break sizes up through the DEGB of the largest pipe in the RCS. This philosophy is consistent with 10 CFR 50.46 (b)(5) and recent recommendations made by the Advisory Committee on Reactor Safeguards (ACRS) in their April 27, 2004, letter to the Chairman. Requiring that mitigative capability be maintained in a realistic and risk-informed evaluation of the PWR sump issue for all LOCA break sizes up through a DEGB of the largest RCS piping ensures that defense-in-depth is maintained.

6.2 ALTERNATE BREAK SIZE

The alternate break size to be applied for alternate evaluation of sump performance is defined in the GR methodology as follows:

- A complete guillotine break of the largest line connected to the reactor coolant system loop piping.
- For main loop piping, a break size will be assumed to be that equivalent to a guillotine break of a 14-inch schedule 160 line. This equates to an effective break area of 196.6 square inches (assuming both sides of the break are pressurized).

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In defining these break sizes, the alternate break size to be considered by each licensee for lines connected to the main loop piping is plant dependent, while the alternate break size to be applied to the main loop piping is identical for each licensee.

The GR also provides guidance for determining whether a double ended guillotine break needs to be considered in attached piping. If sufficient energy for debris generation exists on both sides of the break, a DEGB will be used. The GR criteria for determining whether sufficient energy exists are based on the postulated break distance from a normally closed isolation valve, and are as follows:

- 10 pipe inside diameters for large bore piping (i.e., greater than 2 inch diameter)
- 20 pipe diameters for small bore piping

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If a normally closed isolation valve exists within this number of pipe diameters, than only a single ended break needs to be considered. These GR criteria are based on the low stored energy in the pipe section between the break and isolation valve with respect to significant debris generation.

Additionally, the GR provides guidance for consideration of the ongoing 10 CFR 50.46 rulemaking effort. The GR states that, "In using this GSI-191 alternate break size, it is recognized that when the 50.46 rule is finalized, licensees can re-perform the sump performance evaluations with the final break size specified in 50.46 and modify the plant design and operation. This would assure coherence in the implementation of 50.46."

Staff Evaluation for Section 6.2: The staff has reviewed the alternate break size proposals as described in the GR and finds them to be acceptable. The staff refers to the alternate break size as the "debris generation" break size (DGBS) and will do so throughout the following discussion.

The DGBS to distinguish between customary and more realistic design basis analyses is as follows:

1. All American Society of Mechanical Engineers (ASME) Code Class 1 PWR auxiliary piping (attached to RCS main loop piping) up to and including a double-ended guillotine break of any of these lines - design basis rules apply
2. RCS main loop piping (hot, cold and crossover piping) up to a size equivalent to the area of a DEGB of a 14 inch schedule 160 pipe (approximately 196.6 square inches) - design basis rules apply
3. Breaks in the RCS main loop piping (hot, cold and crossover piping) greater than the above size (approximately 196.6 square inches), and up to the DEGB - licensees must demonstrate mitigative capability, but design basis rules may not necessarily apply.

The technical basis for the staff's acceptance of the division of the pipe break spectrum for the purpose of evaluating debris generation is comprised of several factors. First, the staff considered recent information developed by the NRC's Office of Nuclear Regulatory Research (RES) regarding the frequency of RCS ruptures of various sizes. This information was developed by RES through an expert elicitation process as documented in SECY-04-0060, "Loss-of-Coolant Accident Break Frequencies for the Option III Risk-Informed Reevaluation of 10 CFR 50.46, Appendix K to 10 CFR Part 50, and General Design Criteria (GDC) 35." The RES study determined the frequency of primary pressure boundary failures under normal operational loading and transients. Although the results of the expert elicitation are not yet final, the preliminary results support the observation that the probability of a PWR primary piping system rupture is generally very low and that the break frequency decreases with increasing piping diameter. The selection of a break size equivalent to the area of a DEGB of a 14 inch schedule 160 pipe for RCS main loop piping is consistent with attached auxiliary piping sizes in PWRs, and is also consistent with the ongoing 10 CFR 50.46 rulemaking direction (at this time).

The staff also considered the fact that there is a substantial difference from a deterministic, "margins to failure" or "flaw tolerance" perspective between 30-to-42 inch diameter PWR main coolant loop piping and the next largest ASME Code Class 1 attached auxiliary piping (generally the 12-to-14 inch diameter pressurizer surge line). This difference is evident, for example, in Leak-Before-Break (LBB) evaluations conducted in accordance with NUREG-1061, Volume 3, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee," wherein main coolant loop piping characteristically passes a LBB evaluation more easily than ASME Code Class 1 auxiliary piping systems. Finally, the staff considered the fact that certain ASME Code Class 1 auxiliary piping systems may be more susceptible to failure due to environmental conditions which are conducive to known degradation mechanisms and/or loading conditions which routinely apply significant stresses to the piping system. An example of both of these considerations would be a typical PWR pressurizer surge line in which Alloy 82/182 dissimilar metal welds are subjected to a high temperature operating environment known to abet primary water stress corrosion cracking and which is subjected to significant bending loads during startup/shutdown conditions due to the large temperature gradient between the pressurizer and the hot leg of the main coolant loop.

Based upon the considerations noted above, the staff considers that the division of the pipe break spectrum proposed for the purpose of evaluating debris generation to be acceptable based on operating experience, application of sound engineering judgment, and consideration of risk-informed principles. Licensees using the methods described in Section 6 can apply the defined DGBS for distinguishing between Region I and Region II analyses.

The staff has considered the GR guidance provided regarding the need to consider a DEGB in attached auxiliary piping. The GR provides criteria based on number of pipe diameters, pipe size and distance to a normally closed isolation valve for determining if sufficient energy for debris generation exists on both sides of the break. If a normally closed isolation valve exists within a specified number of pipe diameters from a postulated break location, than only a single ended break needs to be considered. The GR does not provide a technical basis for this criterion. To assess the acceptability of this proposal, the staff considered the fluid volumes available on each side of a DEGB which would fall within the criteria provided in the guidance. Considering that a break occurs at the maximum distance from a normally closed isolation valve, as allowed by the proposed criteria, the staff agrees that there would be an insignificant amount of energy available for destruction from the isolated side of the break when compared to the fluid volume and energy available on the unisolated side of the break. For example, considering a DEGB of a 1 foot diameter auxiliary pipe with a normally closed isolation valve 10 inside pipe diameters away, the fluid volume in the isolated piping portion is less than 10 cubic feet. This fluid volume is insignificant when compared to the RCS fluid volume, which is on the order of 10,000 cubic feet. The fluid and energy blowdown from the isolated side of the break will depressurize and void almost instantaneously, while the blowdown from the RCS side of the break would be significantly larger, on the order of minutes (the staff verified this through a simplified RELAP calculation). Based on this, and considering engineering judgment, the staff finds that the criteria proposed by NEI for evaluating whether a DEGB should be considered in auxiliary piping is acceptable. The staff's engineering judgment takes into consideration that (a) past experiments and analyses have confirmed that debris generation due to initial blast impulse (which would be from both sides of the postulated break) would be minimal, and (b) that debris generation is dominated by jet loading and/or jet erosion. As confirmed by the staff's estimate, blowdown jet impacts would be dominated by the blowdown from the RCS side of the break.

The staff also considered the GR guidance regarding consideration of the ongoing 10 CFR 50.46 rulemaking effort. The staff agrees with the recommended guidance that licensees may re-perform the sump performance evaluations using the final break size specified by rulemaking, and modify the plant design and operation accordingly. This would assure consistency with a new 10 CFR 50.46. The staff expects that the DGBS specified in this section will bound the transition break size specified by a new 10 CFR 50.46.

6.3 REGION I ANALYSIS

The Region I analysis of recirculation sump performance includes evaluation of all break sizes up to and including the DGBS defined in Section 6.2. The majority of the analyses to be performed for the Region I break sizes are to be performed in the same manner as described in Sections 3, 4 and 5 of the GR. For Region I breaks, the GR states that a full range of break locations will be assessed to determine the limiting location considering both debris generation and debris transport. However, as discussed in Section 6.3.2, the GR refers to a Section 4 refinement proposing that Branch Technical Position MEB 3-1 (MEB 3-1) may be used to limit the break locations considered. Additionally, any design basis secondary side breaks (main steam line break, feedwater line break, etc.) which rely on sump recirculation will be analyzed in accordance with the Region I analyses.

With respect to break configuration, circumferential breaks will be assumed to result in pipe severance and separation amounting to at least one diameter lateral displacement of the ruptured piping sections unless physically limited by piping restraints and supports, or other plant structural members that can be shown through analysis to limit pipe movement to less than one diameter lateral displacement. For pipes with a larger diameter than the maximum break size, the maximum attainable break area would be modeled as a partial pipe break with an area equivalent to the DEGB of a pipe with the same diameter as the DGBS. The worst location of the break in terms of orientation around the break location should be considered.

One area where the Section 6.3 guidance differs from the guidance in the baseline analysis of section 3 involves the zone of influence (ZOI) to be considered for debris generation. The guidance in Section 3 regarding the ZOI presumes a DEGB, and for a DEGB, a spherical ZOI is conservatively postulated. A spherical ZOI is appropriate in the Region I analyses for any auxiliary piping attached to the RCS, since a DEGB of any such piping falls within Region I analysis. However, partial breaks of the RCS main loop piping are also included in Region I (breaks up to the DGBS), and would indicate a limited-displacement circumferential break or a longitudinal break, i.e., "split break." The GR proposes that the ZOI for such partial breaks in RCS main loop piping be accounted for by applying one of two methods:

- ZOI Based on a Hemisphere - The ZOI is simulated as a hemisphere radius determined by the destruction pressure of the insulation that would be affected by the postulated break. The break orientation needs to be simulated at various angles around the loop piping to determine maximum debris generation.

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- ZOI Based on a Sphere - Because the worst-case break orientation can be difficult to determine, an alternative to assuming a hemispherical ZOI is to translate the hemispherical volume into an equivalent volume sphere.

The GR also states that the ZOI refinements discussed in Section 4 are available when performing Region I analyses.

The acceptance criteria for containment sump screen performance continues to be core cooling based on available NPSH equal to, or greater than, the required NPSH for all pumps required to operate for long term core cooling. The calculations of required and available NPSH are based on the models and assumptions currently used in design basis analyses of sump and core cooling recirculation performance. Additionally, the GR states that if containment spray is credited in the design basis analyses, the containment sump screen performance also includes NPSH margin for the minimum required containment spray.

The Region I analyses also consider the impact of the DGBS on event timings, thermal-hydraulic conditions and NPSH requirements. For example, use of the DGBS will affect key scenario events such as the timing of transfer from RWST injection to recirculation mode, the containment sump water properties (e.g., temperature), and containment back-pressure (if credited in the design basis analyses). The Region I evaluation will consider these revised timings and parameters as appropriate. The guidance also provides for the impact of operator actions to mitigate containment sump blockage, provided that the operator actions meet the criterion for consideration in design basis analyses. These considerations would include adequate time for operator action per design basis "rules", proceduralized guidance, job-task-analysis, training and other requirements.

__Staff Evaluation for Section 6.3:__ The staff has reviewed the Region I alternate evaluation methodology as described in the GR. The Region I analysis methods described in Section 6.3 are applicable for any break sizes equal to or smaller than the DGBS defined in Section 6.2. The Region I methodology therefore, applies to any ASME Code Class 1 auxiliary piping (attached to RCS main loop piping) up to and including a double-ended guillotine break of any of these lines, and RCS main loop piping (hot, cold and crossover piping) up to and including a size equivalent to the area of a DEGB of a 14 inch schedule 160 pipe. The majority of the Region I analyses are performed in the same manner as the methods described in Sections 3, 4 and 5 of the GR, and as such, those corresponding SER sections are applicable for Region I analysis. For example, the guidance in Sections 3 and 4 is to be used as part of the Region I analyses to determine the debris generation, transport and accumulation on the containment sump screens. The staff evaluation described here will focus on differences from Sections 3, 4 and 5 of the GR.

For Region I breaks, the GR states that a full range of break locations will be assessed to determine the limiting location considering both debris generation and debris transport. Additionally, as discussed in Section 6.3.2, the GR refers to a Section 4.2.1 refinement which proposes that Branch Technical Position MEB 3-1 may be used to limit the break locations considered. As documented in Section 4.2.1 of this SER, the staff concluded that it is inappropriate to cite SRP Section 3.6.2 and Branch Technical Position MEB 3-1 as methodology to be applied for determining break locations to be considered for PWR sump analyses. The staff concludes that for Region I breaks, which are considered as customary design basis analyses, a full range of break locations should be assessed to determine the limiting location considering both debris generation and debris transport. Section 4.2.1 of this SER provides further details regarding the staff's position.

The staff finds that the GR guidance is acceptable with respect to break configuration because the methodology assures that the limiting break location considering debris generation, debris transport and the worst location of the break in terms of orientation around the break location will be evaluated. This methodology provides reasonable assurance that the limiting break conditions for PWR sump analyses will be evaluated. Additionally, considering piping restraints and supports or other plant structural members that can be shown through analysis to limit pipe movement to less than one diameter lateral displacement may be acceptable to the staff; however, because the limiting break location and orientation must be evaluated, these locations may not produce the limiting conditions for sump analyses.

Regarding the ZOI to be considered for attached auxiliary piping breaks, the GR states that a spherical ZOI is postulated for breaks smaller than the DGBS for piping connected to the RCS main loop piping because a DEGB of this piping is postulated. For Region I partial pipe breaks, the GR proposes that one of two methods be applied, either a ZOI based on a hemisphere, or a ZOI based on translating the hemispherical volume into an equivalent volume sphere. The staff evaluated the GR with respect to the ZOI to be considered under these conditions and concludes that applying a hemispherical ZOI is acceptable for such partial breaks, and that when doing so, licensees would need to simulate various directions around the RCS main loop piping to determine the limiting break location. The staff does not accept the proposed approach of a ZOI based on translating the hemispherical volume into an equivalent volume sphere. The GR does not provide any technical justification for this approach except that it is a simplification because the worst-case break orientation can be difficult to determine. The staff does not have a technical basis for accepting a translation of the volumes, which would result in a different ZOI, and the staff has no basis to evaluate whether this would be conservative, nonconservative, or realistic. For simplification, the staff would accept application of a spherical ZOI with a radius equivalent to that of a ZOI based on a hemisphere.

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The application the ZOI refinements for Region I analyses should be in accordance with the staff's position as discussed in Section 4 of this SER.

For the Region I sump analyses, the acceptance criteria for containment sump screen performance continues to be core cooling based on available NPSH equal to, or greater than, the required NPSH for all pumps required to operate for long term core cooling. The calculations of required and available NPSH are based on the models and assumptions currently used in design basis analyses of sump and core cooling recirculation performance, and therefore, the staff finds their continued application for Region I analyses to be acceptable. The staff agrees with the GR that the impact of the DGBS on event timings, thermal-hydraulic conditions and NPSH requirements, and crediting of operator actions for demonstrating that the acceptance criteria are satisfied, can be applied for Region I analyses consistent with customary design basis analysis procedures and requirements. Licensee analyses should consider, at a minimum, the following factors:

1. The accuracy of deterministic analyses performed to calculate DGBS event timings, T/H conditions and NPSH requirements, and their compliance with 10 CFR 50.46. Staff expects that licensees will document, and if necessary, provide to the staff detailed information regarding the analyses and the modeling assumptions. The GR guidance does not explicitly identify which phenomena and parameters will receive time dependent treatment and will be considered in-scope for estimating timing of events.

2. The experimental data used for estimating debris generation, transport and head loss buildup for breaks other than DEGB. In general most of the experimental data was obtained for jet conditions and transport flow rates prototypical of DEGB. For example, most of the debris generation data was obtained for jet durations typical of DEGB (10-30 seconds). Direct use of such data for insulations where erosion is the dominant generation mechanism (e.g., calcium-silicate) may not be appropriate for DGBS breaks. Similar limitations on the applicability of available experimental data to DGBS exist for other phenomena as well, including debris transport and debris buildup – especially when operator actions are to be credited in the mix of the analyses being performed. However, application of Section 3.0 baseline methods ensures conservative treatment of erosion concerns for tabulated materials.

3. Also, due to uncertainties in various phenomena, the staff believes that it is difficult to judge when maximum head loss would occur (e.g., maximum debris accumulation and the minimum NPSH margin may or may not occur simultaneously depending on operator actions). Considerable attention and a broad of spectrum of analyses should be devoted to establish that analyses are customary design basis analyses.

4. If credit is to be taken for containment overpressure, underlying analyses should conform with staff guidance for estimating minimum overpressure as suggested in Regulatory Guide 1.82, Revision 3.

The staff notes that there is a typographical error in the following sentence of Section 6.3.6 of the GR, "In addition, if containment spray is credited in the design basis analyses (containment pressure, radiological consequence, etc.), the containment sump screen performance also includes NPSH margin for operation of the minimum required containment spray." The staff believes that this sentence should state that adequate NPSH margin needs to be available for the maximum required containment spray, or to allow for an overestimate of the required containment spray.

6.4 REGION II ANALYSIS

The Region II analysis of recirculation sump performance includes evaluations of break sizes in the RCS main loop piping (hot, cold and crossover piping) greater than the DGBS specified in Section 6.2 (approximately 196.6 square inches) and up to a DEGB of the largest pipe in the RCS. Only RCS main loop piping is considered in Region II because all primary side attached auxiliary piping and secondary side breaks are fully addressed as part of the Region I analyses. Section 6.4.2 of the GR states that, "[I]f a licensee chooses to use an alternate break size smaller than the largest connected piping to the main coolant loop piping, as discussed in Section 6.2, then connected piping larger than the alternate break size would be addressed as part of the Region II evaluation." The staff finds that this statement is not consistent with the alternate break size as defined in Section 6.2 and should be clarified. NEI and industry representatives informed the staff that this statement is included in the GR to allow for the possibility that the forthcoming 10 CFR 50.46 rulemaking would redefine the design basis LOCA break size to be smaller than the DGBS defined in Section 6.2. As discussed in Section 6.2 of this SER, the staff agrees with the recommended guidance that licensees may re-perform the sump performance evaluations using the final break size specified by rulemaking, and modify the plant design and operation accordingly.

Section 6.4.2 of the GR refers to a Section 4 refinement proposing that Branch Technical Position MEB 3-1 (MEB 3-1) may be used to limit the break locations considered. With respect to break configuration, the Region II analyses are limited to DEGB of the RCS main loop piping. These circumferential breaks are assumed to result in pipe severance and separation amounting to at least one diameter lateral displacement of the ruptured piping sections unless physically limited by piping restraints and supports, other plant structural members, or piping stiffness as may be demonstrated by analysis. The GR states that existing plant-specific dynamic loads analyses for postulated primary side breaks are utilized to assist the determination of the break configuration for Region II analyses.

The ZOI models and assumptions to be applied for Region II analyses are those as described in Sections 3 and 4 of the GR. There are a number of known conservatisms in the ZOI model presented in Sections 3 and 4. However, because development of a technically sound model to more realistically model the ZOI based on existing experimental and analytical data is quite complex and has not been initiated, the GR relies on the models described in Sections 3 and 4.

The guidance in Sections 3 and 4 of the GR is also applied to determine the debris generation, transport and accumulation on the containment sump screens for Region II evaluations. The models presented in Sections 3 and 4 are considered to be bounding models to assure that the debris generation, transport and accumulation are not under-predicted. There are known conservatisms in each portion of these evaluation models in Sections 3 and 4. However, development of more realistic models in these areas is difficult due to the limited amount of experimental and analytical information available, and this work has not yet been initiated.

The acceptance criteria for containment sump screen performance for Region II analyses are continued core and containment cooling. The applicable criteria to demonstrate retained mitigation capability for long-term cooling capability in Region II analyses are:

- Positive NPSH margin is maintained for the minimum number of ECCS pumps necessary to demonstrate adequate core cooling flow, and
- Demonstration of adequate containment cooling capability to provide assurance that the containment boundary remains intact.

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The first criterion (Positive NPSH margin is maintained for the minimum number of ECCS pumps) can be met by ensuring NPSH margin is maintained for one or more moderate to high-capacity ECCS injection pumps. Additionally, for Region II analyses, the GR states that limited operation without NPSH margin is acceptable if it can be shown that the pumps can reasonably be expected to survive during the time period of inadequate available NPSH. Suggested technical justification for this would include vendor information in the form of test data or engineering judgment derived from tests and/or operational events.

The GR states that the second criterion (Demonstration of adequate containment cooling capability) can be met through credit taken for minimal heat removal pathways, including containment fan coolers, permitted by emergency procedures. Additionally, subatmospheric containment plants would not have to demonstrate that the containment remains below atmospheric pressure for the duration of the accident, if permitted by emergency procedures. The GR also states that, "exceeding nominal transient containment design pressure/temperature and environmental qualification (EQ) envelopes is allowed for Region II analysis, if reasonable assurance is provided that containment pressure boundary failure or vital equipment failure would not be expected."

The Region II analyses also consider more realistic modeling of debris generation, transport and accumulation on sump screens based on the timing of debris generation, and transport and accumulation in relation to the timing of the available and required NPSH. More realistic modeling of these items considers:

- debris generation, transport and accumulation is time dependent,
- available NPSH is time dependent, and
- the maximum debris accumulation and the minimum required NPSH may not occur simultaneously.

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The GR also allows credit for operator actions and the operation of non-safety equipment.

<u>Staff Evaluation for Section 6.4:</u> The staff has reviewed the Region II alternate evaluation methodology as described in the GR. The Region II analysis methods described in Section 6.4 are applicable for any breaks in the RCS main loop piping (hot, cold and crossover piping) greater than the DGBS specified in Section 6.2 (approximately 196.6 square inches) and up to a DEGB of the largest pipe in the RCS.

For Region II break locations, Section 6.3.2 of the GR refers to a Section 4.2.1 refinement proposing that Branch Technical Position MEB 3-1 be used to limit the break locations considered. As documented in Section 4.2.1 of this SER, the staff concludes that it is inappropriate to cite SRP Section 3.6.2 and Branch Technical Position MEB 3-1 as methodology to be applied for determining break locations to be considered for PWR sump analyses. The staff concludes that for Region II breaks, a full range of break locations should be assessed to determine the limiting location considering both debris generation and debris transport. Section 4.2.1 of this SER provides further details regarding the staff's position.

The staff finds that the GR guidance is acceptable with respect to break configuration because the limiting break location considering debris generation, debris transport and resulting sump screen head loss will be evaluated. This methodology provides reasonable assurance that the limiting break conditions for PWR sump analyses will be evaluated. Additionally, considering piping restraints and supports or other plant structural members that can be shown through analysis to limit pipe movement to less than one diameter lateral displacement may be acceptable to the staff; however, because the limiting break location must be evaluated, these locations may not produce the limiting conditions for sump analyses.

Certain portions of the Region II analyses are performed in the same manner as the methods described in Sections 3 and 4 of the GR, and as such, those corresponding SER sections are applicable for Region II analyses. The guidance in Sections 3 and 4 is to be used as part of the Region II analyses with respect to ZOI models and assumptions, and for determining debris generation, transport and accumulation on the containment sump screens. There are known conservatisms in each of these models as described in Sections 3 and 4, and as such, the staff finds them to be acceptable for Region II analyses. Sections 3 and 4 of this SER provide further details regarding the staff's position and review of these models.

The GR proposed two acceptance criteria for the Region II analysis. These are:

- Positive NPSH margin is maintained for the minimum number of ECCS pumps necessary to demonstrate adequate core cooling flow.
- Demonstration of adequate containment cooling capability to provide assurance that the containment boundary remains intact.

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The staff considers positive NPSH margin to mean that the available NPSH is greater than the required NPSH for each pump. The GR has not specified the amount of NPSH margin necessary. Since the staff has previously accepted the available NPSH equal to the required NPSH, that is, an NPSH margin of zero, this nonspecificity is acceptable for realistic and risk-informed Region II analyses. The determination of both the available and the required NPSH is addressed in Sections 6.4.7.1 and 6.4.7.2, respectively, of this safety evaluation report.

The GR does not specify what is meant by adequate core cooling. The staff interprets adequate core cooling to mean that the acceptance criteria of 10 CFR 50.46 are satisfied. By maintaining positive NPSH margin to demonstrate adequate core cooling flow, the 10 CFR 50.46 acceptance criteria should not be challenged.

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The GR does not specify what is meant by adequate containment cooling. The staff interprets adequate containment cooling to mean that the containment is in a safe and stable state and preventing risk-significant fission product releases. This will be further taken to mean that the containment has not failed structurally. The GR states that containment design pressure and the containment design temperature may be exceeded for analyses of breaks above the DGBS. The staff will consider this, and licensees should determine, on a plant specific basis, whether exemption and/or license amendment requests are required if the containment design pressure and/or temperature is exceeded. Licensees should determine whether the containment leakage rate exceeds the value of L_a defined in 10 CFR Part 50, Appendix J and given in the plant's technical specifications. An exemption to this regulation and/or a license amendment request might be required if a licensee determines that this is the case. The staff will evaluate these requests on a plant specific basis.

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The GR states that the second criterion can be met through credit taken for minimal heat removal pathways, including containment fan coolers, permitted by the emergency procedures. The staff finds that credit taken for minimal heat removal pathways permitted by the emergency procedures would be acceptable in a realistic and risk-informed Region II analysis. The staff expects that licensees will provide detailed information regarding plant equipment and/or operator actions credited in their Generic Letter responses. The staff will assess credit taken for minimal heat removal pathways as part of the Generic Letter response reviews and closeout process.

The GR also states that it is acceptable to exceed the "nominal" EQ envelopes. The staff finds that applying a more realistic EQ envelope could be acceptable in a realistic and risk-informed Region II analysis. For Region II analyses, the staff does not consider it necessary to comply with the guidance of NUREG-0588, Revision 1, which is the basis for the EQ analyses described in plant Updated Final Safety Analysis Reports (UFSARs). If any equipment exceeds the appropriate EQ envelope, the licensee should consider whether an exemption to 10 CFR 50.49 is required. The staff expects that licensees will provide detailed information with respect to exceeding nominal EQ profiles in their Generic Letter responses. The staff will assess the application of EQ envelopes as part of the Generic Letter response reviews and closeout process.

For the Region II evaluation, the GR criteria would allow limited ECCS and containment heat removal pump operation without NPSH margin. Licensees would need to demonstrate that the pumps can reasonably be expected to survive during the time of inadequate available NPSH margin. Technical justification for this conclusion should be based on test data or engineering judgment derived from tests and/or operating experience.

The GR points out that the guidance for determining adequate NPSH margin is currently provided in Regulatory Guide 1.1 (RG 1.1), which is the licensing basis for some operating reactors, and Regulatory Guide 1.82, Revision 3 (RG 1.82-3), which contains the current staff guidance. The GR suggests that it is not necessary to apply the conservative guidance provided in these Regulatory Guides when analyzing the consequences of breaks larger than the DGBS. The remainder of Section 6.4.7 provides guidance on an alternate, more realistic approach.

Section 6.4.7 discusses the application of Generic Letter (GL) 91-18 (GL 91-18) with respect to determination of realistic NPSH margin. The GR considers that a "nominal" parameter value used in performing Region II analyses could be exceeded. For this situation, the GR proposes that operability assessments in accordance with GL 91-18 are not necessary. The GR establishes a time limit allowing the nominal value to be exceeded for a period of 30 days. LOCA analyses are typically carried out only to 30 days. The staff finds this proposal to be unacceptable because the Region II analyses remain within the design bases. Exceeding the nominal value of a parameter used the Region II analyses may result in decreasing the available NPSH to the degree that there is no longer positive margin for this design basis accident. Therefore, the staff concludes that the same conditions apply as would apply for a Region I analysis and the guidance in GL 91-18 should apply.

The GR discusses the realistic assumptions that may be applied in calculating the available NPSH for breaks larger than the DGBS. These are discussed in Section

6.4.7.1 for each of the factors which contribute to the available NPSH: suction elevation head, absolute pressure head, vapor pressure head, and friction and form head losses. The staff finds the GR discussion for Section 6.4.7.1 to be acceptable with one caveat. The discussion of friction losses notes that experience has shown that calculations of friction loss based on handbook values tend to overestimate the friction loss. The GR states that these values may be reduced based on engineering judgment or test results. To quantify the available margin in these calculations, the staff's position is that a more substantive basis than engineering judgment should be used. Engineering judgment by itself without further technical basis does not provide adequate justification for removing conservatism in handbook friction loss values. The staff will accept a reduction in head loss calculations based on accepted handbook values only if its basis is technically justified.

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The required NPSH of a pump is measured by the pump vendor in accordance with applicable standards. It is usually based on a 3% drop in the pump total head (first stage for a multi-stage pump). This value has been selected as an easily recognized level of cavitation. It is not the level at which cavitation first appears. The GR states that, since total head is not necessarily a critical parameter for a centrifugal pump in the LOCA recirculation mode, the pump vendor may be able to provide relief in the amount of NPSH required to avoid pump damage rather than depend on the formal definition of required NPSH. The staff agrees. The staff has in the past accepted the pump vendor's technical judgment on pump capabilities. In this case, the conditions the pump will experience and the time period that the pump will experience these conditions should be well defined and evaluated by the pump vendor. In addition, staff believes that vendor's technical judgment should take into consideration the fact that recirculation water may include debris of different kinds and sizes (i.e., combined effects of debris ingestion and cavitation should be factored into decision making).

The GR states that accounting for the decrease in required NPSH with an increase in pumped liquid temperature as discussed in ANSI/HI 1.1-1.5-1994 (ANSI/HI 1.1-1.5) should not be used. The staff agrees. This is consistent with the guidance in Regulatory Guide 1.82, Revision 3.

The Calculational Method Section (Section 6.4.7.3) of the GR discusses assumptions that could be applied for more realistic available and required NPSH calculations. It is not clear what is meant by calculating required NPSH since required NPSH is typically measured and specified by the pump vendor. Licensees referencing the GR should clarify this. One of the items listed in this section states, "...Containment pressure head based on absolute pressure rather than vapor pressure." Rather than "absolute pressure," the term "pressure of the containment atmosphere," would be clearer. The staff expects that licensees will provide detailed information regarding the application of more realistic analysis assumptions in their Generic Letter responses. The staff will assess these assumptions as part of the Generic Letter response reviews and closeout process. Additionally, application of certain assumptions may require plant-specific exemptions and/or license amendment requests.

With respect to timing of events, the GR discusses the realistic modeling of debris generation, transport and accumulation on sump screens. One bullet in this section states that, "...the maximum debris accumulation and the minimum required NPSH may not occur simultaneously." It appears that this is referring to minimum available NPSH margin rather than minimum required NPSH. Other than this editorial comment, the staff agrees with the report's proposals in this section. The staff expects that licensees will provide detailed information regarding more realistic modeling of event timing in their Generic Letter responses. The staff will assess this modeling as part of the Generic Letter response reviews and closeout process.

The staff agrees with the GR's proposal of operator actions that may be credited to compensate for the effects of debris generation on the ECCS and the containment spray system. Credit for these actions will be assessed on a plant specific basis and risk calculations supporting the credit should be performed in accordance with Regulatory Guide 1.174.

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The GR does not address the analytical methods to be used for performing the Region II analyses (e.g., computer codes and models). In particular, staff has reservations on how the models and methods described in Sections 3 and 4 could be adopted for these types of analyses. The staff will assess the adequacy of methods used during reviews of any plant-specific licensing submittals and plant-specific audits performed as part of the GSI-191 and Generic Letter closeout process. Part of staff's assessment would include: methods, models and data used to estimate event timings, T/H conditions, and how the debris phenomena treat calculational uncertainties. It is known that all aspects of debris phenomena (including, generation, transport, and head loss) have large uncertainties. In lieu of explicitly treating these uncertainties, staff used engineering judgment to conclude that these uncertainties are typically small compared to conservatism introduced by DEGB type limiting analyses. Licensee evaluations performed under Region-II should be cognizant of such issues and address them explicitly. For example, considerable experimental evidence exists in support of increased head loss due to long-term operation. Very limited, if any, experiments are carried out to quantify such factor mechanistically. Instead traditional correlations developed using short-term tests, corrected based on engineering judgment, were used to account for long-term phenomena. In the past, staff accepted such approximations because of large margin of conservatism implicit in DEGB type analyses.

6.5 RISK INSIGHTS

Section 6.5 of the NEI GR is provided to guide the determination of risk acceptability for cases in which a licensee relies on sump mitigation capability (including crediting operator actions) for the Region II Analysis (i.e., Section 6.4). In Section 6.5 of the NEI Evaluation Guidance, the acceptance guideline from Regulatory Guide (RG) 1.174 that is used to define an acceptably small increase in core damage frequency (CDF) is used to establish a target reliability for the sump mitigation capability. To further ensure the acceptability of this approach, the NEI Evaluation Guidance also uses a conservative value for the large break loss of coolant accident (LBLOCA) initiating event frequency, which is taken from NUREG-1150. Thus, the NEI Evaluation Guidance provides a method by which a licensee can ensure that any increase in CDF resulting from plant modifications, operator actions, etc. that are credited in Section 6.4 will be small and meet the RG 1.174 acceptance guideline by demonstrating that the target reliability of the sump mitigation capability is achieved.

The target reliability is established by first calculating the increase in CDF as the combination of the LBLOCA initiating event frequency (LBLOCA:IEF) and the sump mitigation capability failure probability (SMC:FP). In this calculation there are a number of conservatisms used to make it simple and straightforward, including:

- The base case condition represents the condition in which the current sump meets the regulations without needing credit for mitigation capability and is assumed to not clog (i.e., the sump is perfect, with a clogging probability of 0).

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- The mitigation condition case represents the condition in which the sump takes credit for mitigation capability and assumes if the mitigation capability fails the sump will clog (i.e., the sump always clogs if the mitigation capability fails, with a clogging probability of 1) and a clogged sump results in core damage (i.e., no credit for potential recovery actions).
- The calculation is performed for the entire LBLOCA break spectrum (i.e., all breaks greater than about 6 inches), while the NEI Evaluation Guidelines "Region II" alternate approach is only used for those break sizes greater than the "debris generation" break size, which is only a portion of the LBLOCA break spectrum (i.e., calculation assumes all LBLOCAs require mitigation, not just those greater than the "debris generation" break size).

Based on this approach, the calculation of the increase in CDF can be simplified to:

$$\Delta\text{CDF} = \text{LBLOCA:IEF} \times \text{SMC:FP}$$

Recognizing that the target reliability (TR) is the complement of the sump mitigation capability failure probability (SMC:FP) and resolving the equation results in:

$$\text{TR} = 1 - \text{SMC:FP} = 1 - [\Delta\text{CDF} / \text{LBLOCA:IEF}]$$

The RG 1.174 acceptance guideline for a small change in CDF is less than 1.0E-5/year. This is an appropriate acceptance guideline for plants where the total CDF can be reasonably shown to be less than 1.0E-4/year. The NEI Evaluation Guideline states that the 1.0E-4/year total CDF value bounds the population of PWRs. The staff accepts that this may be true. However, if a licensee's total CDF is significantly greater than 1.0E-4/year, considering all modes and initiators, then that licensee should provide additional justification and meet an appropriately higher target reliability.

The value for the LBLOCA initiating event frequency from NUREG-1150 is 5.0E-4/reactor-year. It is recognized by the staff that this represents a generic bounding value of the LBLOCA frequency and is considerably greater (and thus conservative) than used in plant-specific probabilistic risk assessments (PRAs).

Substituting the above values into the equation for determining the target reliability results in a target reliability for the sump mitigation capability of 0.98 per demand (i.e., SMC:FP equals 2.0E-2/demand).

The staff understands that the reliability of the sump mitigation capability will be determined on a plant-specific basis and ensured with reasonable confidence to be equal to or greater than the above established target reliability. This determination will include evaluations of associated plant modifications as well as credited operator actions, including those modifications and actions credited in Section 6.4 that represent a change from current operations (e.g., crediting operator action to terminate or reduce containment spray flow to assure net positive suction head of the low head pumps).

The staff also accepts that passive components do not need to be considered in the reliability determination, as long as these passive components are demonstrated as being functional by design (e.g., enlarged sump screen areas) or failure is determined to be extremely unlikely (e.g., less than 1.0E-5/demand), even given challenges that passive components might see, such as jet forces or blowdown loads. However, if a measurable and inspectable reliability can be ascribed to a passive component (e.g., passive screen cleaning), then the reliability determination should include these features.

Consistent with the RG 1.174 principles of risk-informed decision-making, the impact of the proposed change must be monitored using performance measurement strategies. Therefore, an implementation and monitoring plan must be developed to ensure that the evaluation conducted to examine the impact of the proposed changes continues to reflect the actual reliability and availability of the SSCs and operator actions that have been evaluated. This will ensure that the conclusions that have been drawn from the evaluation remain valid. Thus, the staff requires licensees to propose, in their plant-specific submittals, a monitoring program that is consistent with RG 1.174 Section 2.3, which includes a means to adequately track the performance of equipment that, when degraded, can affect the conclusions of the licensee's evaluation (i.e., demonstration of the sump mitigative capability to meet its reliability target). The program must be capable of trending equipment performance after a change has been implemented to demonstrate that performance is consistent with that assumed in the traditional engineering and probabilistic analyses that were conducted to justify the change. This must include monitoring associated with non-safety-related SSCs if the analysis determines those SSCs to be relied upon to meet the sump mitigative capability target reliability. The program must also be structured such that feedback of information and corrective actions are accomplished in a timely manner and degradation in performance is detected and corrected before plant safety can be compromised. The staff expects that licensees choosing to apply this methodology would comply with the guidance in RG 1.174 or provide justification for the deviation.

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In summary, the staff finds this portion of the alternate approach acceptable for use in the NEI Evaluation Guidelines "Region II" evaluations for the following reasons:

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- The target reliability determination includes a number of conservative simplifications, including:
- It is performed for the entire LBLOCA break spectrum (i.e., all breaks greater than about 6 inches), while the NEI Evaluation Guidelines "Region II" alternate approach is only used for those break sizes greater than the "debris generation" break size, which is only a portion of the LBLOCA break spectrum.
- The base case condition is assumed not to be susceptible to clogging (i.e., the sump is perfect, with a clogging probability of 0).
- The mitigation condition case assumes if the mitigation capability fails the sump will clog (i.e., the sump always clogs if the mitigation capability fails, with a clogging probability of 1) and that a clogged sump results in core damage (i.e., no credit for potential recovery actions).
- The NUREG-1150 LBLOCA initiating event frequency of 5.0E-4/reactor-year is expected to be much greater than the LBLOCA value derived from the on-going

U.S. Nuclear Regulatory Commission (NRC) Office of Research (RES) expert elicitation process.

- The approach is consistent with RG 1.174 since it uses the acceptance guidelines that define an acceptably small CDF increase in determining the target reliability of the sump mitigation capability.
- Licensees choosing to apply Region II analyses should implement a performance-monitoring program, consistent with Section 2.3 of RG 1.174 to ensure that the conclusions of the licensee's evaluation (i.e., demonstration that the sump mitigative capability meets the established target reliability) are maintained valid.

In considering risk-informing aspects of the resolution of GSI-191, the staff recognized that there is the potential that the containment sump may clog if the mitigation capability credited in the Region II analysis does not function properly. Based on the industry proposed approach in the Region II analysis, which also uses the conservative NUREG-1150 LBLOCA frequency to calculate the target reliability of the mitigation capability, and using the related generic study information, the largest LBLOCA CDF would be 1.4E-5/year. This indicates that at a minimum the risk associated with LBLOCAs will be reduced from the current condition by nearly an order of magnitude.

7.0 ADDITIONAL DESIGN CONSIDERATIONS

Four extenuating design considerations are discussed in this section of the GR that are related to the broad issue of recirculation-sump operability addressed under GSI-191. These topics are (1) structural analysis of the containment sump, (2) upstream effects that limit water flow, (3) downstream effects related to debris penetration of the screen, and (4) potential chemical effects that contribute to head loss either as an additional debris source or by modifying the hydraulic properties of pre-existing beds. Staff evaluations of the GR treatment of these topics follow in corresponding subsections of this SER. The NRC agrees that this list is complete when added to the balance of detail provided in the remainder of the GR, as modified by staff recommendations.

7.1 SUMP STRUCTURAL ANALYSIS

This section of the GR provides general guidance for considerations to be used when performing a structural analysis of the containment sump screen. No specific details are provided in the GR for how to perform this analysis. General items identified for consideration are: verifying maximum differential pressure due to combined clean screen and maximum debris load at rated flow rates, geometry concerns (mesh and frame vs. perforated plate), material selection for post-accident environment, and the addition of hydrodynamic loads due to a seismic event. The GR specifically states that Regulatory Guide 1.82, Revision 3, subsection 1.1.1.8 may need to be referenced for evaluation of hydrodynamic loads on a strainer.

Staff evaluation for Section 7.1: The staff finds the general statements in Section 7.1 pertaining to the analysis of the structural capability of the containment sump strainer to be acceptable. The staff agrees that potential bending and stretching of existing wire mesh may lead to gaps at the points of attachment between wire and framing structures. The staff further agrees that any modifications to existing sump-screen configurations should employ corrosion-resistant materials that will not be affected by post-LOCA containment conditions.

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Consideration of sump structural analysis in the GR and in this SER is limited to the debris loads and the hydraulic loads imposed by water in the sump pool. Dynamic loads imposed on the sump structure and screen by break-jet impingement must be addressed in accordance with GDC 4, including provisions for exclusion of certain breaks from the design basis when analyses reviewed and approved by the NRC demonstrate that the probability of fluid system piping rupture is extremely low.

Paragraph 2(d)(vii) of the Information Request section of Generic Letter 2004-02 requests that addressees verify that trash racks and sump screens are capable of withstanding the loads imposed by expanding jets and missiles. The staff is requesting that addressees verify that the trash racks and sump screens continue to meet the current design basis requirements under GDC 4, as discussed above.

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The GR does not provide detail in its presentation of criteria for sump screen performance and comparisons to predicted head loss. To clarify this information, the staff offers the following discussion. It is true that structural loads on a sump screen should be computed using the total pressure drop across the screen. The total pressure drop is the sum of the head loss computed or measured across the clean screen at a rated flow in the absence of debris and the debris-induced head loss computed or measured under the same volumetric flow rate. The limiting conditions for sump screen structural analysis correspond to a break location and debris source term that induces the maximum total head loss at the sump screen after full consideration of transport and degradation mechanisms. Debris-bed head loss should be calculated for each postulated break scenario according to methods outlined in Section 3.7 and 4.2.5 of the GR as amended by SER recommendations.

Licensing-basis calculations of NPSH margin already include the effects of flow resistance through the clean screen, so it is sufficient to examine the debris-bed head loss separately. For a completely submerged sump screen, if the NPSH margin is smaller than the head loss induced by debris from the limiting break, then the licensing-basis has been exceeded and some form of mitigation, modification, or exemption is warranted. For a partially submerged sump screen, a potentially more restrictive condition may apply. In order to supply adequate water flow through the debris bed, the pressure drop cannot exceed one half of the pool depth in feet of water or the NPSH margin, whichever is smaller. This additional criterion arises because the containment pressure is equal on both sides of the debris bed and the static pressure of the pool is the only way to force water through the bed [RG 1.82-3].

Thus, different criteria may dictate the structural capacity of the sump screen for supporting water flow through a debris bed under recirculation velocities depending on screen geometry. Other considerations like maximum water velocities during fill up and hydrodynamic loads during a seismic event may impose additional design constraints.

The guidance presented in the GR would require each licensee to perform a plant-specific evaluation of their respective sump screen to determine structural capability under post-accident conditions. The staff agrees with the GR reference of Regulatory Guide 1.82 for evaluation of hydrodynamic loads. This plant-specific analysis would be reviewed on a case-by-case study.

7.2 UPSTREAM EFFECTS

This section of the GR provides guidance on evaluating the flowpaths upstream of the containment sump for hold-up of inventory which could reduce flow to and possibly starve the sump. The GR identifies two parameters as being important to the evaluation of upstream effects, they are: (1) containment design and postulated break location; and (2) postulated break size and insulation materials in the ZOI. The GR states that the above two parameters provide a basis to evaluate hold-up or choke points in the flow field within containment upstream of the containment sump. The GR also advises that the containment condition assessment as described in NEI 02-01 provides guidance on this review.

The GR provides users of the document the following examples of locations to evaluate for hold-up of liquid upstream of the sump screen: (1) Narrowing of hallways or passages; (2) gates or screens that restrict access to areas of containment such as behind the bioshield or crane wall; and (3) refueling canal drain. The GR then states that these areas of concern are generally applicable to all containments but advises licensees to evaluate their containment for possible holdup at unique geometric features, and to evaluate any plant-specific insulation installation.

Staff Evaluation for Section 7.2: The staff finds that the above mentioned items of the GR are appropriate as stated and offers the following amplification: Licensees should utilize the results of their debris assessments to estimate the potential for water inventory hold-up. Based on these assessments and the mapping of probable flow paths licensees should utilize methods provided in Chapter 5 of the GR (reducing the source term) for the additional purpose of reducing hold-up of blowdown inventory upstream of the sump. Licensees should evaluate the effect the placement of curbs and debris racks intended to holdup debris may have on the holdup of water en route to the sump.

Staff Conclusions Regarding Section 7.2: The staff finds that the GR provides adequate direction regarding the evaluation of holdup of inventory from the sump. The staff provides the above additional comments as an amplification to the GR.

7.3 DOWNSTREAM EFFECTS

This section of the GR gives licensees guidance on evaluating the flow paths downstream of the containment sump for blockage due to entrained debris. The GR specifies three concerns to be addressed which are: (1) blockage of flow paths in equipment such as containment spray nozzles and tight-clearance valves; (2) wear and abrasion of surfaces such as pump running surfaces, and heat exchanger tubes and orifices; and (3) blockage of flow clearances through fuel assemblies. It is noted here that the NRC is currently conducting research in the area of debris bypass through sump screens and flow blockage of HPSI throttle valves, and that this SER may be supplemented with the results of this research in early CY 2005. The staff would then expect licensees to consider the supplemental information in evaluating their plants for downstream effects.

The GR identifies the starting point for the evaluation to be the flow clearance through the sump screen and states that the maximum size of particulate debris that will pass through the sump screen is determined by the flow clearance through it. The GR states that wear and abrasion of surfaces in the ECC and CS should be evaluated based on flow rates to which the surfaces will be subjected and the grittiness or abrasiveness of the ingested debris. The GR recognizes that the abrasiveness of debris is plant-specific. The GR also states that wear and abrasion of pumps due to ingestion of debris may have been addressed by the pump manufacturer and advises licensees to contact their vendor regarding the ability of the pump to perform with debris in the process fluid.

Staff Evaluation for Section 7.3: The GR states, "If passages and channels in the ECC and CS downstream of the sump screen are larger than the flow clearance through the sump screen, blockage of those passages and channels by ingested debris is not a concern." In addition, the GR states, "Similarly, wear and abrasion of surfaces in the ECC and CS should be evaluated based on flow rates to which the surfaces will be subjected..." The staff finds the GR statements do not fully address the potential safety impact of LOCA generated debris on components downstream of the containment sump. The following represents staff expectations on the review of the effects of debris on components and systems downstream of the containment sump following initiation of containment recirculation. (Refs. 68, 69)

The evaluation of GSI 191 should include a review of the effects of debris on pumps and rotating equipment, piping, valves and heat exchangers downstream of the containment sump related to emergency core cooling (ECC) and containment spray (CS) systems. In particular, any throttle valves installed in the ECC systems for flow balancing, e.g., HPSI throttle valves, should be evaluated for blockage potential. The evaluation should also address the effects of entrained debris on the reactor vessel and internal core components. (Refs. 2, 17)

In general, the downstream review should first define both long term and short term system operating lineups, conditions of operation, and mission times. Where more than one ECC or CS configuration is used during long and short term operation, each line-up should be evaluated with respect to downstream effects. The definition of the design and license bases mission times form the premise from which the short and long term consequences will be determined and evaluated.

Once condition of operation and mission times are established, downstream process fluid conditions should be defined including assumed fiber content, hard materials, soft materials, and various sizes of material particulates. The staff has found that particles larger than the sump screen mesh size will pass through to downstream components. Debris may pass through due to its aspect ratio or because it is 'soft' and differential pressure across the screen pulls it through. No credit may be taken for 'thin bed' filtering effects. (Refs. 68, 69)

Evaluations of systems and components are to be based on the flow rates to which the wetted surfaces will be subjected and the grittiness or abrasiveness of the ingested debris. The abrasiveness of the debris is plant specific, as stated in the GR, and depends on the site-specific materials that may become latent or break-jet-generated debris.

Specific to pumps and rotating equipment, an evaluation should be performed to assess the condition and operability of the component during and following its required mission times. Consideration should be given to wear and abrasion of surfaces; for example, pump running surfaces, bushings, wear rings, etc. Tight clearance components or components where process water is used either to lubricate or cool should be identified and evaluated.

Dirt, dust, and other materials may combine or interact with fiber and cause a matting effect. This matting effect may significantly increase the rate of wear. Test data and operating experience has shown that hard faced components will wear under long-term exposure to post accident 'slurry' conditions. Soft surface materials such as brass, bronze, etc. will wear at much faster rates.

Component rotor dynamics changes and long-term effect on vibrations due to potential wear should be evaluated in the context of pump and rotating equipment operability and reliability. The evaluation should include the potential impact on pump internal loads to address such concerns as rotor and shaft cracking. (Refs. 68, 69)

As stated in the GR, pump manufacturers may have addressed wear and abrasion of pumps due to ingestion of debris. Licensees may consider requesting information and/or test data from the pump vendor regarding the ability of specific pumps to perform with debris in the process fluid. Other sources of information available to licensees include information generated to support the closeout of USI A-43, "Containment Emergency Sump Performance," such as NUREG/CR-2792, "An Assessment of

Residual Heat Removal and Containment Spray Pump Performance Under Air and Debris Ingesting Conditions."

The downstream effects evaluation should also consider system piping, containment spray nozzles and instrumentation tubing. Settling of dusts and fines in low flow / low fluid velocity areas may impact system operating characteristics and should be evaluated. The matting effect may cause blockages and should be addressed. The evaluation should include such tubing connections as provided for differential pressure from flow orifices, elbow taps, and venturis and reactor vessel / RCS leg connections for reactor vessel level and any potential to affect instrumentation necessary for continued long term operation.

Valve (Ref. 70) and heat exchanger wetted materials should be evaluated for susceptibility to wear, surface abrasion, and plugging. Wear may alter the system flow distribution by increasing flow down a path (decreasing resistance due to wear), thus starving another critical path. Or conversely, increased resistance due to plugging of a valve opening, orifice, or heat exchanger tube may cause wear to occur at another path that is taking the balance of the flow thus diverted from the blocked path.

Decreased heat exchanger performance due to plugging, blocking, plating of slurry materials or tube degradation should be evaluated with respect to overall system required hydraulic and heat removal capability.

An overall ECCS or CS system evaluation integrating limiting or worst case pump, valve, piping and heat exchanger conditions should be performed including the potential for reduced pump/system capacity due to internal bypass leakage or through external leakage. Internal leakage of pumps may be through inter-stage supply and discharge wear rings, shaft support and volute bushings, etc. (Refs. 68, 69). Piping systems design bypass flow may increase as bypass valve openings increase or as flow through a heat exchanger is diverted due to plugging or wear. External leakage may occur as a result of leakage through pump seal leak-off lines, from the failure of shaft sealing or bearing components, from the failure of valve packing or through leaks from instrument connections and any other potential fluid paths leading to fluid inventory loss.

Leakage past seals and rings due to wear from debris fines to areas outside containment should be evaluated with respect to fluid inventory and overall accident scenario design and license bases environmental and dose consequences.

Fluids present post LOCA during long and short term recirculation may flow through the reactor vessel and its internal components. The downstream effects evaluation should consider flow passage blockages such as associated with core grid supports, mixing vanes, and debris filters. The evaluation should also consider component binding such as reactor vessel vent valves in B&W designs.

If flow paths between upper downcomer and upper plenum / upper head (such as hot leg nozzle gaps and upper head cooling passages) have an influence on long term cooling, then the potential for plugging these paths should be addressed.

<u>Staff Conclusions Regarding Section 7.3</u>: The staff finds that the GR is non-conservative with respect to its statement that the maximum size of particulate debris that would pass through a sump screen is determined by the flow clearance through the sump screen. As stated above, the staff has seen evidence that particles larger than the flow openings in a screen will deform and flow through or orient axially and flow through (Refs. 68, 69). Licensees should determine, based on their debris generation and transport calculations, what percentage of debris would likely pass through their sump screen and be available for blockage at the downstream locations discussed above.

The evaluation of downstream effects should include consideration of term of operating line-up (long or short), conditions of operation, and mission times as stated above.

Consideration should be given to wear and abrasion of pumps and rotating equipment as discussed above. (Refs. 68, 69) Licensee's downstream effects evaluations should consider system piping, containment spray nozzles, and instrumentation tubing as well. Valve and heat exchanger wetted surfaces should be evaluated for wear, abrasion, and plugging. Wear should be evaluated with respect to the potential to alter system flow distribution. Heat exchanger performance should be evaluated with respect to the potential for blockage or the plating of slurry materials. HPSI throttle valves should be specifically evaluated for their potential to plug and/or wear. (Ref. 70) ECCS and CS overall performance should be evaluated with respect to all conditions discussed above.

Flow blockage such as associated with core grid supports, mixing vanes, and debris filters should be considered. Flow paths between upper downcomer and upper plenum / upper head should be evaluated for long term cooling degradation due to flow interruption from plugging.

As stated above, the staff concludes that the GR recommendations do not fully address the potential safety impact of LOCA generated debris on components downstream of the containment sump. Licensees should address the additional considerations detailed above in the staff's evaluation.

In order to effectively evaluate downstream effects, licensees may need to review equipment specifications, O&M (operations and maintenance) manuals, station drawings such as equipment, piping, isometrics, flow diagrams, etc. Review of previous physical walkdowns of piping and instrument systems may be necessary to verify low points where debris accumulation may occur, potential choke points or other areas of concern not readily verifiable from document reviews. Also leakage past seals and rings due to wear from debris fines to areas outside containment should be evaluated with respect to license bases environmental and dose consequences. Previously issued generic communications regarding downstream effects, HPSI throttle valve clogging, wear of HPI Pump, pipe line clogging, heat exchanger wear due to operation under abrasive or debris laden conditions should also be reviewed.

7.4 CHEMICAL EFFECTS

Section 7.4 of the GR introduces the potential problems of chemical reactions in the post-LOCA environment of PWR containments. The reaction products formed can contribute to blockage of the ECCS sump screens and increase the associated head loss across the screens. The GR notes that a test plan has been developed to study possible interactions among corrosion products and the resultant effects of those products on sump filtration. The GR defers guidance for dealing with these effects until the testing is completed and the data has been appropriately evaluated.

For the purpose of this SER, the issue of chemical effects involves interactions between the post-LOCA PWR containment environment and containment materials that may produce corrosion products, gelatinous material, or other chemical reaction products capable of affecting sump screen head loss. A concern was raised by the Advisory Committee on Reactor Safeguards (ACRS) that an adequate technical basis should be developed to resolve the issues related to chemical reactions (ACRS letter dated 9/30/2003). A "gelatinous" material was observed in a water sample taken from the Three Mile Island (TMI) containment following the accident in 1979. (Oak Ridge National Laboratory Report Memorandum dated September 14, 1979). The relevance of the gelatinous material collected at TMI to the evaluation of potential post-LOCA chemical effects during the ECCS recirculation phase in plants today is uncertain for several reasons. The water sample containing a gelatinous material was collected from the TMI containment approximately 5 months after the accident, which is longer than the typical projected mission time for ECCS recirculation following a modern day PWR LOCA. The source of the water sample collected from the TMI containment was also unique in that some of the water in the TMI containment after the accident was introduced from the Susquehanna River.

A limited scope study was conducted at Los Alamos National Laboratory (LANL) to evaluate potential chemical effects occurring following a LOCA. This study was conducted to assess the potential for chemically induced corrosion products to impede ECCS performance. In some of these tests, metal nitrate salts were added to the test water in concentrations above their solubility limits in order to induce chemical precipitants and assess head loss effects. Although these LANL tests showed that gel formation, with a significant accompanying head loss across a fibrous bed was possible, no integrated testing was performed to demonstrate a progression from initial exposure of metal samples to formation of chemical interaction precipitation products. (LANL Report LA-UR-03-6415, ML033230260). In addition, the test conditions were not intended to be prototypical of a PWR post-LOCA environment. Therefore, a more comprehensive study has been initiated to address potential chemical effects.

An integrated chemical effects test program has been developed through a collaborative effort between the NRC and nuclear industry. The test objective is to characterize any chemical reaction products, including possible gelatinous material, that may develop in a representative plant post-LOCA PWR environment. Test conditions (e.g., pH, temperature, boron concentration) were selected to simulate representative, not necessarily bounding plant conditions. The initial sump conditions experienced during a large break LOCA will not be replicated in order to simplify the experimental test setup and equipment. Instead, the chemical reactions from corrosion and leaching products during the initial LOCA conditions were simulated using the OLI Systems Inc. Suite of Thermodynamic Equilibrium Programs (e.g., Environmental Simulation Program Version 6.6 and Stream Analyzer Version 1.2). The simulations varied the amount of key components, different pH moderators (i.e., sodium hydroxide versus trisodium phosphate), pH, temperature, and pressure and the results indicated large scale corrosion tests using a pressurized test loop were not necessary to capture the period immediately following the LOCA. Thermodynamic simulations and sensitivity analyses of key variables including corrosion products, were developed to rank species that have a potential for causing sump head loss through formation of precipitates. Validation of the appropriate OLI Systems Inc. programs will be performed using available borated water literature and by comparing the Program's initial post-LOCA environment species predictions to results obtained in small scale (e.g., autoclave) corrosion tests in a representative initial post-LOCA environment.

Larger scale corrosion testing will be conducted using facilities at the University of New Mexico. Corrosion test coupon materials include zinc (galvanized steel and inorganic zinc based coatings), aluminum, copper, carbon steel, insulation, and concrete. Relative amounts of test materials were scaled according to plant data provided by the industry based on plant surveys. Test coupons will either be fully immersed or placed above the test loop water line but subjected to a fine spray to simulate exposure to containment spray. The relative distributions of each material were determined based on estimated percentages submerged or subjected to containment sprays following a plant LOCA. If gelatinous material is observed to develop, alternative courses of action will be considered (e.g., head loss tests). Initial testing is expected to begin in September 2004.

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In order to address chemical effects on a plant specific basis, licensees will initially need to evaluate whether the chemical effects test parameters are sufficiently bounding for their plant specific conditions. If plant specific materials are not bounded by the chemical effects test parameters, licensees shall provide technical justification to use any results from the chemical effects tests in their plant specific evaluation. If chemical effects are observed during these tests, licensees will need to evaluate the sump screen head loss consequences of this effect in an integrated manner with other postulated post-LOCA effects. In addition, a licensee who chooses to modify their plant sump screens prior to the completion of chemical effects testing and analysis of the test results should consider potential chemical effects in order to ensure a second plant modification is not necessary should deleterious chemical effects be observed during testing.

8.0 CONDITIONS AND LIMITATIONS

The guidance in the GR and in this SER is offered for all licensees of domestic PWRs for the evaluation of ECCS sump performance. However, the following conditions and limitations apply to its use:

Debris Generation

1) The destruction pressures cited in the GR for determining ZOI radii are based on air jet data and could underestimate debris quantities for a two-phase jet, as discussed in Section 3.4.2.2 of this SER. Therefore the staff position is that destruction pressures based on air jet testing are to be lowered by 40% in order to account for two-phase jet effects.

2) The GR provides calculated and recommended values for ZOI radii for common PWR insulation and coatings materials in Table 3.1. The staff determined that the calculated values were non-conservative at higher destruction pressures but the recommended values are conservative. Therefore the staff position is that only the recommended values be used.

3) The staff agrees with the characterization of debris in GR Section 3.4.3; however, the staff position is that licensees apply insulation-specific debris size information if possible.

Protective Coatings

4) Characterization of failed coatings with the value of 1000 psi as a destruction pressure, with corresponding ZOI of one pipe diameter is not sufficiently justified and may be nonconservative, as discussed in Section 3.4.2. Therefore, the staff position is that licensees should use a spherical coatings ZOI equivalent to 10D, or determined by plant specific analysis, based on experimental data that correlate to plant materials over the range of temperatures and pressures of concern.

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5) The alternative offered to plant-specific data in Section 3.4.3.4, for the determination of coatings thicknesses (i.e., 3 mil equivalent of 10Z), may not be conservative and is therefore not acceptable without adequate plant-specific justification.

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6) For those plants that substantiate no formation of a fibrous thin bed, the assumptions and guidance provided in the GR for coatings may be nonconservative. Therefore, for any such plant, assumptions related to coatings characterization, must be conservative with regard to sump blockage. Consideration should be based upon the plant-specific susceptibility to thin bed formation identified by the licensee. Specifically this includes the plant-specific consideration of larger sized chips, flakes, or other form of break-down which is realistically-conservative, or use of a default area equivalent to the area of the sump screen openings, for coatings size.

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Latent Debris

7) Periodic surveys that monitor changes in latent debris inventory are needed to monitor the effectiveness of cleanliness programs for supporting the overall sump screen blockage vulnerability. The steps presented in the GR for direct assessment of dust thickness are considered by the staff to be impractical and unreliable, and thereby unacceptable. To provide more accurate results, statistical surface sampling should be performed in accordance with the guidance provided in this SER.

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8) If a licensee chooses to take credit for a cleanliness program to account for a fractional surface area for debris accumulation, it is the staff position that documentation be available to verify proper implementation.

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9) In addition to the three categories of miscellaneous debris discussed in the GR, the quantity, characteristics and location of any failed coatings should also be noted in the survey, to the extent available during plant specific walkdowns.

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Transport

10) Those plants with configurations conducive to fast pool velocities should include large piece debris transport in their evaluations. The GR baseline methodology that assumes no transport of large debris to the sump screens is not adequate. A comparison of the characteristic transport velocities to typical debris transport velocities is needed to determine whether or not large piece debris transport is important.

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11) Because (1) the method recommended for determining the quantity of fine debris trapped in inactive pools is over-simplified, (2) a survey of the fractions of inactive pool volumes to total sump pool water volumes is not available to better judge the potential industry wide impact of this assumption, and (3) the comparison of the baseline methodology and a detailed analysis for the volunteer plants was considerably different; a limit on this fraction is needed to limit the impact of this nonconservative methodology assumption. Therefore, the staff concludes that an upper limit on this ratio of 15% should be assumed unless a higher fraction is adequately supported by analyses or experimental data.

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12) The baseline assumption that all debris in the containment bottom floor is uniformly distributed throughout the entire volume of water in containment is also not conservative. This assumption was made in the baseline guidance as justification for the inactive pool volume ratio but otherwise does not directly affect the acceptance of the baseline guidance due to the 100% recirculation pool transport assumption. However, should a plant subsequently perform a pool transport refinement, then this assumption would not apply and at that point alternative approaches such as those detailed in Appendix III would be required.

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Head Loss

13) The licensees should ensure the validity of the NUREG/CR-6224 correlation for their applications of type of insulations and the range of parameters using the guidance provided in Appendix V of this report.

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Alternate Evaluation

14) Consistent with the RG 1.174 principles of risk-informed decision-making, the impact of the proposed change should be monitored using performance measurement strategies. Therefore, the staff position is that licensees develop an implementation and monitoring plan to ensure that the evaluation conducted to examine the impact of the proposed changes continues to reflect the actual reliability and availability of the SSCs and operator actions that have been evaluated.

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This plan should include a means to do the following:

a. Track the performance of equipment that when degraded can affect the conclusions of the licensee's evaluation (i.e., demonstration of the sump mitigative capability to meet its reliability target).
b. Trend equipment performance after a change has been implemented to demonstrate that performance is consistent with that assumed in the traditional engineering and probabilistic analyses that were conducted to justify the change.
c. Monitor non-safety-related SSCs if the analyses determine those SSCs to be relied upon to meet the sump mitigative capability target reliability.

The program should also be structured such that feedback of information and correction actions are accomplished in a timely manner and degradation in performance is detected and corrected before plant safety can be compromised.

Downstream Effects

15) Licensees should consider particles larger than the flow openings in a sump screen will deform and flow through or orient axially and flow through, and determine what percentage of debris would likely pass through their sump screen and be available for blockage at downstream locations.

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16) Licensees should consider term of system operating line-up (short or long), conditions of operation, and mission times.

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17) Licensees should consider wear and abrasion of pumps and rotating equipment, piping, spray nozzles, instrumentation tubing, and HPSI throttle valves. The potential for wear to alter system flow distribution and/or form plating of slurry materials (in heat exchangers) should be included.

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18) An overall ECCS or CS system evaluation should be performed considering the potential for reduced pump/system capacity due to internal bypass leakage or through external leakage.

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19) Licensees should consider flow blockage associated with core grid supports, mixing vanes, and debris filter, and their effects on fuel rod temperature.

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Chemical Effects

20) The staff has considered NEI's response and finds that chemical effects should be addressed on a plant-specific basis. Initially, licensees should evaluate whether the current chemical test parameters, which are available in the test plan for the joint NRC/Industry Integrated Chemical Effects Tests, are sufficiently bounding for their plant specific conditions. If they are not, then licensees should provide technical justification in order to use any of the results from the tests in their plant-specific evaluation. If chemical effects are observed during these tests, then licensees should evaluate the sump screen head loss consequences of this effect. A licensee who chooses to modify their sump screen before tests are complete should consider potential chemical effects in order to avoid additional screen modification should deleterious chemical effects be observed during testing.

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21) Any analytical refinement(s) proposed by a licensee in its plant-specific analysis of sump performance which is not addressed by the staff in this section of the SER, should be presented to the staff for approval.

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9.0 CONCLUSION

The GR provides the PWR industry with an important tool for estimating the head loss across their ECCS sump screens based on the generation, transport, and accumulation of debris in containment to and on the sump screens. The NEI approach is to provide guidance and leave certain areas to be resolved on a plant-specific basis, as opposed to providing a detailed methodology that applies to all PWRs as a stand-alone document (as was done for BWRs with the URG), based on the argument of variability among PWRs. Little testing was done by NEI to support and justify assumptions made in the GR (as opposed to the approach by the BWROG to generate data that supports the URG). However, the NEI guidance provides historical data, considerations, and engineering judgments that can be used by the industry to develop those areas not fully addressed in the GR.

The iterative process used by NEI in this GR also creates some challenges in the overall review. Specifically, although this guidance has been characterized by NEI as extremely conservative, the iterative process allows for the reduction of conservatisms in various areas (identified in each affected section of this evaluation) that could affect other areas of the analysis to produce larger reductions in overall conservatism than were expected.

The approach taken by the staff was to evaluate each area of the GR, and in those areas where there was a lack of supporting data or where conservatism was questioned, provide alternative guidance based on the staff's engineering judgment and/or additional data generated in testing done mainly at LANL. This data is a result of testing specifically contracted by the NRC over the last five years as part of the GSI-191 resolution effort, and involves sump performance research which was completed but in a few cases not published, and is referenced (Ref x) and/or included as appendices in this document. Inclusion of this additional information is also intended to provide valuable insight to the industry in its effort toward evaluating plant-specific vulnerability to sump blockage and related issues.

The staff concludes that the guidance proposed by NEI as approved in accordance with this SER, provides an acceptable evaluation methodology that establishes the necessary basis and provides the realistic-conservatism for an acceptable PWR guidance document. Key conclusions in each area of the analysis are documented below.

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Pipe Break Characterization: The staff finds that the GR guidance is acceptable provided that two outstanding issues, listed below, are adequately addressed by each licensee:

1) The GR does not provide guidance for those plants that can substantiate no thin bed effect, which may impact head loss results and limiting break location.

2) For plants needing to evaluate secondary-side piping such as main steam and feedwater pipe breaks, break locations should be postulated in a manner consistent with the guidance in Section 3.3 of this SER.

To address these issues, the staff provided enhanced guidance in the appropriate sections of this SER. When the guidance provided in the GR is supplemented with the enhanced guidance offered in the SER, the staff finds this section to be acceptable.

Debris Generation/Zone-Of-Influence: The staff has reviewed the use of a spherical model sized in accordance with the ANSI/ANS standard, and finds this approach acceptable. The spherical geometry proposed encompasses a zone which considers multiple jet reflections at targets, offset between broken ends of a guillotine break, and pipe whip.

With regard to the destruction pressures cited for determining ZOI radii, data are referenced from the BWROG URG which were determined using an air jet. However a LOCA jet is a two-phase steam/water jet. Based on staff study of this difference and due to experimental evidence from two-phase jets, the destruction pressures based on air jets could be too high and thus could underestimate debris quantities. Therefore the staff position is that destruction pressures based on air jet testing are to be lowered by 40% in order to account for two-phase jet effects.

The confirmatory analysis performed by the staff (Appendix I) verifies the applicability of the ANSI/ANS standard for determining the size of this zone. Use of a ZOI model is identified as an acceptable approach for analyzing debris generation per RG 1.82, Rev. 3. (This approach was also used and approved by the staff in the BWR sump performance SER.)

The refinement offered in the GR to apply spherical ZOIs that correspond to material-specific destruction pressures for each material that may be affected in the vicinity of a break, is also acceptable.

The staff concurs with the characterization of debris in GR Section 3.4.3. Confirmatory analyses provided in Appendix II, verifies the acceptability of the size distributions recommended in the GR. However, the staff urges application of insulation-specific debris size information if possible.

Protective Coatings: Coating debris generation in the GR is treated separately from other debris types. The GR assumes that coating debris is generated from postulated failure (destruction) of both DBA-qualified and unqualified coatings within the ZOI and from postulated failure of all unqualified coatings outside the ZOI. For coatings, the GR recommends a ZOI destruction pressure of 1000 psi, with a corresponding ZOI radius of one pipe diameter. The GR assumes that all coating debris will fail to a particulate size equivalent to the basic material constituent.

The staff agrees with the approach taken with regard to characterization of coatings; however the staff considers there to be insufficient technical justification to support a value of 1000 psi as a destruction pressure, with corresponding ZOI of one pipe diameter. The staff position is that licensees should use a coatings ZOI spherical equivalent to 10D or a ZOI determined by plant specific analysis, based on experimental data that correlate to plant materials over the range of temperatures and pressures of concern. Note that an equivalent to ten pipe diameters was used for coatings characterization and was approved by the staff in the BWR sump performance SER.

With regard to the characterization of coatings in Section 3.4.3.4 of the GR, an alternative offered to plant-specific data for the determination of coatings thicknesses is an equivalent IOZ thickness of 3 mils. Because this recommended value may be nonconservative and is unsubstantiated as is described in Section 3.4.3.4, the staff finds that this value of 3 mils not to be acceptable without adequate plant-specific justification for any coatings thicknesses used. Plant-specific evaluation of the unqualified coatings within containment is recommended to be performed to determine realistically-conservative coating properties, including thicknesses. Further, it is recommended that means be incorporated into the methodology to periodically assess the amount of unqualified coating identified and used in the sump analysis to ensure the quantity remains bounding and if nonconsevative changes in the amount of unqualified coating occur, that the impact of this change be evaluated.

Also, for those plants that substantiate no formation of a fibrous thin bed, the assumptions and guidance provided in the GR for coatings may be nonconservative. Therefore, for any such plant, assumptions related to coatings characterization, must be conservative with regard to sump blockage. Consideration must be based upon the plant-specific susceptibility to thin bed formation identified by the licensee. Specifically, this includes the plant-specific consideration of larger sized chips, flakes, or other form of break-down which is realistically-conservative, or use of a default area equivalent to the area of the sump screen openings, for coatings size.

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Latent Debris: The staff has reviewed the guidance provided for estimating the impact of latent debris and agrees that it is necessary to determine the types, quantities and locations of latent debris sources. The staff also agrees that it is not appropriate for licensees to claim that their existing foreign material exclusion (FME) programs have entirely eliminated miscellaneous debris. Results from plant specific walkdowns should be used to determine a conservative amount of latent debris in containment and to monitor cleanliness programs for compliance to committed estimates.

The staff further concludes that the guidance provided in the GR for consideration of effects of latent debris is informative and prescriptive, but treats certain attributes in an inconsistent manner, lacks consideration of a number of surfaces and unique phenomena that enhance dust collection, and relies on an impractical and imprecise method for estimating the volume of latent debris on surfaces. Alternate guidance is provided in this section of the SER for statistical sampling and sample analysis to allow licensees to more accurately determine the impact of latent debris on sump screen performance. This revised approach is based on generic characterization of actual PWR debris samples. If desired, a licensee could pursue plant-specific characterization as a refinement.

Debris Transport: The staff finds that the transport guidance for small fines is conservative and acceptable; however, neglect of the large pieces and the neglect of variability and uncertainties due to lack of data, are non-conservative. Therefore, for those plants with configurations conducive to fast pool velocities, consideration of large pieces of debris is necessary. Also, the method recommended for determining the quantity of fine debris trapped in inactive pools, is over-simplified, and therefore the acceptability of this method will be determined on a plant-specific basis depending on whether overall realistic-conservatism is maintained for this portion of the analysis.

Head Loss: Computation of head loss in the GR involves input of design characteristics and reflection of thermal-hydraulic conditions into a head loss correlation (NUREG/CR-6224), which is acceptable to the staff. The licensees should ensure the validity of the NUREG/CR-6224 correlation for their applications of type of insulations and the range of parameters using the guidance provided in Appendix V of this report.

However, the staff finds that the following guidance on fibrous thin bed formation should be considered:

- Use of the appropriate density in the determination of the quantity of debris needed to form a thin bed—i.e., the as-manufactured density.
- Careful evaluation of the limiting porosity for the particular particulate or mixture of particulates in the debris bed.
- Consideration of uncertainties in specifying a one-eighth-inch bed thickness criteria—e.g., the indication that calcium silicate can form a debris bed without supporting fibers.
- Consideration of other uncertainties—e.g., uncertainties associated with mixing of constituents, or uncertainties associated with latent debris data collection.

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Before using the NUREG/CR-6224 correlation that is recommended in the GR or any other head loss correlation, the licensees should ensure that it is applicable for the type of insulation and the range of parameters. If the correlation has been validated for the type of insulations and the range of parameters, the licensees may use it without further validation. If the correlation has not been validated for the type of insulations and the range of parameters, the licensees should validate it using head loss data from tests performed for the particular type of insulations.

Analytical Refinements: Three analytical topics are identified in the GR to be included in this section—i.e., debris generation, debris transport, and head loss. A fourth, break selection, is addressed in Section 6.0.

For debris generation, the GR proposes use of debris-specific ZOI's versus use of the most conservative debris type applied to all. In addition, the GR proposes use of two freely-expanding jets emanating from each broken pipe section versus use of spherical ZOI. The staff finds both debris generation refinements to be acceptable.

For debris transport, two methods for computing flow velocities in a sump pool—i.e., the network method and the computational fluid dynamics method—are provided in the Analytical Refinements section of the GR. However, the staff finds insufficient guidance offered in either option to provide an acceptable alternative to the baseline approach. These refinements are therefore not acceptable.

For head loss, the only refinement cited by the GR is stated to be in GR Section 3.7.2.3.2.3, "Thin Fibrous Beds," where the need for consideration of fibrous thin bed formation, and the alternative consideration of latent debris as the primary contributor to this thin bed for all-RMI plants, are addressed. However, the staff addresses consideration of thin fibrous beds in Section 3.4, "Debris Generation," of this SER pertaining to the baseline, rather than as a refinement.

Therefore, staff finds no specific refinement offered for the head loss analysis.

Physical Refinements To Plant: GR Section 5.0 provides guidance for refinements in the areas of debris source term, debris transport obstructions, and screen modifications.

The staff has reviewed the debris source term refinements involving primarily enhanced housekeeping programs, insulation and/or coatings modifications, and equipment modifications; and finds them to be acceptable. However, with regard to insulation change-out or modification, the staff emphasizes that although maximum debris loadings on the screen may be addressed with this refinement, minimum loadings required to form a thin-bed effect may not be. Also, in regards to coatings, the statement that DBA-qualified coatings have very high destruction pressures, has not been proven (see sections 3.4.2, 3.4.2, and 4.2.2.2.3).

The staff agrees that debris consistent with the materials listed can be effectively trapped with the use of a debris transport obstructions in optimized locations where the local velocities are less than the test results presented. The staff finds the general statements in parts of this section to provide little specific information regarding the methods for determining proper debris transport obstruction design. However, the lack of specific implementation strategies and simplified concepts presented would require each plant to perform a plant specific evaluation of their proposed debris obstruction to determine their effectiveness and structural capability under post-accident conditions. To credit debris transport obstructions for trapping debris, plant specific documentation will also be required to demonstrate an appropriate correlation to the test results in terms of debris type and velocity limits.

With regard to screen modification, those discussed in the GR are found to be acceptable; however, licensees are not limited to those identified in this GR.

Alternate Evaluation: NEI has proposed an alternative evaluation approach which incorporates realistic and risk-informed elements to the PWR sump analysis as described in Section 6.0. In considering risk-informing aspects of the resolution of GSI-191, the staff recognized that there is the potential that the containment sump may clog if the mitigation capability credited in the Region II analysis does not function properly. Based on the industry proposed approach in the Region II analysis, which also uses the conservative NUREG-1150 LBLOCA frequency to calculate the target reliability of the mitigation capability, and using the related generic study information, the largest LBLOCA CDF would be 1.4E-5/year. This indicates that at a minimum the risk associated with LBLOCAs will be reduced from the current condition by nearly an order of magnitude.The staff concludes that GR Section 6.0 provides an acceptable approach for evaluating PWR sump performance. Application of more realistic and risk-informed elements is technically justified based on the low likelihood of such breaks occurring.

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Sump Structural Analysis: The GR does not provide detail in its presentation of criteria for sump screen performance and comparisons to predicted head loss. Therefore, the staff provides additional guidance for assurance that the ECCS sump is able to accommodate both the clean screen head loss and the debris-induced head loss associated with the limiting break while providing adequate flow through both the ECCS injection pumps and the CS pumps if needed. For those structural design considerations mentioned in the GR, each should be assessed for applicability on a plant-specific basis.

Upstream Effects: The GR identifies certain hold-up or choke points which could reduce flow to and possibly cause blockage upstream of the sump. The staff finds the guidance with respect to upstream blockage to be acceptable.

Downstream Effects: This section provides guidance on the evaluation of entrained debris downstream of the sump causing downstream blockage. Because the GR provides limited guidance on how downstream effects should be evaluated, the staff provides the following alternative guidance with regard to downstream blockage:

- Licensees should consider particles larger than the flow openings in a sump screen will deform and flow through or orient axially and flow through, and determine what percentage of debris would likely pass through their sump screen and be available for blockage at downstream locations.
- Licensees should consider term of system operating line-up (short or long), conditions of operation, and mission times.
- Licensees should consider wear and abrasion of pumps and rotating equipment, piping, spray nozzles, instrumentation tubing, and HPSI throttle valves. The potential for wear to alter system flow distribution and/or form plating of slurry materials (in heat exchangers) should be included.

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- An overall ECCS or CS system evaluation should be performed considering the potential for reduced pump/system capacity due to internal bypass leakage or through external leakage.
- Licensees should consider flow blockage associated with core grid supports, mixing vanes, and debris filter, and their effects on fuel rod temperature.

Chemical Effects: The staff has considered NEI's response and finds that chemical effects should be addressed on a plant-specific basis. Initially, licensees should evaluate whether the current chemical test parameters, which are available in the test plan for the joint NRC/Industry Integrated Chemical Effects Tests, are sufficiently bounding for their plant specific conditions. If they are not, then licensees should provide technical justification in order to use any of the results from the tests in their plant-specific evaluation. If chemical effects are observed during these tests, then licensees should evaluate the sump screen head loss consequences of this effect. A licensee who chooses to modify their sump screen before tests are complete should consider potential chemical effects in order to avoid additional screen modification should deleterious chemical effects be observed during testing.

Overall Conclusion: The staff has reviewed the GR and finds portions of the proposed guidance to be acceptable. For those areas found to need additional justification and/or modification due to inadequate detail, lack of supporting data, or lack of analysis to support the technical basis; the staff has provided identified conditions and limitations and required modifications, including alternative guidance, to supplement the guidance in the NEI submittal. The resultant combination of the NEI submittal and staff safety evaluation, provide an acceptable overall guidance methodology for the plant-specific evaluation of ECCS or CSS sump performance following all postulated accidents for which ECCS or CSS recirculation is required, with specific attention given to the potential for debris accumulation that could impede or prevent ECCS or CSS from performing its intended safety functions.

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4	NEI, 2003	Letter from A. R. Pietrangelo, NEI to J. N. Hannon, USNRC, Transmittal of "Draft PWR Containment Sump Evaluation Methodology," Nuclear Energy Institute, dated October 31, 2003.
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13	NRCB, 1995	Bulletin 95-02, "Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode," dated October 17, 1995;
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17	NRCB, 2003	NRC Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at Pressurized-Water Reactors," dated June 9, 2003.
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40	Mtg, 2003b	Summary of Public Meeting Held April 29, 2003, Nuclear Regulatory Commission and Stakeholders Concerning Generic Safety Issue 191, "Assessment of Debris Accumulation on PWR Sump Performance," (TAC No. MA6454) dated May 15, 2003.
41	Mtg, 2003a	Summary of Public Meeting Held March 5, 2003, Nuclear Regulatory Commission and Stakeholders Concerning Generic Safety Issue 191, "Assessment of Debris Accumulation on PWR Sump Performance," (TAC No. MA6454) dated April 24, 2003. Attachments: "Current Status and Plans Resolution of GSI-191," Accession Number ML031180084 "Update of PWR Industry Plan to Address GSI-191," Accession Number ML031180102 "NRC Comments on NEI Draft Evaluation Methodology Ground Rules," Accession Number ML031180095. "GSI-191: Analysis of Recovery Options," Accession Number ML031180119. "Debris Accumulation & CalSil Head Loss Testing – Findings & Preliminary Conclusions," Accession Number ML031180127.

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43	Mtg, 2002a	Summary of Public Meeting Held March 28, 2002, Nuclear Regulatory Commission and Interested Stakeholders Concerning Generic Safety Issue 191, "Assessment of Debris Accumulation on PWR Sump Performance," (TAC No. MA6454), dated April 16, 2002.
44	Mtg, 2002b	Summary of Public Meeting Held May 30, 2002, Nuclear Regulatory Commission and Interested Stakeholders Concerning Generic Safety Issue 191, "Assessment of Debris Accumulation on PWR Sump Performance," (TAC No. MA6454) dated June 6, 2002.
45	Mtg, 2002c	Summary of Public Meeting Held July 2, 2002, Nuclear Regulatory Commission and Interested Stakeholders Concerning Generic Safety Issue 191, "Assessment of Debris Accumulation on PWR Sump Performance," (TAC No. MA6454) dated July 31, 2002.
46	Mtg, 2002d	Summary of Public Meeting Held August 29, 2002, Nuclear Regulatory Commission and Stakeholders Concerning Generic Safety Issue 191, "Assessment of Debris Accumulation on PWR Sump Performance," (TAC No. MA6454) dated September 5, 2002.
47	Mtg, 2002e	Summary of Public Meeting Held October 24, 2002, Nuclear Regulatory Commission, the Nuclear Energy Institute, and Interested Stakeholders Concerning Generic Safety Issue 191, "Assessment of Debris Accumulation on PWR Sump Performance," (TAC No. MA6454) dated October 31, 2002.
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Confirmatory Appendices

APPENDIX I: ANSI/ANS JET MODEL

I.1 INTRODUCTION

Debris generation is the first chronological step in the accident sequence for a postulated high-energy line break. In the idealized case of a double-ended guillotine break (DEGB), high-temperature, high-pressure reactor-cooling fluid may be ejected (from both sides of the broken pipe) that impinges on structures, equipment, piping, insulation, and coatings in the vicinity of the break. The degree of damage induced by the break jets is specific to the materials and structures involved, but the size and shape of the expanding jets and the forces imparted to surrounding objects depend on the thermodynamic conditions of the reactor at the location of the rupture. To maximize the volume of the damage zone, i.e., zone of influence (ZOI), it is conservative to consider free expansion of the break jet to ambient conditions with no perturbation, reflection, or truncation by adjacent structures. Spatial volumes of damage potential, as defined by empirical correlations of local jet pressure and observed damage, for example, can then be integrated over the free-jet conditions and remapped into convenient geometries, such as spheres or cones, that approximate the effects of congested reflection without crediting the associated shadowing, jet dispersion, and energy dissipation.

One reasonably accessible model that is available for computing pressure contours in an expanding jet is presented in Appendices B, C, and D of the American National Standards Institute (ANSI) guidance for the protection of nuclear power plants against the effects of pipe rupture [ANS88]. The ANSI model was used for the evaluation of potential damage volumes in the resolution of the boiling-water-reactor (BWR) strainer-blockage study [URG96, NRC98]. A similar approach suggested for this analysis by [ANS88] is a jet model developed at Sandia National Laboratories [WEI83]. Both the ANSI and the Sandia models were developed specifically for assessing structural loadings on relatively large targets near the jet centerline, so neither offers a true estimate of local pressures within a freely expanding jet. However, these models can be used with appropriate caution to learn a great deal about the spatial extent of and the thermodynamic conditions present within a high-energy jet.

This appendix presents the equation set needed to evaluate the ANSI model describing two-phase expansion of a jet from a broken high-energy line in a pressurized-water-reactor (PWR). To ensure a conservative review of the guidance report (GR), only the conditions related to full separation and full radial offset of a DEGB are developed. Alternative equations are presented in the standard for partial offsets and for longitudinal tears. This discussion is offered to resolve some of the confusion present in the notation of the standard and to provide a self-consistent basis for interpreting computational results relevant to PWR break conditions. The complexity of the jet model is somewhat beyond the scope of manual evaluation, but several investigators have performed successful spreadsheet calculations for discrete conditions. Routines developed in MATLAB and FORTRAN for evaluating the jet model are included at the end of this appendix as a further guide to implementation and for critical review; however, routines obtained from the National Institute of Standards and Technology (NIST) for evaluating thermodynamic state points are not provided.

I.2 JET-MODEL FEATURES AND APPLICABILITY

Despite the apparent complexity of the equation set needed to evaluate the ANSI jet model, it is based on relatively few thermodynamic assumptions and limited comparisons with experimental observation. The bulk of the analytic detail supplies a geometric framework for interpolating jet pressures between assumed or observed transition points. Figure I-1 presents a sample calculation of jet pressure contours for a cold-leg DEGB. Although this calculation represents a relevant bound for evaluation of the GR, to be discussed later, the figure will be used first to introduce geometric features of the model.

The ANSI jet model subdivides the expanding jet into three zones that are delineated by dashed lines in Figure I-1. Zone 1 contains the core region, where it is assumed that liquid extrudes from the pipe under the same stagnation conditions as the upstream reservoir (interior red triangle). Zone 2 represents a zone of continued isentropic expansion, and Zone 3 represents a region of significant mixing with the environment, where the jet boundary is assumed to expand at a fixed, 10-degree, half angle. One group of equations from Appendix C of the standard defines the geometry of the jet envelope, and another group from Appendix D defines the behavior of internal pressure contours. Key geometry features that are determined by the thermodynamic conditions of the break include the length of the core region, the distance to the "asymptotic plane" between Zones 2 and 3, and the radii of the jet envelope at the transition planes between zones. At the asymptotic plane the centerline static pressure is assumed to approach the absolute ambient pressure outside of the jet.

Jet pressures provided by the ANSI model must be interpreted as local impingement gauge pressures. This is a property of the pressure field that is relevant to the interpretation of debris generation data; however, a subtle discrepancy exists between the ANSI model predictions and the desired local pressures. Because target materials may reside anywhere within the jet, fluid impingement can occur from a range of angles. Thus, idealized measurements or calculations of free-field impingement pressure should assume that the fluid stagnates (comes to rest) nonisentropically and parallel to the local flow direction. Note that a further subtlety appears here in the distinction between the classical definition of stagnation pressure that is related to the isentropic deceleration of flow along a streamline and the impingement pressure that includes entropy losses during impaction of a fluid on a physical test object. In general, impingement pressures will be higher than stagnation pressures, but the two terms may be used synonymously at times in this treatise.

In contrast to the desired local impingement pressure, the ANSI model appears to be concerned with total force loadings across relatively large objects placed near the jet centerline. It is stated in Appendix D of the standard that the pressure recovered on a target is related to the component of the flow perpendicular to the target and that, because of the diverging flow in an expanding jet, the pressure distribution on a large flat target will decrease in the radial direction. The pressure equations in the standard produce exactly this effect, and a brief allusion is made to a comparison of the predicted pressures with data taken across the face of large targets placed perpendicular to the jet. Further cautionary notes are given against applying the pressure equations to predict forces on small objects near the edges of the jet where flow velocities are clearly not parallel to the centerline.

These attributes of the model suggest that calculated pressures represent jet impingement conditions that would be experienced in a direction parallel to the midline only. Actual stream lines in a rapidly expanding jet must have a significant radial velocity component in order to create the characteristic envelope shown in Figure I-1, so in a sense, the predicted pressures represent only the longitudinal component of the local, momentum-dominated, total jet pressure. The implication of this interpretation is that true local impingement pressures as measured normal to realistic flow directions in the jet may be underestimated, particularly in Zones 1 and 2, where radial expansion is greatest.

Although a computed pressure isobar may be smaller in radius than that of the corresponding local impingement pressure that is desired for debris generation estimates, it may also be longer in the downstream direction. Comparative elongation of isobars from the jet model occurs because the entire mass flux ejected from the break is assumed to pass through the jet cross section at the asymptotic plane. Thus, the forward momentum of the jet is maximized in a manner that would be considered conservative for structural loading calculations. Unrealistic isobar elongation may also be predicted because the jet-centerline pressure equation for Zone 3 is inherently unbounded; that is, the centerline gauge pressure only falls to zero as the jet diameter grows infinitely large at infinite distance. It is impossible to quantify the net effect on isobar volume of these disparities between the ANSI model and the desired free-expansion impingement pressures without a complete understanding of the experimental measurements on which the model is based; however, the mathematical properties of the pressure equations are certain to exaggerate the length, and hence the volume, of low-pressure isobars.

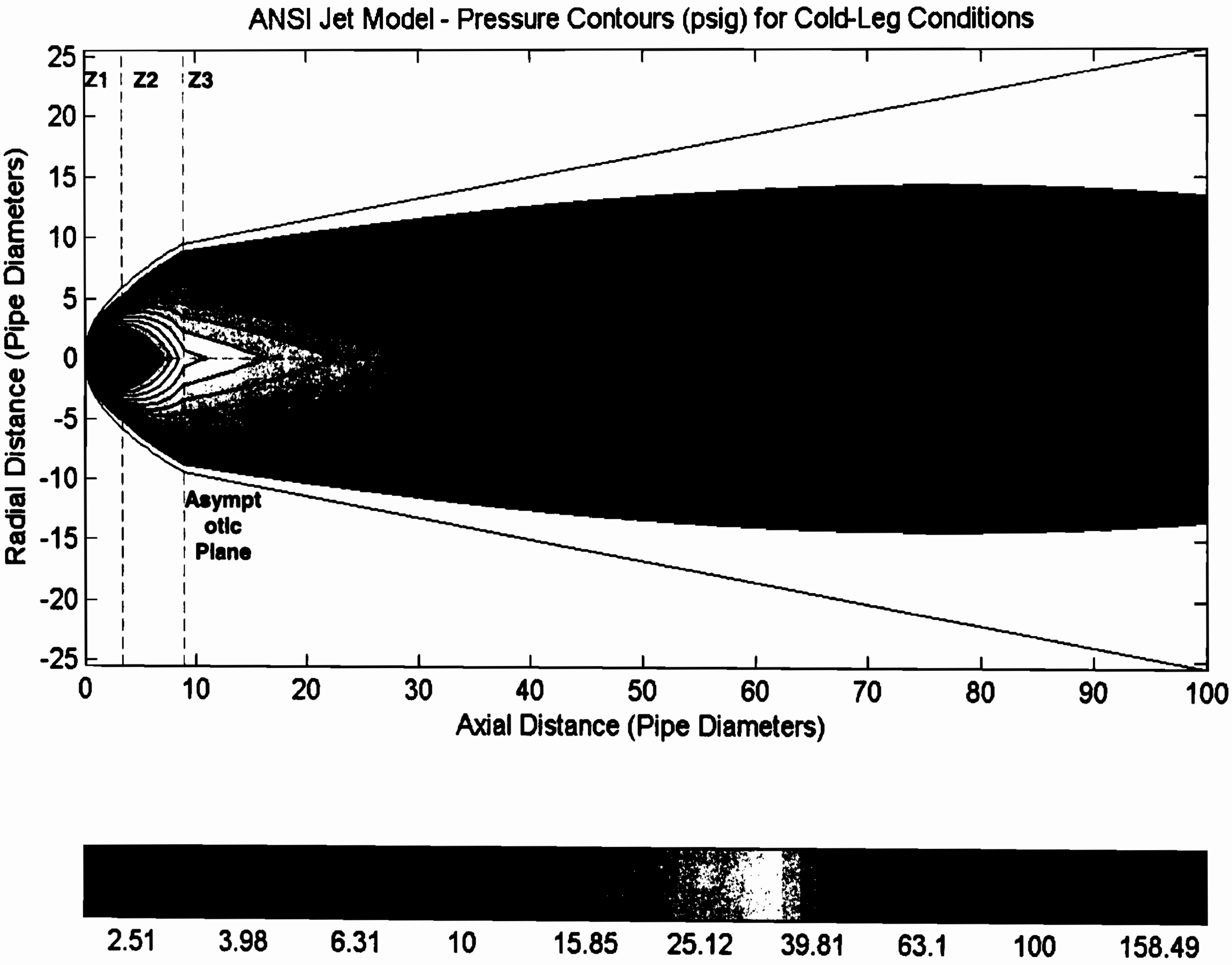

Figure I-1. ANSI Jet-Model Stagnation Pressures for PWR Cold-Leg Break Conditions (530ºF, 2250 psia)

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I.3 JET-MODEL EQUATION SET

I.3.1 Fundamentals

Equations developed in the standard frequently refer to four distinct thermodynamic state points: (1) stagnation conditions of the fluid in the upstream reservoir denoted by subscript "0" (zero), (2) conditions at the exit plane of the pipe denoted by subscript "e", (3) conditions at any point in the jet denoted either with subscript "j" or with no subscript at all, and (4) conditions at the asymptotic plane denoted by subscript "a". These conventions are rigidly applied in the following development to resolve some notation inconsistencies found in the standard. Unless otherwise noted, pressures will refer to the absolute thermodynamic static pressure of the fluid. The first exception to this rule has already been mentioned—that is, the jet-pressure equations that define the local, gauge, longitudinal, impingement pressure.

Figure I-2. Control-Volume Force Balance on a Rigid Plate near the Outlet

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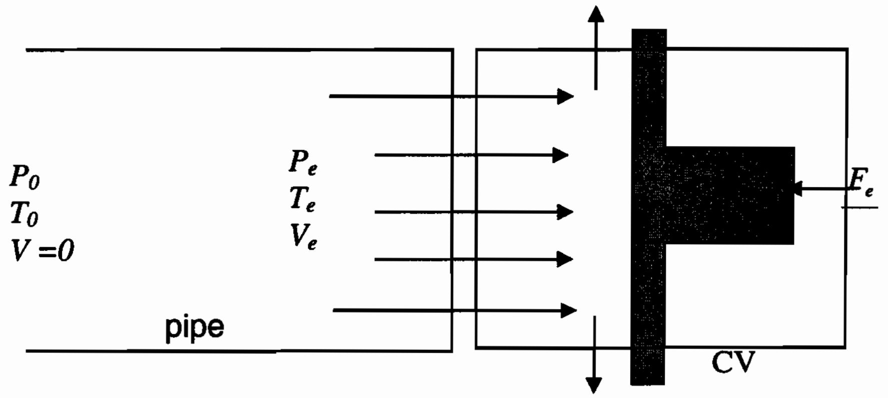

One of the more fundamental relations in the model is actually presented near the end of the standard in Appendix D; it defines the total thrust (force) of the jet at the outlet. If a rigid plate were placed near the outlet, as shown in Figure I-2, the force balance on a control volume (CV) must consider both the static pressures and the rate of change of momentum acting on the boundary. If mass exits the control volume in a symmetric pattern at uniform velocity, the only possible force imbalance is in the *x* direction. The force on a plate near the exit is then

$$F_e = P_e A_e - P_{amb} A_e + \frac{1}{g_c}\frac{d}{dt}(m_e v_e) = (P_e - P_{amb})A_e + \frac{1}{g_c}\left[\left(\frac{d}{dt}m_e\right)v_e + m_e\left(\frac{d}{dt}v_e\right)\right], \qquad \text{(I-1)}$$

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where P_e is the fluid pressure at the exit plane, P_{amb} is the ambient pressure in containment, A_e is the area of the break, and m_e is the mass entering the control volume at velocity v_e. The force-to-mass conversion factor g_c equals 32.2 lbm·ft/lbf·s^2 in English units. Mass enters the control volume at constant velocity $\left(\frac{d}{dt}v_e = 0\right)$ at a rate of $\frac{d}{dt}m_e = \rho_e v_e A_e$, where ρ_e is the fluid density at the exit. Thus, the total thrust generated at the exit plane is

$$F_e = (P_e - P_{amb})A_e + \frac{1}{g_c}\rho_e v_e^2 A_e. \qquad \text{............(I-2)}$$

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Substitution of $G_e = \rho_e v_e$ for the critical mass flux crossing the exit plane yields

$$F_e = \left[(P_e - P_{amb}) + \frac{G_e^2}{g_c \rho_e}\right]A_e, \qquad \text{............(I-3)}$$

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where the first term represents force applied by the static pressure of the fluid and the second term represents force imparted by the momentum of the fluid. The ambient pressure is often assumed to be zero to maximize the available jet thrust conservatively.

Division of Equation (I-2) or (I-3) by the exit area suggests an effective, or area-averaged, jet pressure of $\overline{P}_e = F_e / A_e$. This effective pressure will be greater than the classical stagnation pressure at the exit, which is defined by Bernoulli's equation as $P_e^{stag} = P_e^{static} + \frac{1}{2g_c}\rho_e v_e^2$, because the derivation of Bernoulli's law requires that the fluid be brought to rest in an idealized, reversible manner. Jet impingement on a body is a highly anisentropic process. For an incompressible fluid, the static pressure at the exit equals the ambient pressure, and if friction losses in piping between the reservoir and the break can be neglected, the stagnation pressure at the exit equals the initial pressure. Under these conditions, Bernoulli's equation can be written as

$$\frac{1}{g_c}\rho_e v_e^2 = 2(P_0 - P_{amb}). \qquad \text{............ (I-4)}$$

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Equations (I-2) and (I-3) are often simplified as $F_e = C_T P_0 A_e$, where P_0 is the upstream stagnation pressure and C_T is the thrust coefficient defined by comparison to be

$$C_T = \frac{1}{P_0}\left[\frac{1}{g_c}\rho_e v_e^2 + (P_e - P_{amb})\right] = \frac{1}{P_0}\left[\frac{G_e^2}{g_c \rho_e} + (P_e - P_{amb})\right]. \qquad \text{............(I-5)}$$

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Equation (I-5) emphasizes that the correlation between upstream stagnation pressure and the thrust coefficient is determined by the fluid properties that exist at the exit plane. Several alternative models are available to describe the thermodynamic transitions occurring in a high-energy fluid that is expanding and accelerating, which, in turn, determine the exit density, and the critical mass flux. It is very important that the specification of C_T be consistent with the models used to evaluate G_e and ρ_e. It should be noted that the standard uses inconsistent notation for the thrust coefficient (ex. C_T, C_{T_e}, $C^*_{T_e}$). All forms must refer to a single numeric value if the pressure equations are to be piecewise continuous between jet zones.

Under the conditions of zero friction loss and incompressible flow (solid liquid with no vapor fraction where $P_e = P_{amb}$), Equation (I-4) can be substituted into Equation (I-5) to obtain a theoretical maximum value of $C_T = 2.0$ when ambient pressure is neglected. By treating steam as a perfect gas under isentropic flow to obtain the exit velocity, Shapiro [SHA53] derives a lower theoretical limit of $C_T = 1.26$. Any numeric evaluation of Equation (I-5) using water property tables to derive G_e and ρ_e should be compared to these limits. Although it is clearly most conservative to apply the liquid limit for all state points, numerical evaluation of Equation (I-5) using water tables is sufficiently robust to permit this refinement. Recommendations for computing the thrust coefficient are discussed in Section I.4 later in this appendix, and convenient reference figures are provided.

I.3.2 Jet-Envelope Geometry

The shape and size of the jet envelope predicted by the ANSI model are dictated by the thermodynamic conditions upstream of the break. Except where noted, spatial distances are represented in dimensionless multiples of the broken-pipe inside diameter, D_e. Jet boundaries (and pressure contours) can be scaled in this manner because the equation set is linear with respect to pipe diameter. Linearity can be proven rigorously by factoring and eliminating terms of D_e in every equation. In general, because of potential nonlinearities, it is not sufficient to evaluate a complicated dimensional equation set at a unit value of a candidate scaling parameter and then to assume that the unit result can be multiplied by any desired value of that parameter. To recover physical quantities for a particular pipe size, dimensionless distances must be multiplied by D_e, dimensionless areas must be multiplied by D_e^2, etc.

The distance of extrusion by the jet core is

$$L_c = 0.26\sqrt{\Delta T_{sub}} + 0.5 \quad , \qquad \text{...........(I-6)}$$

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where ΔT_{sub} is the degree of subcooling (ºF) upstream of the break location, i.e., the difference between the saturation temperature T_{sat} at the system pressure P_0 and the system temperature T_0. The jet core is shown by the interior red triangle in Figure I-1. Note that L_c takes on a value of 0.5 for saturated or superheated conditions. Also, if $L_c > L_a$, the distance to the asymptotic plane defined below, L_c should be set to zero and the jet pressure should be assumed to be uniform across the break area at a value of $P_j = (F_e / A_e)/C_T$, where the ratio F_e / A_e is computed from Equation (I-2) or (I-3). This can occur for low-pressure nonexpanding jets. A jet can be treated as nonexpanding when the initial temperature of a liquid reservoir is less than the saturation temperature at P_{amb} or the initial pressure of a gas reservoir is equal to ambient pressure, $P_0 = P_{amb}$.

The diameter of the jet at the exit plane is defined to be

$$D_{je} = \sqrt{C_T}\,, \qquad \text{............(I-7)}$$

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which is slightly larger than the diameter of the pipe because $1.26 \le C_T \le 2.0$.

The diameter of the jet at the asymptotic plane (Zone 2 to Zone 3 boundary) is defined by the relation

$$D_a^2 = \frac{G_e^2}{g_c \rho_a C_T P_0}\,, \qquad \text{............(I-8)}$$

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where ρ_a is the homogeneous fluid density at the centerline distance to this plane, which is given by

$$L_a = \frac{1}{2}(D_a - 1). \qquad \text{............(I-9)}$$

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Note that some care must be taken to keep pressure and mass flux dimensionally consistent in Equation (I-8). The density ρ_a is to be evaluated at a state point defined by the system enthalpy h_0 and an asymptotic-plane static pressure defined by

$$P_a = \left\{1 - 0.5\left(1 - \frac{2P_{amb}}{P_o}\right) f(h_o)\right\} P_{amb}\,, \qquad \text{............(I-10)}$$

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where

$$f(h_0) = \sqrt{0.1 + \frac{h_o - h_f}{h_{fg}}} \quad \text{for} \quad \frac{h_o - h_f}{h_{fg}} > -0.1, \text{ and } f(h_0) = 0 \text{ otherwise.} \qquad \text{(I-11)}$$

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Within the condition stated by Equation (I-11), h_f and h_g are the saturated fluid enthalpy and saturated vapor enthalpy at P_0, respectively, and $h_{fg} = h_g - h_f$ is the heat of vaporization. Further conditions on Equation (I-10) are that if the ratio $P_{amb} / P_0 > 1/2$, it should be set equal to 1/2 and that, as a static pressure, $P_a \ge 0$.

The first criterion on $f(h_0)$ simply checks whether the initial quality $x_0 = \frac{h_o - h_f}{h_{fg}}$ is greater than negative 10%. When considered as a whole, these conditions imply that $0 \le P_a \le P_{amb}$. If the initial fluid is more than 10% subcooled, the jet static pressure equals ambient pressure at the asymptotic plane. If the jet is less than 10% subcooled, the jet static pressure at the asymptotic plane can be lower than ambient pressure. Equation (I-10) suggests that the asymptotic plane is placed at the distance where the jet static pressure approaches ambient pressure. The distance to this plane given by Equation (I-9) may simply have been chosen by geometric comparison with observed jets.

The state point defined by the asymptotic pressure P_a and the system enthalpy h_0 may be a two-phase condition. In this case, it is necessary to evaluate the asymptotic density ρ_a using the quality $x_a = \frac{h_o - h_{fa}}{h_{ga} - h_{fa}}$, where h_{fa} and h_{ga} are the saturated fluid and vapor enthalpies at P_a, respectively. Then $\rho_a = \left[\frac{x_a}{\rho_{ga}} + \frac{1 - x_a}{\rho_{fa}}\right]^{-1}$, where ρ_{fa} and ρ_{ga} are the saturated fluid and vapor densities at P_a, respectively. Automated steam tables generally give mixture densities directly for a two-phase state point, so this complication may be unnecessary.

The similarity of terms in Equation (I-8) to the force-balance equations derived in the previous section suggests a different interpretation for the asymptotic plane. For convenient reference, the jet diameter at the asymptotic plane is again given by

$$D_a^2 = \frac{G_e^2}{g_c \rho_a C_T P_0}, \qquad \text{............(I-12)}$$

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Given the discussion following Equation (I-3) and the definition of the thrust coefficient, the factors $C_T P_0$ in Equation (I-12) are immediately recognized as $\overline{P}_e = F_e / A_e$, the average total jet pressure at the exit. If a relation similar to Equation (I-3) is written to describe the area-averaged pressure across the jet cross section at the asymptotic plane,

$$\overline{P}_a = \frac{F_a}{A_a} = \left[(P_a - P_{amb}) + \frac{G_e^2}{g_c \rho_a}\right], \qquad \text{............(I-13)}$$

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then the term $G_e^2 / g_c \rho_a$ in Equation (I-12) is recognized to be $\overline{P}_a - (P_a - P_{amb})$. If the static pressure at the asymptotic plane P_a is not much different than the ambient pressure P_{amb}, then Equation (I-12) reduces to the ratio of average pressures computed over the jet cross section at the asymptotic plane and over the jet cross section at the exit,

$$D_a^2 = \frac{F_a/A_a}{F_e/A_e} = \frac{\overline{P}_a}{\overline{P}_e}. \tag{I-14}$$

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Writing explicitly the definition of the dimensionless asymptotic-plane area as $D_a^2 = \frac{A_a}{A_e}$ illustrates that the diameter of the jet given by Equation (I-8) has been chosen at the point where the ratio of average pressures approaches the ratio of cross sectional areas, and for this to be true, the total force across each area must be the same. Hence, the ANSI model implicitly assumes that the jet force available at the outlet is conserved across the jet cross section at the asymptotic plane. At this distance, the jet is presumed to begin interacting with the environment.[*] This development also shows that the ANSI model projects the entire mass flux across the asymptotic plane rather than following more realistic stream lines across the jet boundary in Zones 1 and 2. Equation (I-8) is derived more rigorously in Section I-4 to further emphasize these points.

The remainder of the jet envelope is simply interpolated as a function of centerline distance *L* between the transition diameters discussed above. Within Zone 1, the diameter of the jet core is given by

$$D_c = \sqrt{C_T}\left(1 - \frac{L}{L_c}\right) . \tag{I-15}$$

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For Zone 1 and 2 $(0 < L \leq L_a)$, the jet diameter is given by

$$D_j^2 = \left[1 + \frac{L}{L_a}\left(\frac{D_a^2}{D_{je}^2} - 1\right)\right] D_{je}^2 . \tag{I-16}$$

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In Zone 3 $(L > L_a)$, the jet diameter expands at a 10-degree half angle beginning from the diameter at the asymptotic plane. The Zone-3 diameter is specified by

$$D_j^2 = \left[1 + \frac{2(L - L_a)}{D_a}\tan(10°)\right]^2 D_a^2 . \tag{I-17}$$

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I.3.3 Jet Pressures

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[*] This observation was derived from the jet equations and is not expounded as part of any derivation in the standard. It is simply an implication of the definitions.

Pressure contours also appear to be interpolated from a limited number of geometric reference points, but the basis for this interpolation is not evident from the standard. It can be shown that all equations are piecewise continuous at the separation planes between zones; however, no effort was made to match first-derivative slopes. This deficiency admits the possibility of "kinks" in the contours, as observed in Figure I-1 across the boundary between Zones 2 and 3. Pressure contours in Zone 1 $(0 \le L \le L_c)$ depend on the following discriminant. If

$$D_j^2 + 2D_jD_c + 3D_c^2 \le 6C_T, \qquad \text{............(I-18)}$$

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then the jet pressures are given as a function of radius $(r_c < r \le r_j)$ for jet diameters $D_j = 2r_j$ as

$$P_j = \left(\frac{D_j - 2r}{D_j - D_c}\right)\left[1 - \frac{2\left(D_j^2 + D_jD_c + D_c^2 - 3C_T\right)}{D_j^2 - D_c^2}\left(\frac{2r - D_c}{D_j - D_c}\right)\right]P_0. \qquad \text{............(I-19)}$$

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Otherwise,

$$P_j = \left(\frac{D_j - 2r}{D_j - D_c}\right)^2\left[\frac{6\left(C_T - D_c^2\right)}{\left(D_j - D_c\right)\left(D_j + 3D_c\right)}\right]P_0. \qquad \text{............(I-20)}$$

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It is important to note that the leading term $(D_j - 2r)$ vanishes in both Equation (I-19) and (I-20) as the radius approaches the jet envelope where the absolute pressure equals P_{amb}. Therefore, evaluations of P_j must be interpreted as gauge pressures. In Equation (I-19), the term $\left(\frac{2r - D_c}{D_j - D_c}\right)$ ensures that the jet pressure matches P_0 on the boundary of the core. There is no similar constraint provided in Equation (I-20), so there will be a sharp discontinuity in pressure at the boundary of the jet core when this condition is invoked, as shown in Figure I-1. Equations (I-19) and (I-20) were not intended to be evaluated inside of the core region. Within the core, the system stagnation conditions are presumed to hold.

In Zones 2 and 3, jet pressures are parameterized in terms of the jet centerline pressure P_{jc}. In Zone 2 $(L_c < L \le L_a)$,

$$P_{jc} = \left\{F_c - \left(F_c - \frac{3C_T}{D_a^2}\right)\frac{L_a}{L}\frac{(L - L_c)}{(L_a - L_c)}\right\}P_0, \qquad \text{............(I-21)}$$

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where the parameter $F_c = 1.0$ if $D_j^2 \le 6C_T$ at distance L_c and $F_c = 6C_T / D_j^2$ otherwise. When $L = L_c$, Equation (I-21) reduces to $P_{jc} = F_c P_0$. If $F_c = 1.0$, the centerline pressure will match the assumed pressure in the core region, but otherwise, there will again be a discontinuity. Given the centerline pressure, jet pressures in Zone 2 are specified by

$$P_j = \left(1-\frac{2r}{D_j}\right)\left\{1-2\left(\frac{2r}{Dj}\right)\left[1-\frac{3C_T}{D_j^2}\frac{P_0}{P_{jc}}\right]\right\}P_{jc}. \qquad \text{............(I-22)}$$

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It can be shown by integration that (I-22) is essentially a geometric rather than physical condition: it leads to full recovery of the jet force anywhere in Zone 2 regardless of the value assigned to the jet diameter. In Zone 3, centerline pressures are given by

$$P_{jc} = \frac{3C_T P_0}{D_a^2\left[1+\frac{2(L-L_a)}{D_a}\tan(10°)\right]^2} \qquad \text{............(I-23)}$$

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and jet pressures are given by

$$P_j = \left(\frac{D_j - 2r}{D_j}\right)P_{jc}. \qquad \text{............(I-24)}$$

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Pressures on the transition between Zones 2 and 3 are piecewise continuous, including on the centerline.

I.3.4 Pressure-Contour Characteristic Equations

Equations presented in the previous section can be used to evaluate longitudinal impingement pressures at any location in the jet. However, in the present forms, they are not particularly convenient for identifying geometric characteristics such as isobar boundaries. Similarly, when numerically computing volumes under a given isobar, it is convenient to know the downstream range of the contour, which always begins at $L = 0$ and terminates in a cusp on the centerline at some distance $L = L_t(P_j)$. Relationships presented in this section are not developed in the ANSI standard; they are offered to facilitate some of the many practical details involved with implementing the standard.

Figure I-1 illustrates the typical behavior of jet-pressure isobars generated by the ANSI model. The isobars outlined in black represent lines of constant pressure that can be found by solving the pressure Equations (I-19), (I-20), (I-22), and (I-24) for the radii at a constant pressure P_j. Remember that the downstream distance L is implicitly specified by the jet diameter D_j. Each pressure equation can be reduced to a general quadratic expression for the radius of the form $Ar^2 + Br + C = 0$.

The coefficients from Equation (I-19) for Zone 1 are

$$A = 4H,\ \ B = -2\left[1 + H\left(D_j + D_c\right)\right],\text{ and } C = D_j + HD_jD_c + \left(D_c - D_j\right)\frac{P_j}{P_0},$$

............(I-25)

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where

$$H = 2\frac{\left(D_j^2 + D_jD_c + D_c^2 - 3C_T\right)}{\left(D_j^2 - D_c^2\right)\left(D_j - D_c\right)}.$$

............(I-26)

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The coefficients from Equation (I-20) for Zone 1 are

$$A = 4,\ B = -4D_j,\text{ and } C = D_j^2 - \left(D_j - D_c\right)^2 \frac{P_j}{P_0} I,$$

............(I-27)

where

$$I = \frac{6\left(C_T - D_c^2\right)}{\left(D_j - D_c\right)\left(D_j + 3D_c\right)}.$$

............(I-28)

A special case occurs in Zone 1 at $L = 0$, where $D_j = D_c$ and $r = D_j / 2 = D_c / 2$ for all P_j.

Equation (I-22) yields the following coefficients for Zone 2:

$$A = 8\frac{J}{D_j^2},\ B = -\left(\frac{2}{D_j} + \frac{4J}{D_j}\right),\text{ and } C = 1 - \frac{P_j}{P_{jc}},\text{ where } J = \left(1 - \frac{3C_T}{D_j^2}\frac{P_0}{P_{jc}}\right).$$

............(I-29)

Finally, Equation (I-24) yields for Zone 3 the coefficients

$$A = 0,\ B = -2 / D_j,\text{ and } C = 1 - \frac{P_j}{P_{jc}}.$$

............(I-30)

The analytic solution for radius in Zone 3 is

$$r = \frac{1}{2} D_j \left(1 - \frac{P_j}{P_{jc}}\right).$$

............(I-31)

The sharp tip of each contour shown in Figure I-1 is another nonphysical feature of the ANSI model that arises from lack of attention to matching spatial first derivatives. It might be expected that each isobar be smoothly bounded and have infinite slope at the terminal point, especially at very low pressures where the jet returns to ambient conditions. It is helpful to know the distance to the terminal point of each contour for iterative integration of spatial volumes. These points can be found by solving the centerline pressure Equations (I-21) and (I-23) for distances $\underline{L_t}$ corresponding to the desired pressure. Note that there are no terminal points in Zone 1 except for the jet core.

For Zone 2 from Equation (I-21) comes the relation

$$L_t = \frac{L_c}{1 - \frac{L_a - L_c}{RL_a}\left(F_c - \frac{P_j}{P_0}\right)}, \qquad \text{............(I-32)}$$

where

$$R = F_c - \frac{3C_T}{D_a^2}, \qquad \text{............(I-33)}$$

and for Zone 3 from Equation (I-23) comes the relation

$$L_t = \frac{1}{2}\left[\left(\frac{3C_T P_0}{D_a^2 P_j}\right)^{1/2} - 1\right]\frac{D_a}{\tan(10°)} + L_a. \qquad \text{............(I-34)}$$

One remaining practicality is the numerical integration of pressure isobars defined by Equations (25), (27), (29), and (31). If these equations are evaluated at a set of discrete distances L_i, the corresponding radii r_i define adjacent conical frusta with unique slopes as shown in Figure I-3. The analytic formula for the frustum of a cone is given by

$$V_i = \pi\left[\frac{1}{3}m_i^2 L^3 + m_i\left(r_{i+1} - m_i L_{i+1}\right)L^2 + \left(m_i^2 L_{i+1}^2 - 2r_{i+1} m_i L_{i+1} + r_{i+1}^2\right)L\right]_{L_i}^{L_{i+1}} \qquad \text{.........(I-35)}$$

where the linear slope of the sides of the conical segment $m = \frac{r_{i+1} - r_i}{L_{i+1} - L_i}$. The total volume under the isobar is approximated by the sum $V_{isobar} = \sum V_i$ and can be refined to any desired accuracy by evaluating the pressure-isobar equations at finer resolution.

The total volume of an isobar should be multiplied by a factor of 2 when double-ended breaks of equivalent upstream pressure are being considered, and finally, converted to a volume-equivalent sphere by the formula

$$R_{sphere} = \left(\frac{3}{4\pi}V_{isobar}\right)^{1/3}. \qquad \text{............(I-36)}$$

I.4 DERIVATION OF ASYMPTOTIC-PLANE AREA

To obtain Equation (I-8) for the jet diameter D_a at the asymptotic plane, force balances are applied to the two control volumes shown in Figure I-4 in a manner analogous to the derivation of the thrust force given by Equation (I-2). In the figure, a plate is positioned normal to the flow at the asymptotic plane. The force required to hold the plate in static equilibrium is notated F_e. The fluid deflected by the plate is assumed to exit the control volume isotropically in a plane oriented parallel to the face of the plate. Exit flow is not represented in the figure.

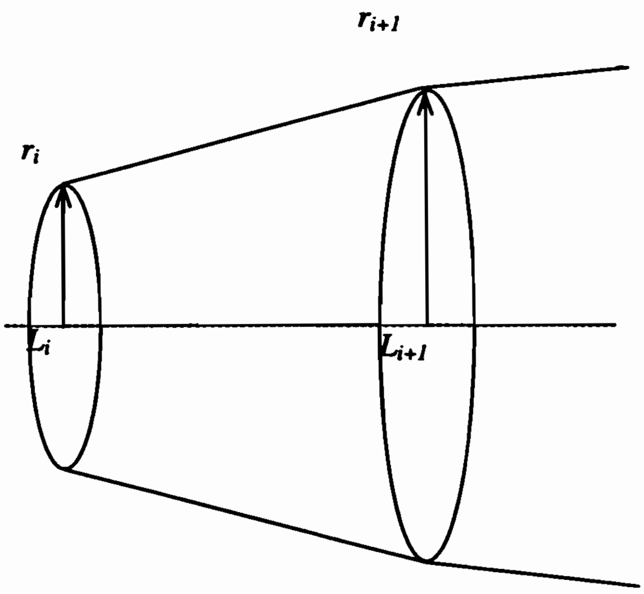

Figure I-3. Linear Segmentation of Jet Cross Sections for Numerical Volume Integration

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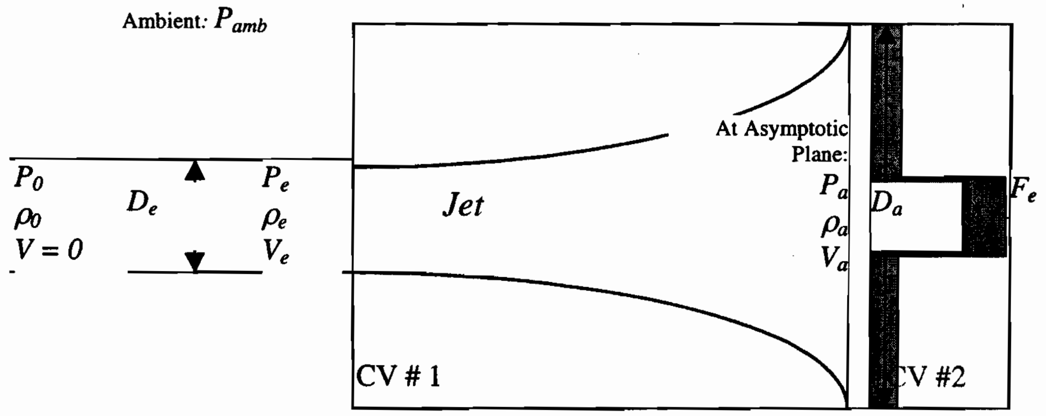

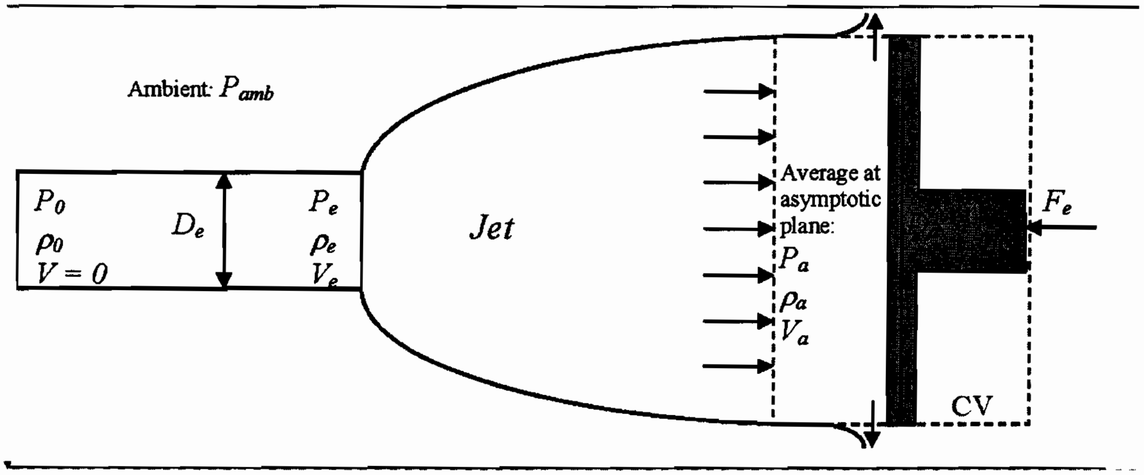

Figure I-4. Control-Volume Force Balance on a Rigid Plate at the Asymptotic Plane Used to Derive Eq. (C-3) in the ANSI Standard

It is assumed in Appendix C of the standard (p. 52) that the fluid does not begin to interact with the surrounding environment until after it crosses the asymptotic plane. Hence, no energy is supplied to or removed from the jet in the region upstream of the control volume in Figure (I-4). Therefore, the entire jet force will be recovered on a target at this distance.

The jet characteristics at the asymptotic plane – fluid density ρ_a, velocity v_a, and static pressure P_a – are not expected to be uniform, so to render the force balance for the control volume tractable, these properties are averaged over the jet cross section. The force balance in the direction of the jet flow may hence be written as

$$F_e = \left(P_a - P_{amb}\right)A_a + \frac{1}{g_c}\left(v_a \frac{d}{dt}m_a + m_a \frac{d}{dt}v_a\right), \qquad \text{...........(I-37)}$$

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where $A_a = \pi D_a^2 / 4$ is the jet area at the asymptotic plane and m_a is the mass of the fluid located within the control volume.

For steady flow, $dv_a/dt = 0$. The rate at which mass enters the control volume, dm_a/dt, is simply the total mass flow crossing the asymptotic plane and is given by

$$\frac{dm_a}{dt} = \rho_a v_a A_a. \qquad \text{............(I-38)}$$

Hence, the force balance simplifies to

$$F_e = (P_a - P_{amb})A_a + \frac{1}{g_c}\rho_a v_a^2 A_a. \qquad \text{............(I-39)}$$

Since no mass escapes the jet between the break location and the asymptotic plane, the mass flow rates at the break and at the asymptotic plane must be equal, i.e.,

$$\rho_a v_a A_a = \rho_e v_e A_e. \qquad \text{............(I-40)}$$

This relation may be employed to eliminate v_a in the force balance.

As mentioned in the discussion following Equation (I-11), the static pressure at the asymptotic plane is generally taken to be equal to P_{amb}. Setting P_a equal to P_{amb} yields

$$F_e = \frac{1}{g_c}\frac{\rho_e^2 v_e^2 A_e^2}{\rho_a A_a}. \qquad \text{............(I-41)}$$

Since the the full jet thrust force is recovered, this evaluation of F_e may be set equal to that obtained Equation (I-2) to give the result

$$\frac{A_a}{A_e} = \frac{\rho_e^2 v_e^2}{g_c \rho_a} \cdot \frac{1}{(P_e - P_{amb}) + \frac{1}{g_c}\rho_e v_e^2}. \qquad \text{............(I-42)}$$

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The second fraction in this equation is recognized by comparison with Equation (I-5) as being equal to $1/(C_T P_0)$. Making use of the mass flux definition $G_e = \rho_e v_e$, leads to the expression for the jet area at the asymptotic plane given in the standard,

$$\frac{A_a}{A_e} = \frac{G_e^2}{g_c \rho_a C_T P_0}. \qquad \text{............(I-43)}$$

The standard recommends that the density ρ_a at the asymptotic plane be evaluated using the local static pressure P_a and the system stagnation enthalpy h_0 rather than the local static enthalpy h_a. Therefore, it is implicitly assumed that the dynamic enthalpy at the asymptotic plane, $v_a^2/2$, is small.

An inconsistency is noted here because P_a in the ANSI jet model – as governed by equation (I-10) – is not always equal to P_{amb}, yet the asymptotic plane area is always computed as if this were the case. For slightly subcooled, saturated, or two-phase upstream conditions, application of Equation (I-10) leads to a value for P_a that is less than P_{amb}. Although the physical reasoning behind Equation (I-10) is not documented in the standard, it appears to correct for cases in which the dynamic enthalpy is non-negligible. This development further confirms that only longitudinal pressures are being computed for P_{jet}, at least at the asymptotic plane, and probably everywhere within the jet envelope.

I.5 CRITICAL FLOW MODELS

I.5.1 Discharge Mass Flux

Results produced by the jet model are sensitive to the value assigned to the mass flux discharged from the break plane, G_e [lbm/ft^2/s]. The area of the jet at the asymptotic plane A_a [ft^2], i.e., the cross sectional area reached by the jet following free (isentropic) expansion, is proportional to G_e^2. Thus, G_e is indirectly specified via Figures C-4 and C-5 in the standard, which plot the ratio of the asymptotic area to the break plane area A_a/A_e for upstream conditions ranging from 50°F subcooled liquid to saturated vapor. Aside from difficulties inherent in recovering numerical values from coarsely resolved plots, use of these figures is not recommended for two reasons:

1. The range of upstream stagnation conditions covered by the plot – extending only to 50oF subcooling – is insufficient. Typical cold-leg conditions in a PWR might entail subcooling of 100oF or more.
2. The origin of the results is unclear. Which model was used to evaluate the relevant mass fluxes and thrust coefficients? Without this information, there can be no confidence that the rest of the model will be applied in a self-consistent manner.

Therefore, we strongly concur with the recommendation given in the standard (p. 57) that a two-phase critical flow model be employed to evaluate G_e. Two models that are in widespread use are cited: the homogeneous equilibrium model (HEM)† and the

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† For a discussion of practical considerations surrounding implementation of the HEM, as well as a tabulation of results for a wide range of upstream conditions, see Ref. [HAL80].

Henry-Fauske model [HEN71]. The standard provides a loose recommendation regarding the applicability of the models as a function of upstream stagnation properties: the HEM for saturated or two-phase and Henry-Fauske for subcooled conditions.

Several pitfalls await a naïve application of this guidance. To facilitate the exposition of these pitfalls, it is useful first to provide a simplified description of the physics inherent in each of the models.

Under the HEM, the phases are assumed to be in thermodynamic equilibrium and to remain well mixed. The relative velocity between the phases is therefore assumed to be zero. External heat transfer, wall roughness, and other interactions with the environment are neglected so that the expansion is isentropic.

Given these assumptions, the first law of thermodynamics is applied to the homogenized fluid. Combined with the definition of the mass flux, the first law yields an expression for G_e in terms of the mixture's static properties at the choked point. The critical mass flux is defined as the value of G_e that maximizes this expression. Numerical solution of the HEM is thus an iterative process, entailing a search over the space of static state points that preserve the upstream stagnation entropy.

The Henry-Fauske model preserves some of the assumptions made under the HEM, namely that the mass flux may be expressed as a function of the thermodynamic state at the throat, that the critical mass flux can be obtained by maximizing this function, and that the expansion is isentropic. However, Henry and Fauske argue that the assumptions of homogeneous mixing and thermodynamic equilibrium during the expansion are unrealistic given the short time scales involved. Rather, interphase mass transfer is constrained such that the quality x_t at the throat is equal to the upstream stagnation quality x_0. Heat transfer during the expansion is also assumed negligible; the liquid-phase temperature T_{ft} at the throat is held fixed at the upstream liquid temperature T_{f0}. The temperature of the vapor phase, if it is present, is allowed to vary. The heat- and mass-transfer rates at the throat are treated as significant, and expressions for these are developed assuming polytropic vapor behavior.

In practice, the Henry-Fauske model is implemented by solving a transcendental equation for the static pressure at the throat that maximizes mass flux. Both Henry-Fauske and the HEM are evaluated through iterative procedures, with thermodynamic properties being queried upon each iteration. Therefore, the models were coded as a series of FORTRAN subroutines, driven by a MATLAB control function, that directly couple with the FORTRAN implementation of the NIST/American Society of Mechanical Engineers (ASME) steam tables [HAR96] when fluid properties are required. The results obtained from the software were successfully validated against those presented in Refs. [Hal80] and [Hen71]. These programmed routines allow a thorough assessment of the practical ramifications of using each model within the ANSI jet-modeling framework.

The standard does not provide guidance with regard to critical flow modeling for superheated conditions. The simplest approach would be to treat the steam as an ideal gas and apply the appropriate equation of state. This treatment was attempted and found to be highly inadvisable for the slightly superheated states that are of most relevance to the present application. Two qualitative observations support this conclusion: first, when the upstream superheat is small the flow at the choked location is in fact two phase; second, slightly superheated high-pressure steam does not exhibit the typically assumed idealized properties (e.g., a specific heat ratio of 1.3) so that transitions evaluated using the ideal gas law would not preserve entropy. These considerations lead us to recommend that the HEM be used treat the superheated state points that may arise in this application.

As mentioned above, the standard does provide guidance for two-phase and single-phase liquid stagnation state points. Specifically, it recommends the use of HEM for saturated and Henry-Fauske for subcooled upstream conditions. We believe that the Henry-Fauske model should, in fact, be employed for both of these regimes. This recommendation stems from several considerations, as outlined below.

Critical mass fluxes predicted by the HEM and Henry-Fauske models exhibit their most significant disagreement at precisely the transition point recommended in the standard, i.e., for saturated-liquid upstream conditions. Figure I-5 and Figure I-6 provide contour plots of G_e as obtained from the two models for subcooled vessel stagnation conditions.

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In figures showing flow properties for subcooled state points, the stagnation temperature is varied on the x axis and pressure on the y axis. The regions between contour lines of constant G_e are shaded for ease of delineation. Because the domain of validity of the flow models does not extend to superheated conditions, pressure and temperature combinations that lie within this regime are blanked out on the plots. Mass fluxes for saturated upstream conditions are shown in Figure I-7 and Figure I-8. In these plots, G_e is calculated at several saturated (temperature, pressure) state points as a function of the vessel quality.

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Figure I-9 and Figure I-10 display the variation between the HEM and Henry-Fauske mass fluxes. It can be seen from these figures that discrepancies of 50% or more exist for saturated liquid upstream conditions and that significant variations persist for slightly subcooled and low-quality two-phase stagnation conditions. This disagreement follows from a variation in the assumptions regarding interphase mass transfer. Because the quality is held fixed under the Henry-Fauske model, the discharge is almost entirely in the liquid phase. Under the HEM, however, heat and mass transfer between the phases is allowed and the discharge has a quality that is significantly greater than zero. This discharge possesses a lower density and higher velocity than that predicted by Henry-Fauske. It can be shown numerically that the HEM mass flux prediction will be lower than that of Henry-Fauske for the slightly subcooled, saturated liquid, and low-quality upstream conditions in which the HEM prediction of discharge quality is markedly higher than that of Henry-Fauske.

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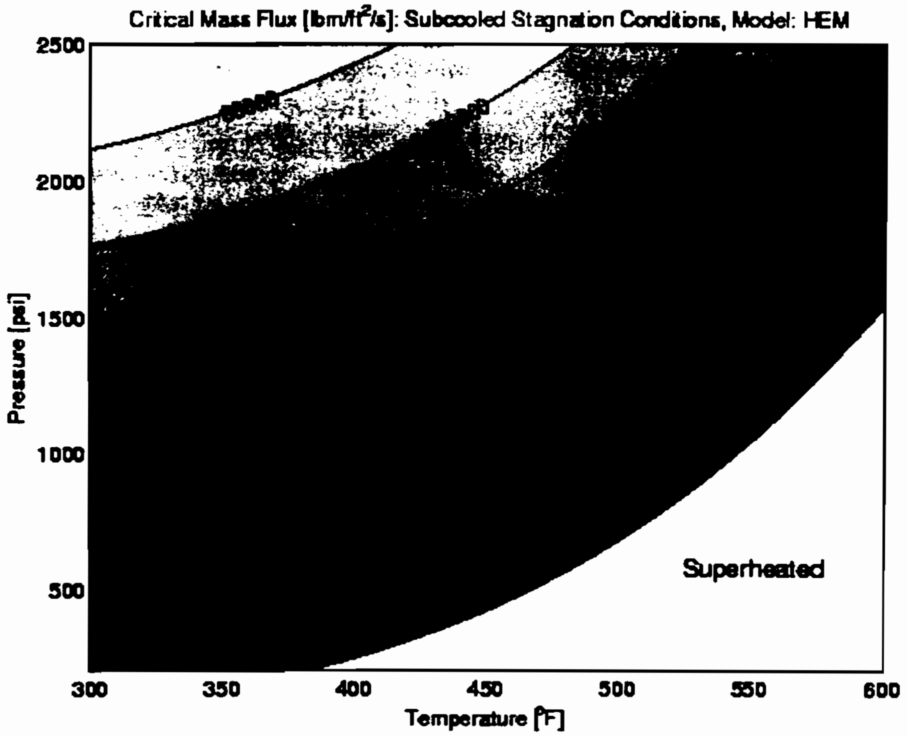

Figure I-5. HEM Critical Mass Flux, Subcooled Stagnation

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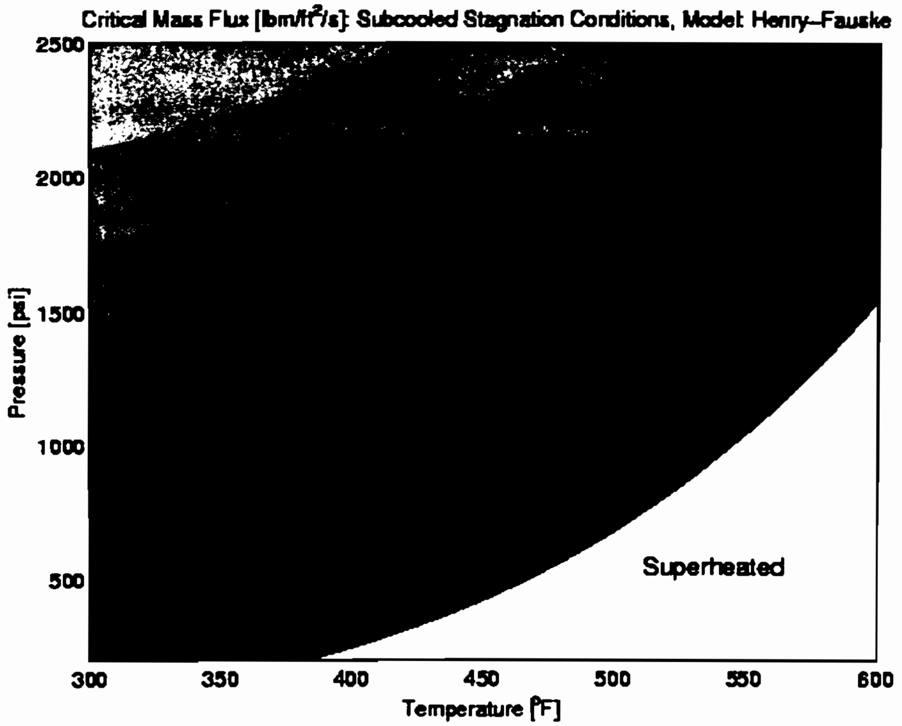

Figure I-6. Henry-Fauske Critical Mass Flux, Subcooled Stagnation

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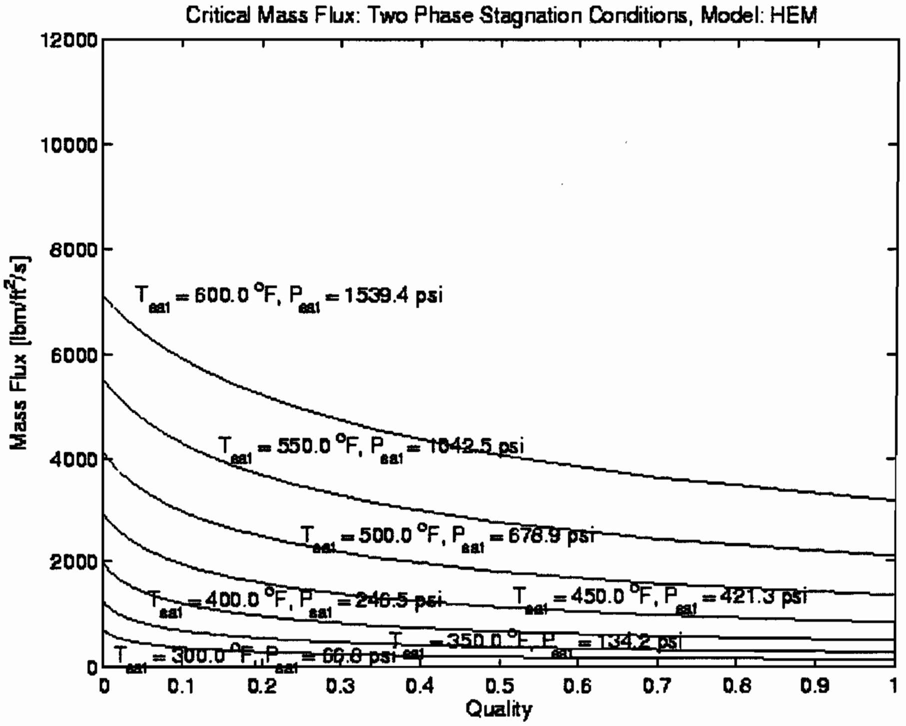

Figure I-7. HEM Critical Mass Flux, Saturated Stagnation

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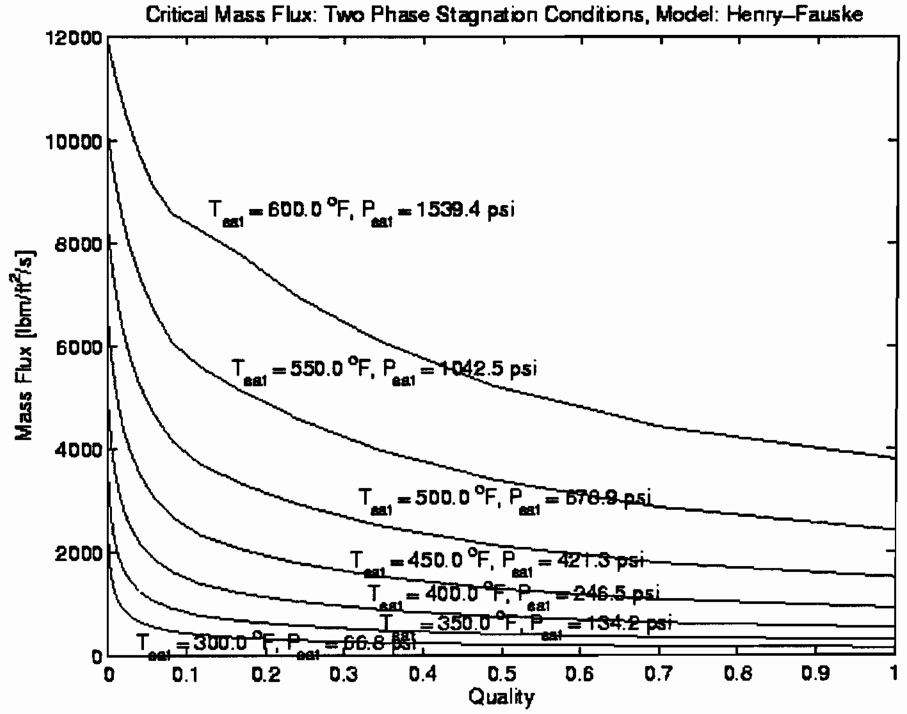

Figure I-8. Henry-Fauske Critical Mass Flux, Saturated Stagnation

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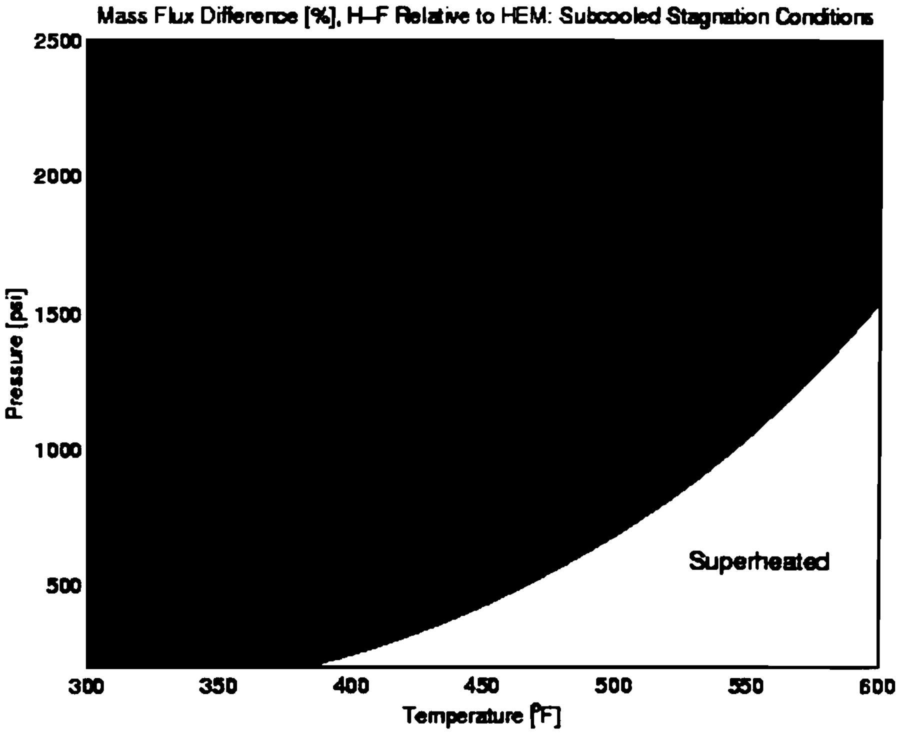

Figure I-9. Mass Flux Difference, Subcooled Stagnation

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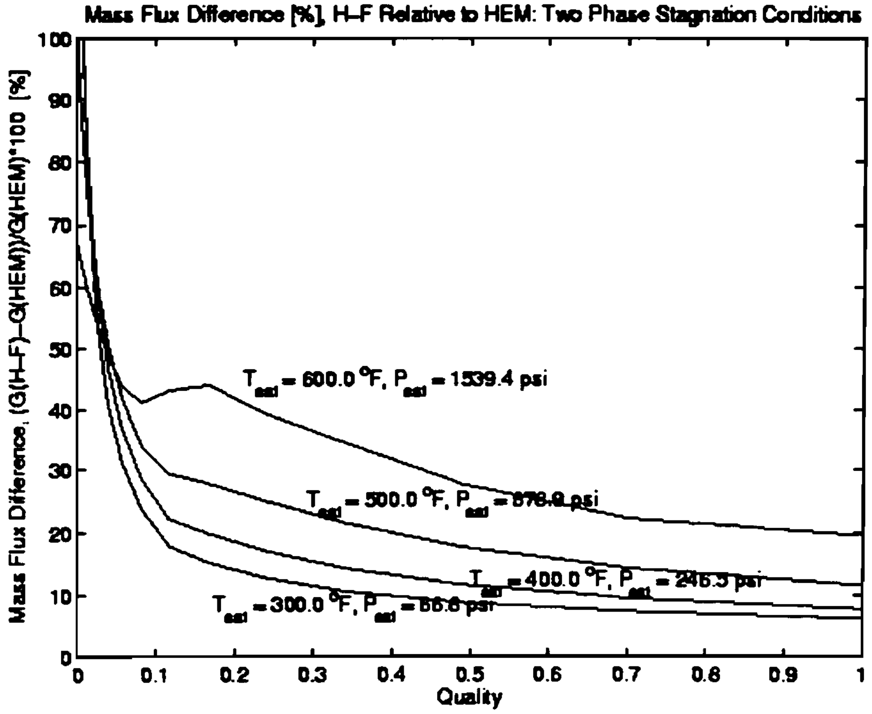

Figure I-10. Mass Flux Difference, Saturated Stagnation

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If the advice of the standard is followed, then a significant discontinuity would be observed when the critical flow model transitions from the HEM to Henry-Fauske. The nature and magnitude of this discontinuity is explored further below. Although users of the jet model are in practice unlikely to observe this discontinuity, because during a blowdown, the transition might only occur after significant pressure drops, we see no compelling reason to preserve it. The issue then becomes one of selecting the model that offers the best fidelity to available data. The figures show that the HEM and Henry-Fauske offer comparable predictions for highly subcooled as well as high-quality two-phase conditions. This is to be expected because under these conditions, both models predict essentially monophasic fluid properties at the throat and the detailed treatment of the interphase heat- and mass-transfer rates offered by Henry-Fauske does not come into play. The benchmarking results reported in Ref. [Hen71] lead us to conclude that the Henry-Fauske model exhibits superior agreement to the data under low-quality two-phase and saturated liquid conditions. This alone is sufficient reason to adopt Henry-Fauske; further evidence may be found from an examination of a second major input to the ANSI jet model, the thrust coefficient.

I.5.2 Direct Evaluation of Thrust Coefficients

The thrust coefficient $\underline{C_T}$ acts as a surrogate for the jet thrust force, which is not explicitly called for as an input to the ANSI model. This discussion will address only the steady-state thrust coefficient for frictionless, unrestricted flow, but its conclusions can be generalized to include those cases as well. Regardless of upstream conditions, the thrust coefficient is used to correlate the thrust force $\underline{T}$, upstream stagnation absolute pressure $\underline{P_0}$, ambient pressure $\underline{P_{amb}}$, and break area $\underline{A_e}$ by the expression

$$\underline{T = C_T(P_o - P_{amb})A_e.} \qquad \text{...........(I-44)}$$

Calculation of the thrust coefficient requires knowledge of local flow conditions at the break. Because these are unknown unless a critical flow model such as the HEM or Henry-Fauske is used to compute them, pp. 35 – 45 of the standard provide a series of correlations and figures that may be used as surrogates. Because both Henry-Fauske and the HEM were implemented for the current review, the results obtained from these models will be compared with the recommendations provided in the standard.

The thrust force may be computed by calculating the force that must be exerted to hold in static equilibrium a plate positioned normal to the flow directly at the break point. This thrust is given by

$$\underline{T = (P_e - P_{amb})A_e + \frac{1}{g_c}\rho_e v_e^2 A_e,} \qquad \text{...........(I-45)}$$

where the static pressure $\underline{P_e}$, fluid density $\underline{\rho_e}$, and flow velocity $\underline{v_e}$ are evaluated at the exit. Combining the above equations yields an expression for the thrust coefficient:

$$C_T = \frac{1}{P_0 - P_{amb}}\left(\frac{1}{g_c}\rho_e v_e^2 + (P_e - P_{amb})\right). \qquad \text{............(I-46)}$$

Figure I-4 through Figure I-7 show thrust coefficients computed using pressures and fluid properties evaluated from the HEM and Henry-Fauske models. Regardless of the model, the value of C_T approaches 2.0 for incompressible, highly subcooled liquid and ~1.26 for saturated steam. These results agree with theory and are recommended for use in the standard.

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For subcooled flashing upstream conditions, the standard on p. 42 recommends use of the curve fits presented by Webb [WEB76]. Based on an enthalpy normalization factor

$$h^* = \frac{h_0 - 180}{h_{sat} - 180}, \qquad \text{............(I-47)}$$

where h_0 [Btu/lbm] is the upstream stagnation enthalpy and h_{sat} [Btu/lbm] is the saturated water enthalpy at the stagnation pressure, the correlation is evaluated as

$$C_T = 2.0 - 0.861 h^{*2} \text{ for } 0 \le h^* < 0.75 \qquad \text{............(I-48)}$$

and

$$C_T = 3.22 - 3.0 h^* + 0.97 h^{*2} \text{ for } 0.75 \le h^* \le 1.0. \qquad \text{............(I-49)}$$

For saturated or superheated steam, the standard recommends a thrust coefficient of

$$C_T = 1.26 - P_{amb}/P_0 . \qquad \text{............(I-50)}$$

For two-phase steam-water mixtures, the standard provides only a figure that does not address relevant PWR break conditions, and for nonflashing water jets with temperatures less than the saturation temperature at ambient pressure and pressures greater than ambient, the standard recommends that

$$C_T = \frac{2}{1 + fL/D}, \qquad \text{............(I-51)}$$

where the Fanning friction factor f is normally assumed to be zero for conservatism. The ratio L/D represents a dimensionless flow-path length based on the characteristic length and diameter of the piping between the assumed thermodynamic reservoir and the break location.

Webb claims, and our calculations verify, that his correlations agree with values computed from the Henry-Fauske model to within 3% for upstream stagnation pressures ranging from 300 to 2400 psia. The standard does not clearly state this range of applicability. Webb's correlation is recommended when a computational implementation of a critical flow model is unavailable, but two inconsistencies require clarification.

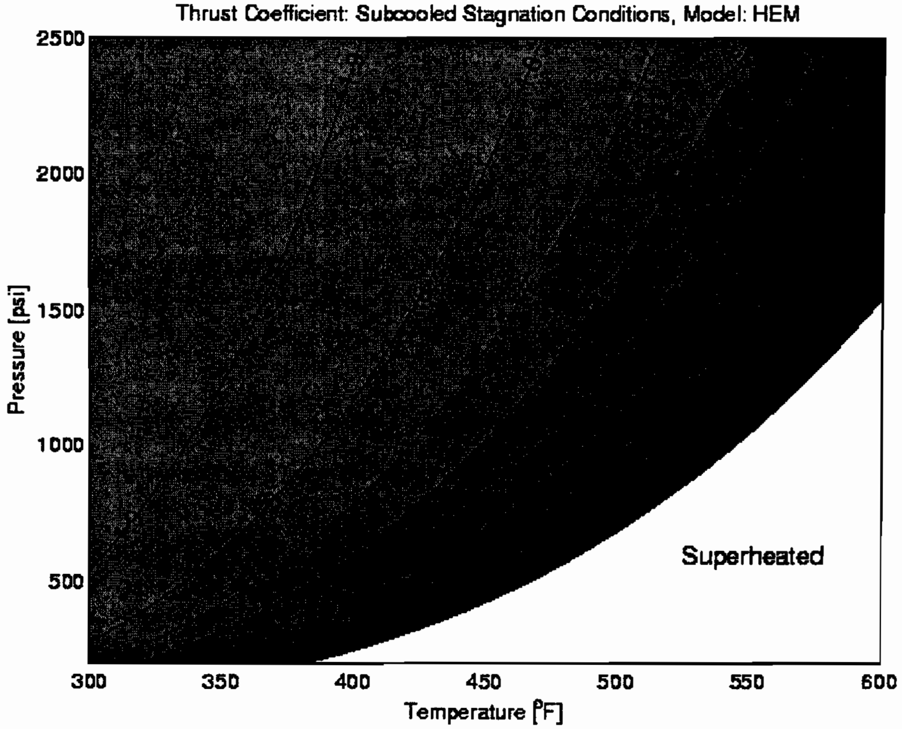

Figure I-11, HEM Thrust Coefficient, Subcooled Stagnation

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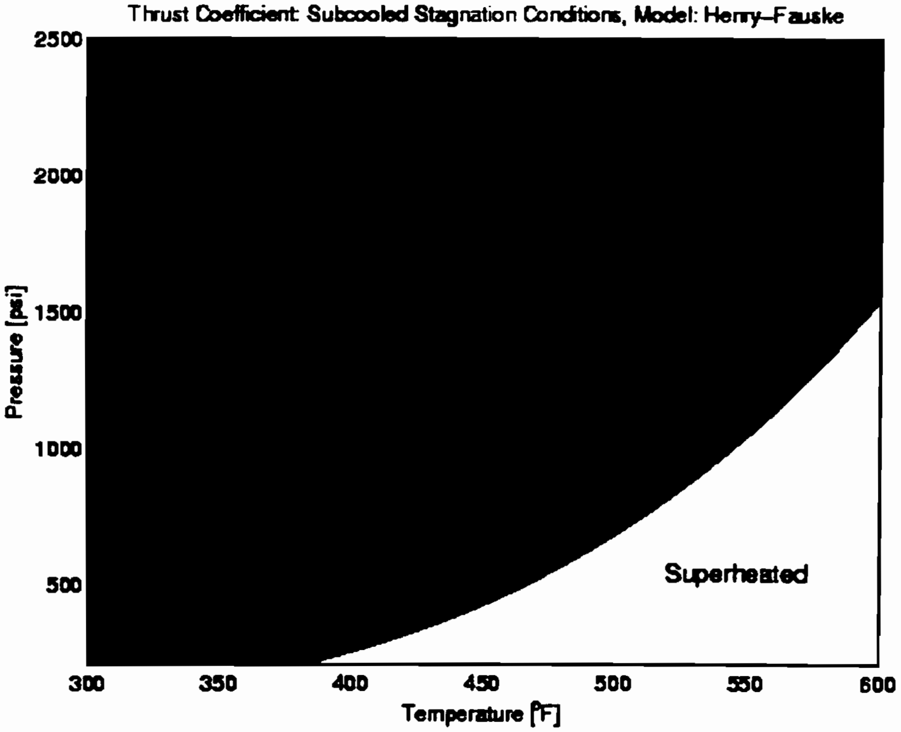

Figure I-12, Henry-Fauske Thrust Coefficient, Subcooled Stagnation

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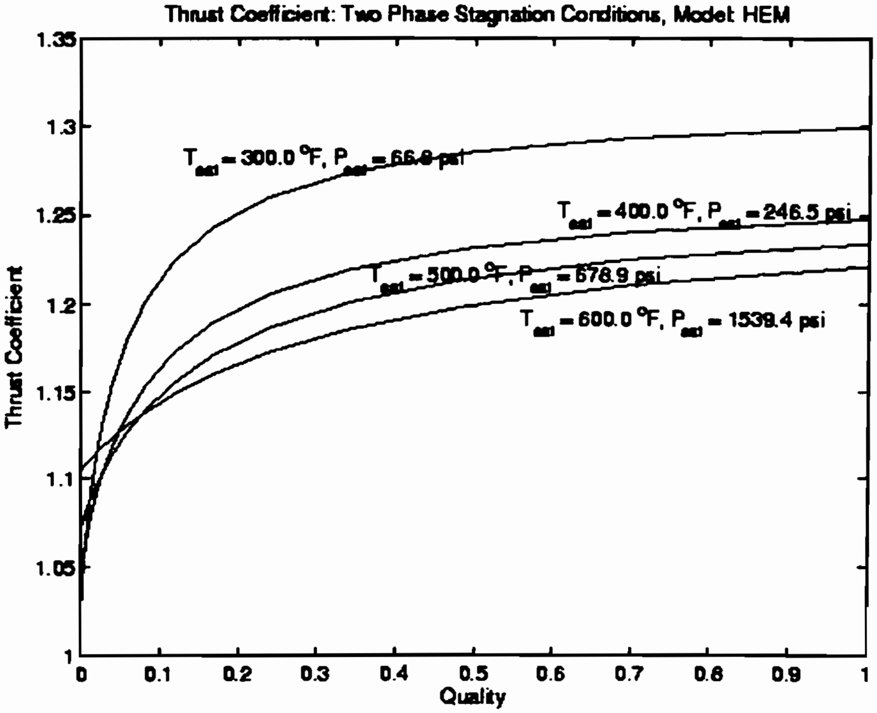

Figure I-13, HEM Thrust Coefficient, Saturated Stagnation

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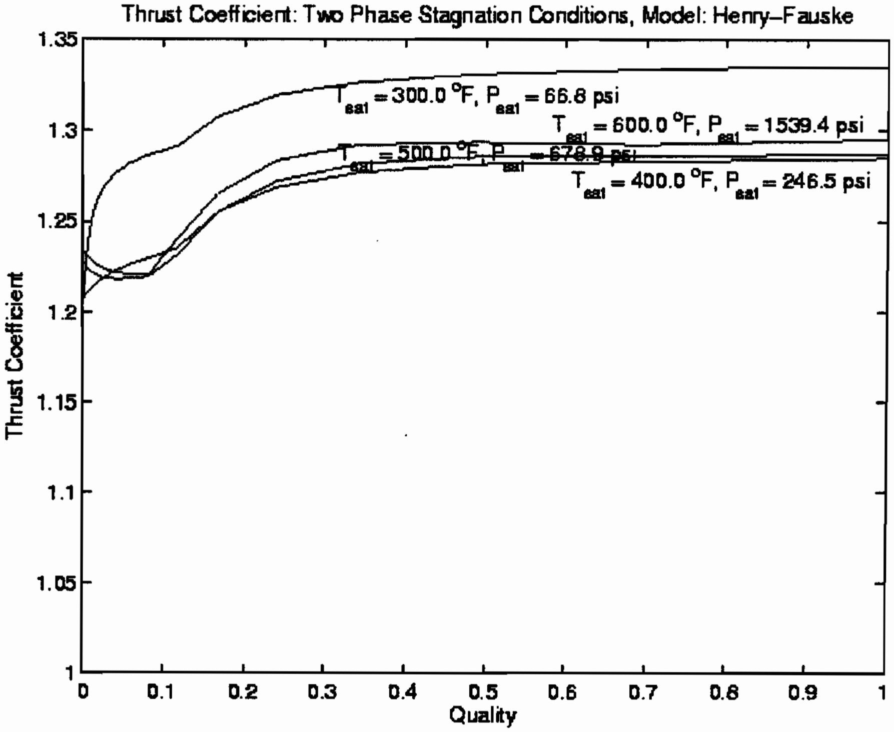

Figure I-14. Henry-Fauske Thrust Coefficient, Saturated Stagnation

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In presenting Webb's model, the standard neglects to clarify the "180" figure against which the enthalpy is nondimensionalized. This is, in fact, the enthalpy of saturated water at atmospheric pressure, 14.7 psi. It may be justifiably claimed that, during a blowdown, the ambient containment pressure might vary from below atmospheric to significantly above atmospheric. Changes in P_{amb} cannot be accounted for by Webb's model; however, C_T evaluated from the force balance varies weakly with P_{amb}. This effect is not large: even for highly subcooled conditions at the lower end of the range of validity of Webb's correlation, $P_0 = 300$ psia, neglecting P_{amb} altogether changes the thrust coefficient evaluated from the force balance by less than 5%.

The standard also places insufficient emphasis on the fact that Webb's correlation is obtained from calculations using the Henry-Fauske model. Because this is the case, employing HEM-derived mass fluxes with thrust coefficients obtained from this correlation propagates of a significant inconsistency. Figure I-8 shows that significant deviation exists between thrust coefficients computed from the outlet conditions provided by the two critical flow models. The use of Henry-Fauske-derived thrust coefficients with HEM mass fluxes will result in overprediction of damage radii. This follows because the larger Henry-Fauske thrust coefficient implicitly imposes a higher flow density, velocity, and/or static pressure at the break plane.

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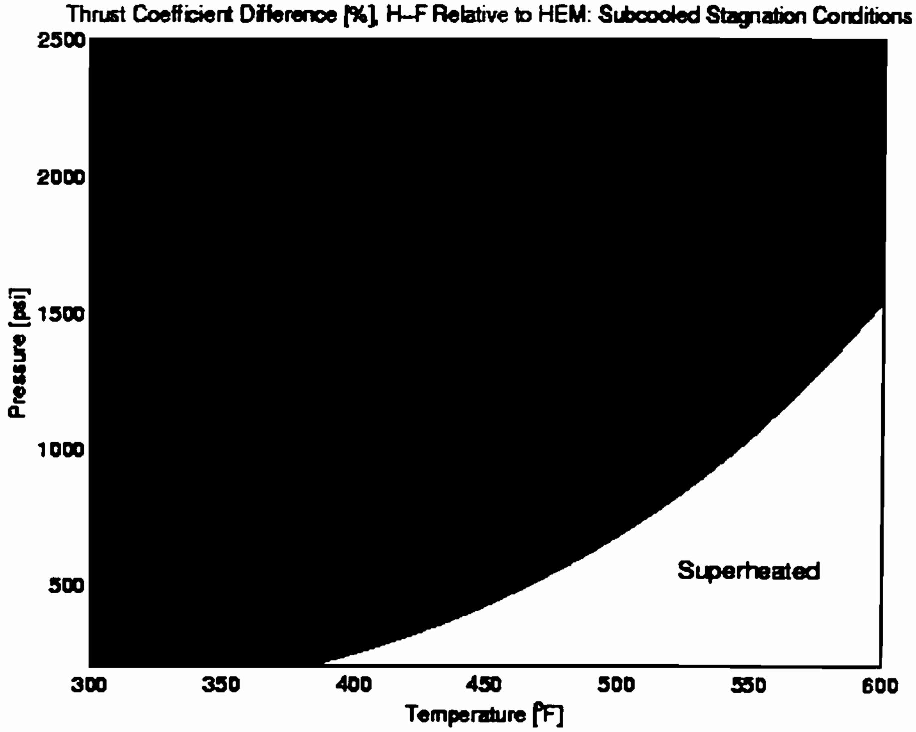

Figure I-15, Thrust Coefficient Difference, Subcooled Stagnation

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I.5.3 Effects of Flow Models on Jet Behavior

While the sensitivity of the jet pressure contour map in its entirety to variations in C_T is too complicated to permit analytic treatment, the effect of variation of C_T on conditions at the asymptotic plane can be used for illustration. Equation (I-43) shows that the jet area A_a at the asymptotic plane is inversely proportional to C_T. However, from conservation of mass, Equation (I-40), the average flow velocity at the asymptotic plane v_a is inversely proportional to A_a and, thus, directly proportional to C_T. This conclusion can be drawn because the average fluid density ρ_a at the asymptotic plane depends, in the ANSI formulation, only upon upstream stagnation conditions. The dynamic pressure of the fluid, which is proportional to the square of its velocity, thus varies as C_T^2. The results of decreased jet cross sectional area and increased velocity from the larger Henry-Fauske thrust coefficient will be a narrower, more penetrating jet and larger volume-equivalent radii at a given damage pressure.

In fact, it can be seen from Figure I-6 that the thrust coefficient for upstream conditions at or near saturation as derived from the HEM is significantly lower than the value of 1.26 recommended in Figure B-5 of the Standard. The inconsistency inherent in use of the 1.26 value with the HEM mass flux would again result in overprediction of volume-equivalent radii. This additional consideration strengthens our recommendation

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that the Henry-Fauske method be employed for all flow regimes when performing the calculations outlined in the standard.

As mentioned above, the critical mass flux G_e derived from the HEM will be smaller, significantly so for stagnation conditions lying near the liquid saturation line in (P,h) space, than that obtained from Henry-Fauske. Because this is the case, it is also useful to address the behavior at the asymptotic plane when G_e is varied with C_T being held constant. Following the same reasoning pursued above when the thrust coefficient was varied, we see that the jet area at the asymptotic plane varies as G_e^2. The average jet velocity at that location v_a, on the other hand, behaves as $v_a = kG_e/A_a$ so that $v_a \sim 1/G_e$. Thus, a seemingly paradoxical conclusion is reached, namely that reducing the mass flux while holding the thrust coefficient constant increases the velocity at the asymptotic plane and might *increase* the volume-equivalent radii.

Although this thought experiment is not conclusive or comprehensive – the location of the asymptotic plane, for instance, also depends on G_e and C_T and has not been taken into account – numerical computations verify its conclusions. Table I-1 shows critical flow model results for five of the upstream conditions given in Table I-2. The conditions selected from that table are #8, PWR Hot Leg Initial; #1, PWR Cold Leg Initial; #2, PWR Cold-Leg Blowdown; #9, BWR Hot-Leg; and #11, Main Steam Line. All three PWR stagnation states are subcooled; the BWR state is two-phase with a quality of 0.15 and the steam line case is superheated by 35ºF. In addition to the mass flux G_e, thrust coefficient C_T and discharge velocity v_e obtained, the volume-equivalent damage radii for the 10 and 150 psig contours are also shown. It might be intuitively expected that the Henry-Fauske model is the more conservative when calculating damage radii because it predicts critical mass fluxes and thrust coefficients that are greater than those of the HEM, but, as shown in the table, particularly for initial conditions nearing saturation, this is not the case.

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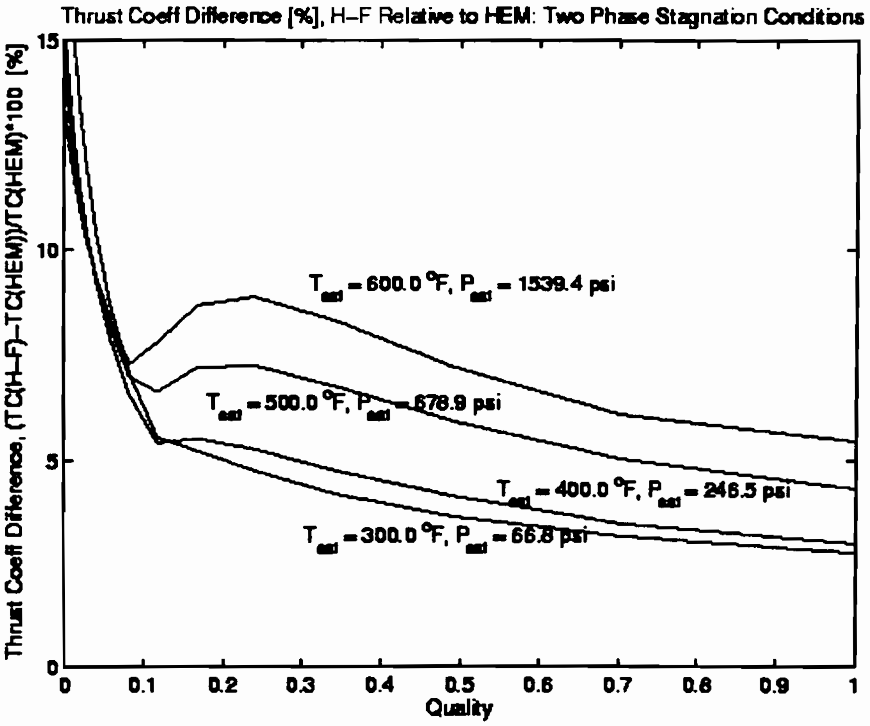

Figure I-16, Thrust Coefficient Difference, Saturated Stagnation

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Table I-1. Critical Flow Model Results and Their Effect on Volume-Equivalent Damage Radii

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	Critical Mass Flux $\underline{G_e}$ [lbm/ft^2/s]		Thrust Coefficient $\underline{C_T}$ [--]			Break Flow Velocity $\underline{v_e}$ [ft/s]		150-psig* Damage-Pressure Radius [pipe diameters]		10-psig* Damage-Pressure Radius [pipe diameters]	
	HEM	**H-F**	**HEM**	**H-F**	**Webb****	**HEM**	**H-F**	**HEM**	**H-F**	**HEM**	**H-F**
1. Cold Leg Initial (2250 psia, 530 F)	**24850**	**25330**	**1.62**	**1.64**	**1.63**	**522**	**527**	**1.48**	**1.48**	**12.00**	**12.04**
2. Cold-Leg Blowdown (393 psia, 291 F)	**13370**	**13390**	**1.88**	**1.89**	**1.90**	**232**	**232**	**0.96**	**0.96**	**4.42**	**4.43**
8. Hot Leg Initial (2250 psia, 630 F)	**11840**	**15400**	**1.17**	**1.28**	**1.28**	**296**	**382**	**1.60**	**1.59**	**11.14**	**11.07**
9. BWR Hot Leg (1040 psia, 550 F, X = 0.15)	**3920**	**5260**	**1.16**	**1.26**	**N/A**	**178**	**158**	**1.11**	**1.12**	**7.81**	**7.80**
11. Main Steam Line (910 psia, 570 F)	**1800**	**N/A**	**1.24**	**N/A**	**N/A**	**464**	**N/A**	**1.08**	**N/A**	**7.58**	**N/A**

* Damage-pressure radii are given as multiples of the break diameter. They are obtained by constructing spheres with volume equal to the volume enclosed by a given jet stagnation pressure contour. See Section I.3 for further elaboration.

** Shown for purposes of comparison only; not used in damage-pressure-radius calculations given in this table.

I.6 SAMPLE CALCULATIONS

The ANSI model presented in the previous sections for predicting stagnation pressures in an expanding jet was implemented in a MATLAB routine called ANSIJet (see Attachment 1 to this appendix). This programming language was selected for convenient interface with steam-table routines available from NIST. Several cases relevant to both PWR initial break and blowdown conditions were evaluated. Two generic BWR state points were also evaluated, as were three cases applicable to steam line flow in secondary loops. Two of these relate to a single-pass Babcock & Wilcox steam generator discharging superheated (by ca. 35° F) steam; the third applies to a Combustion Engineering U-tube heat exchanger and is assumed to yield saturated steam. These conditions are defined in Table I-2 for later reference by case number. Note that Figure I-1 corresponds to the cold-leg initial break condition defined as Case #1.

Table I-2. Comparative Calculation Set Using ANSI Jet Model

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[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
1	cold leg initial[1]	2250	530	Subcooled
2	cold-leg blowdown[1]	393	291	Subcooled
3	cold-leg blowdown[1]	857	351	Subcooled
4	cold-leg blowdown[1]	1321	411	Subcooled
5	cold-leg blowdown[1]	1786	471	Subcooled
6	10% greater pressure than Case 1	2475	530	Subcooled
7	cold leg initial[2]	2250	540	Subcooled
8	hot leg initial[3]	2250	630	Subcooled
9	BWR hot leg[4]	1040	550	0.15
10	BWR cold leg[4]	1040	420	Subcooled
11	main steam line (MSL): Babcock & Wilcox (B&W)[4] – full power	910	570	Superheated
12	B&W MSL: design conditions[4]	1075	603	Superheated
13	MSL: Combustion Engr. Calvert Cliffs[5]	846	525	1.0

[1] From reference [RAO0]

[2] From reference [NEI04]

[3] From reference [DUD76]

[4] From reference [RAH92]

[5] From reference [LOB90]

Jet-pressure isobars for Cases 1 through 6 were integrated over a wide range of values and converted to equivalent spherical diameters. These results are presented in Fig. I-17. Recall that the ANSI-model stagnation pressure is being used as a correlation parameter that corresponds to observed damage in debris generation tests. Use of this correlation is the reason that the Figure I-17 abscissa is labeled as "Damage Pressure." Case 1 represents a previously studied hydraulic condition [RAO02] that will be used as the reference case. Reading from the figure, a damage pressure of 10 psig corresponds to an equivalent jet radius of approximately 12 pipe diameters. Note that equivalent radii climb sharply for damage pressures below 20 psig.

This set of calculations suggests that the state-point pressure of the jet dominates the determination of isobar volumes. Other cases that are not shown in Figure I-17 were bounded by Case 1. Case 7, the nominal PWR cold-leg condition recommended in the GR, was almost indistinguishable from Case 1. Case 8, a nominal hot-leg break condition, was also bounded by the reference case except at damage pressures greater than 120 psig. Hot-leg conditions are much closer to saturation (630ºF vs. 653ºF); therefore, the shapes of the pressure contours change near the core. Case 6 was run as a perturbation check for plants that may at times have higher operating pressures than the nominal value of 2250 psig. Although the pressure increase was 10% higher than the reference, the maximum deviation in spherical volume was only 8%; therefore, a linear adjustment for higher pressure would be conservative in the absence of a full jet-model analysis.

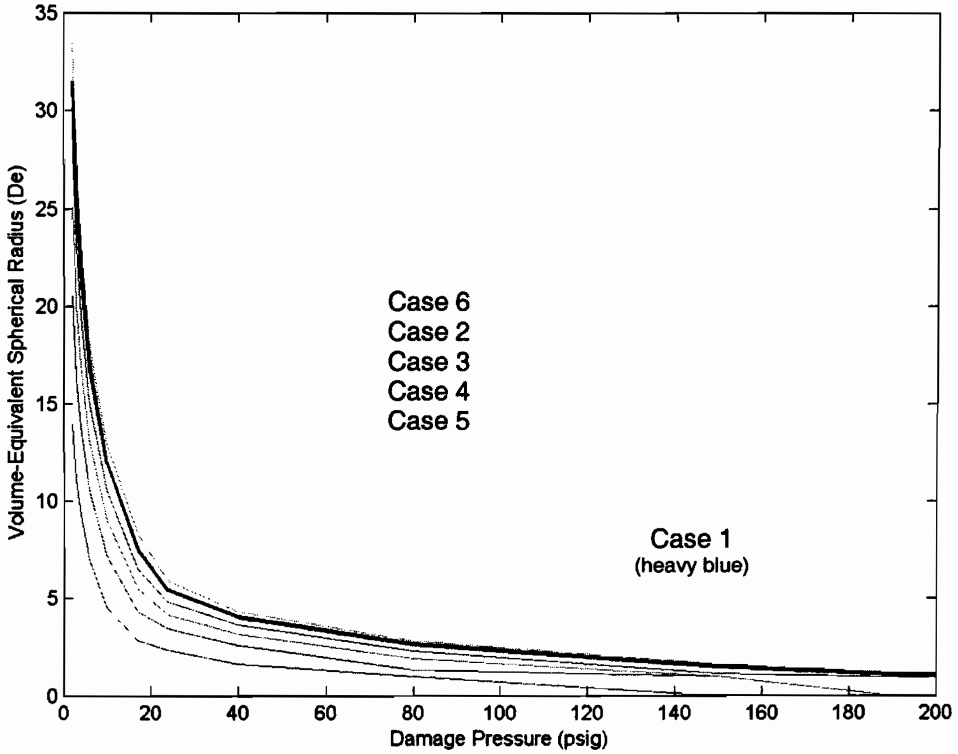

Figure I-17. Comparison of ANSI Jet-Model Equivalent Spherical Radii for Six Initial Break Conditions

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The damage radii associated with the BWR hot-leg and cold-leg conditions of Cases 9 and 10 are shown in Figure I-18. Given the lower stagnation pressures pertinent to BWR coolant, the equivalent radii are, as expected, smaller than was the case for PWR conditions at comparable values of damage pressure. The radii obtained for the three steam line cases are given in Figure I-19. Two of these, Cases 11 and 13, are specified as representative of full power operating conditions. The third, Case 12, is a design specification included to serve as a conservative bounding scenario. Given that the thrust coefficient is nearly invariant at a value near 1.26 for high-quality two-phase and superheated upstream conditions, it appears reasonable to expect damage radii in such regimes to respond linearly to variation in the stagnation pressure. A pressure contour plot for the steam line break condition is provided in Figure I-20. This figure compares to Figure I-1 for PWR cold-leg stagnation conditions. One of the subtle differences between these figures is the higher centerline pressure exhibited by the MSL case to axial distances of about 30 pipe diameters. The steam flow exhibits a narrower jet that is higher-velocity at the centerline, leading to a greater dynamic contribution to the stagnation pressure. Differences in the initial pressure should also be considered when visually comparing Figures I-1 and I-20.

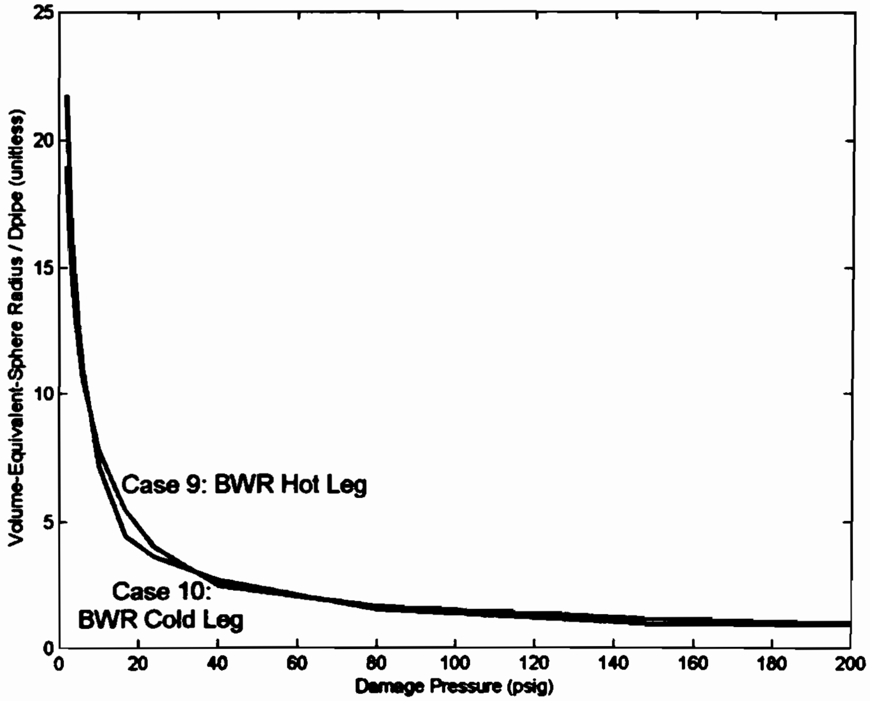

Figure I-18. Comparison of ANSI Jet-Model Equivalent Spherical Radii for BWR Break Conditions

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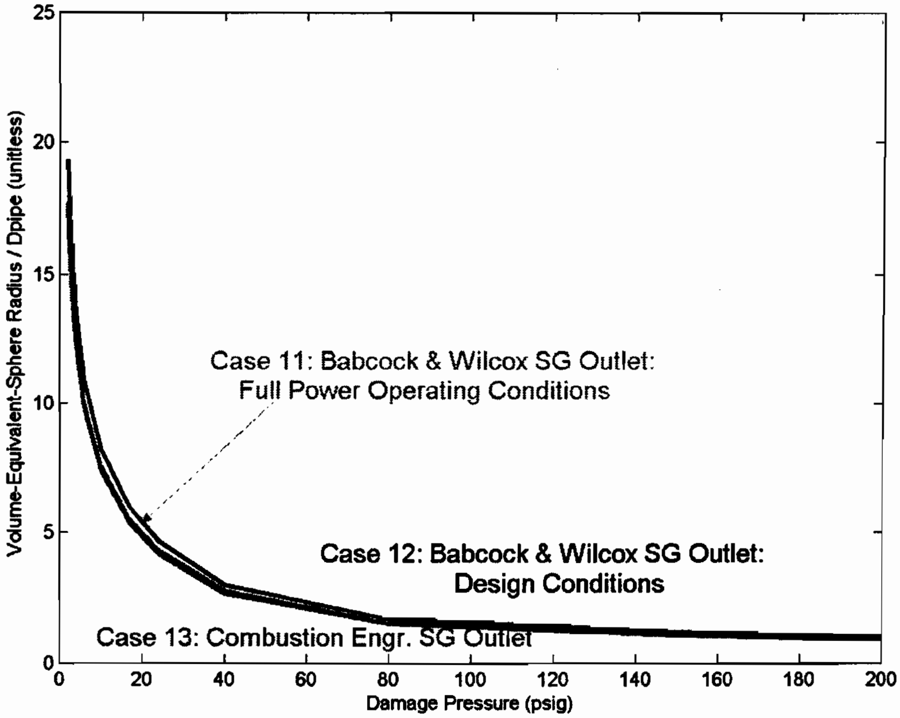

Figure I-19. Comparison of ANSI Jet-Model Equivalent Spherical Radii for Main Steam Line Break Conditions

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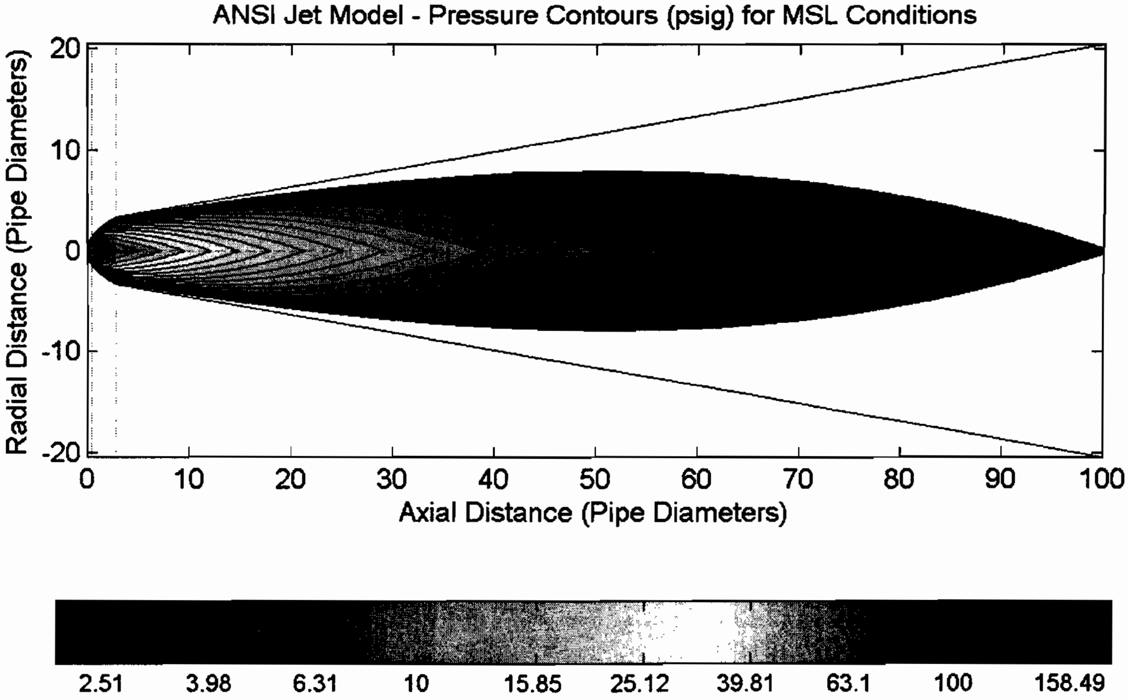

Figure I-20. ANSI Jet-Model Stagnation Pressures for MSL Break Conditions (570ºF, 910 psia)

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Other useful information can be extracted from the jet model in addition to equivalent spherical diameters derived from spatial volume integrals. Appendix D of the ANSI standard suggests that target temperatures can be estimated by evaluating a thermodynamic state point using the jet pressures $\overline{P_j}$ and the initial enthalpy $\overline{h_0}$. Presuming that the model supplies realistic, nonisentropic impingement pressures (at least in the longitudinal direction), this approach will give the temperature of the stationary fluid striking the surface of a large target. Actual target temperatures might vary with internal heat conduction properties and external drag coefficients that affect aerodynamic heating, but it is instructive to compute this approximation nonetheless. Figure I-21 illustrates the isotherm plot corresponding to Case 1 for the reference cold-leg break.

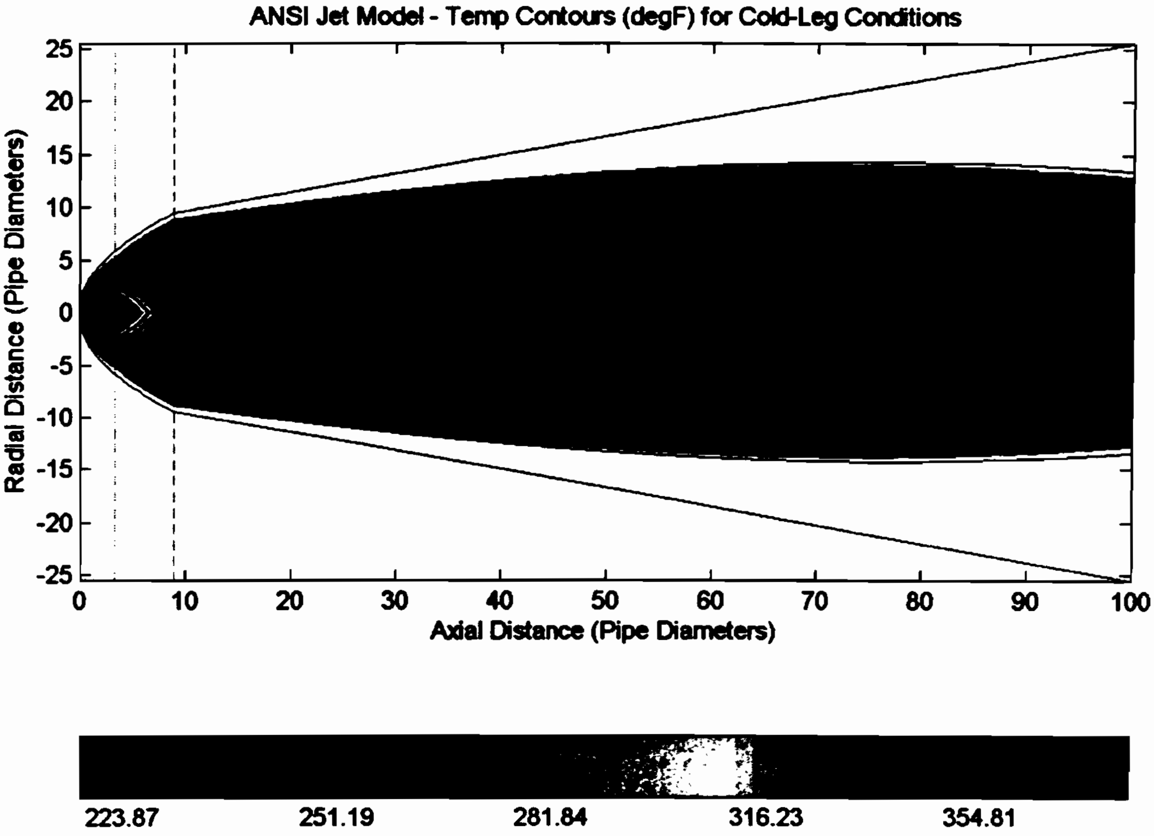

Figure I-21 Isotherm Contours for the Reference Cold-Leg Break at 2250 psia and 530ºF

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The somewhat surprising attribute of the isotherm map is how slowly the impingement temperature changes beyond the range of 10 to 15 pipe diameters downstream of the break. For potential debris-generation mechanisms that are suspected to have important thermal responses, this information can directly benefit both the specification of relevant test parameters and the interpretation of existing test data. For example, a test performed at 280ºF that exhibits good damage resistance demonstrates substantially less spatial vulnerability to high-temperature jets than a test performed at 220ºF. As with pressure contours, isotherm volumes can also be mapped to equivalent spherical volumes, and because the ANSI model exhibits spatial monotonicity (uniformly increasing or decreasing in every direction) in all physical jet properties, there is a unique correspondence between pressure, temperature, and contour volume.

Another impingement-state parameter of interest is the fluid quality. There has been a long-standing debate regarding the potential for enhanced debris generation in the presence of entrained water droplets compared with that observed for high-quality steam and for air-jet surrogates. While the ANSI model cannot answer this concern, it may offer information on the spatial extent of the phenomena. Subject to the same interpretations and approximations as those discussed for impingement temperature, the jet quality can also be evaluated at $\underline{P_0}$ and $\underline{h_0}$. Contours of equal two-phase steam quality are illustrated in Figure I-22 for the reference cold-leg break. Similar to temperature, the fluid quality changes slowly beyond a range of 10 to 15 pipe diameters and maintains a nominal value between 0.25 and 0.35. This range would be considered low-quality steam for turbine generator applications and might be viewed with concern for its potential erosion effects on stainless steel rotor blades. Certainly, the time regimes of jet impact and in-service steam components are drastically different, but the potential damage mechanisms are the same.

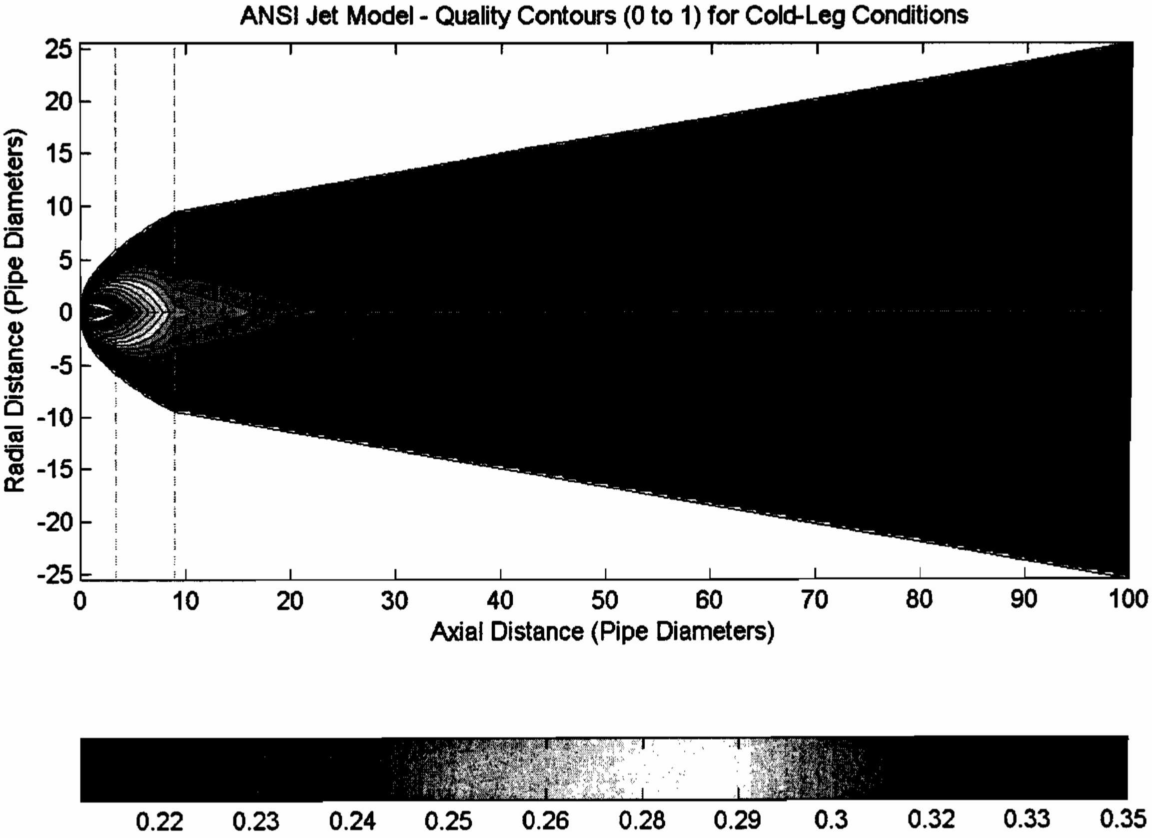

Figure I-22. Contours of Equivalent Steam Quality for the Reference Cold-Leg Break at 2250 psia and 530ºF

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The thermodynamic treatment of two-phase saturated conditions in the ANSI standard is inherently a homogeneous mass-mixture model. That is, the two-phase mixture is considered to be a single fluid with equivalent mass-weighted thermodynamic properties. This assumption, along with that of equal phase velocities in the jet, is justified by Lahey and Moody [LAH84]. Therefore, void fractions could be estimated from the local pressures and qualities. Under this assumption, it was found that the qualities shown in Figure I-19 would correspond to void fractions greater than 0.95 for all regions of the jet apart from the core. While Figure I-19 could be separated into the fluid and vapor mass fractions using the saturation properties and the definition of quality, the real issue of momentum transfer to a target could not be addressed with convincing accuracy. Theoretical treatments of two-phase transport introduce concepts of condensate nucleation, interphase velocities, droplet drag coefficients, and void fraction (space between droplets) that are difficult to measure experimentally. Pursuing this analysis with the present ANSI model would exceed the scope of its purpose and fidelity.

In summary, Table I-3 presents a set of concomitant values for pressure, temperature, quality and equivalent spherical radius that characterize the approximate impingement conditions in an expanding jet generated by a cold-leg break at 2250 psia and 530ºF. With respect to equivalent spherical diameter, this reference case is observed to bound all break conditions of interest for a PWR accident analysis. Table I-4 lists intermediate parameter values computed by ANSIJet for the reference break conditions. This information may be useful for comparisons of independent implementations of the jet model.

Table I-3. Summary of Jet Properties for the Reference Cold-Leg Break

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[redacted]	[redacted]	[redacted]	[redacted]
2	218.7	0.35	31.5
3	221.8	0.34	25.4
4	224.6	0.34	21.6
6	230.0	0.34	17.0
10	239.6	0.33	11.9
17	253.7	0.32	7.5
24	265.5	0.31	5.4
40	287.0	0.29	4.0
80	324.2	0.26	2.6
150	366.1	0.21	1.5
190	384.0	0.20	1.1
2250	530.0	0.00	0.9

Table I-4. Intermediate Parameters Computed by the ANSI Jet Routine for the Reference Cold-Leg Break Conditions

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Vessel Pressure	P0	[psia]	2250
Vessel Temp	T0	[deg F]	530
Vessel Quality	X0	[-]	-0.430084
Vessel Density	r0	[lbm/ft^3]	48.0879
Vessel Enthalpy	h0	[Btu/lbm]	522.455
Sat Temp at P0	Tsat	[deg F]	653.014
Liq Sat Enth at P0	hf	[Btu/lbm]	700.946
Vap Sat Enth at P0	hg	[Btu/lbm]	1115.96
Ambient Pressure	Pamb	[psia]	14.7
Pres at Asym Plane	Pa	[psia]	14.7
Dens at Pa, h0	rma	[lbm/ft^3]	0.105653
Computed Thrust Coeff	TC	[-]	1.64413
Crit Mass Flux	Ge	[lbm/ft^2/s]	25329.2
Tsat at Pamb	Tsatamb	[deg F]	212.238
Liq Sat Enth at Pamb	hfamb	[Btu/lbm]	180.176
Vap Sat Enth at Pamb	hgamb	[Btu/lbm]	1150.28
Degrees Subcooling	delTsub	[deg F]	123.014

I.7 SUMMARY CRITIQUE OF THE ANSI JET MODEL

Appendix I provides an exposition of the ANSI model and addresses several points where the model may be insufficiently clear or may suffer from an inconsistency. The major issues raised in the Appendix are summarized below; where applicable, recommendations for remediation are provided.

- The pressure distribution produced by the model exhibits a discontinuity across the boundary of the core. Within the core, the stagnation pressure is assumed to equal the upstream pressure P_0; the discontinuity has been observed to reach an order of magnitude for certain upstream conditions.

- Although not explicitly stated in the model, the jet pressure distribution, which falls to zero in the far field, must be interpreted as representative of local impingement gauge pressures.

- The jet pressure at the centerline, however, remains nonzero for any finite value of the axial penetration distance. This exaggerates pressure isobar volumes and causes volume-equivalent spherical damage radii to approach infinity as the damage pressure goes to zero.

- The pressure distribution has evidently been formulated such that the thrust force is correctly recovered only for targets oriented normal to the flow direction at the orifice. Therefore, the model may not be a good approximation to free-field expansion: it may not accurately predict local conditions at points away from the jet centerline, where the flow velocity on such a normally-oriented plate would exhibit a significant tangential component. This concern is not addressed by the application of a shape factor as outlined in Appendix D of the ANSI report.

- The above point has further ramifications for the applicability of the model to small targets. Since the stagnation pressure field produced by the model was developed to reproduce loadings on large flat targets, it is inaccurate to apply the stagnation pressures to small and/or non-flat objects. One could bound the true conditions by computing local static pressures as well; however, knowledge of the local velocity field and of the characteristics of the two-phase jet flow that are beyond the scope of the ANSI model would be required.

- A discontinuity in the slope of the isobars exists between Zones 2 and 3. This discontinuity is clearly evident in Figure I-1. The sharp terminal points of pressure isobars at the axial centerline also suggest that more attention could be given to the behavior of first spatial derivatives.

- The assumption of isentropic and/or isenthalpic expansion should be made with caution. For instance, stagnation conditions at the asymptotic plane are evaluated assuming isenthalpic behavior, implying no energy loss to the environment. In general, however, the isentropic assumption appears to be applied to the expanding jet. For a discussion of the limitations of these assumptions see Ref. [WIT02].

- Although it was analytically confirmed that all characteristic lengths in the problem scale linearly with the break diameter $\underline{D_e}$, it is recommended that users implement the formulation of the model presented herein, as it has been nondimensionalized with respect to this quantity.

- The notation adopted by the standard for the thrust coefficient is evidently inconsistent: $\underline{C_T}$, $\underline{C_{Te}}$, and $\underline{C_{Te}^*}$ all appear in the equations describing the pressure distribution for the various jet zones. These forms must all refer to a single numeric value if the pressure equations are to be piecewise continuous between zones.

- The ANSI model presents an expression for the jet area at the asymptotic plane that rests upon the assumption that the average flow static pressure at that location equals the ambient pressure $\underline{P_{amb}}$. Elsewhere in the ANSI model, however, the asymptotic plane static pressure is assigned a value that may be less than $\underline{P_{amb}}$.

- The standard advises users to implement a critical flow model, either the homogeneous equilibrium model (HEM) or the Henry-Fauske model, to obtain the jet mass flux $\underline{G_e}$. Users not having such a model available may estimate $\underline{G_e}$ from Figure C-4 of the ANSI report; however this figure only covers stagnation conditions extending to 2000 psi and 50°F of subcooling, leaving certain states (*e.g.*, cold leg conditions in many PWRs) unaddressed. Given the additional inaccuracies that may be introduced by reading from the figure, it is strongly recommended that a critical flow model be implemented for use with the jet model.

- The standard recommends that Henry-Fauske critical flow model be used for subcooled vessel conditions and the HEM for saturated conditions. This would introduce a strong discontinuity as the liquid saturation point is crossed. Therefore, since Henry-Fauske is evidently in better agreement with the data for both subcooled and two-phase conditions, exclusive use of this model is recommended.

- An implied discontinuity exists across the break plane, as the ANSI model assumes that fluid in the core is in equilibrium at the upstream stagnation pressure and quality. This assumption contradicts aspects of both the HEM and Henry-Fauske models.

- The correlation recommended by the standard for use in calculating the thrust coefficient $\underline{C_T}$ for subcooled conditions applies only to Henry-Fauske derived mass fluxes. This is not made clear in the standard. Also left unclear is the assumption inherent in the correlation that ambient conditions are at standard pressure. Therefore, this correlation should not be used in conjunction with HEM mass fluxes, and users of the standard should bear in mind that the correlation is not strictly validated for ambient conditions deviating from those of the standard atmosphere. The error is small, though, for most upstream pressures of interest in the present analysis.

- No analytic correlation is provided by the standard for the thrust coefficient relevant to saturated steam-water mixtures. Within the standard, users may only consult Figure B-5 to visually gauge an approximate value. Another recourse would be to consult the thrust coefficient contour plots presented in this appendix, or better, implement a critical mass flux model to enable direct calculation of mass flux and thrust coefficient via the Henry-Fauske model.

- Users should be aware that one desired result of the model, volume-equivalent spherical damage-pressure radii, can behave nonintuitively as certain upstream conditions are varied. For instance, the PWR hot leg and cold leg results presented in Table I-1 of this appendix show that the flow from the hot leg break exhibits a lower mass flux and thrust coefficient than that from the cold leg. Nonetheless, the damage radii are roughly comparable, with radii for the hot leg break being greater than those of the cold leg for higher damage pressures and smaller for lower damage pressures. These results, which follow from variations in the flow velocity and density at the break, reinforce the importance of not eliminating lower-energy break points *a priori* when conducting ZOI analyses.

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Attachment 1 to APPENDIX I

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C                 mexFunction                 C
C  MATLAB-executable subroutine that serves as the jumping-off point  C
C  for calls to the ASME steam tables via the subroutines located in  C
C  INTPK.FOR and elsewhere.                 C
C                                   C
C  The subroutine MUST be named mexFunction and it MUST contain the   C
C  four arguments                    C
C  NLHS:  number of elements contained in the array PLHS        C
C  PLHS:  an array of pointers to the output values to be returned    C
C  to MATLAB by mexFunction.  It is stored as an array of integer     C
C  memory references; MATLAB handles extraction of the outputs from   C
C  these references, but PLHS itself must be populated via the       C
C  mxCopyReal8ToPtr function (see below).             C
C  NRHS:  number of elements contained in the array PRHS        C
C  PRHS:  array of pointers to input values, as described above for   C
C  PLHS.  The input values themselves are extracted by calling       C
C  mxCopyPtrToReal8.                     C
C                                   C
C  For this application, the inputs in PRHS and outputs in PLHS are   C
C  organized as follows:                  C
C                                   C
C  Say that there are y (Po,To,Xo) state points describing a blowdown C
C  history.  Po(y) [psia] is an array defining the pressures at these C
C  points, To(y) [deg F] defines the pressures, and Xo(y) gives the   C
C  qualities.  Only two of these would be specified for each state    C
C  point; mexFunction will return the third.  During the subcooled    C
C  period, T and P would be given and mexFunction would populate      C
C  the quality by evaluating X = (h - h_f)/h_fg.  Under saturated     C
C  conditions, either T or P would be given along with X and the model C
C  would return the unspecified quantity as P = P_sat or T = T_sat.   C
C  Other outputs to be returned are                C
C                                   C
C  Tsat(y) [deg F], saturation temperature             C
C  Hf(y)   [Btu/lbm], enthalpy of liquid phase at Tsat        C
C  Hg(y)   [Btu/lbm], enthalpy of vapor phase at Tsat         C
C  Rhoo(y) [lbm/ft^3], fluid density               C
C  Ho(y)   [Btu/lbm], fluid enthalpy               C
C                                   C
C  Note that for saturated fluid there are redundancies, e.g.,       C
```

```
C  To=Tsat and only three of Hf, Hg, Ho, X are needed.          C
C                                          C
C  Say that z pressure/temperature contours are to be evaluated.     C
C  The input would be an array Pj(z) of pressures.  The temperatures  C
C  evaluated for the corresponding y state points would be stored in   C
C  the array Tj(y*z) indexed by Tj[y_i,z_j]=Tj[z*(y_i-1)+z_j].       C
C                                          C
C  The critical mass flux Ge(y) [lbm/ft^2/s] is computed for each     C
C  state point, as is the thrust coefficient K (notated TC in the     C
C  code).  This is evaluated using a force balance at the orifice,     C
C  with the mass flux and thermodynamic state at the orifice being    C
C  obtained from the model being implemented (the HEM, Henry-Fauske,  C
C  or the ideal gas law).                          C
C                                          C
C  The pressure Pa(y) and density Rhoa(y) [lbm/ft^3] at the asymptotic C
C  plane are also computed.                         C
C                                          C
C  The Ge calculations are carried out using one of three models,     C
C  homogeneous equilibrium (HEM), Henry-Fauske (H-F), or the ideal gas C
C  equations of state.  The model used is governed by the value of the C
C  USE_LOGIC integer input flag:                          C
C                                          C
C   USE_LOGIC    Subcooled    Saturated   Superheated           C
C     0           HEM          HEM          HEM           C
C     1           H-F          H-F        HEM           C
C     2           HEM          HEM        Ideal Gas          C
C     3           H-F          H-F      Ideal Gas          C
C                                          C
C  Finally, the saturation temperature Tsat,amb at Pamb is computed,  C
C  as are the saturated liquid and vapor enthalpies Hf,amb and Hg,amb. C
C                                          C
C  The input PRHS is thus                           C
C                                          C
C  CONCAT(Po, To, Xo, Pj, Pamb, y, z, USE_LOGIC)              C
C                                          C
C  and the output PLHS is                           C
C                                          C
C  CONCAT(Po, To, Xo, Tsat, Hf, Hg, Rhoo, Ho, Pa, Rhoa, Ge, TC,       C
C  Tj, Tsat,amb, Hf,amb, Hg,amb).                        C
C                                          C
C  AUTHOR: Erich Schneider                        C
C  DRAFTED: July 5, 2004                         C
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```

```
      SUBROUTINE mexFunction(NLHS, PLHS, NRHS, PRHS)

C   Initialize and dimension other arguments to subroutines located in
C   INTPK.FOR.  See EXAM.FOR for another example.

      IMPLICIT DOUBLE PRECISION (A-H,O-Z)
      INCLUDE 'nprop.cmn'
      DIMENSION IWORK(NPROP), IWANT(NPROP), PROPR(NPROP),
PROPSI(NPROP)
      DIMENSION WAVRI(NRIMAX), RI(NRIMAX), IRIFLG(NRIMAX)
      DIMENSION IPCHK(5), IPFLG(5)
      INTEGER mxGetM, mxGetN, mxGetPr
      INTEGER NLHS, NRHS
      INTEGER PLHS(*), PRHS(*)
      INTEGER M, N, SIZE, IN_COUNT, OUT_COUNT, IY, IZ, I, J, MODE

      INTEGER IN_PTR, OUT_PTR, USE_LOGIC, USE_MODEL

C number of outputs per state point.  Currently 12: the three inputs
C P, T and X (recall that 2 are specified and this subroutine finds
C the third), Tsat, Hf, Hg, Rhoo, Ho, Pa, Rhoa, Ge, TC
          PARAMETER (NUM_OUTPUTS = 12)

C Dimension input and output arrays given that no more than 50 state
C points and 50 contours may be accepted as inputs.
C This would be a lot more elegant if FORTRAN supported dynamic memory
C allocation, but that's the price one pays for a fast language!
          PARAMETER(MAX_STATE_PTS=50, MAX_CONTOURS=50)

          REAL*8 PAMB, TSATAMB, FOFH0, RATIO, MIX_PROP
          REAL*8 HFAMB, HGAMB
          REAL*8 IN_VALS(MAX_STATE_PTS*3+MAX_CONTOURS+4)
          REAL*8 OUT_VALS(MAX_STATE_PTS*NUM_OUTPUTS+
&
     & MAX_STATE_PTS*MAX_CONTOURS+1)
          REAL*8 P0(MAX_STATE_PTS), T0(MAX_STATE_PTS),
X0(MAX_STATE_PTS)
          REAL*8 T_SAT(MAX_STATE_PTS), HF(MAX_STATE_PTS),
HG(MAX_STATE_PTS)
          REAL*8 RHO0(MAX_STATE_PTS), H0(MAX_STATE_PTS),
PA(MAX_STATE_PTS)
      REAL*8 RHOA(MAX_STATE_PTS), GE(MAX_STATE_PTS),
PJ(MAX_CONTOURS)
          REAL*8 TJ(MAX_STATE_PTS,MAX_CONTOURS), TC(MAX_STATE_PTS)
```

```
C  Create array of reals from the array PRHS of pointers

        M = mxGetM(PRHS(1))
        N = mxGetN(PRHS(1))

        SIZE = M*N

        IN_PTR=mxGetPr(PRHS(1))

        CALL mxCopyPtrToReal8(IN_PTR,IN_VALS,SIZE)

C  Disassemble and parse input value array: get number of state points
C  to be examined and number of P/T contours to be obtained for each
C  state point.  Note that at least one state point must exist for the
C  contour calculation to take place.  If this is not the case,
C  the contour calculations will be skipped even if Pj input values
C  are supplied.

        IY=IN_VALS(SIZE-2)
        IZ=IN_VALS(SIZE-1)

C  Obtain value of integer USE_LOGIC flag
        USE_LOGIC=IN_VALS(SIZE)

C Verify that user is not trying to evaluate more than MAX_STATE_PTS
C state points or more than MAX_CONTOURS contours.  If this is the
C case, return a soft landing

        IF (IY .GT. MAX_STATE_PTS) THEN
                CALL mexErrMsgTxt('Number of state points passed to QUERYST is
greater&
   & than MAX_STATE_PTS.  Decrease number of points to be analyzed or      &
   &increase MAX_STATE_PTS in QUERYST.for.')
        ENDIF
        IF (IZ .GT. MAX_CONTOURS) THEN
                CALL mexErrMsgTxt('Number of contour points passed to QUERYST is
great&
   &er than MAX_CONTOURS.  Decrease number of points to be analyzed or      &
   & increase MAX_CONTOURS in QUERYST.for.')
        ENDIF

        IN_COUNT=1
        OUT_COUNT=1
```

```
C  Prepare IWANT vector to harvest enthalpies

     DO 110 I=1,NPROP
                  IWANT(I) = 0
110   CONTINUE
     IWANT(6) = 1
     DO 111 I=1,5
        IPCHK(I) = 0
111   CONTINUE

C  read P_amb and obtain Tsat and enthalpies at P_amb
        PAMB=IN_VALS(SIZE-3)
        CALL TSAT(PAMB, TSATAMB, RHOL, RHOV, IWORK, PROPR, IERR)

C  Compute liquid and vapor enthalpies HF & HG
     CALL PROPS(IWANT, TSATAMB, RHOL, PROPSI, PROPR,0,I2PH,0,
    &   ISFLG,0, ICFLG, IPCHK, IPFLG, 0, 0, WAVRI, RI, IRIFLG)
        HFAMB = PROPSI(6)

     CALL PROPS(IWANT, TSATAMB, RHOV, PROPSI, PROPR,0,I2PH,0,
    &   ISFLG,0, ICFLG, IPCHK, IPFLG, 0, 0, WAVRI, RI, IRIFLG)
        HGAMB = PROPSI(6)

        IF (IY .GT. 0) THEN

C  Read (P, T, X) values for IY state points
                  DO 100 I=1,IY
                          P0(I)=IN_VALS(I)
                          T0(I)=IN_VALS(I+IY)
                          X0(I)=IN_VALS(I+2*IY)
                          GE(I)=0.
                          TC(I)=0.

C  Compute properties for this point: first check if fluid is saturated
C  QUERYST.FOR treats the fluid as saturated if the input pressure is < 0,
C  in which case the input quality should be in [0,1]

                          IF ((P0(I) .LT. 0) .OR. (T0(I) .LT. 0)) THEN
C  Saturated conditions with only one of P and T specified; calculate the other

                                  IF (P0(I) .LT. 0) THEN
                                          T_SAT(I) = T0(I)
C  Find saturation pressure
                                          CALL PSAT(T0(I), PMPA, RHOL, RHOV,
IWORK, PROPR,
```

```
     &                          IERR)
                                P0(I) = PMPA
                          ELSE IF (T0(I) .LT. 0) THEN
C  Find saturation temperature
                                CALL TSAT(P0(I), TK, RHOL, RHOV, IWORK,
PROPR,
     &          IERR)
                                T0(I) = TK
                                T_SAT(I) = T0(I)
                          ENDIF
C  Find mixture density
                          RHO0(I) = MIX_PROP(X0(I), RHOL, RHOV)

C  Compute liquid and vapor enthalpies HF & HG
          CALL PROPS(IWANT, T0(I), RHOL, PROPSI, PROPR,0,I2PH,0,
     &      ISFLG,0, ICFLG, IPCHK, IPFLG, 0, 0, WAVRI, RI, IRIFLG)
                          HF(I) = PROPSI(6)

          CALL PROPS(IWANT, T0(I), RHOV, PROPSI, PROPR,0,I2PH,0,
     &      ISFLG,0, ICFLG, IPCHK, IPFLG, 0, 0, WAVRI, RI, IRIFLG)
                          HG(I) = PROPSI(6)

C  Compute mixture enthalpy
                          H0(I) = MIX_PROP(X0(I), HF(I), HG(I))

                    ELSE

C  Find sauration temperature at P0
                          CALL TSAT(P0(I), TK, RHOL, RHOV, IWORK, PROPR,
IERR)
                          T_SAT(I) = TK

C  Obtain enthalpies at (Tsat, P0)
          CALL PROPS(IWANT, TK, RHOL, PROPSI, PROPR, 0, I2PH, 0,
     &      ISFLG,0, ICFLG, IPCHK, IPFLG, 0, 0, WAVRI, RI, IRIFLG)
                          HF(I) = PROPSI(6)
          CALL PROPS(IWANT, TK, RHOV, PROPSI, PROPR, 0, I2PH, 0,
     &      ISFLG,0, ICFLG, IPCHK, IPFLG, 0, 0, WAVRI, RI, IRIFLG)
                          HG(I) = PROPSI(6)

C Find density and enthalpy at (T0, P0)
                          CALL DENS0(DOUT, P0(I),T0(I), DPD, IWORK,
PROPR, IERR)
                          RHO0(I)=DOUT
          CALL PROPS(IWANT,T0(I),DOUT,PROPSI, PROPR, 0, I2PH, 0,
```

```
     &          ISFLG,0, ICFLG, IPCHK, IPFLG, 0, 0, WAVRI, RI, IRIFLG)
                                  H0(I) = PROPSI(6)

                                  X0(I) = (H0(I) - HF(I))/(HG(I) - HF(I))
                        ENDIF

C Given the initial quality, determine the pressure at the asymptotic plane
                        IF (X0(I) .GT. -0.1) THEN
                                  FOFH0 = SQRT(0.1 + X0(I))
                        ELSE
                                  FOFH0 = 0.0
                        ENDIF
                        RATIO = PAMB/P0(I)
                        IF (RATIO .GT. 0.5) THEN
                                  RATIO=0.5
                        ENDIF

                        PA(I) = (1. - 1./2.*(1.-2.*RATIO)*FOFH0)*PAMB

C  Now find the density.  Set MODE = 1 for HSSOLV (Inp: P, H, Out: T, rho)
                        MODE = 1
                        CALL HSSOLV(MODE, PA(I), H0(I), TPOUT, D1, DV, DL,
     &          I2PH, Q, IWORK, PROPR, IERR)

C  Format of result in HSSOLV dependent on phase of fluid as signified by I2PH
                        IF ((I2PH .EQ. 2) .OR. (I2PH .EQ. 4)) THEN
                                  RHOA(I) = 1./MIX_PROP(Q, 1.0/DL, 1.0/DV)
                        ELSE
                                  RHOA(I) = D1
                        ENDIF

C  The code block below sets USE_MODEL based upon user-specified USE_LOGIC
C  and upstream stagnation conditions.


C  Based upon the specification provided by the user, USE_MODEL is below
C  assigned a value of zero (HEM), one (H-F) or two (ideal gas)
C  prior to being passed to CRIT_MASS_FLUX. However: if USE_MODEL = 2 and
C  upstream stagnation conditions are insufficiently superheated such that
C  the ideal gas law yields a static state that is in the two-phase regime,
C  CRIT_MASS_FLUX automatically defaults to the HEM.  In general, since the
C  HEM reduces to the ideal gas law as the superheating increases, USE_LOGIC
C  = 2 and 3 should be avoided.  This is doubly so since truly ideal gas-like
C  behavior is not likely to be observed for any of the problems that are
C  being studied with ANSIJET.
```

```
                    IF (X0(I) .GT. 1.0) THEN
                          IF ((USE_LOGIC .EQ. 2) .OR. (USE_LOGIC .EQ. 3))
THEN
                                USE_MODEL = 2
                          ELSE
                                USE_MODEL = 0
                          ENDIF
                    ELSE
                          IF ((USE_LOGIC .EQ. 0) .OR. (USE_LOGIC .EQ. 2))
THEN
                                USE_MODEL = 0
                          ELSE
                                USE_MODEL = 1
                          ENDIF
                    ENDIF
                    CALL CRIT_MASS_FLUX(GE(I),TC(I),P0(I),H0(I),PAMB,
   &        USE_MODEL)

100             CONTINUE
                IN_COUNT=IY*3

                IF(IZ .GT. 0) THEN
C  Read Pj values for IZ contours
                      DO 101 J=1, IZ
                            PJ(J)=IN_VALS(IN_COUNT+J)
101                   CONTINUE

C  Compute Tj at each Pj value for every state point
                      DO 102 I=1, IY
                            DO 103 J=1, IZ
C  MODE = 1 -> HSSOLV expects P and H as inputs and returns T
                                  MODE = 1
                                  CALL HSSOLV(MODE, PJ(J), H0(I), TPOUT, D1,
DV, DL,
   &           I2PH, Q, IWORK, PROPR, IERR)
                                  TJ(I,J)=TPOUT
103                         CONTINUE
102                   CONTINUE
                ENDIF
          ENDIF

C  Create output array

          DO 104 I=1,IY
```

```
                OUT_VALS(I)=P0(I)
                OUT_VALS(I+IY)=T0(I)
                OUT_VALS(I+2*IY)=X0(I)
                OUT_VALS(I+3*IY)=T_SAT(I)
                OUT_VALS(I+4*IY)=HF(I)
                OUT_VALS(I+5*IY)=HG(I)
                OUT_VALS(I+6*IY)=RHO0(I)
                OUT_VALS(I+7*IY)=H0(I)
                OUT_VALS(I+8*IY)=PA(I)
                OUT_VALS(I+9*IY)=RHOA(I)
                OUT_VALS(I+10*IY)=GE(I)
                OUT_VALS(I+11*IY)=TC(I)
                DO 105 J=1,IZ
                        OUT_VALS(NUM_OUTPUTS*IY+(I-1)*IZ+J)=TJ(I,J)
105             CONTINUE
104     CONTINUE
        OUT_VALS(NUM_OUTPUTS*IY+IZ*IY+1)=TSATAMB
        OUT_VALS(NUM_OUTPUTS*IY+IZ*IY+2)=HFAMB
        OUT_VALS(NUM_OUTPUTS*IY+IZ*IY+3)=HGAMB

C Create pointer to output array and size it

        SIZE=MAX_STATE_PTS*NUM_OUTPUTS+MAX_STATE_PTS*MAX_CON
TOURS+3
        PLHS(1)=mxCreateDoubleMatrix(SIZE,1,0)
        OUT_PTR=mxGetPr(PLHS(1))

C Populate pointer to output for MATLAB use

        CALL mxCopyReal8ToPtr(OUT_VALS,OUT_PTR,SIZE)

        RETURN
        END

        FUNCTION MIX_PROP(QUALITY, PROP_F, PROP_G)

CCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCC
CCCCCCCCCCCCCCCCCCC
C                       MIX_PROP                    C
C  Given an input quality and phase properties at saturation, MIX_PROP C
C  computes the value of the property for the mixture.  Any mass-     C
C  specific property that has meaning for a two phase mixture         C
C  (1/density, enthalpy, etc.) may be computed.                  C
C                                                   C
C  Inputs:                                             C
```

```
C  QUALITY   Does not necessarily lie in [0,1]                 C
C  PROP_F, PROP_G: saturation values of the property to be computed    C
C                                          C
C  Output:                                    C
C  MIX_PROP = QUALITY*PROP_G + (1-QUALITY)*PROP_F                 C
C                                          C
CCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCC
CCCCCCCCCCCCCCCCCCC

       REAL*8 QUALITY, PROP_F, PROP_G, MIX_PROP

       MIX_PROP = QUALITY*PROP_G + (1.0-QUALITY)*PROP_F

       RETURN
       END


       SUBROUTINE CRIT_MASS_FLUX(G_CALC, TC_CALC, P0, H0,
    &  P_AMB, USE_MODEL)

CCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCC
CCCCCCCCCCCCCCCCCCC
C                  CRIT_MASS_FLUX                  C
C  Given a state point (P0, H0), derives the critical mass flow G     C
C  (kg/m^2-s) as per the homogeneous equil. method (see Hall & Czapary,C
C  "Tables of Homogeneous Equilibrium Critical Flow Parameters for     C
C  Water In SI Units," EG&G Idaho Report EGG-2056,1980), or the method C
C  of Henry and Fauske, "The Two-Phase Critical Flow of One-Component  C
C  Mixtures in Nozzles, Orifices, and Short Tubes," J. Heat Transfer   C
C  May 1971, p. 179, or the ideal gas equation of state.              C
C                                          C
C  HEM Notes                                    C
C                                          C
C  The HEM model maximizes the mass flux G = V/v (V = flow velocity    C
C  [m/s], v = specific volume [m^3/kg].  Applying the First Law, this  C
C  is equivalent to                               C
C  G = 2(h0 - h)^1/2 / v,                           C
C  where h0 is the stagnation enthalpy of the fluid, and h and v are   C
C  the enthalpy and specific volume to be adjusted isoentropically     C
C  such that G is maximized.  The optimum or critical state at which   C
C  the tradeoff between decreased static enthalpy (and thus higher     C
C  velocity) and increased specific volume may be represented by a     C
C  critical pressure, p*.  It is convenient to optimize on the above   C
C  using p* as the independent variable.                     C
C                                          C
```

```
C The maximization method used is a golden-mean bisection after     C
C Teukolsky et. al, "Numerical Recipes in Fortran."  The algorithm    C
C has been altered somewhat to take into account the nature of the    C
C function G = f(p*): since f(p*) is undefined for p* > P0 and also   C
C for very low values of p*, some care must be taken in bracketing    C
C the root.   The algorithm generally converges within 50 iterations, C
C with 1% accuracy being obtained after about 20 evaluations of G.    C
C                                                                    C
C The algorithm may be tuned by adjusting the fractional tolerance    C
C TOLER and the interval size STRETCH_FACTOR over which the root      C
C bracketing algorithm searches.                                     C
C                                                                    C
C Henry-Fauske Notes                                                 C
C                                                                    C
C The Henry-Fauske model also searches for the critical pressure p*   C
C for which the mass flux is maximized.  However, the computational   C
C technique is modified somewhat for the Henry-Fauske formulation as  C
C it requires solution of a transcendental equation for p*.  The      C
C equation only requires knowledge of upstream stagnation conditions  C
C and is described in greater detail in subroutine HENRY_FAUSKE below.C
C                                                                    C
C To use the same computational engine as that applied to the HEM    C
C above, the Henry-Fauske formulation is recast as a maximization     C
C problem by writing the transcendental equation for p*,             C
C f (p*) = g (p*),                                                   C
C in a form amenable to solution via golden-mean maximization:        C
C A = - ( g (p*) - f (p*))^2,                                        C
C where the problem becomes one of finding the value of p* that       C
C maximizes A, with perfect convergence of course resulting in A = 0. C
C The quantity A is notated P_ROOT in the code below.                C
C                                                                    C
C The same considerations as described for the HEM above apply in    C
C connection with root bracketing.  Convergence slows for highly      C
C subcooled upstream stagnation conditions.  Under these conditions,  C
C the quantity dG/dp* is very large in the vicinity of the root, and  C
C the bracketing of the root becomes an increasingly difficult        C
C problem.  This is evidenced by the behavior of the equilibrium      C
C quality at the thorat (notated x_E in Henry and Fauske's paper),    C
C which approaches zero for the p* that solves the model as upstream  C
C subcooling increases.                                              C
C                                                                    C
C Ideal Gas Notes                                                    C
C                                                                    C
C Application of the ideal gas equation of state is generally not     C
C advisable since the ideal gas approximation is not a good one for   C
```

```
C  upstream stagnation conditions that are only slightly superheated.  C
C  In fact, it is possible for an evaluation using this method to      C
C  result in static (choked) conditions that in fact lie within the    C
C  two-phase regime.  If the user has selected ideal gas evaluation    C
C  and this is found to occur, CRIT_MASS_FLUX defaults to the HEM for  C
C  G_e and thrust coefficient calculations.  Since conditions of       C
C  interest for ANSIJET, e.g., main steam line break, are not greatly  C
C  superheated, it is recommended that users adopt the HEM instead     C
C  for all superheated evaluations of G_e.                             C
C                                                                      C
C  The HEM in fact reduces to the ideal gas eqn. for heavily           C
C  superheated conditions. This provides an additional reason to       C
C  adopt it.  The only disadvantage to the HEM is that the method is   C
C  somewhat more computationally intensive.                            C
C                                                                      C
C                                                                      C
C                                                      EAS 7/7/04  C
C  ********* MODIFICATIONS *********                                   C
C  EAS 7/14/04:  Fixed bug that caused code to use improperly          C
C               initialized values for GE                              C
C  EAS 7/18/04:  Deployed Henry-Fauske model solver; added switch to   C
C               allow user selection of model to use                   C
C  EAS 7/28/04:  Fixed root bracketing interval for H-F so that        C
C               routine will search the entire space of state points C
C               allowed by the steam tables.                           C
C  EAS 7/28/04:  The model to be used is now an input from the MATLAB C
C               function call.  See MexFunction for documentation.   C
C  EAS 7/30/04:  Changed to subroutine that computes thrust coeffs.  C
C               subsequent to critical mass flux calculations          C
C  EAS 8/17/04   Added ideal gas eqn. of state evaluation              C
C                                                                      C
CCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCC
CCCCCCCCCCCCCCCCCCC


    IMPLICIT DOUBLE PRECISION (A-H,O-Z)
        INCLUDE 'nprop.cmn'
C  Minimum temperature at which steam tables calculate thermo. properties
C  Used for bounding of space in which root is to be sought
    PARAMETER (T_MIN = 273.2)
    DIMENSION IWORK(NPROP), IWANT(NPROP), PROPR(NPROP),
PROPSI(NPROP)
    DIMENSION WAVRI(NRIMAX), RI(NRIMAX), IRIFLG(NRIMAX)
    DIMENSION IPCHK(5), IPFLG(5)
```

```
        REAL*8
H0,P0,T0,S0,X0,STRETCH_FACTOR,P_HOLD,G_CALC,TC_CALC,R
        REAL*8 G_HOLD, MIX_PROP, DENS, VG0, VF0, ROOT_HOLD,
P_MIN,RHO_HOLD
        REAL*8 T_HOLD
C  0 = HEM, 1 = Henry-Fauske
        INTEGER USE_MODEL

C  Bracketing triplet p_low=PSTAR(1), p_mid=PSTAR(2), p_hi=PSTAR(3)
C  and values of G at each p*
        REAL*8 PSTAR(3), GE(3), P_ROOT(3), RHO_T(3)

        PARAMETER (GOLD = 1.618034, TOLER=1.0e-5)

C  Get temperature T0 and density Rho0 corresponding to (P0, H0)
        CALL HSSOLV(1, P0, H0, TPOUT, D1, DV, DL,
   &         I2PH, Q, IWORK, PROPR, IERR)


        IF ((I2PH .EQ. 2) .OR. (I2PH .EQ. 4)) THEN
C  2 phase stagnation conditions
                DENS = 1./MIX_PROP(Q, 1.0/DL, 1.0/DV)
                X0=Q
                VF0 = 1./DL
                VG0 = 1./DV
        ELSE IF(I2PH .EQ. -1) THEN
C  Saturated / subcooled liquid stagnation
                DENS = D1
                X0= (1./DENS - 1./DL)/(1./DV-1./DL)
                VF0 = 1./DENS
                VG0 = 0.
        ELSE IF(I2PH .EQ. 1) THEN
C  Saturated vapor stagnation
                DENS = D1
                X0= (1./DENS - 1./DL)/(1./DV-1./DL)
C  Note: although the model functions without crashing for
C  superheated vapor, this does not imply its validity!!
                VF0 = 1./DL
                VG0 = 1./DENS
        ENDIF
        T0=TPOUT

C  Get entropy S0 corresponding to (T0, Rho0)
    DO 202 I=1,NPROP
                IWANT(I) = 0
```

```
202  CONTINUE
   IWANT(7) = 1

   CALL PROPS(IWANT,T0,DENS,PROPSI, PROPR, 1, I2PH, 0,
  &ISFLG,0, ICFLG, IPCHK, IPFLG, 0, 0, WAVRI, RI, IRIFLG)
   S0 = PROPSI(7)

        IF (USE_MODEL .EQ. 2) THEN

C  Using the ideal gas law for superheated steam:
C  T_HOLD and RHO_HOLD will contain the static temperature and density
C  at the throat.  These are used to verify that the ideal gas law offers
C  a reasonably valid model of the expansion.

                T_HOLD = T0
                RHO_HOLD = DENS

                IWANT(8) = 1
                IWANT(9) = 1

C  Additional upstream properties are needed:
C  Obtain ratio of specific heats, GAMMA = C_P/C_V.  GAMMA ~ 1.3 for steam.

111             CALL PROPS(IWANT,T_HOLD,RHO_HOLD,PROPSI, PROPR, 1,
I2PH, 0,
  &      ISFLG,0, ICFLG, IPCHK, IPFLG, 0, 0, WAVRI, RI, IRIFLG)

                IF ((I2PH .EQ. 2) .OR. (I2PH .EQ. 4)) THEN

C  If this is true, static conditions at the exit are two-phase.  The ideal
C  gas equation of state is obviously not applicable.  Break and evaluate
C  using the HEM.

                        USE_MODEL = 0
                        GOTO 110
                ENDIF

                GAMMA = PROPSI(9)/PROPSI(8)
C  Gas constant R = C_P - C_V  [J/kg/K]
                R = 1000.*(PROPSI(9) - PROPSI(8))

C  Compute the mass flux noting that P0 is stored in MPa and must be converted
C  Hence G_CALC is has units [kg/m^2/s]
                G_CALC =
(GAMMA/R*(2./(GAMMA+1.))**((GAMMA+1.)/(GAMMA-1.)))**
```

```
   &     0.5 * P0 / T0**0.5 *1.e6
C  Compute the static discharge density rho = G / V
                 RHO_HOLD = G_CALC / (2.*GAMMA*R*T0/(GAMMA+1.))**0.5

C  The static pressure at the exit is also needed to compute the thurst coeff:
C  This pressure is given in MPa.
                 P_HOLD = P0 * (2./(GAMMA+1.))**(GAMMA/(GAMMA-1.))


C  Obtain the static temperature, check for consistency, and recompute
C  static properties to check validity of ideal gas law eqn. of state:
                 T_STAT = T0 * 2./(GAMMA + 1.)
                 IF ((T_HOLD - T_STAT)**2. .GT. .01) THEN
                      T_HOLD = T_STAT

C Re-evaluate state equation to verify that the correct property values were
C used.  If the fluid were behaving as a perfect gas this loop would not be
C necessary. Entropy, for instance, is not conserved between the stagnation and
C static states, although it is approximately constant through the expansion
C for highly superheated conditions.
                      GOTO 111
                 ENDIF

C  Success: the ideal gas law results for G, RHO and P will be used below
C  to obtain the thrust coefficient (theoretically 1.26 for steam)

110     ENDIF

        IF(USE_MODEL. EQ. 0) THEN

C  Using the HEM:
C  Bracket the maximum.  It is assumed that Ge(p*) is well-behaved in that
C  there will be only one local maximum.  Ge(p*=P0) = 0, so we sweep downward
C  in p* until we have a triplet (p_low,p_mid,p_hi=P0) in which the root is
C  bracketed.

                 STRETCH_FACTOR=0.75
                 GE(1)=1
                 GE(2)=0
                 PSTAR(1)=P0

C  Increase P* interval to be searched for root until root is bracketed
200              IF (GE(1) .GT. GE(2)) THEN
                      PSTAR(1)=PSTAR(1)*STRETCH_FACTOR
                      PSTAR(3)=P0
```

```
                        PSTAR(2)=PSTAR(1)+1./GOLD*(PSTAR(3)-PSTAR(1))
                        GE(3)=0.
                        CALL EVAL_G(GE(1), RHO_T(1), H0, PSTAR(1), S0)
                        CALL EVAL_G(GE(2), RHO_T(2), H0, PSTAR(2), S0)
                        IF(GE(2) .LT. TOLER) THEN
C  If this is true, we've fallen into an area where both p_low and p_mid
C  are undefined.  Resize the bracketing interval and try again
                                PSTAR(1)=PSTAR(1)/STRETCH_FACTOR
                                STRETCH_FACTOR=SQRT(STRETCH_FACTOR)
                                GE(1)=GE(2)+TOLER
                        ENDIF
                        GOTO 200
                ENDIF

201             IF((PSTAR(3)-PSTAR(1))/PSTAR(3) .GT. TOLER) THEN
C  Check which interval of (p_low,p_mid),(p_mid,p_hi) is larger and bisect it
                        IF(PSTAR(3)-PSTAR(2) .GT. PSTAR(2)-PSTAR(1)) THEN
C  The p_mid to p_hi interval is larger; bisect this one
                                P_HOLD=PSTAR(3)
                                G_HOLD=GE(3)
                                RHO_HOLD=RHO_T(3)
                                PSTAR(3)=PSTAR(3)-1.0/GOLD*(PSTAR(3)-PSTAR(2))
                                CALL EVAL_G(GE(3), RHO_T(3), H0, PSTAR(3), S0)
                        ELSE
                                P_HOLD=PSTAR(1)
                                G_HOLD=GE(1)
                                RHO_HOLD=RHO_T(1)
                                PSTAR(1)=PSTAR(1)+1.0/GOLD*(PSTAR(2)-
PSTAR(1))
                                CALL EVAL_G(GE(1), RHO_T(1), H0, PSTAR(1), S0)
                        ENDIF
                        IF (GE(2) .LT. GE(1)) THEN
C G at p_mid is not at large as G at p_low.  Shift bisection interval
C so that old p_mid is now p_hi
                                PSTAR(3)=PSTAR(2)
                                GE(3)=GE(2)
                                RHO_T(3)=RHO_T(2)
                                PSTAR(2)=PSTAR(1)
                                GE(2)=GE(1)
                                RHO_T(2)=RHO_T(1)
                                PSTAR(1)=P_HOLD
                                GE(1)=G_HOLD
                                RHO_T(1)=RHO_HOLD
                        ELSE IF(GE(2) .LT. GE(3)) THEN
C G at p_mid is not at large as G at p_hi.  Shift bisection interval
```

```
C so that old p_mid is now p_low
                  PSTAR(1)=PSTAR(2)
                  GE(1)=GE(2)
                  RHO_T(1)=RHO_T(2)
                  PSTAR(2)=PSTAR(3)
                  GE(2)=GE(3)
                  RHO_T(2)=RHO_T(3)
                  PSTAR(3)=P_HOLD
                  GE(3)=G_HOLD
                  RHO_T(3)=RHO_HOLD
              ELSE
C bisection interval is fine; continue
              ENDIF
              GOTO 201
          ENDIF

C all done, interval size has decrased to specified tolerance.
          G_CALC=GE(2)
          RHO_HOLD=RHO_T(2)
          P_HOLD=PSTAR(2)
      ELSE IF (USE_MODEL .EQ. 1) THEN

C Using Henry - Fauske:
          GE(1)=0
          GE(2)=0
          GE(3)=0
          P_ROOT(1)=-1
          P_ROOT(2)=-1
          P_ROOT(3)=-1

C Bracketing the root and ensuring that the intermediate guess for the
C throat pressure PSTAR results in a larger P_ROOT than the other two
C guesses is a challenge, because the Henry-Fauske evaluation exhibits
C markedly different behavior in the subcooled region as compared to 2-
C phase initial conditions (observe discontinuities in the derivatives
C of the critical mass fluxes shown in Figs 12 through 14 of Henry &
C Fauske).  Under subcooled initial conditions, some guesses for throat
C pressure may be invalid as they result in subcooled conditions at the
C throat (see subroutine HENRY_FAUSKE for more).  Hence, bracket the root
C by starting with initial bounds of P_low ~ 0, P_high ~ P0.  Evaluate
C RHS of 0 = P_ROOT at P_mid, P_low and P_high.  If P_ROOT (P_low) is
C closer to zero than at P_mid, try again with P_hi now equal to P_mid.
C If P_ROOT (P_high) is closer to zero than at P_mid, pursue a similar
C strategy by setting P_low = P_hi.  Repeat until the P_mid guess gives
C a root evaluation that is closer to zero than either P_low or P_hi.
```

```
              PSTAR(3)=P0 - TOLER

C Find the lowest saturation pressure at which steam tables can
C obtain the necessary thermo. properties.  This will serve as a
C lower bound for throat pressure derived by the root-finding routine

              CALL PSAT(T_MIN, PMPA, RHOL, RHOV, IWORK, PROPR, IERR)
              PSTAR(1)=PMPA

196           PSTAR(2)=PSTAR(1)+1./GOLD*(PSTAR(3)-PSTAR(1))
              CALL HENRY_FAUSKE(H0, PSTAR(1), T0, S0, X0, P0,
  &                 VF0, VG0, P_ROOT(1), GE(1), RHO_T(1))
              CALL HENRY_FAUSKE(H0, PSTAR(2), T0, S0, X0, P0,
  &                 VF0, VG0, P_ROOT(2), GE(2), RHO_T(2))
              CALL HENRY_FAUSKE(H0, PSTAR(3), T0, S0, X0, P0,
  &                 VF0, VG0, P_ROOT(3), GE(3), RHO_T(3))
              IF (P_ROOT(2) .EQ. 0.0) THEN
                   P_ROOT(2)=-1.
                   P_ROOT(3)=-1.+TOLER
              ELSE IF (P_ROOT(3) .EQ. 0.0) THEN
                   P_ROOT(3)=P_ROOT(2)-TOLER
              ENDIF
              IF (P_ROOT(2) .LT. P_ROOT(1)) THEN
                   PSTAR(3)=PSTAR(2)
                   GOTO 196
              ELSE IF(P_ROOT(2). LT. P_ROOT(3)) THEN
                   PSTAR(1)=PSTAR(2)
                   GOTO 196
              ENDIF
203           IF((PSTAR(3)-PSTAR(1))/PSTAR(3) .GT. TOLER) THEN
                   IF(PSTAR(3)-PSTAR(2) .GT. PSTAR(2)-PSTAR(1)) THEN
C the p_mid to p_hi interval is larger; bisect this one
                        P_HOLD=PSTAR(3)
                        G_HOLD=GE(3)
                        ROOT_HOLD=P_ROOT(3)
                        RHO_HOLD=RHO_T(3)
                        PSTAR(3)=PSTAR(3)-1.0/GOLD*(PSTAR(3)-PSTAR(2))
                        CALL HENRY_FAUSKE(H0, PSTAR(3), T0, S0, X0, P0,
  &                     VF0, VG0, P_ROOT(3), GE(3), RHO_T(3))
                        IF (P_ROOT(3) .EQ. 0.0) THEN
                             P_ROOT(3)=P_ROOT(2)-TOLER
                        ENDIF
                   ELSE
                        P_HOLD=PSTAR(1)
```

```
                                G_HOLD=GE(1)
                                ROOT_HOLD=P_ROOT(1)
                                RHO_HOLD=RHO_T(1)
                                PSTAR(1)=PSTAR(1)+1.0/GOLD*(PSTAR(2)-
PSTAR(1))
                                CALL HENRY_FAUSKE(H0, PSTAR(1), T0, S0, X0, P0,
   &                            VF0, VG0, P_ROOT(1), GE(1), RHO_T(1))
                                IF (P_ROOT(1) .EQ. 0.0) THEN
                                     P_ROOT(1)=P_ROOT(2)-TOLER
                                ENDIF
                           ENDIF
                           IF (P_ROOT(2) .LT. P_ROOT(1)) THEN
C RHS of 0 = f(PSTAR) as evaluated in HENRY_FAUSKE is farther from zero at
C p_mid than at p_low.  Shift bisection interval so that old p_mid is now p_hi
                                PSTAR(3)=PSTAR(2)
                                GE(3)=GE(2)
                                P_ROOT(3)=P_ROOT(2)
                                RHO_T(3)=RHO_T(2)
                                PSTAR(2)=PSTAR(1)
                                P_ROOT(2)=P_ROOT(1)
                                GE(2)=GE(1)
                                RHO_T(2)=RHO_T(1)
                                PSTAR(1)=P_HOLD
                                GE(1)=G_HOLD
                                P_ROOT(1)=ROOT_HOLD
                                RHO_T(1)=RHO_HOLD
                           ELSE IF(P_ROOT(2) .LT. P_ROOT(3)) THEN
C RHS of 0 = f(PSTAR) as evaluated in HENRY_FAUSKE is farther from zero at
C p_mid than at p_hi.  Shift bisection interval so that old p_mid is now p_low
                                PSTAR(1)=PSTAR(2)
                                GE(1)=GE(2)
                                P_ROOT(1)=P_ROOT(2)
                                RHO_T(1)=RHO_T(2)
                                PSTAR(2)=PSTAR(3)
                                GE(2)=GE(3)
                                P_ROOT(2)=P_ROOT(3)
                                RHO_T(2)=RHO_T(3)
                                PSTAR(3)=P_HOLD
                                GE(3)=G_HOLD
                                P_ROOT(3)=ROOT_HOLD
                                RHO_T(3)=RHO_HOLD
                           ELSE
C bisection interval is fine; continue
                           ENDIF
                           GOTO 203
```

```
          ENDIF
          G_CALC=GE(2)
          RHO_HOLD=RHO_T(2)
          P_HOLD=PSTAR(2)
      ENDIF

      IF(G_CALC .GT. 0) THEN
          TC_CALC = THRUST_COEFF(G_CALC, RHO_HOLD, P_AMB,
P_HOLD, P0)
      ELSE
          TC_CALC = 0.
      ENDIF
      RETURN
      END

      SUBROUTINE EVAL_G(GCRIT, RHO_T, H0, PSTAR, S0)

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CCCCCCCCCCCCCCCCCCC
C                    EVAL_G                    C
C  Evaluates the HEM expression for the critical mass flux,        C
C                                              C
C  G = (2 * (H0 - H(S0,PSTAR)))^1/2 / v(S0,PSTAR)            C
C                                              C
C  where G is the mass flux [kg/m^2/s], H0 is the stagnation enthalpy  C
C  [J/kg], H is the enthalpy at the current pressure guess PSTAR [MPa],C
C  and v is the specific volume [m^3/kg] at PSTAR.             C
C                                              C
C  Returns zero if PSTAR > P0(S0, H0) or if the pressure guess is so   C
C  low that no solution exists (i.e., if H > H0).            C
C                                              C
C  If the calculation is successful, the value of the static density  C
C  at the throat, RHO_T [kg/m^3] = 1/v(S0,PSTAR), is also returned.   C
C                                              C
C                                 EAS 7/8/04    C
CCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCC
CCCCCCCCCCCCCCCCCCC


    IMPLICIT DOUBLE PRECISION (A-H,O-Z)
      INCLUDE 'nprop.cmn'
      REAL*8 H0, PSTAR, H, SV, S0, DENS, MIX_PROP, GCRIT,RHO_T
    DIMENSION IWORK(NPROP), IWANT(NPROP), PROPR(NPROP),
PROPSI(NPROP)
    DIMENSION WAVRI(NRIMAX), RI(NRIMAX), IRIFLG(NRIMAX)
```

```
    DIMENSION IPCHK(5), IPFLG(5)

    DO 310 I=1,NPROP
          IWANT(I) = 0
310  CONTINUE
    IWANT(6) = 1

       CALL HSSOLV(2, PSTAR, S0, TPOUT, D1, DV, DL,
   &       I2PH, Q, IWORK, PROPR, IERR)

C  Find the density at (S0, PSTAR)
       IF ((I2PH .EQ. 2) .OR. (I2PH .EQ. 4)) THEN
          DENS = 1./MIX_PROP(Q, 1.0/DL, 1.0/DV)
       ELSE
          DENS = D1
       ENDIF

C  Obtain H(S0,PSTAR)
       CALL PROPS(IWANT,TPOUT,DENS,PROPSI, PROPR, 1, I2PH, 0,
   &ISFLG,0, ICFLG, IPCHK, IPFLG, 0, 0, WAVRI, RI, IRIFLG)

       H = PROPSI(6)
       SV=1.0/DENS

       IF(H .GT. H0) THEN
C  Trial point leads to undefined G
          GCRIT = 0
          RHO_T = 0
       ELSE
C  Make sure to convert from kJ/kg to J/kg so that G has expected units
          GCRIT = SQRT(2000.*(H0-H))/SV
          RHO_T = DENS
       ENDIF

       RETURN
       END

       SUBROUTINE
HENRY_FAUSKE(H0,PT,T0,S0,X0,P0,VF0,VG0,ROOT,GCRIT,
   &   RHO_T)

CCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCC
CCCCCCCCCCCCCCCCCCC
C                  HENRY_FAUSKE                  C
C  Given a state point (P0, H0), derives the critical mass flux as    C
```

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C given by the Henry Fauske model. See Henry and Fauske, "The Two C
C Phase Critical Flow of One-Component Mixtures in Nozzles, Orifices C
C and Short Tubes," J. Heat Transfer, May 1971, p. 179. C
C C
C INPUTS C
C In addition to the upstream stagnation pressure and enthalpy P0 and C
C H0, several other quantities describing the state point are pre- C
C computed and passed to the subroutine. These are C
C S0, the entropy, X0, the quality (in (-inf,inf)), the temperature C
C T0, the liquid and vapor specific volumes VF0 and VG0 (if the C
C upstream conditions are subcooled, only VF0 is defined and used). C
C The current guess for the thorat pressure PT is the final input. C
C C
C OUTPUTS C
C Three outputs are populated. These are C
C ROOT, defined as the quantity -(RHS-LHS)^2. RHS and LHS are the C
C right and left hand sides of <EQ. 33> for 2 phase flow and <EQ. 46> C
C for subcooled flow. This is returned for use in the root finding C
C algorithm in CRIT_MASS_FLUX, wherein the throat pressure PT is C
C varied such that ROOT is maximized. C
C The second output, GCRIT, is the mass flux evaluation as per C
C <EQ. 29> (2-phase) or <EQ. 45> (subcooled). C
C The final output, RHO_T, is the static density at the throat. C
C C
C COMMENTS C
C C
C - discontinuity in result for high pressures when crossing from C
C subcooled to 2 phase region. Possible explanation: assumption C
C in model regarding vapor specific volume at throat C
C COMMENT 7/28/04: The above seems to be the case. The assumption C
C that the vapor formed in the orifice is saturated at the local C
C pressure seems good for lower pressures, but breaks down somewhat C
C at higher pressures. C
C C
C - a discontinuity in slope dG/dS when crossing from subcooled to C
C saturated has been observed. See Figs. 12-14 of paper. C
C - assumption in model that equilibrium throat quality is *always* C
C greater than zero, while discharge quality = upstream stagnation C
C quality. C
C COMMENT 7/28/04: Result is well-behaved even for heavily subcooled C
C stagnation conditions. In fact it approaches the HEM result. C
C However, the numerical situation is complicted in that the equil. C
C pseudo-quality x_E (as defined in <EQ. 23>) that solves <EQ. 33> is C
C very close to zero, and <EQ. 33> is undefined for x_E <= 0. Hence C
C it's heavy going for the bisection routine, and the mass flux C
```

```
C  results, while still meeting convergence criteria, are somewhat    C
C  'raggedy' since d(G_crit)/dx_E is very large in this area.  This   C
C  raggedness can be observed in the mass flux contour plots.         C
C                                                                     C
C  - short tube / orifice correction might be addressed, but it's     C
C  always conservative to assume a thoroughly dispersed mixture       C
C  - the factor N relating the equil. quality x_E  to the static      C
C  quality at the throat varies from 0 to 1 as x_E ranges from 0 to   C
C  0.14.  This factor controls the rate of mass transfer between the  C
C  phases at the throat, sliding between zero (frozen) and unity      C
C  (equilibrium). Could be advantageous to examine other means of     C
C  correlating the two.                                               C
C                                                                     C
C  - the H-F model's results are strictly larger than those of the HEM C
C  in the regime tested, but results, but the predictions converge    C
C  when upstream condition is  1) significantly subcooled, or         C
C  2) 2-phase at higher qualities (See Figs. 5 and 7 in               C
C  Henry & Fauske; HEM underpredicts H-F most significantly at higher  C
C  pressures and near the liquid saturation point                     C
C                                                                     C
C                                               EAS 7/18/04    C
CCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCC
CCCCCCCCCCCCCCCCCCCC

    IMPLICIT DOUBLE PRECISION (A-H,O-Z)
        INCLUDE 'nprop.cmn'
        REAL*8 FUNC,H0,PT,TT,T0,H,SV,S0,X0,DENS,P0,C_VF,C_PG,XE,SEF,SEG
        REAL*8
ALPHA_0,ALPHA_T,VFT,VGT,BETA,POLYTROPIC,N,GAMMA,ETA,VF0,VG0
    DIMENSION IWORK(NPROP), IWANT(NPROP), PROPR(NPROP),
PROPSI(NPROP)
    DIMENSION WAVRI(NRIMAX), RI(NRIMAX), IRIFLG(NRIMAX)
    DIMENSION IPCHK(5), IPFLG(5)
        REAL*8 MIX_PROP,S0F,S0G,RHS,GG,ROOT,GCRIT,DSDP,VFE,VGE
        REAL*8 FUDGE, ORIFICE_C,PERTURBED_T,RHO_T

        PARAMETER (DP = 1.e-5)

C Orifice and short tube discharge coefficient as defined on p. 185
C of Henry and Fauske.  This fudge factor modifies the critical pressure
C ratio and mass flow rate (see <EQ. 47>).  It should be set to unity for
C subcooled flows; for two phase flow, if the flow regime may be considered
C compressible, it may be justified to take a lower value (0.84 is recommended
C in the paper).  However, in the interest of conservatism it may be best
C to leave this set to 1.0 throughout.
```

```
        ORIFICE_C=1.0
C Define the throat to upstream pressure ratio as per <EQ. 34>
        ETA = PT/P0

C In H-F, liquid phase density at throat equal to upstream stagnation density
C See discussion preceeding and following <EQ. 17>.
        VFT = VF0

C Vapor phase static density at throat (to be computed below if X0 > 0)
        VGT = 0

    DO 310 I=1,NPROP
            IWANT(I) = 0
310   CONTINUE
        IWANT(7) = 1
C Isochoric heat capacity, C_V
        IWANT(8) = 1

        CALL HSSOLV(2, PT, S0, TPOUT, D1, DV, DL,
   &         I2PH, Q, IWORK, PROPR, IERR)

        TT = TPOUT
        IF ((I2PH .EQ. 2) .OR. (I2PH .EQ. 4)) THEN
            DENS = 1./MIX_PROP(Q, 1.0/DL, 1.0/DV)
        ELSE
            DENS = D1
        ENDIF
C Saturated liquid and vapor specific volumes at (PT,S0)
        VFE=1./DL
        VGE=1./DV

        CALL PROPS(IWANT,TT,1./VFE,PROPSI, PROPR, 1, I2PH, 0,
   &ISFLG,0, ICFLG, IPCHK, IPFLG, 0, 0, WAVRI, RI, IRIFLG)

C Obtain properties of saturated liquid at S0: heat capacity and entropy
        C_VF = PROPSI(8)
        SEF = PROPSI(7)

C Approximate pressure derivative of saturated liquid enthalpy by evaluating
C first the change in temperature and liquid density following from a
C change DP in PT with entropy held constant
        CALL HSSOLV(2, PT+DP, S0, TPOUT, D1, DV, DL,
   &         I2PH, Q, IWORK, PROPR, IERR)
        PERTURBED_T = TPOUT
C evaluate saturated liquid enthalpy at this new temperature and density
```

```
        CALL PROPS(IWANT,PERTURBED_T,DL,PROPSI, PROPR, 1, I2PH, 0,
     &ISFLG,0, ICFLG, IPCHK, IPFLG, 0, 0, WAVRI, RI, IRIFLG)

C approximate the derivative by (S(PT+DP)- S(PT))/DP and convert from MPa to Pa
        DSDP = (PROPSI(7)-SEF)/DP*1.e-6

C Obtain entropy of saturated vapor at S0
        CALL PROPS(IWANT,TT,1./VGE,PROPSI, PROPR, 1, I2PH, 0,
     &  ISFLG,0, ICFLG, IPCHK, IPFLG, 0, 0, WAVRI, RI, IRIFLG)

        SEG = PROPSI(7)

C Write the local equilibrium quality at the throat in terms of phase
C entropies at the throat.  Represents quality that fluid would
C possess if phases were allowed to equilibrate <EQ. 23>
        XE = (S0 - SEF)/(SEG - SEF)

C Define the fudge factor correlating dX/dPT to dXE/dPT as per <EQ. 30>
        IF (XE. GT. 0.14) THEN
                N = 1.
        ELSE IF (XE .GT. 0) THEN
                N = XE/0.14
        ELSE
                N = 0.
        ENDIF

        IF (X0 .GT. 0) THEN
C Isobaric heat capacity, C_p
                IWANT(9) = 1
                I2PH = 2
                FUDGE=0

124             CALL PROPS(IWANT,TT,1./VGE+FUDGE,PROPSI, PROPR, 1, I2PH,
0,
     &          ISFLG,0, ICFLG, IPCHK, IPFLG, 0, 0, WAVRI, RI, IRIFLG)

C This loop is necessary because of apparent fluctuation
C in the least significant digit of the saturated vapor specific volume at P_E,
C VGE.  If one calculates this volume using HSSOLV, then feeds it back into PROPS
C to obtain the isobaric heat capacity C_p of the vapor, an error *sometimes*
C results.  C_p is not defined for a two-phase mixture; occasionally, the least
C significant digit of VGE varies such that PROPS believes the fluid being passed
C has quality very slightly less than 1.0.  C_p, undefined in this regime, is
C returned as zero.  Hence, if PROPS indicates that it believes the fluid is 2-phase,
C adjust the density very slightly to return to the vapor-only regime and try the
```

```
C calculation again:

          IF(I2PH .EQ. 2) THEN
               FUDGE=FUDGE-(1./VGE)*1.e-8
               GOTO 124
          ENDIF
C Isentropic exponent = C_p/C_V evaluated for saturated vapor at the throat
C (PROPS input arguments are T and sat vapor density at this state point)
          GAMMA = PROPSI(9)/PROPSI(8)
          C_PG=PROPSI(9)
          IWANT(8)=0
          IWANT(9)=0

C Finally, obtain liquid and vapor saturatation enthalpies for upstream stagnation
C conditions.  Arguments: stagnation temperature and saturated liquid and vapor
C densities
          CALL PROPS(IWANT,T0,1./VG0,PROPSI, PROPR, 1, I2PH, 0,
  &       ISFLG,0, ICFLG, IPCHK, IPFLG, 0, 0, WAVRI, RI, IRIFLG)
          S0G = PROPSI(7)

          CALL PROPS(IWANT,T0,1./VF0,PROPSI, PROPR, 1, I2PH, 0,
  &       ISFLG,0, ICFLG, IPCHK, IPFLG, 0, 0, WAVRI, RI, IRIFLG)
          S0F = PROPSI(7)

C Express the polytropic exponent at the throat.  Recall that X_throat = X0 and
C the expansion is isentropic.   <EQ. 19>
          POLYTROPIC = ((1.-X0)*C_VF/C_PG + 1)/((1.-X0)*C_VF/C_PG+
  &       1/GAMMA)

C The vapor specific volume at the throat obtained assuming polytropic behavior:
C See <EQS. 18, 19, 38>
          VGT = VG0*(ETA**(-1./GAMMA))

C Collecting terms into ALPHA_0 <EQ. 36>, ALPHA_T <EQ. 37> and BETA <EQ.
38>:
          ALPHA_0 = X0*VG0/((1.-X0)*VF0+X0*VG0)
          ALPHA_T = X0*VGT/((1.-X0)*VF0+X0*VGT)

          BETA = (1./POLYTROPIC+(1.-VF0/VGT)*((1.-X0)*N*PT*1.e6*DSDP/
  &       (X0*(SEG-SEF))-C_PG*(1./POLYTROPIC-1./GAMMA)/(S0G-S0F))

C Compute the RHS of <EQ. 33>
          GG = GAMMA/(GAMMA-1.)
          RHS = (((1.-ALPHA_0)*(1.-ETA)/ALPHA_0+GG)/
  &       (1./(2.*BETA*ORIFICE_C*ORIFICE_C*
```

```
   &           ALPHA_T*ALPHA_T)+GG))**(GG)

C To make this amenable to numerical solution via maximization, return the
C value -(RHS-ETA)^2, which will exhibit a maximum value of zero when the
C correct guess for the root, PT, is supplied.
               RHS = RHS-ETA
               ROOT=-(RHS*RHS)

C At last, evaluate the critical mass flux as a function of PT
               GCRIT=(X0*VGT/(POLYTROPIC*PT*1.e6)+(VGT-VF0)*((1.-
X0)*N/(SEG-
   &           SEF)*DSDP-(X0*C_PG*(1./POLYTROPIC-1./GAMMA)/(PT*1.e6*
   &           (S0G-S0F)))))**(-0.5)

          ELSE
C Subcooled upstream stagnation conditions

               IF (XE. GT. 0) THEN

C Pressure ratio guess was valid in that quality at the nozzle is nonzero
                    GCRIT=((VGE-VF0)*N*DSDP/(SEG-SEF))**(-0.5)
                    RHS = 1. -
VF0*GCRIT*GCRIT/2/(P0*1.e6)/ORIFICE_C/ORIFICE_C
                    RHS = RHS-ETA
                    ROOT = -(RHS*RHS)
               ELSE

C Pressure ratio guess was not good, resulting in subcooled nozzle conditions.
C H-F does not function under these conditions -- see <EQ. 45> w/ N = 0.
C Return zero mass flux so that the root finder knows to discard this attempt.
                    GCRIT = 0
               ENDIF
          ENDIF

C Return the fluid density at the throat to CRIT_MASS_FLUX for use in calculating
C the thrust coefficient.  Specifying the mass flux and fluid density also
C specifies the velocity of the homogenized fluid.
          IF (GCRIT .GT. 0) THEN
               IF (X0. LE. 0) THEN
                    RHO_T = 1./VFT
               ELSE IF (X0 .LE. 1) THEN
                    RHO_T = 1./MIX_PROP(X0,VFT,VGT)
               ELSE
                    RHO_T = 1./VGT
               ENDIF
```

```
      ELSE
            RHO_T = 0
      ENDIF
      END

      FUNCTION THRUST_COEFF(GE, RHO_T, P_AMB, PT, P0)

CCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCC
CCCCCCCCCCCCCCCCCCC
C                 THRUST_COEFF                  C
C  Calculates the thrust coefficient by            C
C                                            C
C               K = PT/P0*(1+G^2/RHO_T/PT),             C
C  where                                     C
C  P0 and PT are the stagnation and throat pressures [Pa_gauge]      C
C  G is the mass flux [kg/m^2/s] as evaluted by H-F or the HEM,      C
C  RHO_T is the fluid density [kg/m^3] at the throat             C
C                                   EAS 7/30/04 C
CCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCC
CCCCCCCCCCCCCCCCCCC

      REAL*8 GE, RHO_T, P_AMB, PT, P0, THRUST_COEFF

      THRUST_COEFF=(PT-P_AMB)/(P0-P_AMB)*(1.0+GE*GE/
   &  RHO_T/((PT-P_AMB)*1.e6))

      RETURN
      END
```

Page Break

```
function [p,t,x,ts,hf,hg,d,h0,pa,da,ge,tc,tj,ta,hfa,hga] = ...
 GetProperties(pres, temp, qual, pdmg, pamb, use_model)

% Given N (pres, temp, qual) state points and M damage
% pressures, GetProperties invokes the NIST steam tables
% to obtain
% p[N] state point pressures      [psi]
% t[N] temperatures          [deg F]
% x[N] qualities          [-]
% ts[N] saturation temperatures     [deg F]
% hf[N] liquid phase enthalpies at ts  [Btu/lbm]
% hg[N] vapor phase enthalpies at ts  [Btu/lbm]
% d[N] densities            [lbm/ft^3]
% h0[N] enthalpies           [Btu/lbm]
% pa[N] pressures at aysmptotic plane  [psi]
% da[N] densities at asymptotic plane  [lbm/ft^3]
```

```
% ge[N] critical mass fluxes      [lbm/ft^2/s]
% tc[N] thrust coefficients       [-]
% tj[N][M] temperatures at pdmg[M]   [deg F]
% ta saturation temp at ambient pres.  [deg F]
% hfa liquid phase enthalpy at ta    [Btu/lbm]
% hga vapor phase enthalpy at ta     [Btu/lbm]
%
% EAS 7/12/04
%
% 7/28/04:
% To obtain the ge and tc, GetProperties uses the HEM or H-F according to
% the rule specified by the integer flag use_model. See ansijet.m and
% QUERYST.FOR for documentation regarding use_model.

n_outputs=12;

% define conversion factors for use in UnitConverter
pres_conv = [0 0 0 0 1 0];
temp_conv = [1 0 0 0 0 0];
enth_conv = [0 -1 0 1 0 0];
dens_conv = [0 1 -3 0 0 0];
mflx_conv = [0 1 -2 0 0 0];

% convert from english to SI units
pres=UnitConverter(pres,pres_conv);
temp=UnitConverter(temp,temp_conv);
pdmg=UnitConverter(pdmg,pres_conv);
pamb=UnitConverter(pamb,pres_conv);

% concatenate inputs. This is not strictly necessary
% since the mex routine can handle multiple input variables,
% but the functionality would be identical regardless
% the last two inputs should be the number of state points
% and the number of contours per state point

npts=size(pres);
npts=npts(:,2);

nconts=size(pdmg);
nconts=nconts(:,2);

mex_input=[pres temp qual pdmg pamb npts nconts use_model];

% call QUERYST.FOR via QUERYST.DLL
mex_output=queryst(mex_input);
```

```
% reconstruct outputs
for i=1 : npts
  p(i)=mex_output(i);
  t(i)=mex_output(i+npts);
  x(i)=mex_output(i+npts*2);
  ts(i)=mex_output(i+npts*3);
  hf(i)=mex_output(i+npts*4);
  hg(i)=mex_output(i+npts*5);
  d(i)=mex_output(i+npts*6);
  h0(i)=mex_output(i+npts*7);
  pa(i)=mex_output(i+npts*8);
  da(i)=mex_output(i+npts*9);
  ge(i)=mex_output(i+npts*10);
  tc(i)=mex_output(i+npts*11);
  for j=1 : nconts
    tj(i,j)=mex_output(npts*n_outputs+(i-1)*nconts+j);
  end
end
ta=mex_output(npts*n_outputs+npts*nconts+1);
hfa=mex_output(npts*n_outputs+npts*nconts+2);
hga=mex_output(npts*n_outputs+npts*nconts+3);

% if no pressure / temperature contours were requested, create an empty
% dummy tj to avoid undefined variable contours later on
if(nconts==0)
  tj=[];
end

% convert back from SI to english units
p=UnitConverter(p,-pres_conv);
t=UnitConverter(t,-temp_conv);
ts=UnitConverter(ts,-temp_conv);
hf=UnitConverter(hf,-enth_conv);
hg=UnitConverter(hg,-enth_conv);
h0=UnitConverter(h0,-enth_conv);
pa=UnitConverter(pa,-pres_conv);
d=UnitConverter(d,-dens_conv);
da=UnitConverter(da,-dens_conv);
ge=UnitConverter(ge,-mflx_conv);
ta=UnitConverter(ta,-temp_conv);
tj=UnitConverter(tj,-temp_conv);
hga=UnitConverter(hga,-enth_conv);
hfa=UnitConverter(hfa,-enth_conv);
return
```

Page Break

```
function conv_x = UnitConverter(x_o, dimension_array)

conv_factors = zeros(size(dimension_array));

% converts a quantity from the english unit system to SI
% or vice versa, returning the converted value as conv_x.
% x_o quantity (in english or SI units) to be converted
% default operation is english -> SI
% dimension_array of integers describing the english system
% unit to be converted to SI, viz:

% (1) temperature conversion F -> K
conv_factors(1)=5./9.;
% (1) can only take on the values 1 (F->K) and -1 (K->F).
% If (1) is nonzero, all other members of dimension_array
% will be ignored!
% (2) lbm -> kg
conv_factors(2)=0.45359237;
% (3) ft -> m
conv_factors(3)=0.3048;
% (4) Btu -> kJ
conv_factors(4)=1.05505585;
% (5) psi -> MPa
conv_factors(5)=0.00689475729;
% (6) deg F -> deg K
conv_factors(6)=5./9.;
%
% The value passed is the exponent of the unit to be converted.
% Example usage: to convert density [lbm/ft^3] to SI units,
% one would pass dimension_array = [0 1 -3 0 0 0].
% To convert [kg/m^3] back to [lbm/ft^3], one would simply
% obtain the inverse of the conversion factor applied previously
% by passing [0 -1 3 0 0 0].

result = x_o;
if dimension_array(1) == 1
  result = (result + 459.4) * conv_factors(1);
elseif dimension_array(1) == -1
  result = result / conv_factors(1) - 459.4;
else
  for i = 2:length(conv_factors)
    if dimension_array(i) ~= 0
      result = result * conv_factors(i)^dimension_array(i);
    end
  end
end
```

```
conv_x = result;
```

APPENDIX II: CONFIRMATORY DEBRIS GENERATION ANALYSES

The Nuclear Energy Institute (NEI) guidance contains recommendations that will determine the quantities of insulation debris generated with the zone of influence (ZOI). These recommendations include the size of the ZOI based on the insulation destruction pressure and the fraction of the insulation located within the ZOI that subsequently is damaged into the small-fine-debris category. Confirmatory research was performed to ascertain whether the NEI recommendation would reliably result in conservative estimates for the volumes of debris generated within the ZOI. This appendix documents the confirmatory research estimates for the volumes of small fine debris. The confirmatory research for determining the size of the ZOI is the subject of Appendix I. Both the NEI guidance and the confirmatory research used the ANSI/ANS-58.2-1988 standard to calculate the jet isobar volumes with very similar results. The confirmatory research issues addressed herein include the following.

- The NEI guidance recommends the assumption that 60% of the fibrous and 75% of the reflective metal insulation (RMI) volume contained within the ZOI becomes small fine debris. Confirmatory research was performed that integrated the insulation damage versus jet pressures over the ZOI volume to determine the fraction of the insulation within the ZOI that would become small fine debris based on available debris generation data.

- The NEI guidance recommends adapting the debris-size distribution for NUKON™ to other types of fibrous insulation that have a destruction pressure higher than that of NUKON™. The size distribution confirmatory research provides partial justification that supports that NEI recommendation.

- The applicability of air-jet-determined destruction pressures to two-phase pressurized-water-reactor (PWR) loss-of-coolant-accident (LOCA) jets has been questioned. NUREG/CR-6762 (Vol. 3) noted that data from the Ontario Power Generation (OPG) two-phase debris generation tests indicated that the destruction pressure could be lower for a two-phase jet than for an air jet and that the resultant debris could be finer. Therefore, it may be prudent to apply a safety factor to accommodate the uncertainty. This confirmatory analysis estimates the volume fractions for small fine debris if an alternate lower destruction pressure were used than those in the NEI guidance.

II.1 COMPARISON OF JET ISOBAR VOLUME CALCULATIONS

Three calculations of the jet isobar volumes were available for comparison.* The calculations were the following.

- The volumes determined from the NEI guidance recommended values for ZOI radii versus the destruction pressures in NEI baseline guidance's Table 3-1. The destruction pressure represents the jet isobar pressure for each particular ZOI radii.
- The volumes determined from the confirmatory research (Appendix I) for the ZOI radii versus the jet pressure.

---

* The volumes are actually presented in terms of the break diameter cubed (D^3) corresponding to an equivalent spherical radius in terms of r/D (i.e., $4/3\ \pi\ r^3/D^3$).

- The volumes determined from the boiling-water-reactor owners' group (BWROG) recommendation documented in their utility resolution guidance (URG). Although these volumes apply to a BWR steam jet rather than a PWR two-phase jet, the volumes are compared here to demonstrate the differences between PWR and BWR LOCA jets.

Both the NEI guidance and the confirmatory research volume calculations used the ANSI/ANS-58.2-1988 standard method, whereas the BWROG URG method used the computational-fluid-dynamics (CFD) code, NPARC, to evaluate the volumes. The equivalent spherical radii for these three methods are compared in Figure II-1.

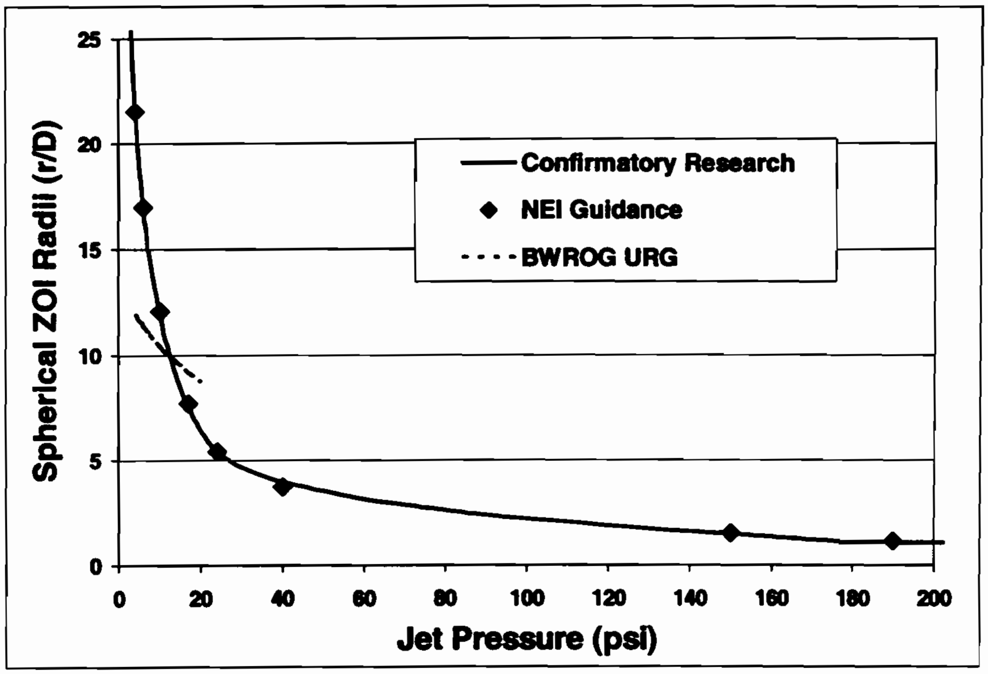

Figure VI-1. Comparison of Jet Isobar Volumes.

As shown, at the lower jet pressures, the pressure isobar volumes are much larger for the PWR two-phase LOCA jet than for the BWR steam jet. A principal reason for this difference is the higher energy associated with the higher pressure of a PWR reactor coolant system (RCS) than with a BWR RCS; however, another consideration is the accuracy of the ANSI/ANS-58.2-1988 standard at the lower pressures. For example, the validity of the assumption in the ANSI/ANS-58.2-1988 standard that the jet expands at a half angle of 10 degrees once the jet expansion has reached the asymptotic plane becomes more important at the lower expansion pressures. The accuracy of the debris volumes of insulations that damage significantly at the lower jet pressures is subject to the accuracy of this assumption. Note that the confirmatory research and NEI-recommended-equivalent spherical ZOI radii are in good agreement.

II.2 METHOD OF DETERMINING ZOI DEBRIS-SIZE DISTRIBUTIONS

The volume of debris generated within a ZOI depends on the following three factors: (1) the size of the ZOI defined by the spherical radius, (2) the concentration of a particular insulation within the ZOI, and (3) the fraction of the ZOI insulation that is damaged into a particular debris-size classification. The size distribution and spherical ZOI radius are interdependent. The threshold damage pressure and the jet volumes determine the size of the ZOI (Appendix I). The insulation concentration within a ZOI is determined by plant-specific information (i.e., the volume of a particular insulation within the ZOI divided by the volume of the ZOI).

Integration of experimental debris generation data is required to determine the fraction of the ZOI insulation that is damaged into a particular debris-size classification (e.g., NEI small fine debris). A generalized equation was offered in NUREG/CR-6808 for this integration. A slightly expanded version of this equation is

$$F_{ZOI} = \frac{3}{r_{ZOI}^3} \int_0^{r_{ZOI}} f_d\left(P_{jet}(r)\right) r^2\, dr \quad ,$$

where

F_{ZOI} = the fraction of the ZOI insulation type i that is damaged into a particular debris-size classification;

f_d = the fraction of debris damaged into a particular debris size as a function of the jet pressure P_{jet}, which is a function of the spherical radius, r, within the ZOI; and

r_{ZOI} = the outer radius of the ZOI.

Implicit in this integration is the assumption that the insulation is uniformly distributed within the ZOI, which may not be realistic. Because the functional information needed for this integration is not available in an equation form simple enough for a formal integration to proceed, the following simplification is used:

$$F_{ZOI} = \frac{1}{r_{ZOI}^3} \sum_j \left[\frac{f_{fines}\left(P_{jet}(r_j)\right) + f_{fines}\left(P_{jet}(r_{j-1})\right)}{2} \left(r_j^3 - r_{j-1}^3\right) \right] \quad ,$$

where

f_{fines} = the fraction of debris damaged into a particular debris size as a function of the jet pressure P_{jet} at a radius of r_j.

The spherical ZOI is first subdivided into numerous spherical shells (j). The precision of the integration increases with the number of subdivisions. In a spreadsheet, the jet pressure is listed in increasing values and then the spherical radii are determined, followed by the damage fraction evaluated at each r_j. For the intervals, the average damage across the interval and the volume of the interval is determined. Multiplying the average interval damage by the interval volume, summing, and dividing by the total ZOI volume results in the debris fraction for the ZOI.

II.3 EVALUATION OF DEBRIS SPECIFIC DAMAGE FRACTIONS AND POTENTIAL DEBRIS VOLUME

Potential debris volumes were calculated for fibrous, RMI, and particulate debris types and compared with the NEI baseline model to determine whether the baseline is conservative. The potential volume of debris is defined as the fraction of the ZOI debris damaged into a particular debris size multiplied by the total volume of the sphere, as

$$V_{Potential} = F_{ZOI} \left(\frac{4}{3} \pi \right) r_{ZOI}^{3} \quad .$$

Note that to calculate the volume of small fine debris generated, the potential volume must be multiplied by the concentration of insulation ($C_{insulation}$), i.e., the fraction of the ZOI actually occupied by the insulation, and by the pipe break diameter cubed. Again, it is assumed that the insulation type in question is uniformly distributed over the ZOI, regardless of the size of the ZOI, as

$$V_{Fines} = C_{Insulation} \, V_{Potential} \, D^{3} \quad .$$

II.3.1 Fibrous Debris

The fibrous insulation types evaluated include NUKON™, Transco (Transco Products, Inc., or TPI), Temp-Mat, K-wool, and Knauf. Table II-1 shows the NEI-guidance-recommended destruction pressures and an alternate set of values used herein to test the sensitivity of the potential debris volumes to the destruction pressures.

Table VI-1. Fibrous Insulation Destruction Pressures

Insulation	NEI Recommendation	Alternate Lower Pressure
NUKON™	10 psi	6 psi
TPI	10 psi[*]	6 psi
Knauf	10 psi	6 psi
Temp-Mat	17 psi	10 psi
K-wool	40 psi	17 psi

II.3.1.1 Low-Density Fiberglass (LDFG) Debris

A review of the air jet testing debris generation data, both the BWROG Air Jet Impact Testing (AJIT) data (BWROG URG) and the drywell debris transport study (DDTS) data [NUREG/CR-6369, 1999], demonstrates that NUKON™, TPI, and Knauf fiberglass insulations underwent similar damage. These insulations have approximately the same as-manufactured density (~2.4 lb/ft^3), and their recommended minimum pressures for destruction are usually taken to be the same pressure. Therefore, these insulations have been grouped together as LDFG insulation.

The fractions for the small fines from the AJIT debris generation test data are plotted in Figure II-2 as a function of the jet centerline pressure for these three types of LDFG insulations. A curve was drawn through the data to continuously represent the damage for use in the damage integration over the ZOI. One set of seven data points was from tests (in the DDTS) that used a

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[*] NEI guidance considers TPI fiber blankets to behave similarly to NUKON™ blankets.

4-in. nozzle, whereas the remainder used a 3-in. nozzle. The 4-in. nozzle data from the DDTS generally shows more damage than do the 3-in. nozzle tests. The basic reason for the higher damage was that with the larger-diameter jet, more of the target insulation blanket was exposed to higher pressures. Note that the data were correlated by the estimated jet centerline pressure but that the pressure on the blanket decreased outward from the centerline. When the blanket was placed in close to the jet, the ends of the blanket were hit with substantially less force of flow than the centerline for which the data were correlated. For example, for the 3-in. nozzle data point for NUKON™ at a jet pressure of 20 psi, only ~7% of the insulation was damaged into small fine debris, whereas the TPI blankets in the 4-in. nozzle were totally destroyed at this pressure. Apparently, testing blanket destruction for insulations requiring a pressure higher than ~17 psi needs a jet nozzle larger than 3 in. For LDFG, any jet pressure larger than ~17 psi will totally destroy the blanket into small fine debris, whereas the NEI guidance cited an OPG two-phase jet test with 52% of the insulation damaged into small fine debris as their basis of conservatism.

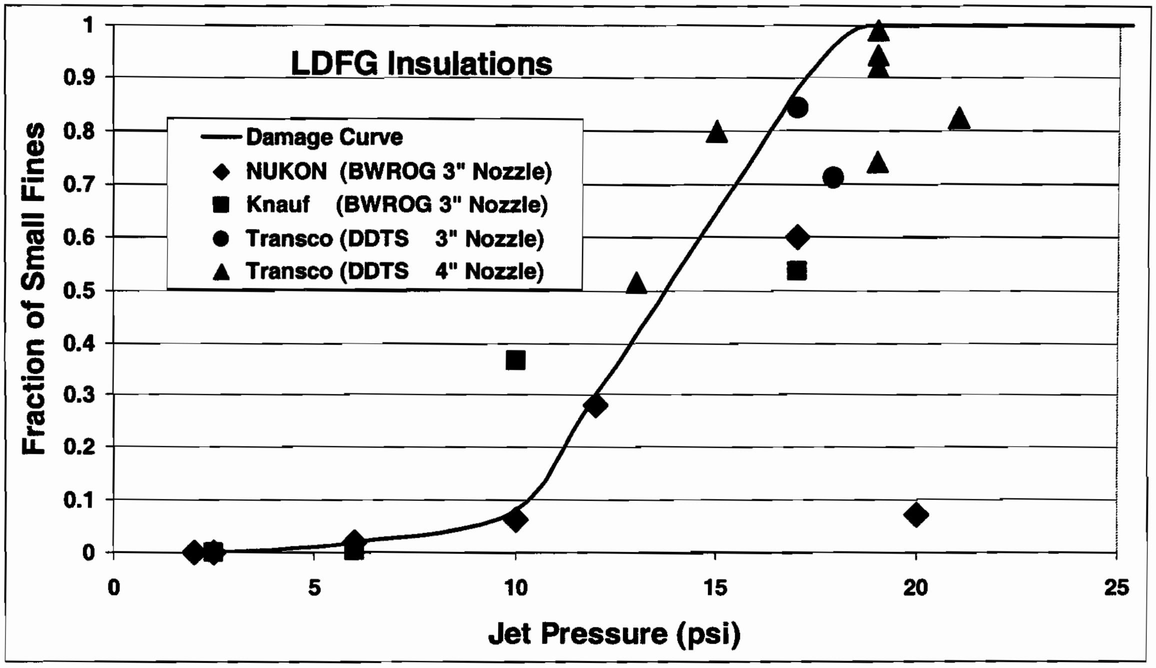

Figure VI-2. LDFG Damage Curve for Small Fine Debris.

Another significant point of discussion is that the threshold of damage for LDFG insulation has been specified as 10 psi, where Figure II-2 clearly shows damage at jet pressures <10 psi. Apparently, neglecting the tail of the damage curve was considered acceptable for the BWR strainer resolution because of the lesser BWR jet volumes at lower pressures, as shown in Figure II-1. However, the much larger jet volumes below 10 psi for the PWR jet shown in Figure II-1 make the neglect of the tail less acceptable.

The results of debris-size distribution integration over the ZOI are provided in Table II-2. A lower alternate damage pressure results in a larger equivalent spherical ZOI; however, a lesser fraction of the debris is damaged into small fine debris. The use of the alternate damage pressures over the NEI-recommended damage pressures for PWR analyses would result in

~16% more small fine debris. The potential debris volumes are compared in Figure II-3, along with an estimate using the baseline guidance. The baseline estimate is simply 60% of 4/3 π $(12.1/D)^3$. As shown, the baseline guidance appears to be conservative, but not overly so.

Table VI-2. Results of Debris-Size Distribution Integration for LDFG Insulations

Jet Pressure Isobar Volume Calculation	Radius of Sphere (r/D)	Fraction Small Fines	Potential Debris Volumes (V/D^3)
NEI-Recommended Damage Pressures			
BWROG Steam Jet	10.4	0.83	3910
PWR Two-Phase Jet (Confirmatory)	11.9	0.53	3790
Alternate Damage Pressures			
BWROG Steam Jet	11.4	0.65	3980
PWR Two-Phase Jet (Confirmatory)	17.0	0.22	4410

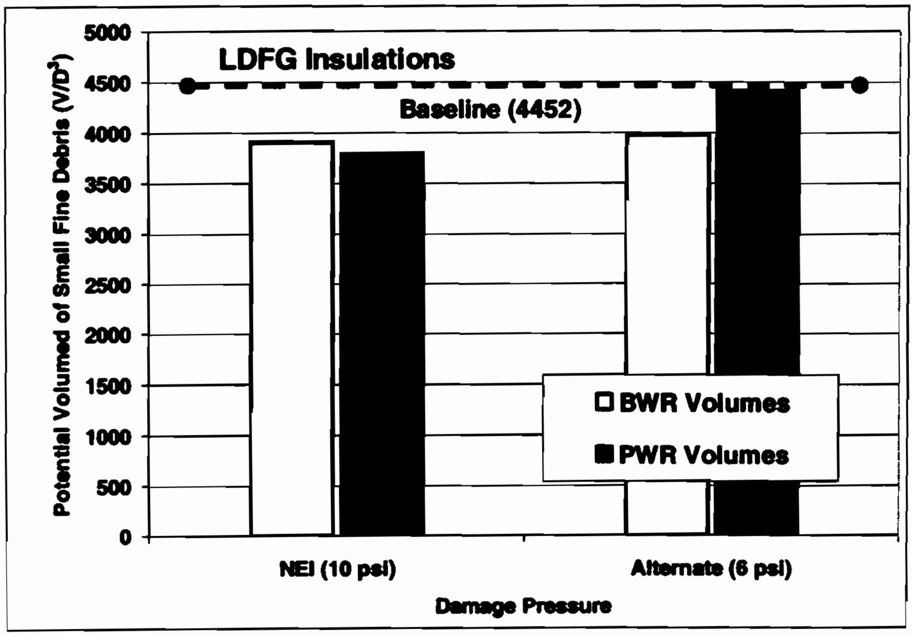

Figure VI-3. Potential Volumes of Small Fines LDFG Debris. fix y axis label (volumes)

The NEI baseline guidance completely neglects the transport of large debris to the sump screen; however, some plants will likely need to consider large debris transport as part of a more realistic evaluation. Therefore, the following equation is provided to estimate the volume of large debris generated within the ZOI:

$$V_{Large} = C_{Insulation}\left(1 - F_{ZOI}\right)\left(\frac{4}{3}\pi\right) r_{ZOI}^{3}\, D^{3} \quad .$$

Also, plants that must perform more realistic evaluations may need to subdivide the baseline small-fine-debris class into fines and small-piece debris, where the fines (e.g., individual fibers) remain suspended in the pool and the small-piece debris sinks to the pool floor where the debris may or may not transport to the sump screen. The baseline guidance has the inherent assumption that all of its small fine debris essentially remains suspended.

During the debris generation tests conducted during the DDTS, 15% to 25% of the debris from a completely disintegrated TPI fiberglass blanket was classified as nonrecoverable. The nonrecoverable debris either exited the test chamber through a fine-mesh catch screen or deposited onto surfaces in such a fine form that it could not be collected by hand (it was collected by hosing off the surfaces). Therefore, it would be reasonable to assume that 25% of the baseline small fine debris (i.e., F_{ZOI}) is in the form of individual fibers and that the other 75% is in the form of small-piece debris.

II.3.1.2 Temp-Mat Debris

Temp-Mat is much higher-density insulation (~11.8 lb/ft^3) than the LDFG insulation and requires a significantly higher-pressure jet pressure to damage the insulation. The Temp-Mat insulation debris fractions for the small fine debris from the AJIT tests are shown in Figure II-4. This figure shows six data points for Temp-Mat, two of which were tests where no significant damage was noted. The test with the maximum damage had ~36% of the insulation damaged into small fine debris, with the remainder of the insulation forming large-piece debris. Unfortunately, no tests were conducted with jet pressures high enough to complete the damage curve to total destruction into small fine debris, as was done for the LDFG insulations. Therefore, a conservative extrapolation of the data is required to perform the debris generation integration over the equivalent ZOI sphere. The extrapolation used herein is shown as a dashed line in Figure II-4. The selection of the NEI-guidance damage pressure of 17 psi is also illustrated in Figure II-4, where it is seen that significant small fine debris is generated at jet pressures below 17 psi.

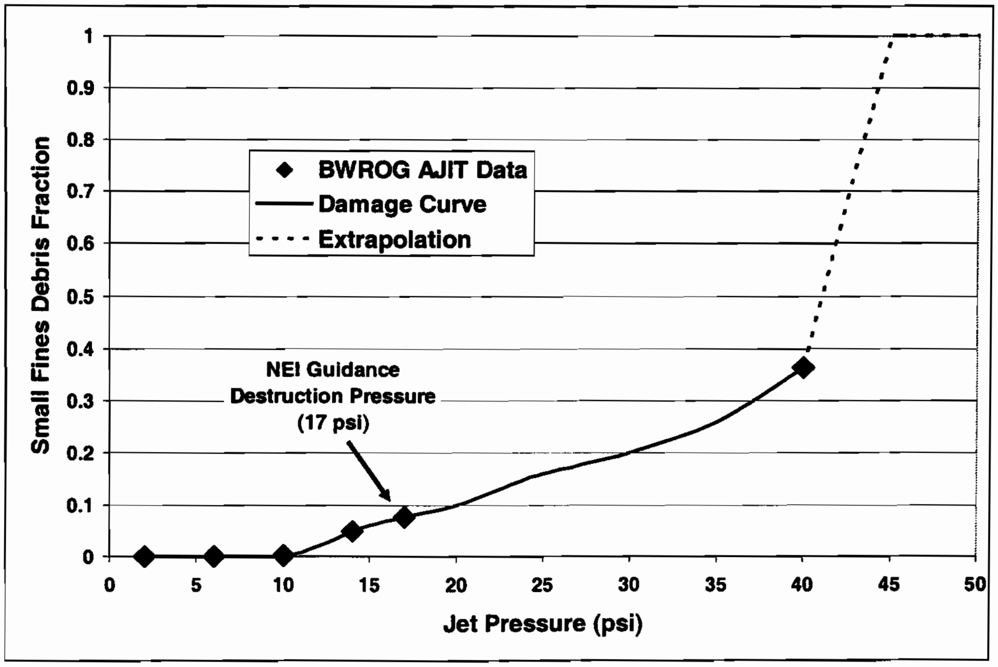

Figure VI-4. Temp-Mat Damage Curve for Small Fine Debris. fix x axis to "fine"

The results of the Temp-Mat debris-size distribution integration over the ZOI are provided in Table II-3. The potential debris volumes are compared in Figure II-5, along with an estimate using the baseline guidance [60% of 4/3 π $(7.8/D)^3$]. A lower alternate damage pressure results in a larger equivalent spherical ZOI; however, a lesser fraction of the debris is damaged into small fine debris. The use of the alternate damage pressures over the NEI-recommended damage pressures for PWR analyses would result in ~36% more estimated small fine debris. For Temp-Mat insulation, the baseline is conservative with respect to both the NEI-guidance damage pressure of 17 psi and the alternate pressure of 10 psi.

The debris-size estimate for Temp-Mat has more uncertainty associated with the estimate than does the similar calculation for LDFG, primarily because of more limited data. The negative uncertainties include the neglect of the damage curve tail by the NEI-recommended damage pressure (quantified using the alternate damage pressure) and the fact that the BWROG AJIT tests used the small 3-in. nozzle, which makes it difficult to subject the entire target blanket to the characteristic jet pressure (near the centerline pressure) when the blanket is located close to the nozzle. The positive uncertainty is the sharp extrapolation of the damage curve to 100% destruction at 45 psi. In this case, it is possible that the positive uncertainty overshadows the negative uncertainties.

Table VI-3. Results of Debris-Size Distribution Integration for Temp-Mat Insulation

Jet Pressure Isobar Volume Calculation	Radius of Sphere(r/D)	Fraction Small Fines	Potential Debris Volumes (V/D^3)
NEI Recommended Damage Pressures			
PWR Two-Phase Jet (Confirmatory)	7.5	0.25	448
Alternate Damage Pressures			
PWR Two-Phase Jet (Confirmatory)	11.9	0.086	608

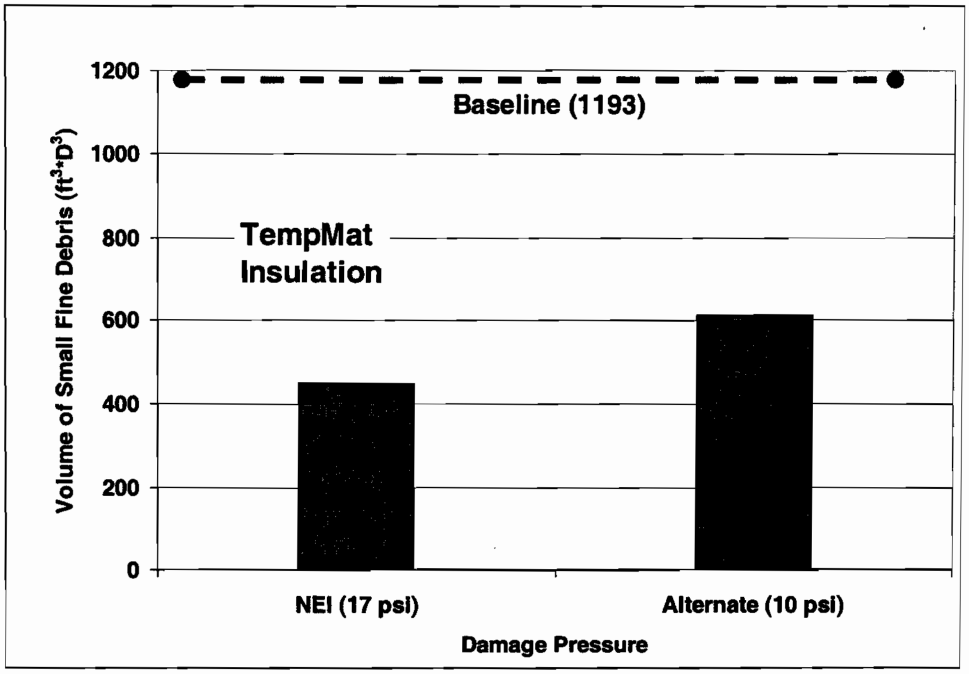

Figure VI-5. Potential Volumes of Small Fine Temp-Mat Debris.

II.3.1.3K-wool Debris

K-wool is also higher-density insulation (~10 lb/ft^3) than the LDFG insulation and requires an even higher-pressure jet pressure to damage the insulation. The NEI-recommended damage pressure for K-wool is 40 psi. The K-wool insulation debris fractions for the small fine debris from the AJIT tests are shown in Figure II-6. This figure shows only four data points for K-wool, two of which were tests where no significant damage was noted. The test with the maximum damage had ~7.1% of the insulation damaged into small fine debris, with much of the remainder of the insulation still contained in the blanket cover and still attached to the target mount. As with the Temp-Mat data, the K-wool damage curve is incomplete because the highest jet pressure tested was that of the NEI-recommended damage pressure. To perform the debris generation integration over the equivalent ZOI sphere, the test data were conservatively extrapolated, as shown in Figure II-6.

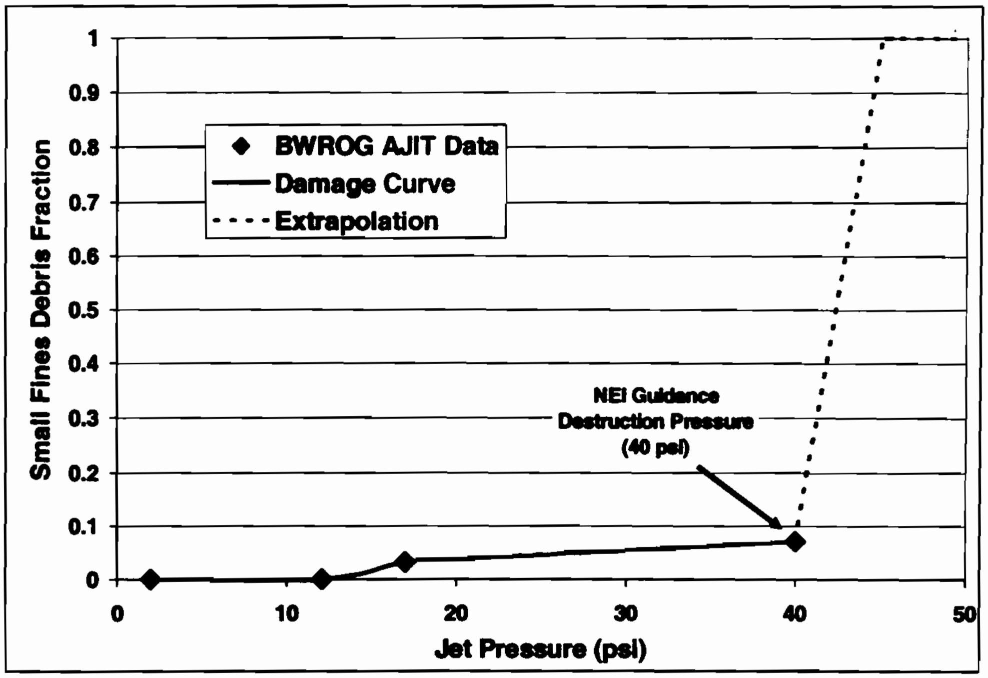

Figure VI-6. K-wool Damage Curve for Small Fine Debris. fix x axis label to "fine"

The results of the K-wool debris-size distribution integration over the ZOI are provided in Table II-4. The potential debris volumes are compared in Figure II-7, along with an estimate using the baseline guidance [60% of 4/3 π $(3.8/D)^3$]. The difficulty with the K-wool integration is that there is no debris generation data for a jet pressure higher than the NEI-recommended destruction pressure of 40 psi. Therefore, to ensure conservative debris-size integration, it must necessarily be assumed that the insulation is completely destroyed at a pressure higher than 40 psi (the integration herein assumed 100% at 45 psi). However, this assumption may be overly conservative. For K-wool insulation, the baseline is not conservative with respect to either the NEI-guidance damage pressure of 40 psi or the alternate pressure of 17 psi.

Table VI-4. Results of Debris-Size Distribution Integration for K-wool Insulation

Delete

Jet Pressure Isobar Volume Calculation	**Radius of Sphere(r/D)**	**Fraction Small Fines**	**Potential Debris Volumes (V/D^3)**
NEI-Recommended Damage Pressures			
PWR Two-Phase Jet (Confirmatory)	4.0	0.92	246
Alternate Damage Pressures			
PWR Two-Phase Jet (Confirmatory)	7.5	0.17	307

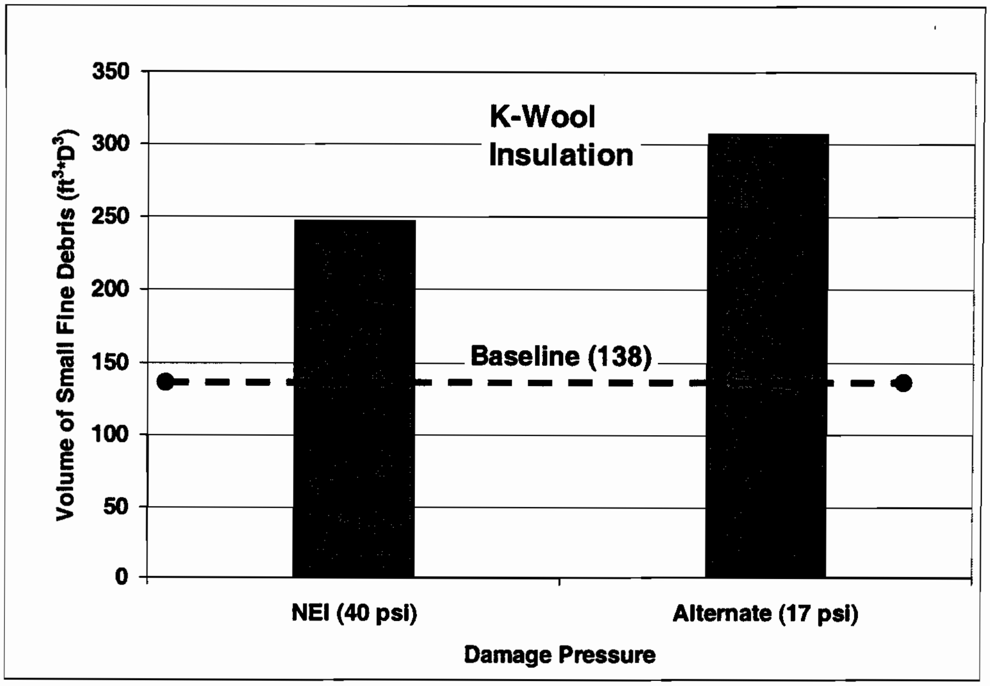

Figure VI-7. Potential Volumes of Small Fine K-wool Debris.

II.3.1.4 Correlation between Debris Size and Destruction Pressure

The NEI guidance contains the assumption that it is conservative to adapt the debris-size distribution for NUKON™ to other types of insulations that have a higher destruction pressure than NUKON™ (e.g., Temp-Mat and K-wool). This assumption is examined by comparing the debris generation data for LDFG, Temp-Mat, and K-wool, as shown in Figure II-8.

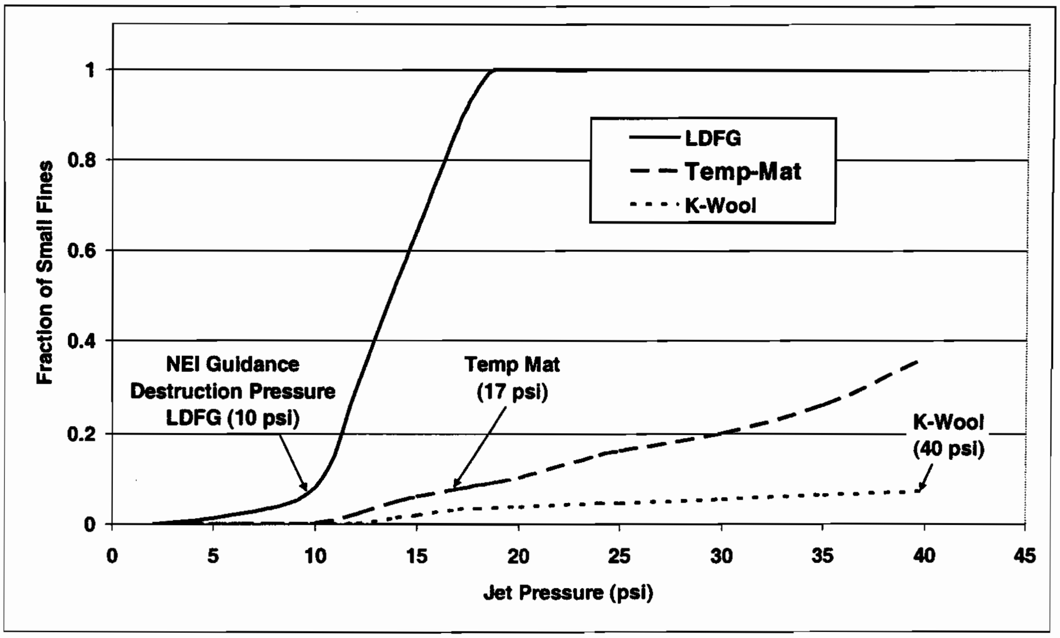

Figure VI-8. Comparison of Fibrous Insulation Damage Curves.

This damage curve comparison for LDFG, Temp-Mat, and K-wool does seem to support the concept that a higher destruction pressure results in the fractions of small fines being increasingly smaller as the destruction pressure increases. Certainly this is the case for Temp-Mat, where the baseline guidance is conservative relative to the integration herein where both the fractions of small fine debris and the potential debris volumes are smaller than the baseline guidance. Although this case is likely true for K-wool as well, it cannot be conclusively proven because of the complete lack of data beyond the NEI-recommended destruction pressure.

II.3.2 RMI Debris

The NEI guidance contains recommendations for three types of RMI insulation:

1. DARMET® manufactured by Darchem Engineering, Ltd.;
2. RMI manufactured by TPI; and
3. Mirror® marketed by Diamond Power Specialty Company (DPSC).

The NEI recommends that 75% of the RMI insulation contained in the equivalent spherical ZOI should be assumed to be turned into small fine debris. Table II-5 shows the NEI-recommended destruction pressures and the corresponding NEI-recommended radii for those pressures. Note that the ZOI for DARMET® and TPI are quite small compared with the ZOI for DPSC Mirror®.

Table VI-5, NEI-Recommended RMI Insulation Destruction Pressures and ZOI Radii

RMI Insulation	Destruction Pressures (psi)	ZOI Radius (L/D)
DARMET®	190 psi	1.3
TPI	190 psi	1.3
DPSC Mirror®	4 psi	21.6

Nearly all the debris generation data used to justify the NEI recommendations came from the BWROG Air Jet Impact Testing (AJIT) data [BWROG URG]; therefore, the NEI recommendations must be anchored to the insulation types as tested. Besides the BWROG AJIT tests, a single Nuclear Regulatory Commission (NRC)-sponsored test* was conducted using a stainless-steel DPSC Mirror® RMI cassette at the Siemens AG Power Generation Group (KWU) test facility in Karlstein am Main, Germany (1994 and 1995**can't find these as references in this section**) [SEA-95-970-01-A:2, 1996]. The cassettes and their closures, as tested in the AJIT tests with the cassettes mounted perpendicular to the jet centerline,† are provided in Table II-6. All of the cassettes tested had stainless-steel sheaths.

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*The NRC-sponsored test involved a stainless-steel Mirror® cassette mounted directly on a device designed to simulate a double-ended guillotine break (DEGB) such that the discharge impinged on the inner surface of the RMI target as it would an insulation cassette surrounding a postulated pipe break. This NRC-sponsored test was performed with a high-pressure blast of two-phase water/steam flow from a pressurized vessel connected to a target mount by a blowdown line with a double-rupture disk. In this test, the cassette was completely destroyed into debris that can be considered small fine debris.

†Two tests were conducted, with the cassette mounted parallel to the jet centerline.

A review of the data indicates that the stainless-steel sheaths were not directly penetrated by the air jet; rather, the sheaths disassembled at the seams, such as with rivet failures. Those cassettes secured by stainless-steel bands in addition to latches and strikes generally remained relatively intact. The severity of the damage, in terms of the generation of small fine debris, depends on the degree or ease of disassembling the cassette. However, when considering large-piece debris, all detached cassettes, disassembled or not, become large-piece debris.

Table VI-6. BWROG AJIT RMI Insulations Tested

Delet

Insulation	RMI Foils Tested	Cassette Closures
DARMET®	Stainless-Steel Foils	Darchem Stainless-Steel Bands and CamLoc® Latches and Strikes
TPI	Aluminum Foils	Latch and Strike Closures
TPI	Stainless-Steel Foils	Latch and Strike Closures
DPSC Mirror®	Aluminum Foils	Latch and Strike Closures
DPSC Mirror®	Stainless-Steel Foils	Latch and Strike Closures
DPSC Mirror®	Stainless-Steel Foils	Latch and Strike Closures and Sure-Hold Band Closures

II.3.2.1 DARMET®, Manufactured by Darchem Engineering, Ltd.

The NEI-recommended destruction pressure of 190 psi for stainless-steel DARMET®, manufactured by Darchem Engineering, Ltd. and held in place by Darchem stainless-steel bands and CamLoc® latches and strikes, is based on two AJIT tests, Tests 25-1 and 25-2 with jet centerline pressures on target of 190 and 590 psi, respectively. In both of these tests, the cassettes, although deformed, remained intact and attached to the target mount. In effect, no debris was generated. This result indicates that a pressure greater than 590 psi is required to generate debris, with the exception of a cassette mounted over the break, where the jet would enter the inside of the cassette. This scenario would almost certainly result in complete destruction of that cassette. Another possible exception could be a jet approximately parallel to the cassette sheath that could penetrate through the ends—a configuration that has not been tested. It is apparent that the baseline recommendation of assuming 75% of this insulation within a 1.3/D spherical radius becomes small fine debris is conservative.

II.3.2.2 RMI Manufactured by Transco Products, Inc.

TPI manufactures stainless-steel and aluminum RMI insulation. The NEI guidance recommends a destruction pressure of 190 psi for the TPI RMI. The TPI cassettes tested included both aluminum and stainless-steel foils encased in stainless-steel sheaths secured with latches and strikes (no bands were used). Although the recommended destruction pressure is 190 psi, a small amount of fine debris was noted for jet pressures as low as 10 psi (Test 21-3). On the other hand, only small quantities of fine debris (i.e., <0.5%) were found for tests with jet pressures as high as 600 psi. Figure II-9 shows the debris generation fractions for TPI stainless-steel RMI small fines.

Table II-8 **Table II-7?** shows a comparison of potential debris volumes when estimated using the NEI baseline guidance and when acknowledging debris generation at jet pressures as low as 10 psi. Recall that to get actual volumes of debris, the potential volumes must be multiplied by the insulation concentration and again by D^3. For the baseline estimate, the volume associated with a ZOI radius of 1.3/D is multiplied by 75% to get the baseline potential volume. For the alternate estimate, the ZOI volume out to a jet pressure of 10 psi was multiplied by 0.5% to get the alternate potential volumes. The application of the alternate pressure results in approximately three times as much small fine debris as using the baseline guidance. However, even these quantities are not very large compared with such insulations as LDFG.

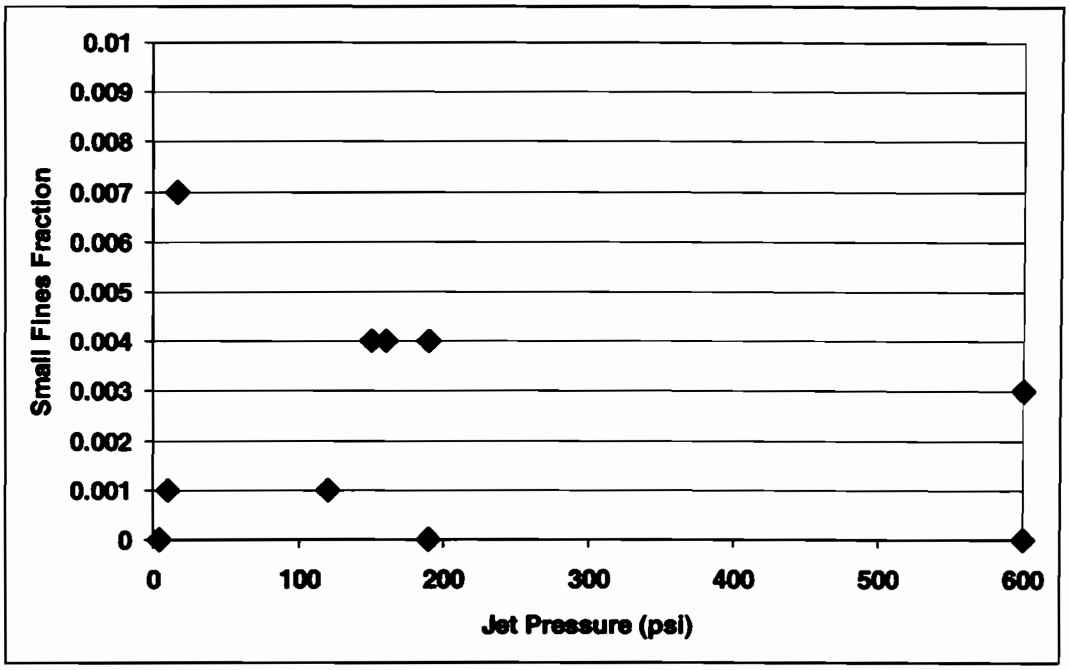

Figure VI-9. TPI Stainless-Steel RMI Small-Fine-Debris Fractions. fix y axis label (should be small-fine-debris fraction, not fines)

Table VI-7, Comparison of TPI Potential Debris Volumes

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Guidance	Damage Pressure (psi)	Radius of ZOI (r/D)	Damage Fraction	Potential Volume of Debris (V/D^3)
Confirmatory Recommended Jet Isobar Volumes				
NEI Guidance	190	1.5	0.75	10.6
Alternate	10	11.9	0.005	35.3

However, if the transport of large-piece TPI RMI debris becomes necessary to the strainer blockage evaluation, the use of 190 psi to define the ZOI is totally inadequate. Although the TPI stainless-steel sheaths may effectively contain the foils, their latches and strikes do not effectively keep the cassettes attached to the mounts (or pipes). AJIT Test 21-2, with a jet pressure of only 4 psi, shows the two cassette half sections detached from the target mount, i.e., the cassettes become large-piece debris. At 4 psi, the ZOI radius would be ~21.6/D;

therefore, numerous cassettes in various degrees of damage would be expected on the break-room floor. If the transport flow velocities were sufficient to move cassettes, then these cassettes could become a significant problem.

II.3.2.3 DPSC Mirror®, Manufactured by Diamond Power Specialty Company

DPSC manufactures stainless-steel and aluminum RMI insulations marketed as Mirror® insulations. The Mirror® cassettes tested included both aluminum and stainless-steel foils encased in stainless-steel sheaths secured with latches and strikes with or without "Sure-Hold" bands. The NEI guidance recommends a destruction pressure of 4 psi for the DPSC Mirror® insulations. The apparent reason that Mirror® cassettes form debris at much lower pressures than does the TPI RMI is the construction of the sheaths, i.e., the cassette integrity depends on strength of the seams.

The debris fractions for the small fine debris from the AJIT tests are shown in Figure II-10. The small fine debris was correlated here as pieces <6 in., although the NEI guidance specified RMI small fines as <4 in.; therefore, a small measure of conservatism was added to the comparison. Figure II-10 shows six data points for Mirror®, with two of those tests generating very minor quantities of small fines. It should perhaps be noted that with the lower pressure test where the RMI cassette was exposed to a jet pressure of only 2 psi (AJIT Test 18-3), the cassette was still detached from the target mount, leaving two half cassettes on the chamber floor. The test with the largest quantity of small fine debris (AJIT Test 17-1) had only 10.6% of the foils turned into pieces <6 in., with the remaining foils becoming large-piece debris. The conservative extrapolation shown in Figure II-10 to complete the spherical ZOI debris fraction integration assumes complete destruction at a jet pressure of 130 psi. Note that in the single NRC-sponsored Mirror® debris generation test conducted at the KWU test facility, the test article was completely destroyed.

The results of the Mirror® debris-size distribution integration over the ZOI are provided in Table II-8. The potential debris volume of $661/D^3$ is quite low compared with an estimate using the baseline guidance [75% of $4/3\ \pi\ (21.6/D)^3$] of $31660/D^3$. Although this insulation is damaged at jet pressures as low as 4 psi, a relatively small amount of small debris is formed at pressures less than ~120 psi, and when the debris damage data are applied to the larger ZOI radius of 21.6/D, only a small fraction of the insulation in that sphere becomes small fine debris. For DPSC Mirror® RMI insulation, the NEI-baseline-guidance assumption that 75% of the insulation within a 21.6/D ZOI sphere would become debris <4 in. in size (i.e., $31,660/D^3$) is overly conservative. However, the quantities of large-piece debris, including nearly intact cassettes, could be very large because even 2 psi can detach the cassettes, which could become very important in containments where the transport velocities are high enough to move this heavier debris significantly.

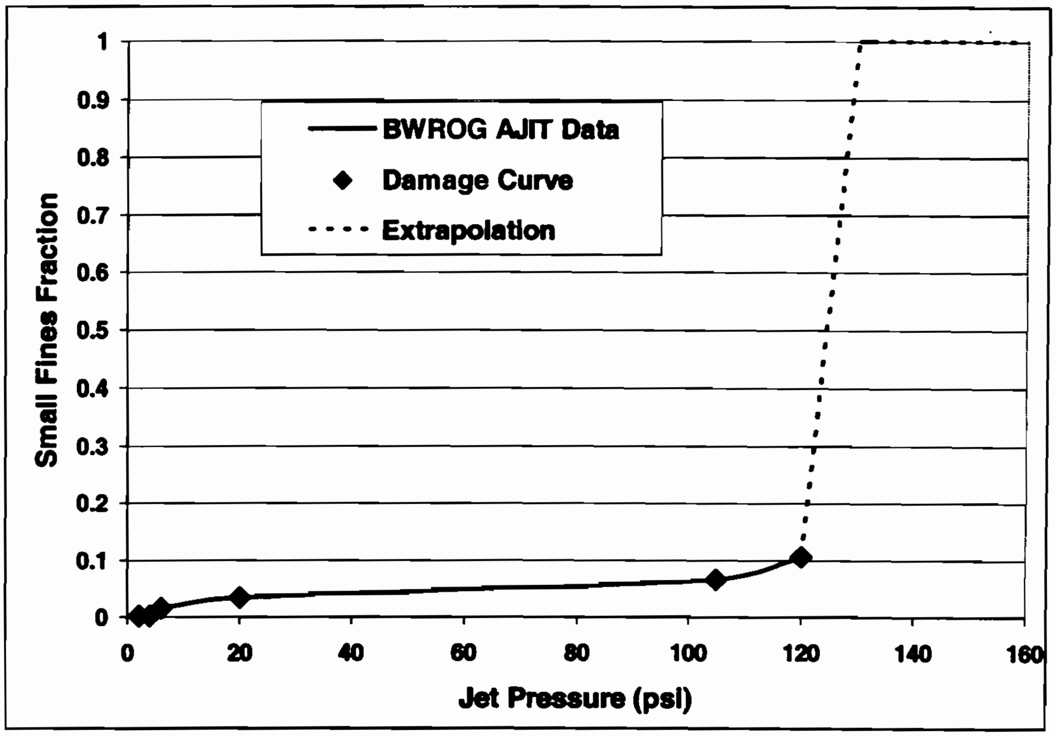

Figure VI-10. DPSC Mirror Damage Curve for Small Fine Debris.

Table VI-8. Results of Debris-Size Distribution Integration for DPSC Mirror® Insulation

Jet Pressure Isobar Volume Calculation	Radius of Sphere (r/D)	Fraction Small Fines	Potential Debris Volumes (V/D^3)
NEI-Recommended Damage Pressures			
PWR Two-Phase Jet (Confirmatory)	21.6	0.016	658

II.3.3 Particulate Insulation Debris

II.3.3.1 Min-K Debris

The NEI baseline guidance recommends the assumption that 100% of the Min-K insulation located inside a ZOI defined by the destruction pressure of 4 psi, corresponding to a radius of 21.6/D, becomes small fine debris. The basis for this recommendation apparently is the single Min-K BWROG AJIT debris generation test, Test 9-1. In this test, ~70% of the Min-K insulation became small fine debris. In fact, most of this debris was not recovered, apparently because it was too fine.* Based on the extensive damage to this Min-K blanket at 4 psi, it does not seem reasonable to assume that the threshold of damage is 4 psi.

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* It was noted that a cloud of debris was observed to exit the test chamber through the exhaust screen and that the venting of the chamber to clear the dust required more than 15 minutes.

At jet pressures substantially higher than 4 psi, it seems likely that the Min-K would be totally destroyed. At jet pressures <4 psi, the damage to Min-K would continue but would decrease in severity until the pressure became insufficient to cause damage. However, that pressure is not known. It is unlikely that the NEI baseline guidance is conservative with respect to the Min-K blanket tested. On the other hand, Min-K insulation protected by a metal jacket secured with steel bands would most likely be substantially less damaged than the unjacketed blanket tested.

II.3.3.2 Calcium Silicate Debris

The NEI baseline guidance recommends the assumption that 100% of the calcium silicate insulation located inside a ZOI defined by the destruction pressure of 24 psi (corresponding to a radius of 5.5/D) becomes small fine debris. The OPG debris generation tests [N-REP-34320-10000-R00] were cited to justify the 24-psi destruction pressure. The OPG tests involved impacting aluminum-jacketed calcium silicate insulation targets with a two-phase water/steam jet. The jacketing was secured with stainless-steel bands, and the jacketing seams were typically oriented at 45 degrees from the jet centerline—an orientation that appeared to maximize damage. The OPG data, illustrated in Figure II-11, only cover a limited range of damage pressures (~24 to 65 psi).

The damage curve shown in Figure II-12 was generated by summing all four debris categories in Figure II-11 to get the OPG debris fractions shown and then constructing a plausible curve through the data that was conservatively extrapolated at both ends. The results of the calcium silicate debris-size distribution integration over the ZOI are provided in Table II-9. The potential debris volumes are compared in Figure II-13, along with an estimate using the baseline guidance [100% of $4/3\ \pi\ (5.45/D)^3$]. A lower alternate damage pressure results in a larger equivalent spherical ZOI, but a lesser fraction of the debris is damaged into small fine debris. The use of the alternate damage pressures over the NEI-recommended damage pressures for PWR analyses would result in ~43% more estimated small fine debris. For calcium silicate insulation, the baseline is conservative with respect to both the NEI guidance damage pressure of 24 psi and the alternate pressure of 20 psi.

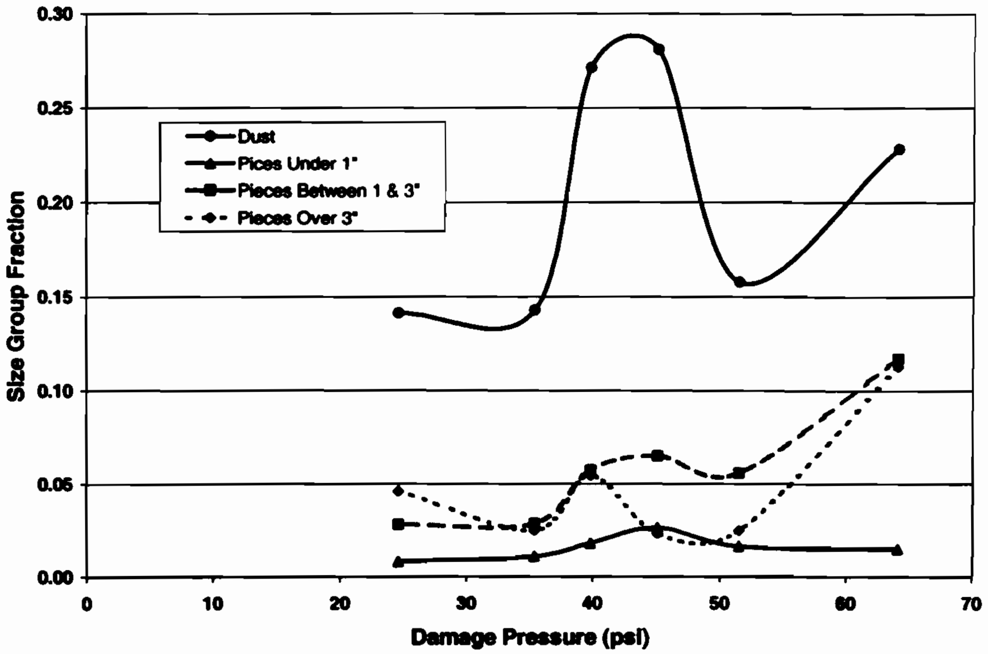

Figure VI-11. Debris-Size Distributions for OPG Calcium Silicate Tests.

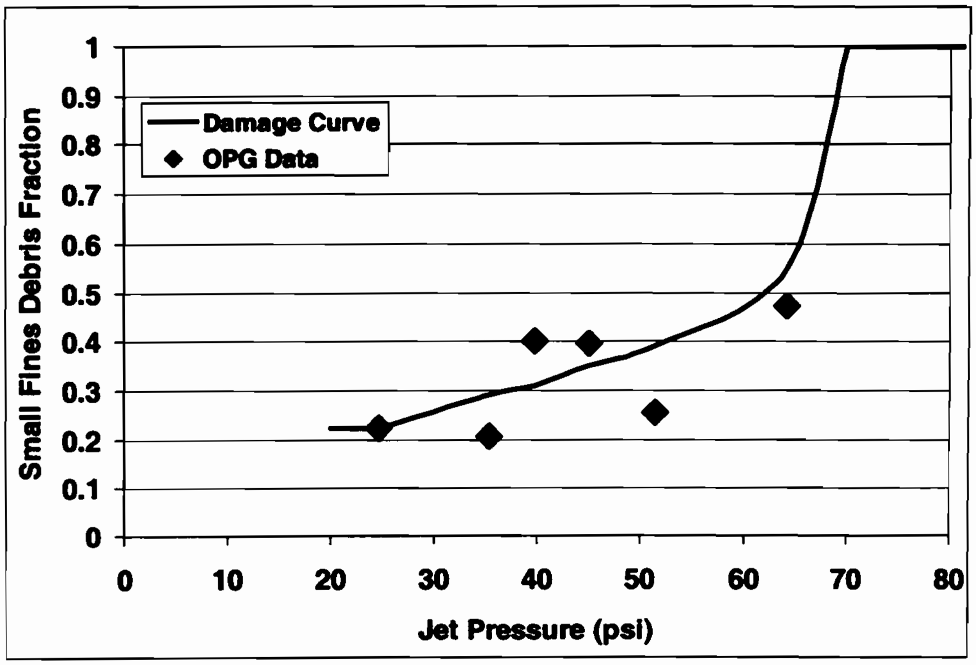

Figure VI-12. Calcium Silicate Damage Curve for Small Fine Debris.

Table VI-9, Results of Debris-Size Distribution Integration for Calcium Silicate Insulation

Delet

Jet Pressure Isobar Volume Calculation	Radius of Sphere (r/D)	Fraction Small Fines	Potential Debris Volumes (V/D^3)
NEI-Recommended Damage Pressures			
PWR Two-Phase Jet (Confirmatory)	5.4	0.42	273
Alternate Damage Pressures			
PWR Two-Phase Jet (Confirmatory)	6.4	0.34	372

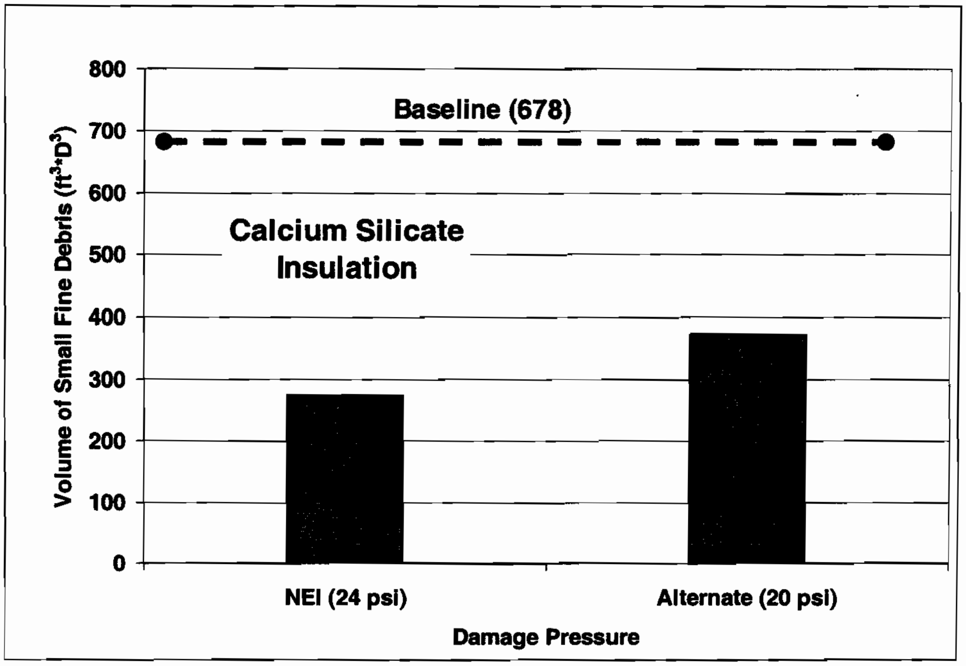

Figure VI-13. Potential Volumes of Small Fine Calcium Silicate Debris.

The BWROG AJIT tests also contain four tests of calcium silicate with aluminum jacketing secured by four ¾-in. stainless steel bands; however, these tests indicated that a jet of 150 psi was needed to cause significant damage. The reason(s) that a much higher pressure was needed to cause significant damage in the AJIT calcium tests than in the OPG tests has not been determined but is likely due to the differences in jacketing thickness, seam orientation, and strength of the bands. Here the destruction pressure depends more on the pressure needed to remove the jacket and expose the insulation than on the pressure required to erode the calcium silicate.

II.4 SUMMARY AND CONCLUSIONS

Confirmatory research was performed to ascertain whether the NEI recommendations for ZOI destruction pressures and debris fractions would reliably result in conservative estimates for the volumes of debris generated within the ZOI. Specifically, the NEI guidance recommends the assumption that 60% of the fibrous and 75% of the RMI insulation volume contained within the ZOI becomes small fine debris for ZOI radii defined by their recommended destruction pressures. The NEI guidance recommends adapting the debris-size distribution for NUKON™ to other types of fibrous insulation that have a destruction pressure higher than that of NUKON™.

Available debris generation data were used to define debris fractions versus jet pressure curves for the insulations examined. Difficulties encountered when correlating these data include aspects of protective jacketing and banding, as well as the variability in insulations. Before the insulation is subjected directly to jet flow forces, the flow must penetrate the protective coverings. Steel bands securing a metal jacket can require a rather high jet pressure to open the jacket before insulation debris is generated. The seam orientation affects the ease with which an edge of the jacket can be peeled back; it appeared that a seam orientation of ~45 degrees from the oncoming jet maximizes the potential for jacket opening. Another factor affecting the quality of debris generation data was the size of the jet nozzle relative to the insulation destruction pressure. If the target insulation had to be placed close to the nozzle to get the required destruction pressure, then the jet pressure became uneven along the length of the target; in fact, in some tests the target ends were likely located outside the influence of the jet. To test insulations with a higher destruction pressure, either larger nozzles or shorter targets are required. All of these considerations are factored into the evaluation of debris fractions.

ZOI debris fractions and insulation destruction pressures are interdependent; that is, the larger the ZOI, the smaller the fraction of the insulation within the ZOI that becomes small fine debris. Therefore, when the lower alternate pressure is used in the integration process, the resultant debris fraction will be less than that corresponding to the NEI-recommended destruction pressure.

The results and conclusions regarding relative conservatism of this confirmatory debris generation analyses are summarized in Table II-10 for the insulations examined. These results are relative to the NEI baseline guidance for the small-fine-debris-size category.

Table VI-10. Summary Comparison of Confirmatory and Baseline Potential Debris Volumes

Delet

Insulation	Confirmatory Research Result	Relative Conservatism of Baseline Guidance
Fibrous Insulations		
NUKON™	Baseline guidance results compare well with confirmatory results.	Baseline guidance for NUKON™ provides realistic results that are only slightly conservative.
Temp-Mat	Baseline results are approximately twice the confirmatory results (based on limited data).	Baseline guidance is conservative for Temp-Mat insulation.

Insulation	Confirmatory Research Result	Relative Conservatism of Baseline Guidance
K-wool	Baseline results are only about half that of the confirmatory results (based on limited data).	Baseline guidance is likely conservative for K-wool, despite the nonconservative comparison with confirmatory analysis. The poor nonconservative comparison is due to the extreme extrapolation of data required by the lack of data for pressures greater than the NEI destruction pressure. Still, conservatism cannot be proven with existing data.
RMI Insulations		
DARMET®	No confirmatory analysis for this insulation. Rather, a review of the debris generation data illustrated substantially less small fine debris than would be estimated using the baseline guidance methodology.	Baseline guidance is conservative for DARMET® insulation.
TPI	Baseline results account for only one-third of the confirmatory debris estimate, which includes the small quantities of debris generated at lower pressures but that are neglected when the baseline destruction pressure is used.	Baseline guidance is not conservative, but the quantities of this debris are relatively low; therefore, this nonconservative estimate is not a major issue.
DPSC Mirror®	Baseline results were almost 50 times that of the confirmatory result. The baseline minimum destruction pressure of 4 psi results is a very large ZOI volume, but the damage to the insulation is relatively minor at the lower pressures, thus the large differences in results.	Baseline guidance is conservative for Mirror® insulation.
Particulate Insulations		
Min-K	No confirmatory analysis for this insulation. Rather, the data from the single Min-K debris generation test were examined, i.e., approximately 2/3 of the insulation was turned into fine dust debris at a jet pressure of only 4 psi.	Baseline guidance is not conservative because the one test indicated that substantial damage would occur to Min-K insulation at significantly lower pressures than the destruction pressure of 4 psi and that the damage at 4 psi was extreme.
Calcium Silicate	Baseline results are approximately twice the confirmatory results, even when the lower jet pressure of 20 psi (recommended in NUREG/CR-6808) is considered instead of the baseline destruction pressure of 24 psi.	Baseline guidance appears to be conservative for calcium silicate insulation, but the debris generation data are not sufficient to determine the threshold jet pressure for generating small fine debris, i.e., the threshold destruction pressure could actually be less than the 20 psi alternate pressure used in the confirmatory analysis.

The following additional comments should be noted:

- The use of the alternate destruction pressure provides some quantification of the uncertainty associated with the selection of the destruction pressures. These uncertainties include the neglect of the tails of the debris damage curves and the uncertainty associated with the potential two-phase effect on debris generation relative to the available air-jet-generated data.

- A comparison of the NUKON™ results with the BWROG URG steam jet model illustrates that the neglect of the tails of the debris damage curve has a larger impact for PWRs than for BWRs (see Figure II-3).

- The NEI guidance recommendation that adapts the debris-size distribution for NUKON™ to other types of fibrous insulation that have a destruction pressure higher than that of NUKON™ has been partially supported (see Figure II-8), although it cannot be conclusively ensured.

- The ZOI for large debris generation in some cases does not correlate with the ZOI for small-fine-debris generation. A case in point is the analysis for TPI RMI, where most of the small fine debris would be generated inside jet pressures of 190 psi but large debris was generated (in the form of detached cassettes) at pressures as low as 4 psi. Therefore, rather larger quantities of large debris could be formed than were predicted using the baseline guidance ZOI sizes.

- It should be emphasized that the typical debris generation analyses were performed for insulations where the debris generation data were very limited. The data for the LDFG insulations (see Figure II-2) illustrate the potential variability in such data. Therefore, the limited debris generation data cause substantial uncertainty with debris generation estimations.

II.5 REFERENCES

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[NEDO-32686, Rev. 0, 1996] "Utility Resolution Guidance for ECCS Suction Strainer Blockage," BWROG, NEDO-32686, Rev. 0, November 1996. **not used in this section**

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APPENDIX III: VOLUNTEER-PLANT CONTAINMENT POOL COMPUTATIONAL-FLUID-DYNAMICS ANALYSIS

III.1 INTRODUCTION

A three dimensional computational fluid dynamics (CFD) model was developed to analyze the flow patterns developed in the Nuclear Regulatory Commission's (NRC's) volunteer-plant reactor containment during loss-of-coolant accidents (LOCAs). The purpose of the CFD modeling was to assess the water velocities and flow patterns developed during sump pump operation to support estimates of subsequent LOCA-generated sump pool debris transport. Water sources to the sump pool included effluents from the LOCA break and containment spray drainage. The locations and flow rates of each of these water sources and the recirculation pumping rates determined the characteristics of the sump pool that subsequently determined whether, and what fraction of, the debris deposited into the pool could transport to the recirculation sump screens. Threshold transport velocities were determined by experiments conducted at the University of New Mexico (UNM) for debris from pressurized-water-reactor (PWR) insulating materials [NUREG/CR-6772]; therefore, these threshold velocities were used to set the velocity contours of the CFD flow diagrams to facilitate the determination of whether debris would likely transport. The CFD simulations are discussed in Section III.2.

A logic chart debris-transport model was developed to supplement the CFD analyses so that information from the CFD simulations could be used with the blowdown/washdown transport analyses documented in Appendix VI to determine estimates of debris transport to the recirculation sump screens. The pool velocity and turbulence characteristics determine areas of the pool where debris entrapment may occur. The flow streamlines can be used to determine whether debris entering the pool at a discrete location would likely pass through one of the potential entrapment locations. The debris transport process was decomposed using a logic chart approach to facilitate the individual transport steps—steps that could be determined analytically, experimentally, or simply judged. The subsequent quantification of the chart then provided an estimate of the overall sump pool debris transport. The debris transport estimates are discussed in Section III.3.

III.2 ANALYSIS OF THE CFD SIMULATION

III.2.1 Modeling Methodology, Assumptions, and Conditions Simulated

The commercial CFD program Fluent™ was used to compute the volunteer-plant containment pool flows for large and small LOCA breaks. The containment geometry was available in Autocad™ format and was imported into the Fluent™ preprocessor and grid generator. As shown in Figure III.2-1, all of the structures, stairwells, and sumps were included in the model geometry, but the containment pool was modeled only to a depth of 6 ft. This is the maximum anticipated depth of water during steady-state operation of the spray system and sump pump operating in the recirculation mode.

The splash locations are shown in Figure III.2-2 and can be seen as the extruded volumes above the containment pool in Figure III.2-1. The splash locations and flow rates shown in Figure III.2-2 are explained in detail in Appendix VI. A few modifications to the splash locations and flow rates were made in the CFD model.

1. One of the four "yellow" floor drains from Level 832, with a total flow rate of 397 gpm in Figure III.2-2, is located on top of a wall. Thus, the adjacent yellow splash located in the corridor had double the individual flow rate. (Note: for all the Level 832 floor drains, the total mass flow was evenly distributed to all locations, with the exceptions noted here.)
2. The uniformly distributed "liner film flow" of 700 gpm and
3. The "Level 808 sprays" of 1080 gpm were neglected entirely.

Thirteen LOCA break conditions were simulated: eight large LOCA conditions (four break locations each considered with and without the spray flows) and five small LOCA conditions (four break locations without spray flows and one location with spray flows). Both large and small LOCA breaks were considered because each can cause the sump screens to become clogged in a different way. The large LOCA break and spray flows will result in a large pool depth and all of the screen surface area to be come wetted. The large LOCA break will likely generate more debris that can migrate to the sump screens causing an unacceptable head loss due to the amount of debris collected. The small LOCA break may not cause the spray systems to be activated, and could result in a water depth wetting only the lower portion of the sump screens. This has the potential of forming a thin bed debris mat over a small portion of the screen area resulting in an unacceptable head loss. If the spray flow systems are not activated, depending on break location, a larger portion of the pool flows do not have velocities in excess of the debris threshold velocities and do not participate in the recirculation flow. Therefore the debris generated in those regions does not migrate to the sump screens and provide information on areas to divert debris into during the break and pool fill up.

The four break locations considered correspond to a break occurring in one of the four quadrants [steam generator (SG) compartments] in Figure III.2-2. The total break flow was assumed to be 7400 and 1611 gpm for the large and small LOCA break flows, respectively. It was assumed that the upper two SG compartments were physically separate from the lower two compartments; thus, if the break were postulated to occur in the upper left quadrant, 75% of the break flow would be partitioned to the upper left and 25% to the upper right quadrants; none of the break flow was considered in the lower two quadrants. The 75%/25% partitioning was determined arbitrarily, but it seemed to be a realistic assumption. Additionally, a transient pool fill-up simulation was initiated for a large LOCA break in the upper-left quadrant. Only the break flows were simulated in the upper half of the SG compartments with the break flow partitioned as described above. It should be noted that the above apportionment of the flow represents an estimate of the volunteer plant break due to the steam generator compartment configuration. The steam generators are raised above the pool floor level and do not participate in the recirculation flow, thus the break flow enters the pool by flowing down the steam generator stairwells and thus the water sheets across the steam generator compartment and does not pool to any significant depth. Thus the 75/25% apportionment was assumed, but a thorough analysis of how the break flow would enter the pool would be required. These analyses would be required by each plant, using their expert knowledge of the containment configuration, thus the above apportionment is illustrative of the types of flows that would enter the pool.

Three boundary condition types were used in the simulation. All hard surfaces (walls, floors, etc.) were specified to be a no-slip wall condition. The spray system splash and LOCA break flows were specified as a mass flow inlet condition, and the sumps were set to a pressure outflow boundary condition. Because the break flow sheeting described previously was not included, the break and spray flows present in the SG compartment were applied as a mass inflow boundary on a vertical surface at the exit of the SG entrance steps of each quadrant (i.e., a mass flow boundary condition located at the "door" of the SG entrance steps, for instance). The spray/splash mass flow boundary conditions were placed on the "top" of each extruded spray location, as shown in Figure III.2-1. This extruded volume was found to be easier to handle in Fluent™ than trying to set the boundary condition on the "top" of the pool surface.

The combination of mass inflow and pressure outflow satisfies the mass continuity condition without unnecessary complications due to numeric and other boundary condition errors. In theory, a mass outflow condition at the bottom of the sump could be specified, but there are numerical instabilities when that condition is prescribed. By using a pressure outflow condition at the sumps, the pressure is allowed to "float" to satisfy the incompressible continuity equation. In other words, the pressure at the bottom of the sump is adjusted by the code to balance the mass flow entering and exiting the pool. In this way the introduction of artificial pressure waves in the solution that can be created by specifying mass inflow and outflow conditions were avoided.

A second-order-accurate numerical method was used to solve the incompressible Navier-Stokes equations, in conjunction with a renormalized group-theory turbulent-kinetic-energy and dissipation (RNG κ-ε) turbulence closure. This closure was chosen because of its ability to treat swirling flows, but in practice, little difference was found between the RNG κ-ε and the more traditional κ-ε closure for these simulations. The pressure equation was solved using a PISO method, as described in the Fluent™ documentation. For the steady state pool flow analyses, the pool volume was assumed to be completely full of liquid water and initialized to zero velocity. The inflow boundary conditions were flowing from the start, and the solution was allowed to proceed until a steady-state condition was achieved. The normalized residuals of the continuity, momentum, and κ and ε equations were monitored until convergence was achieved, typically about 400 iterations. For the steady state pool flow analysis, an additional convergence criterion was to integrate the mass flow rate at the two sump pressure outflow boundaries and compare it with the mass inflow. A mass balance had to be achieved, in addition to a drop in the normalized residuals, for the simulation to be deemed converged.

III.2.2 Results and Discussion

This section contains the results of the CFD simulations. These simulations illustrate what can be achieved with a CFD analysis of the containment pool flows. For application to a particular plant containment, a more rigorous set of simulations should be performed, including grid convergence tests (e.g., does doubling the number of grid points change the results significantly).

One figure of merit was to determine the fraction of the pool flow volume that produced velocities in excess of the debris migration threshold velocities. Based on the experimental measurements reported in NUREG/CR-6772, the RMI and fiber flock transport threshold velocities were determined to be 0.085 and 0.037 m/s, respectively. Note that only one debris transport threshold velocity for fiber and one for small RMI were used for the following analyses.

III.2.2.1 Transient Containment Pool Fill-Up

For this simulation, a volume-of-fluid (VOF) method was used. The containment pool was initially filled with air, and water was allowed to enter the pool from the SG entrance stairs. Only the break flows for a large LOCA break, located in the upper-left quadrant, were included. As noted in Section III.2.1, the break flow is partitioned such that 75% of the water leaves the upper-left SG compartment stairwell and 25% leaves the upper-right SG compartment stairwell. This condition corresponds to the time immediately after a break occurs and before the spray system is activated. All walls were treated as no-slip surfaces, and because the fill-up phase is being simulated, the sumps were also treated with no-slip surfaces instead of pressure outflow boundary conditions. The top boundary of the simulated pool was prescribed as a pressure outflow boundary condition instead of as a no-slip wall. This treatment allows the air to leave the domain as the water displaces it. The containment pressurization that occurs during a LOCA was not modeled because it has minimal effect on pool transport.

Figures III.2-4 through III.2-12 show the volume fraction of water, at a height of 0.01 m above the containment floor, as the containment pool fills at 0.34, 0.94, 11.4, 21.4, 31.4, 41.4, 51.4, 71.4, and 111.4 seconds after the water leaves the SG compartment stairwells. The color scheme shown corresponds to a red color for 100% water in the computational cell and blue for 100% air in the cell. Other colors indicate that the computational cell has both air and water partially filling the cell. From Figures III.2-4 to III.2-12, the areas that are first swept by the water can be seen, as well as how the containment pool fills. This simulation shows the areas that fill first and thus provides information needed to design systems to divert debris to areas of the pool that do not participate in recirculation flow. In general, the water leaves the SG compartment, flows out the doorway, and hits the circular outer wall. Then the water flows circumferentially around the containment until the two water streams meet near the sumps. Then the water starts to enter the areas between the upper and lower SG compartments. For this plant configuration, these two areas between the upper and lower SG compartments are the only "quiet" zones (i.e. flow velocities much lower than the debris threshold) in the pool when all break locations are considered in the subsequent steady-state pool flow analysis.

Figures III.2-13 through III.2-21 show the fluid velocity during the fill-up at the same set of time increments previously discussed for volume fractions. Note that when the water volume fraction and fluid velocity plots are compared, there is motion ahead of the water. This motion is the air moving in response to the approaching front of water. During fill-up, the water velocity near the front is in the range of 2–3 m/s, well in excess of the debris transport threshold velocities of 0.037 and 0.085 m/s for fiber and RMI, respectively.

III.2.2.2 Steady-State-Flow Analysis

To study the containment pool's steady-state-flow dynamics, the simulated volume was considered to be completely full of water. In the case of a small LOCA break, the spray flows were not included; however, for the large LOCA break, spray flows were included in the simulations. With the simulated pool full of water, the break and spray flows were introduced as mass inflow boundary conditions and the sumps were set to a pressure outflow boundary condition. These simulations produced a stimulated steady-state-flow condition for further debris transport analysis, which will be discussed in Section III.3.

Figures III.2-22 through III.2-29 show the steady-state-flow pattern developed for a small LOCA break condition, without spray flows, and Figures III.2-30 to III.2-37 show large LOCA break conditions, including spray flows. These figures show contours of water velocity at a height of 0.01 m above the containment floor and show a velocity range from 0 m/s up to the threshold velocity for fiber or RMI, 0.037 and 0.085 m/s, respectively. From these plots, the area enclosed by the threshold velocity contour can be computed, and by dividing by the entire available flow area in the containment, a percentage of area in excess of the threshold velocity may be computed. These percentages, or fractional areas in excess of the threshold velocity, are summarized in Table III.2-1 for both large and small LOCA break conditions.

Figures III.2-38 through III.2-47 show streamlines for origins near the splash locations for a large LOCA break at two different locations: an upper-left break and a lower-right break. A rake of particles was released from ($-15 < X < -5$, $Y=10$), and also from ($0 < X < 5$, $Y=15$) and allowed to follow the flow. From these streamlines, debris trajectories can be determined and their fate postulated. Figures III.2-38 and III.2-39 show the streamlines superimposed on the background velocity map that were color coded using the fiber (0.037 m/s) and RMI (0.085 m/s) threshold velocity, respectively. An oblique view showing the three-dimensionality of the streamlines is shown in Figures III.2-41 and III.2-42, color coded according to the flow speed, using the fiber and RMI threshold velocity, respectively. Thus, it could be deduced that if the velocity (speed) along a particular streamline became smaller than the debris type threshold velocity, it would not be so likely to migrate to the sump screen. By using rakes and streamline analysis at potential debris entry locations, a method for determining whether the debris will transport to the sump screens could be developed.

A similar set of plots are shown in Figures III.2-42 through III.2.45 for the large LOCA break located in the lower-right quadrant. Notice that the streamline patterns are quite different for the lower-right break location when compared to the upper-left break location.

Shown in Figure III.2-46 is a vortex induced by the splash located in the upper-right quadrant in Figure III.2-42. Here the streamlines are color coded by velocity using the fiber velocity threshold. Because the water enters the pool from above and penetrates to the containment floor, a vortex with significant vertical motion is created. Figure III.2-47 shows the streamlines color coded by turbulent kinetic energy (TKE). This type of information would be useful in determining debris degradation mechanisms, particularly for fibrous debris. In Figures III.2-46 to III.2-47, the streamlines show the type of rotation that debris can encounter near the entry of a splash into the pool. The water flow produces vortices around the splash entry and could potentially shred debris into finer particles/pieces than those generated by the break itself. No attempt was made in this document to quantify the debris shredding mechanisms; rather, this document simply illustrates what can be gleaned from a CFD analysis of the pool dynamics.

Table III.2-1 Percentage of Containment Pool Flow Area in Excess of the Debris Transport Threshold Velocity. Total Pool Area = 767.7 m^2

Break Location	Break Size	RMI (%)	Fiber Flocks (%)
Upper Right	Large	35	60
Upper Left	Large	30	54
Lower Left	Large	22	43

Break Location	Break Size	RMI (%)	Fiber Flocks (%)
Lower Right	Large	22	41
Upper Right	Small	5	31
Upper Left	Small	2	25
Lower Left	Small	5	14
Lower Right	Small	5	19

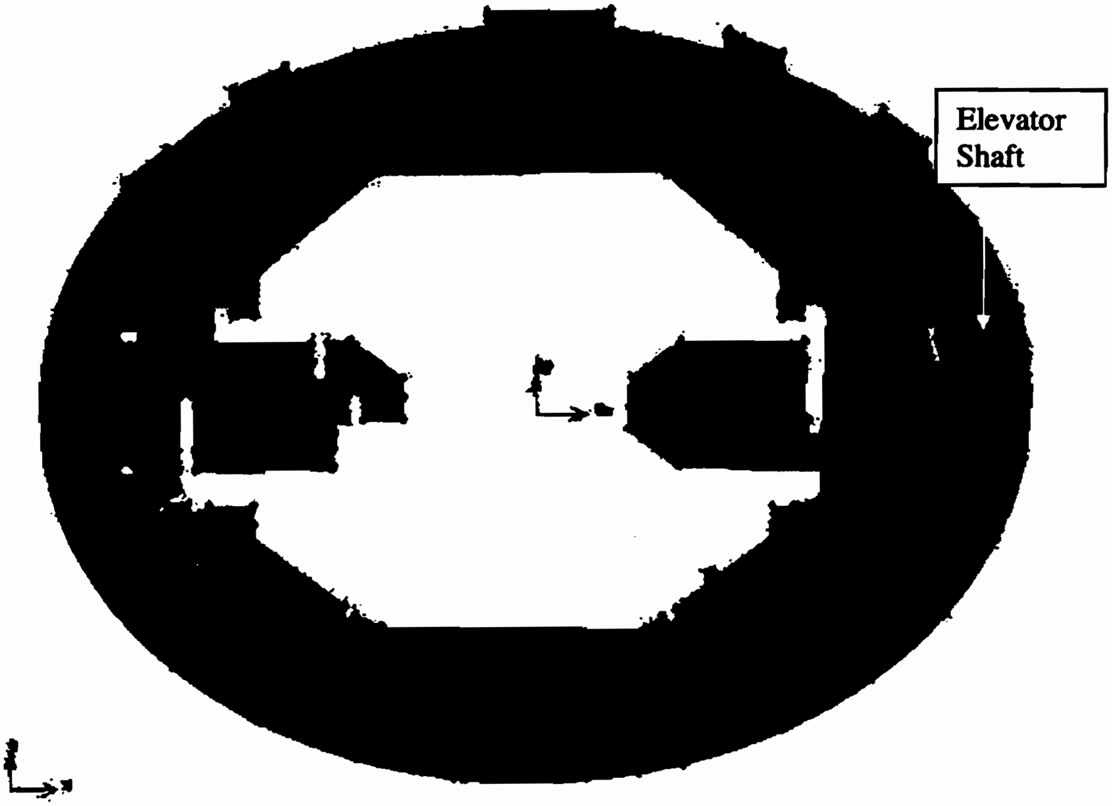


Figure III.2-1. Volunteer plant geometry and flow region modeled. (Note: Splash Locations Are Shown Extruded above the Nominal Pool Depth.)

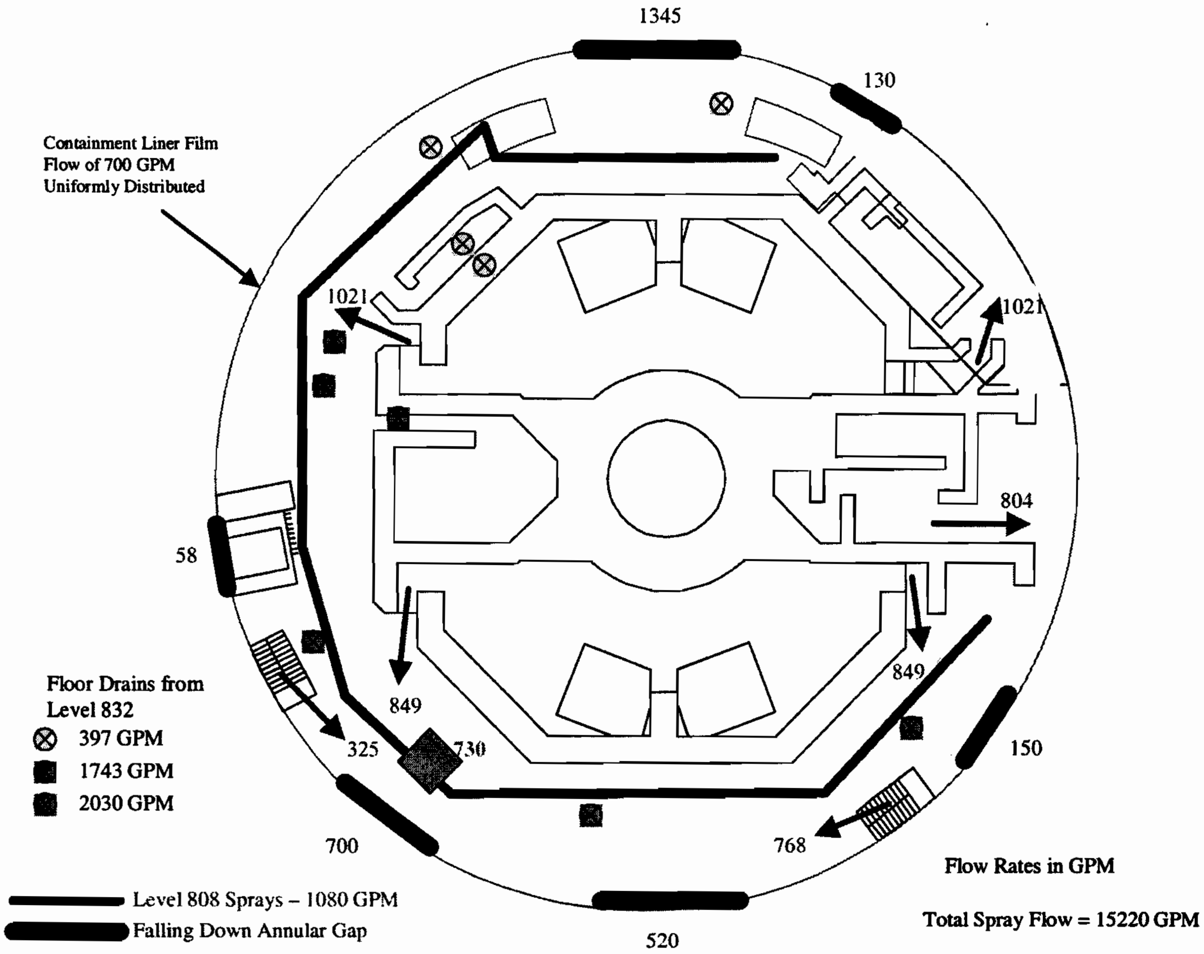


Figure III.2-2. Spray Flow Rates (gpm) and Locations for the Volunteer-Plant Pool Flow Calculations.

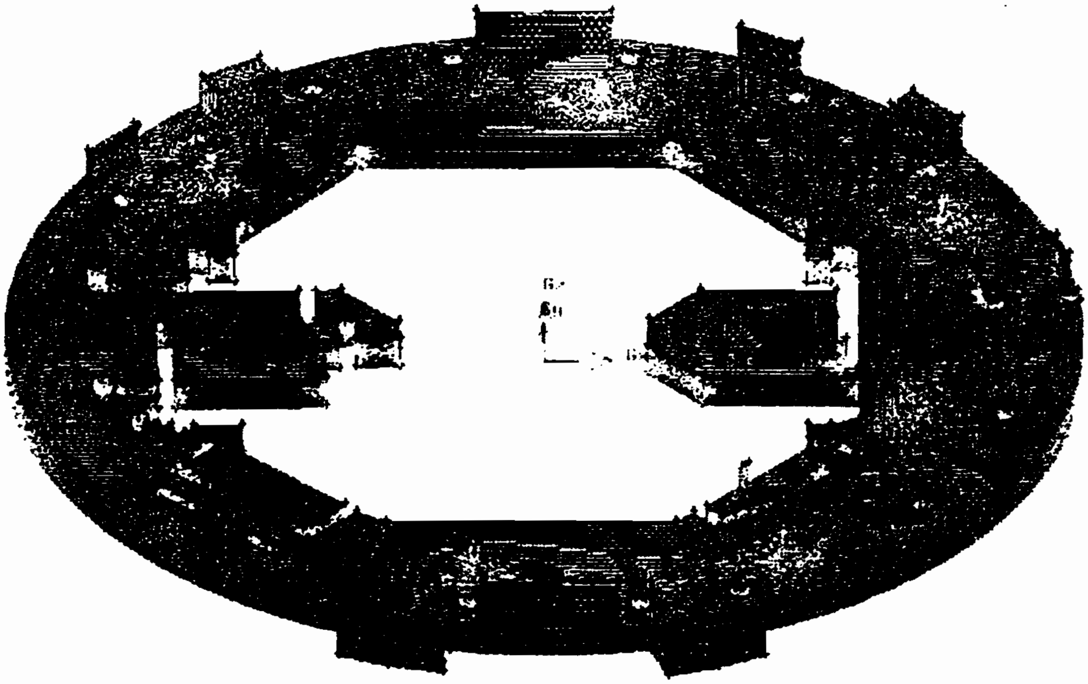


Figure III.2-3. Unstructured Mesh Created for Containment Pool Flow Calculations.

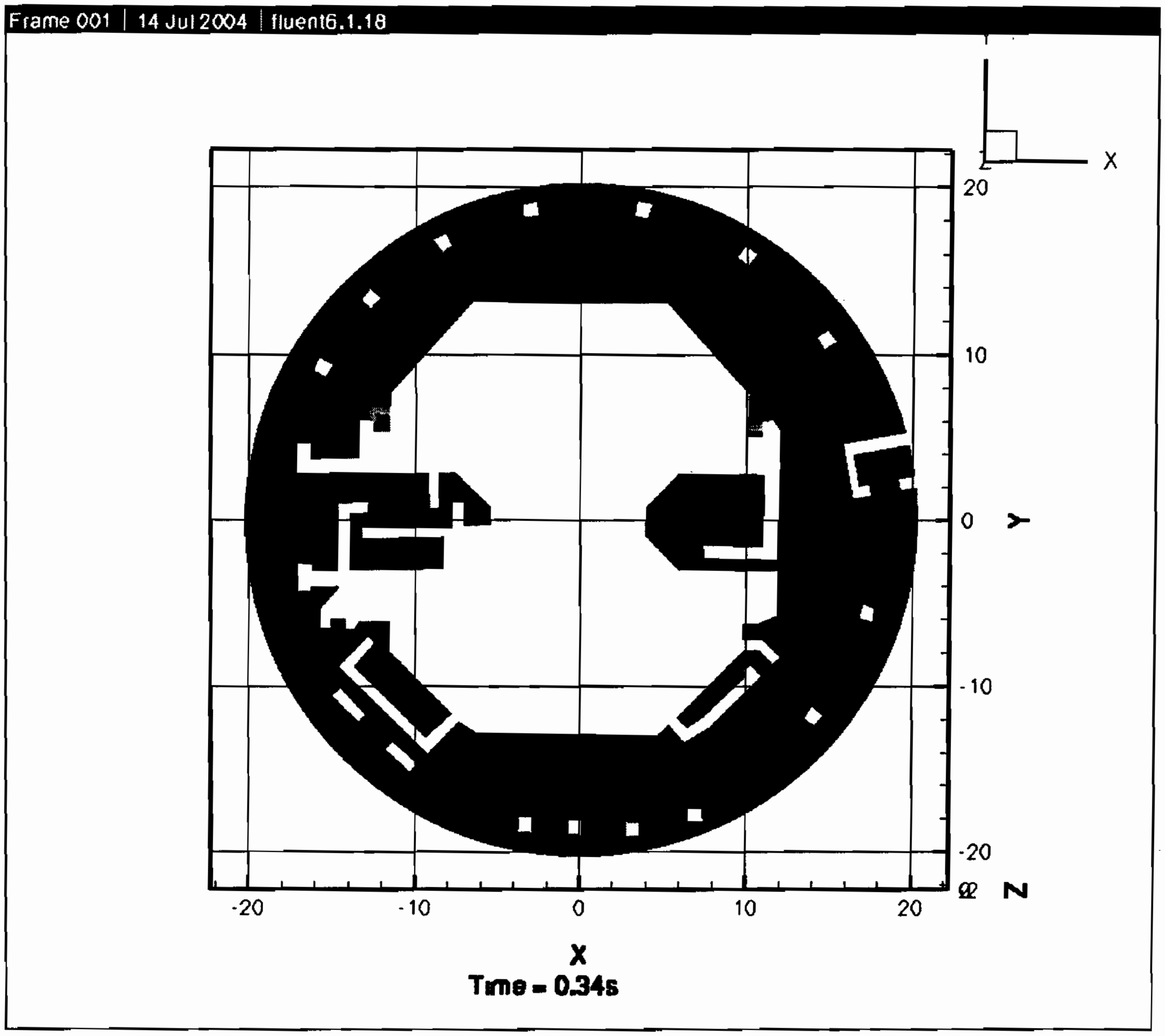


Figure III.2-4. Transient volume of fluid during the simulation of containment pool fill up. Computational cell volume fraction of water is shown at a height of 0.01m above the containment floor. Red is 100% water (0% air), blue 0% water (100% air). Time of the snapshot in seconds after the break flow is initiated is shown in the bottom of the figure.

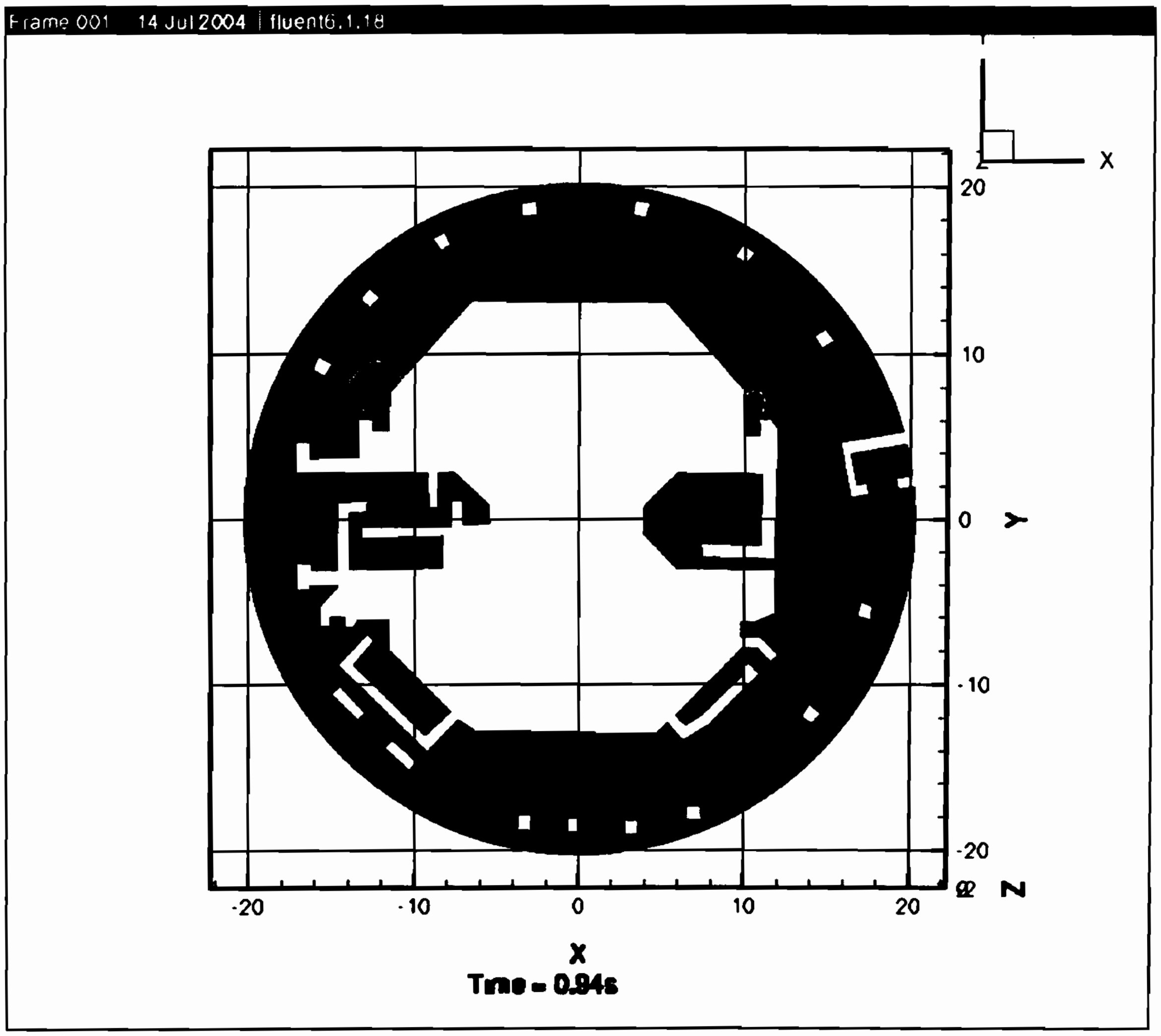


Figure III.2-5. Same as Figure III.2-4 for t = 0.94 Seconds.

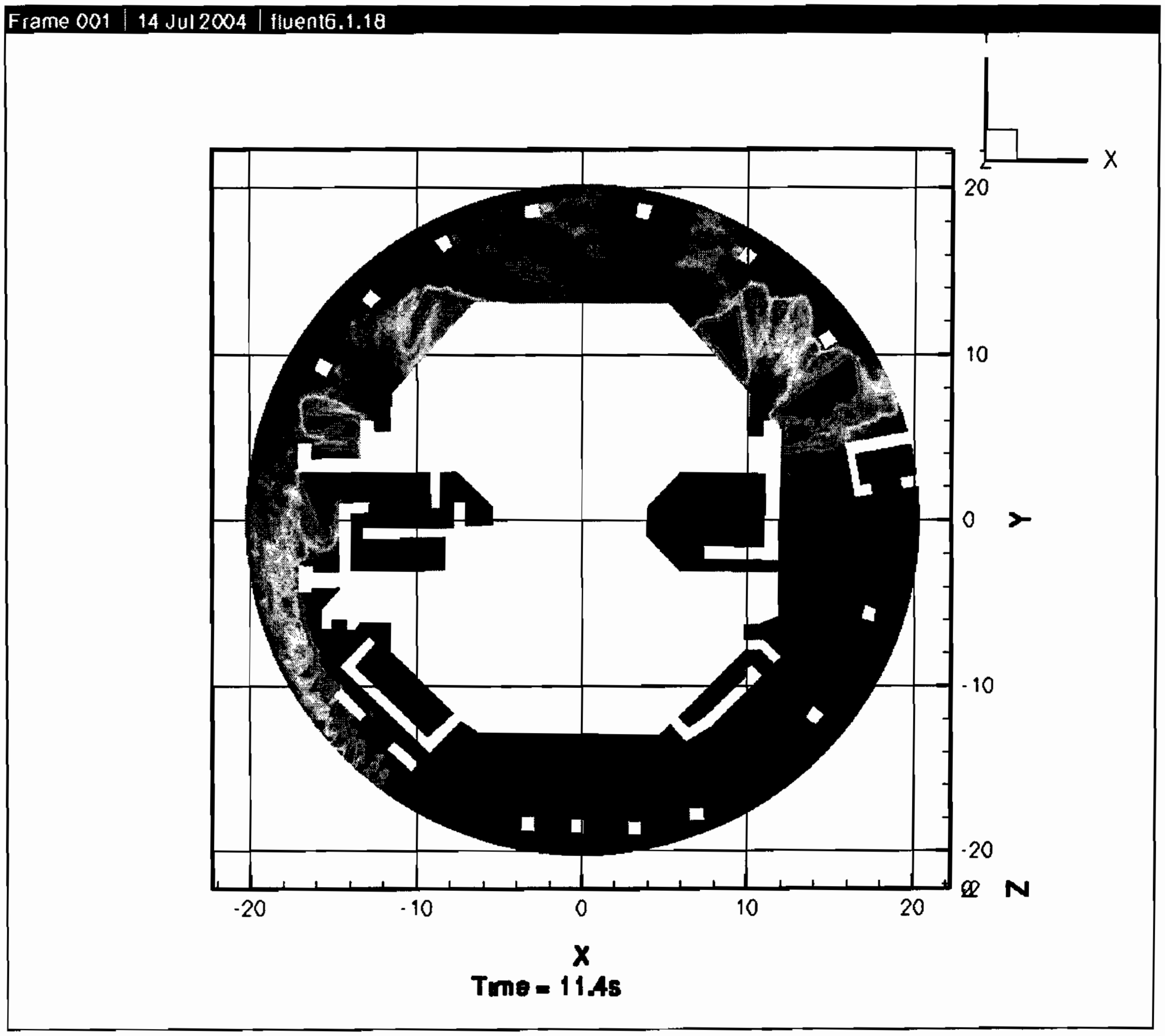


Figure III.2-6. Same as Figure III.2-4 for t = 11.4 Seconds.

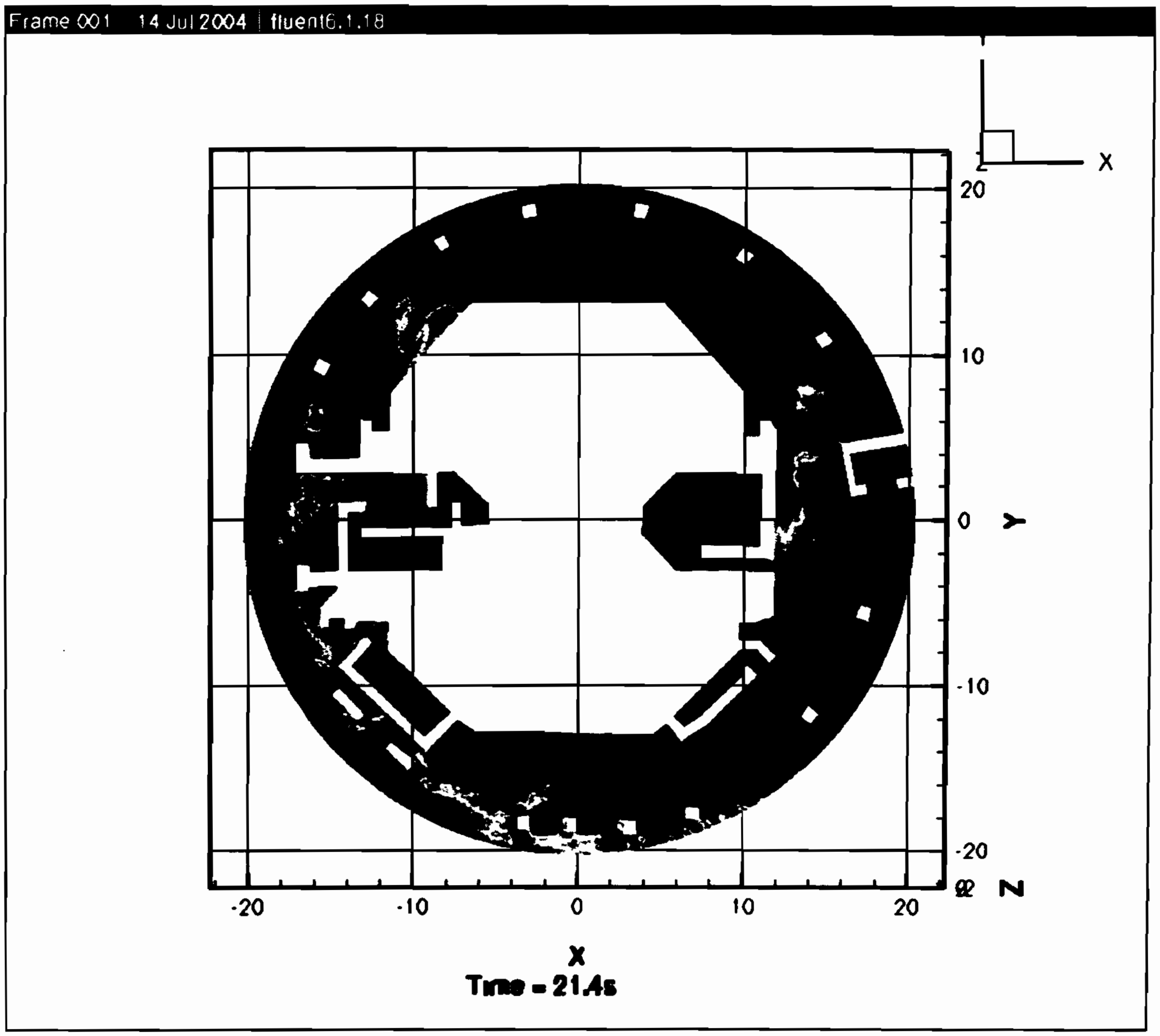


Figure III.2-7. Same as Figure III.2-4 for t = 21.4 Seconds.

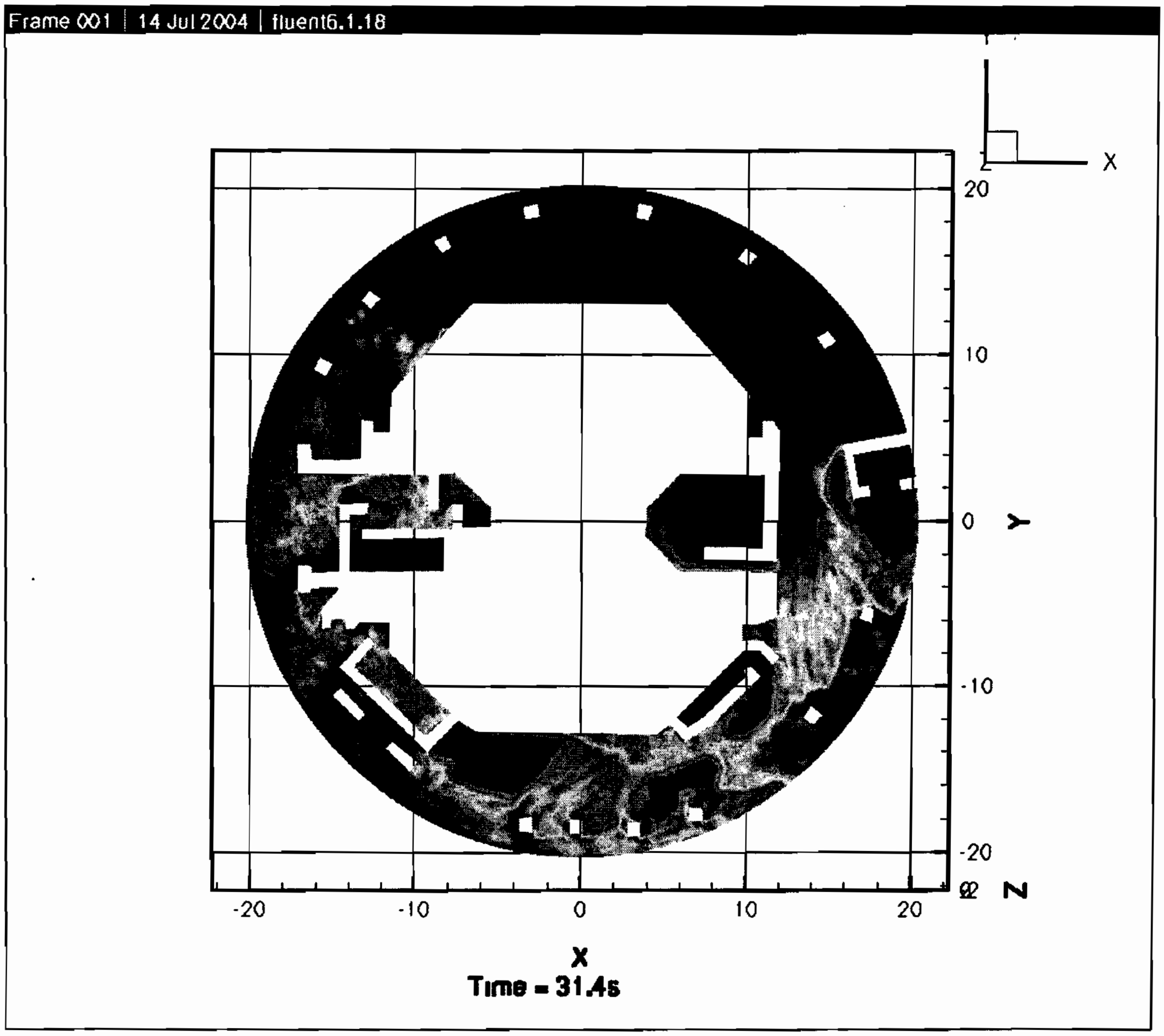


Figure III.2-8. Same as Figure III.2-4 for t = 31.4 Seconds.

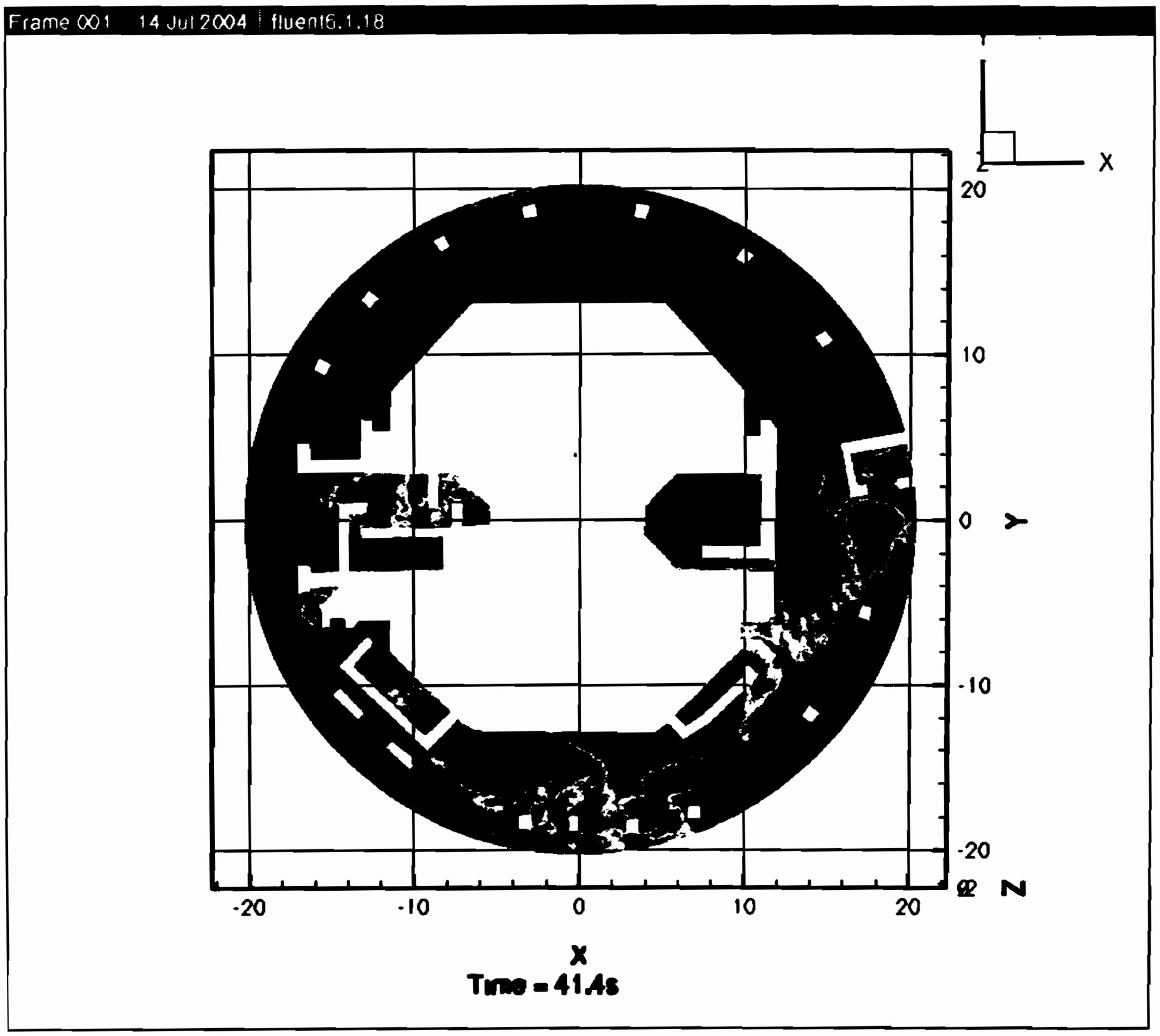


Figure III.2-9. Same as Figure III.2-4 for t = 41.4 Seconds.

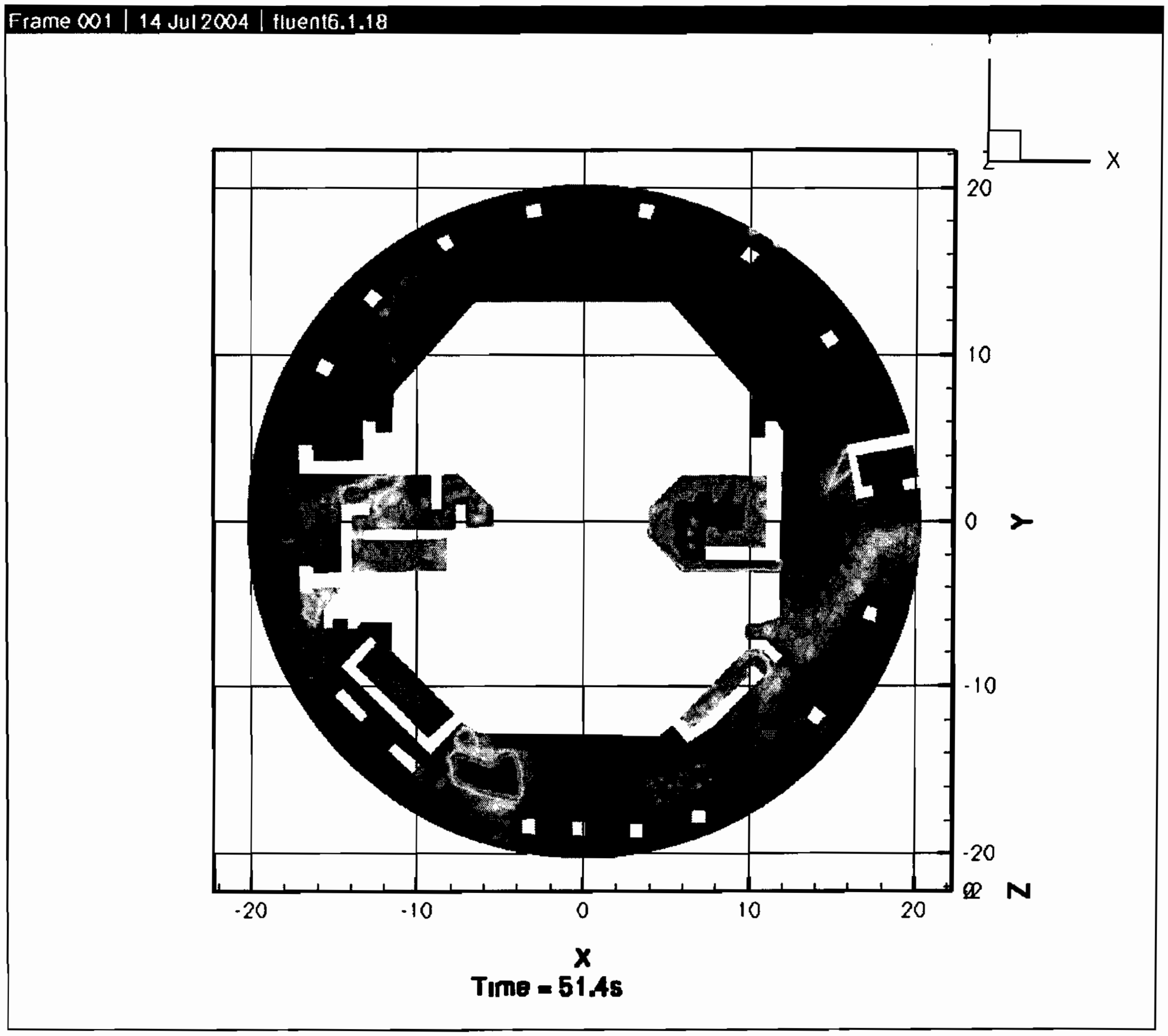


Figure III.2-10. Same as Figure III.2-4 for t = 51.4 Seconds.

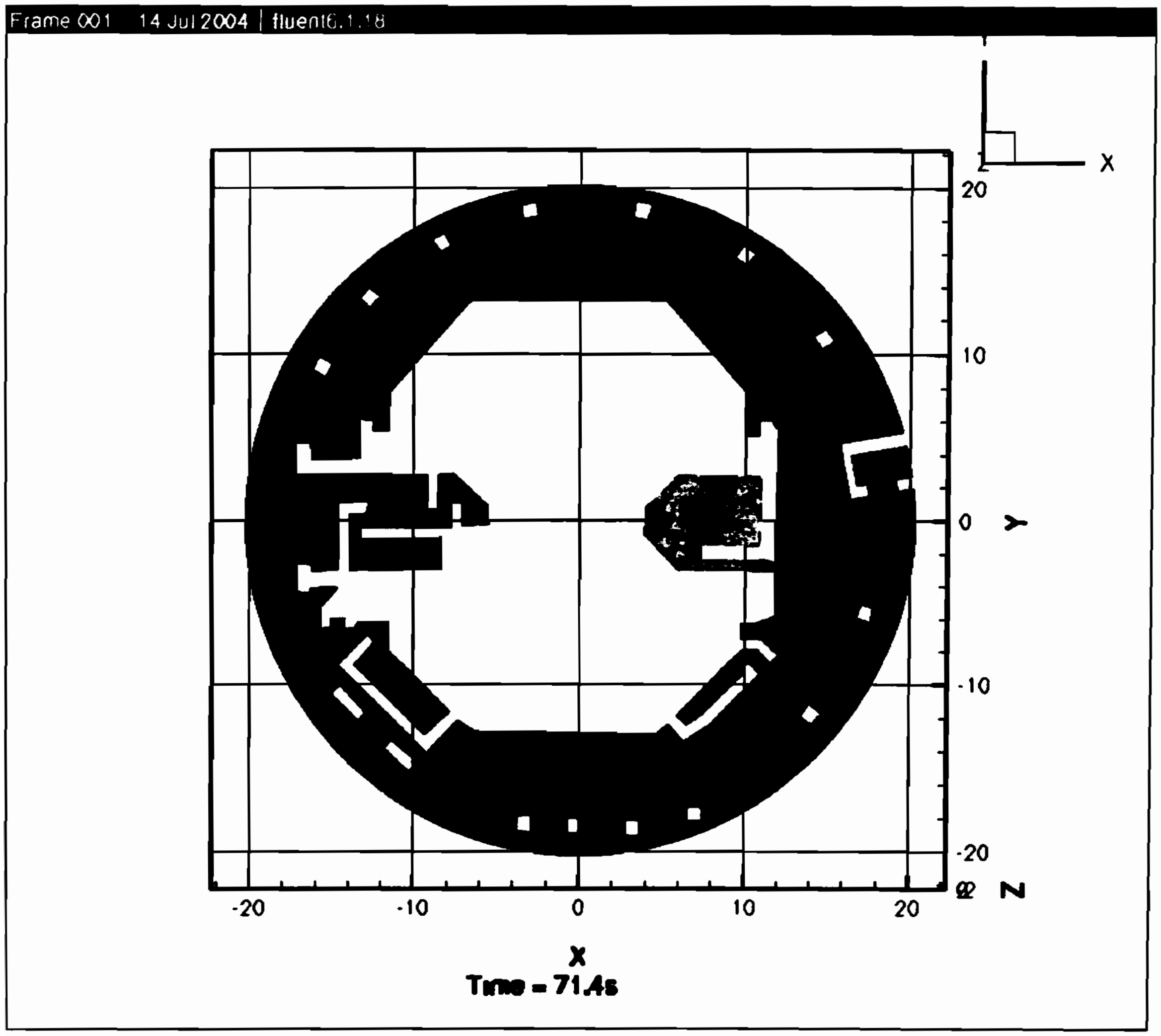


Figure III.2-11. Same as Figure III.2-4 for t = 71.4 Seconds.

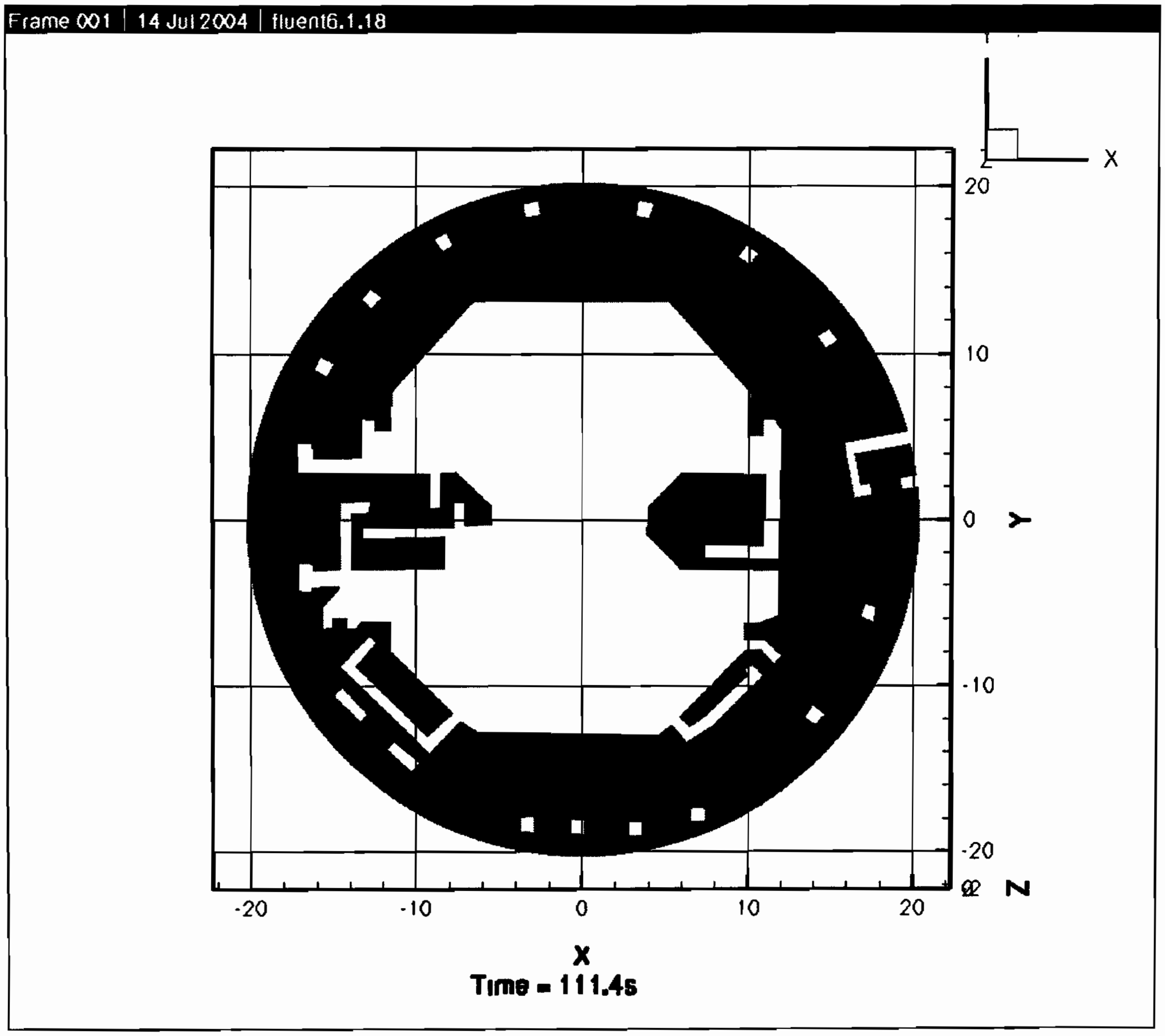


Figure III.2-12. Same as Figure III.2-4 for t = 111.4 Seconds. Note That the Solid Red Color Indicates That the Cells Adjacent to the Floor Are Full of Water, Not That the Entire Pool Is Full of Water.

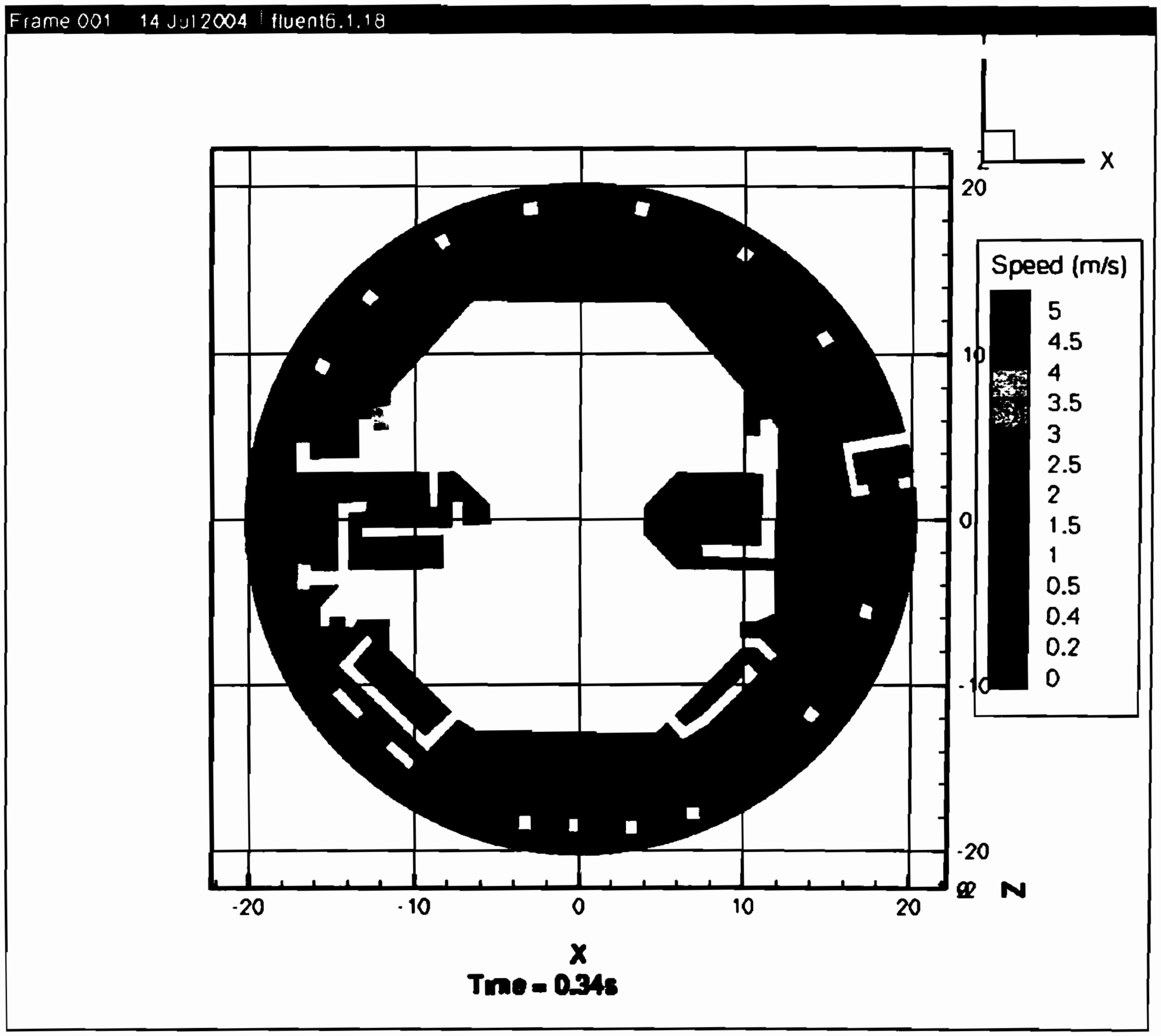


Figure III.2-13. Transient VOF Simulation of Containment Pool Fill-Up. Contours of Fluid Velocity Are Shown. Time Snapshot Shown in the Figure Is Seconds after the Break Flow Is Initiated. Note That the Fluid Velocity May Be Water or Air; Figures Showing the Volume Fraction of Water (Figures III.2-4 to III.2-12) Should Be Used to Determine the Actual Water Velocity.

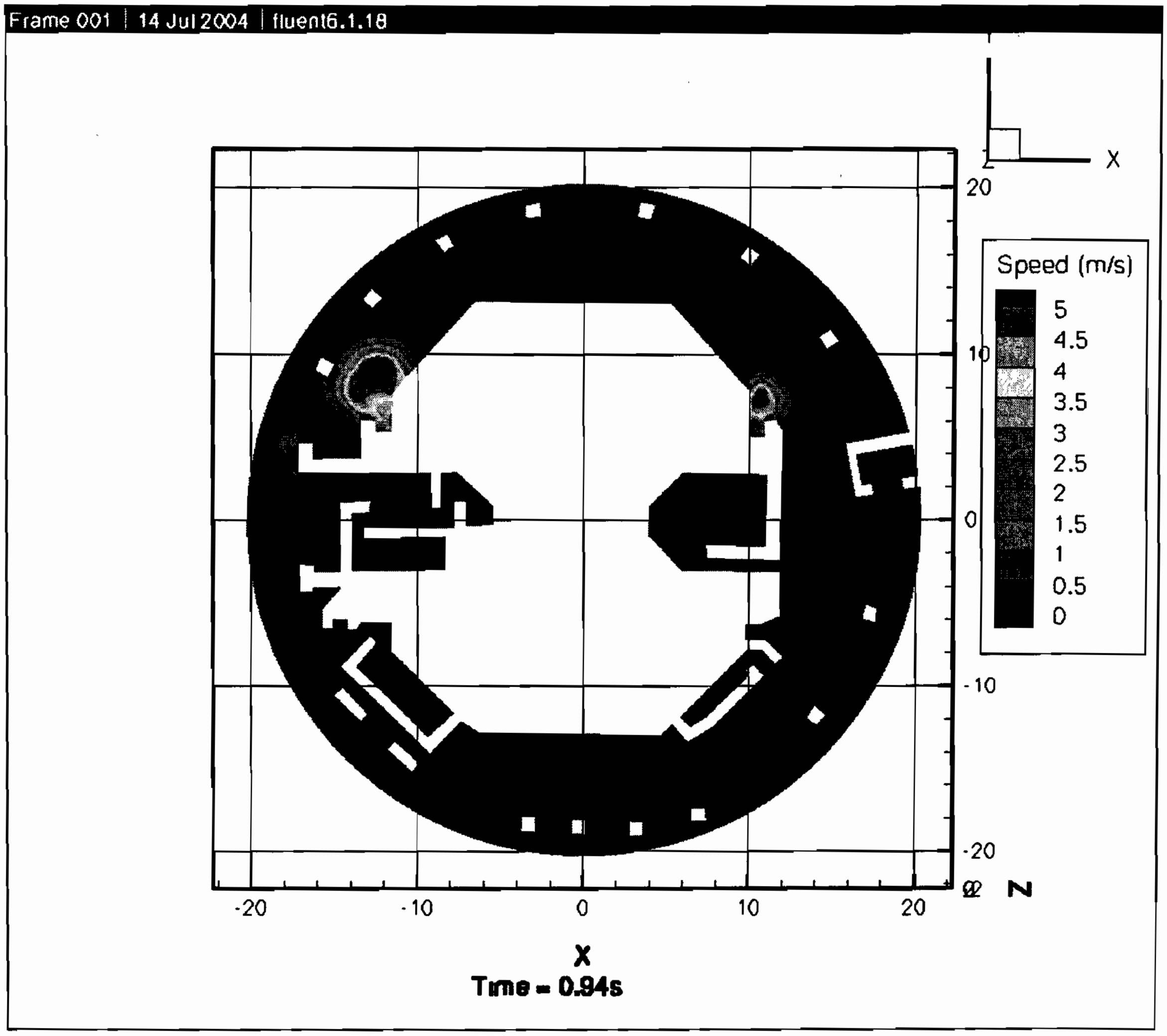


Figure III.2-14. Same as Figure III.2-13 for t = 0.94 Seconds.

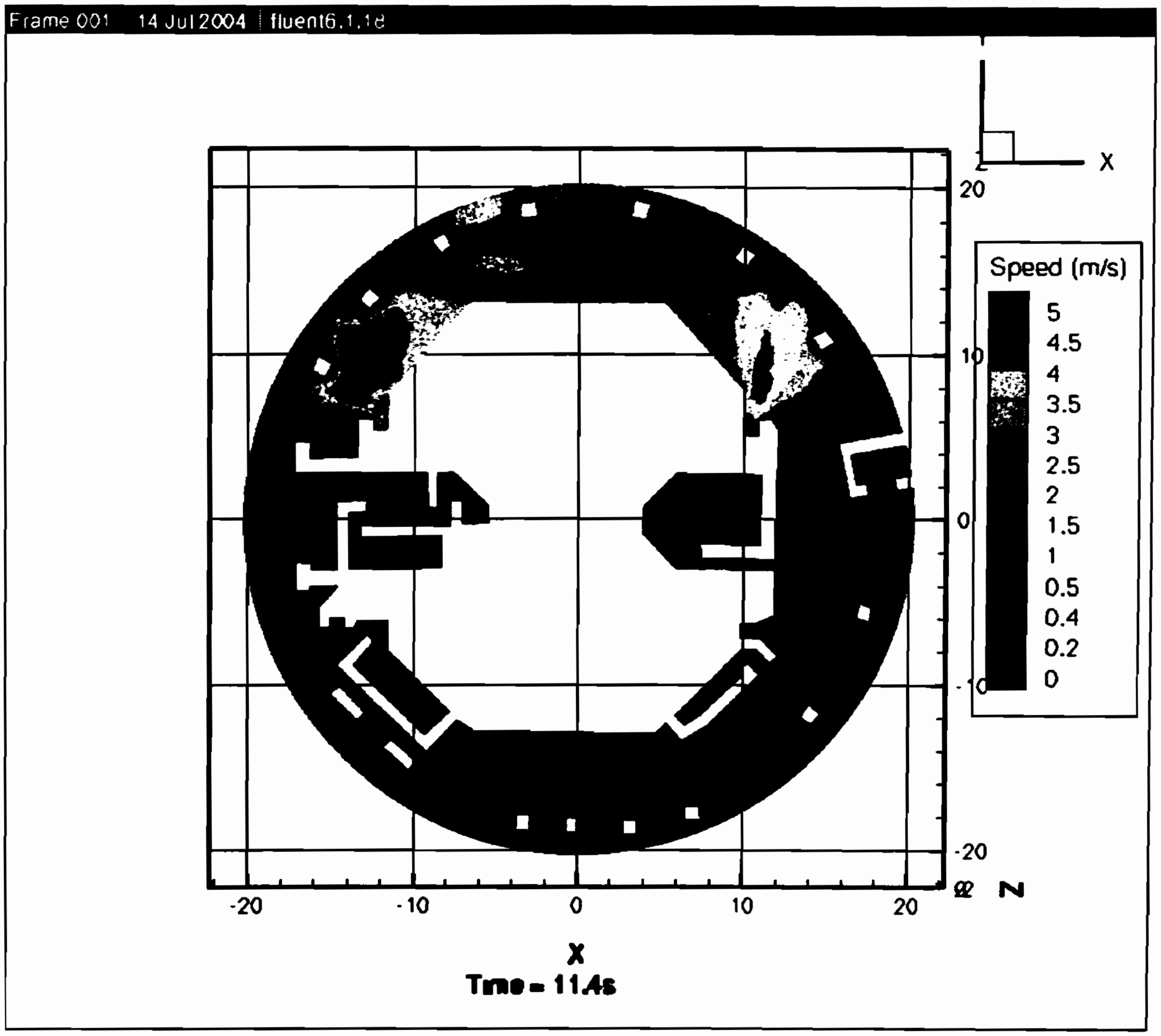


Figure III.2-15. Same as Figure III.2-13 for t = 11.4 Seconds.

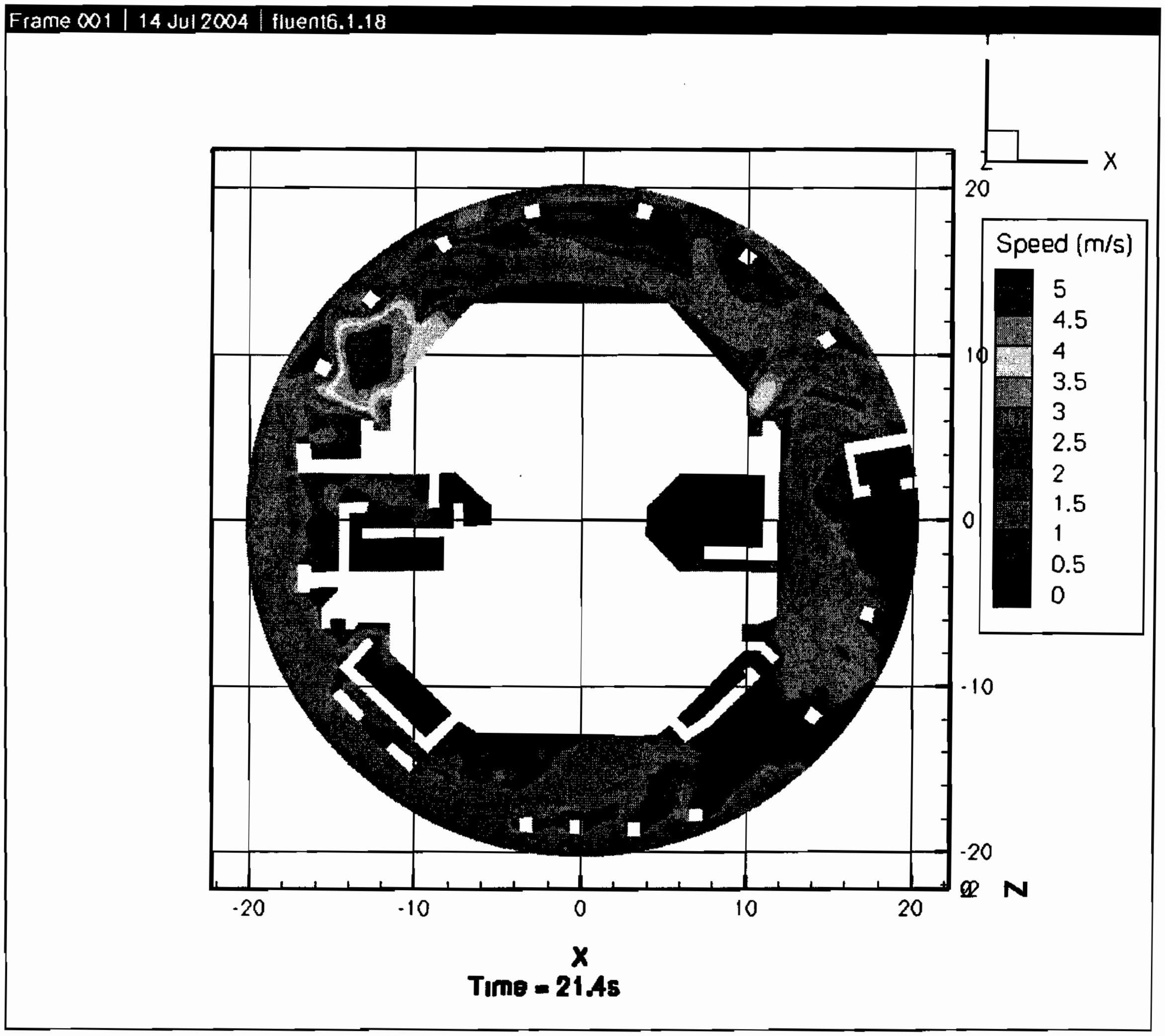


Figure III.2-16. Same as Figure III.2-13 for t = 21.4 Seconds.

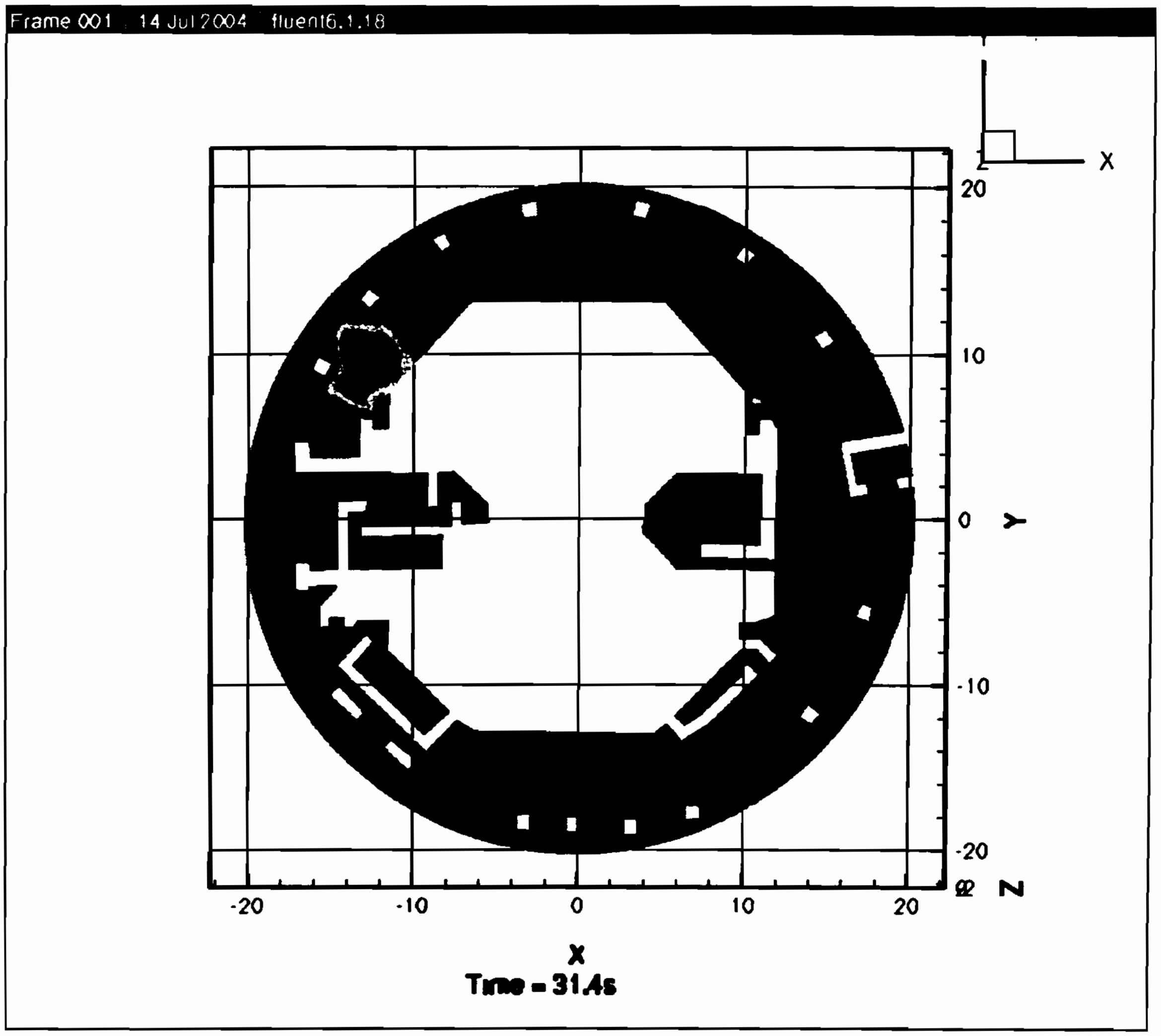


Figure III.2-17. Same as Figure III.2-13 for t = 31.4 Seconds.

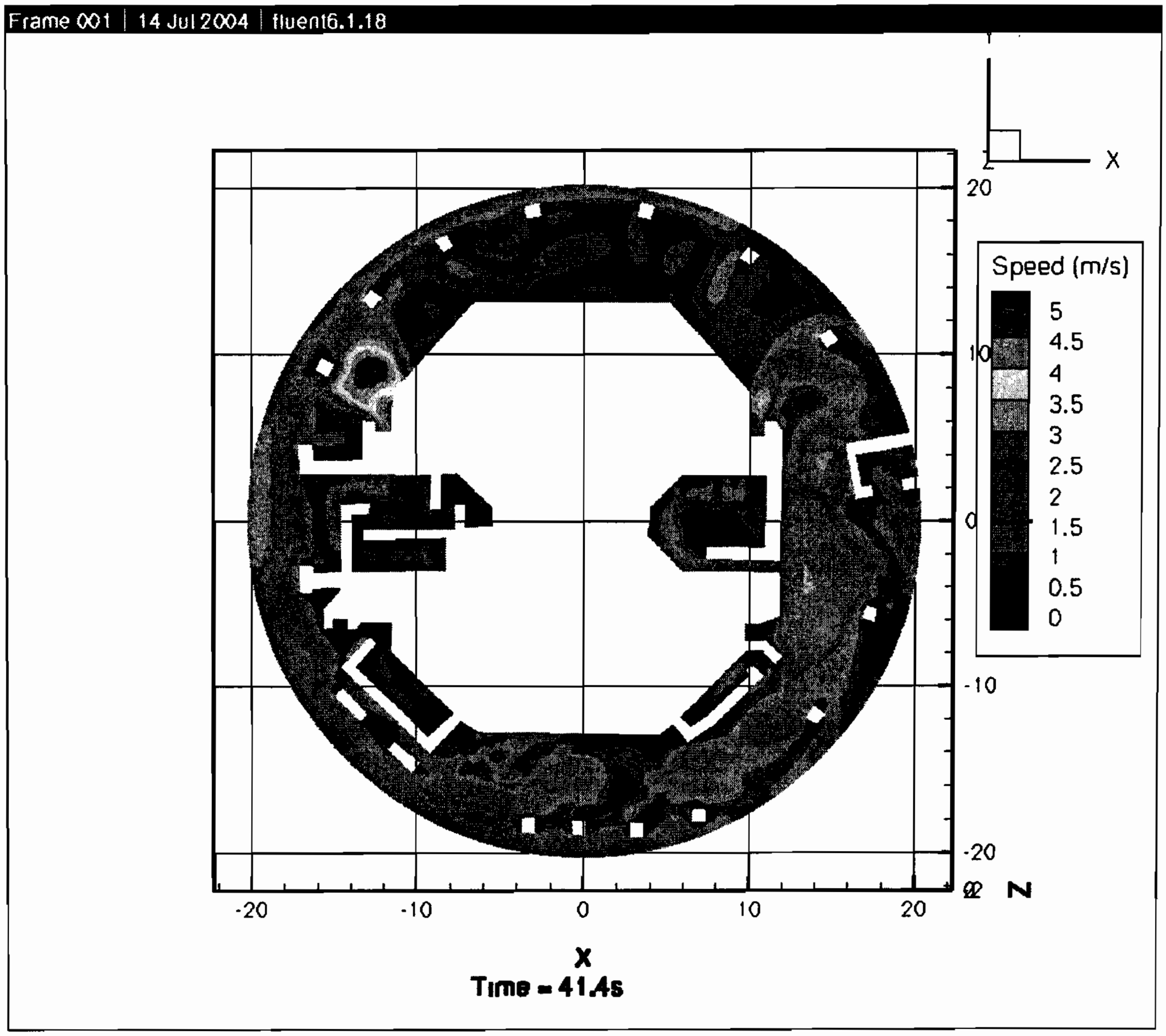


Figure III.2-18. Same as Figure III.2-13 for t = 41.4 Seconds.

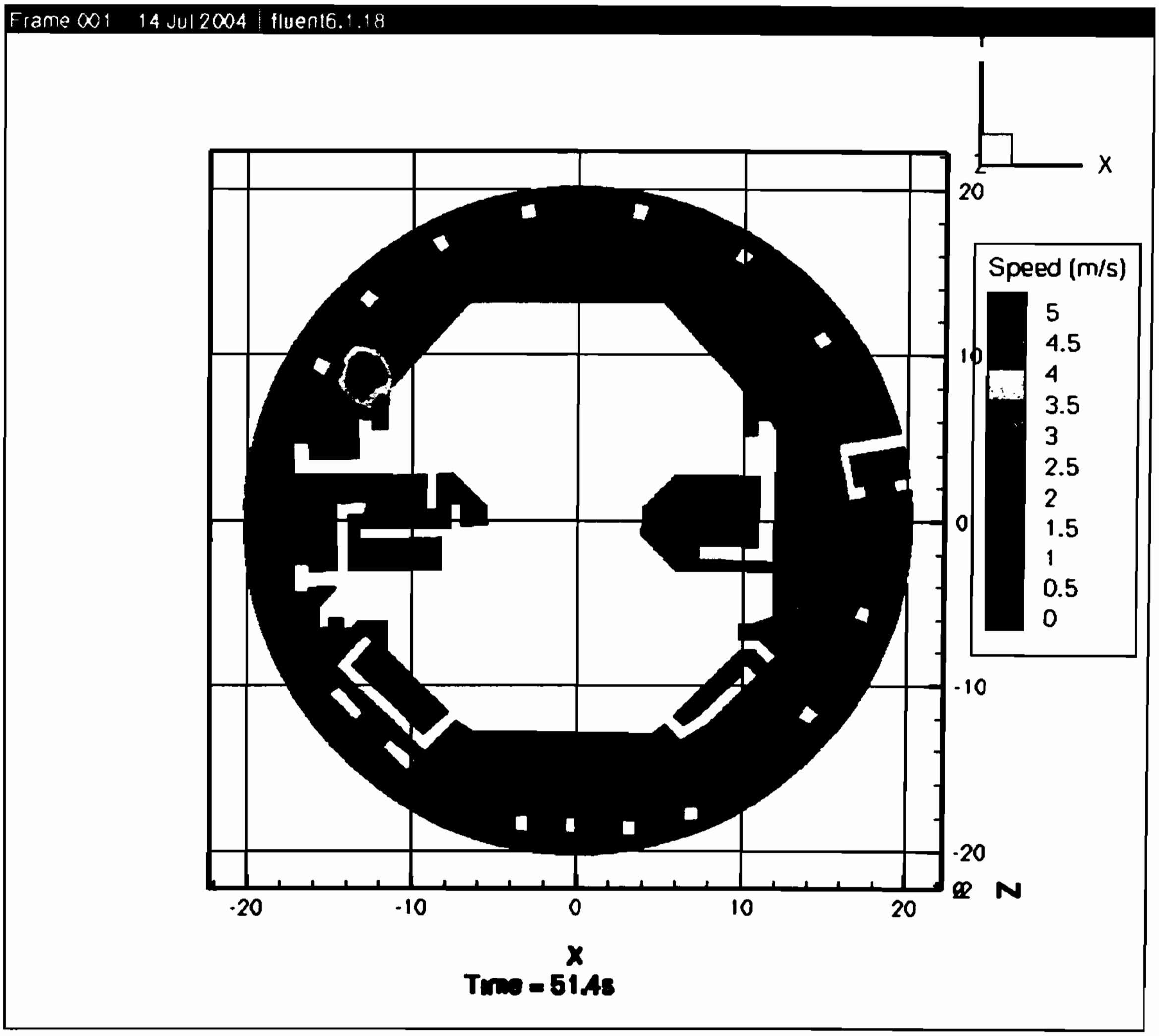


Figure III.2-19. Same as Figure III.2-13 for t = 51.4 Seconds.

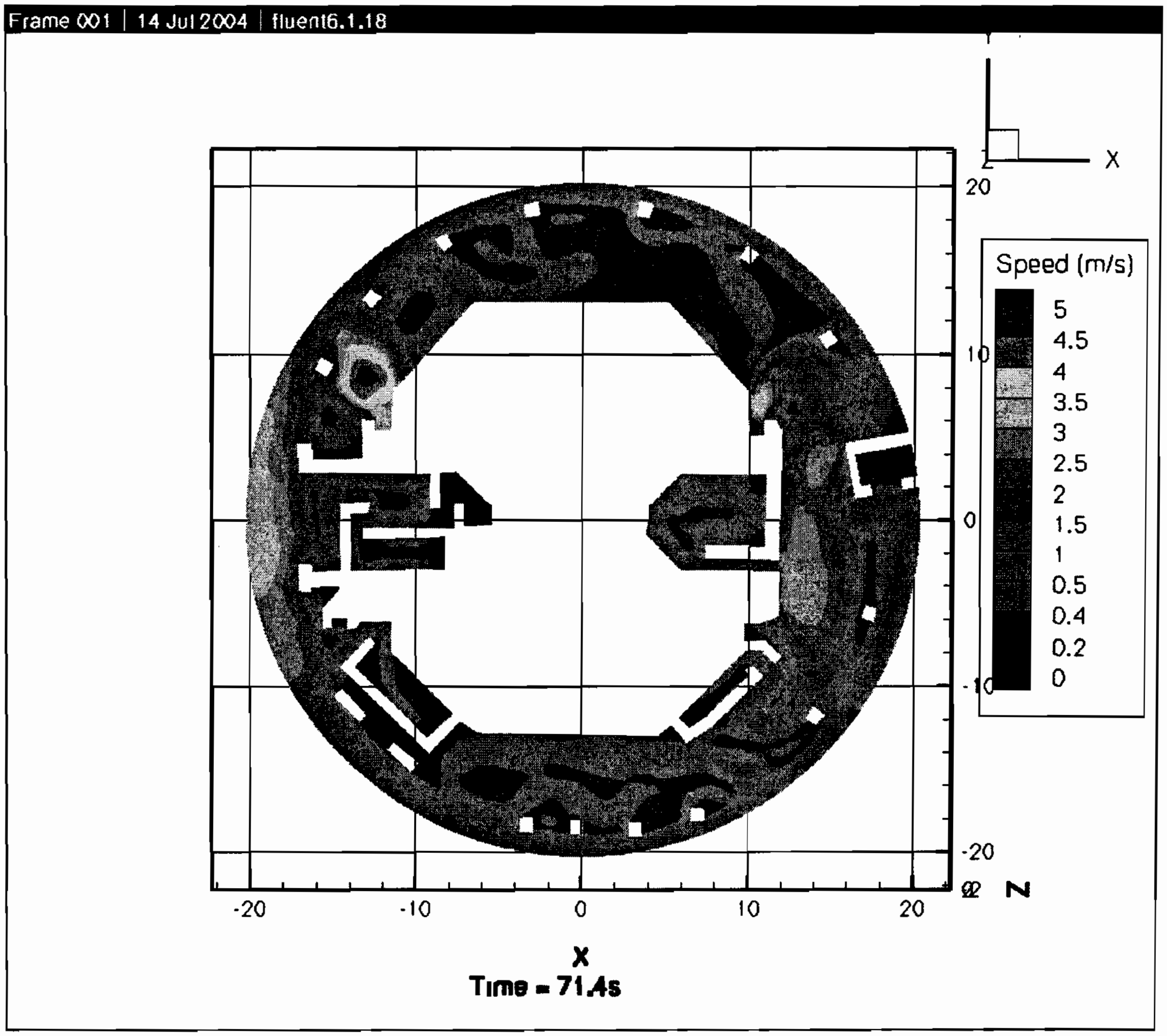


Figure III.2-20. Same as Figure III.2-13 for t = 71.4 Seconds.

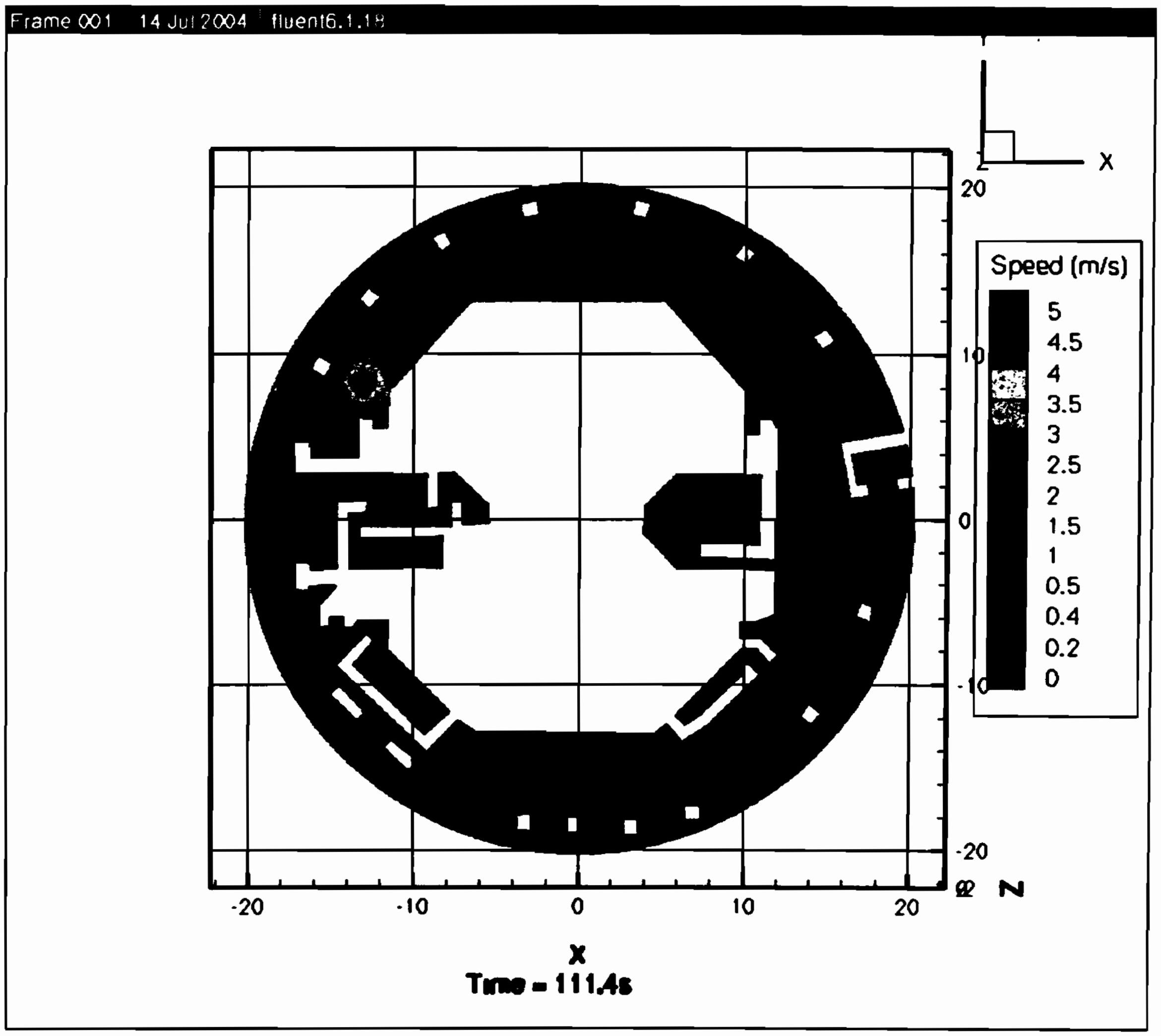


Figure III.2-21. Same as Figure III.2-13 for t = 111.4 Seconds.

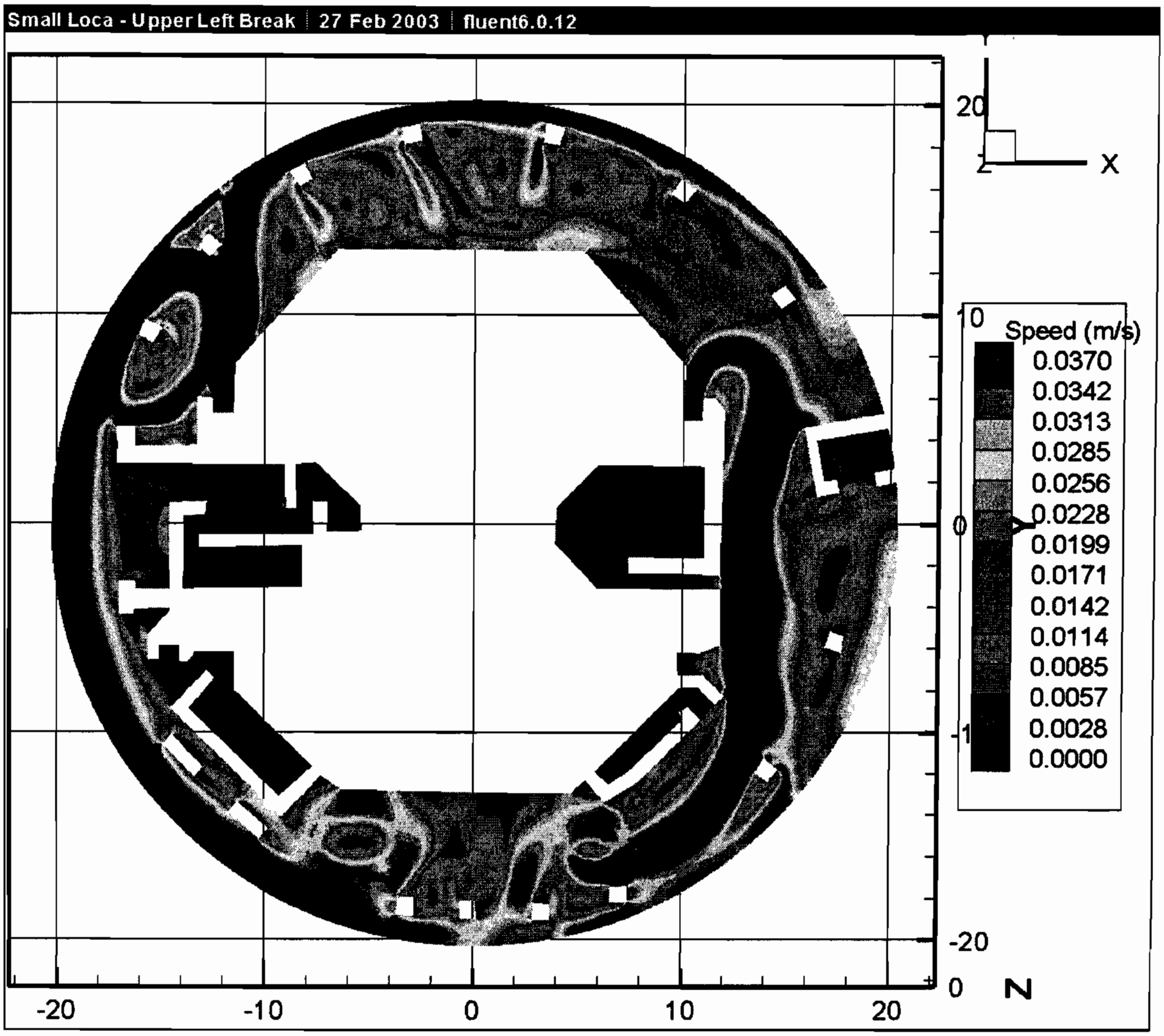


Figure III.2-22. Small LOCA Break Located in the Upper-Left Quadrant. Speeds Greater Than or Equal to the Fiber Threshold (0.037 m/s) Are Colored RED.

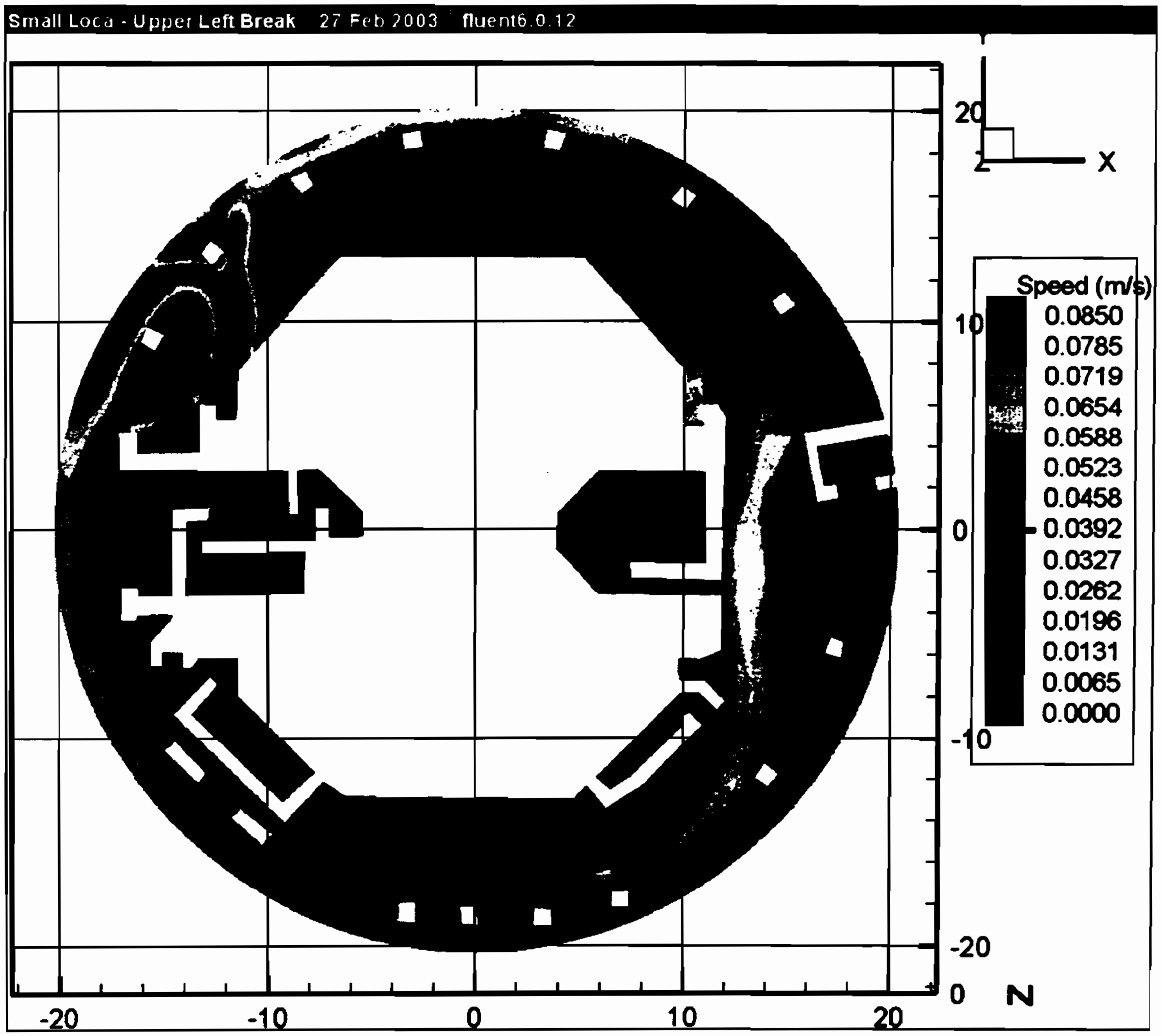


Figure III.2-23. Small LOCA Break Located in the Upper-Left Quadrant. Speeds Greater Than or Equal to the RMI Threshold (0.085 m/s) Are Colored RED.

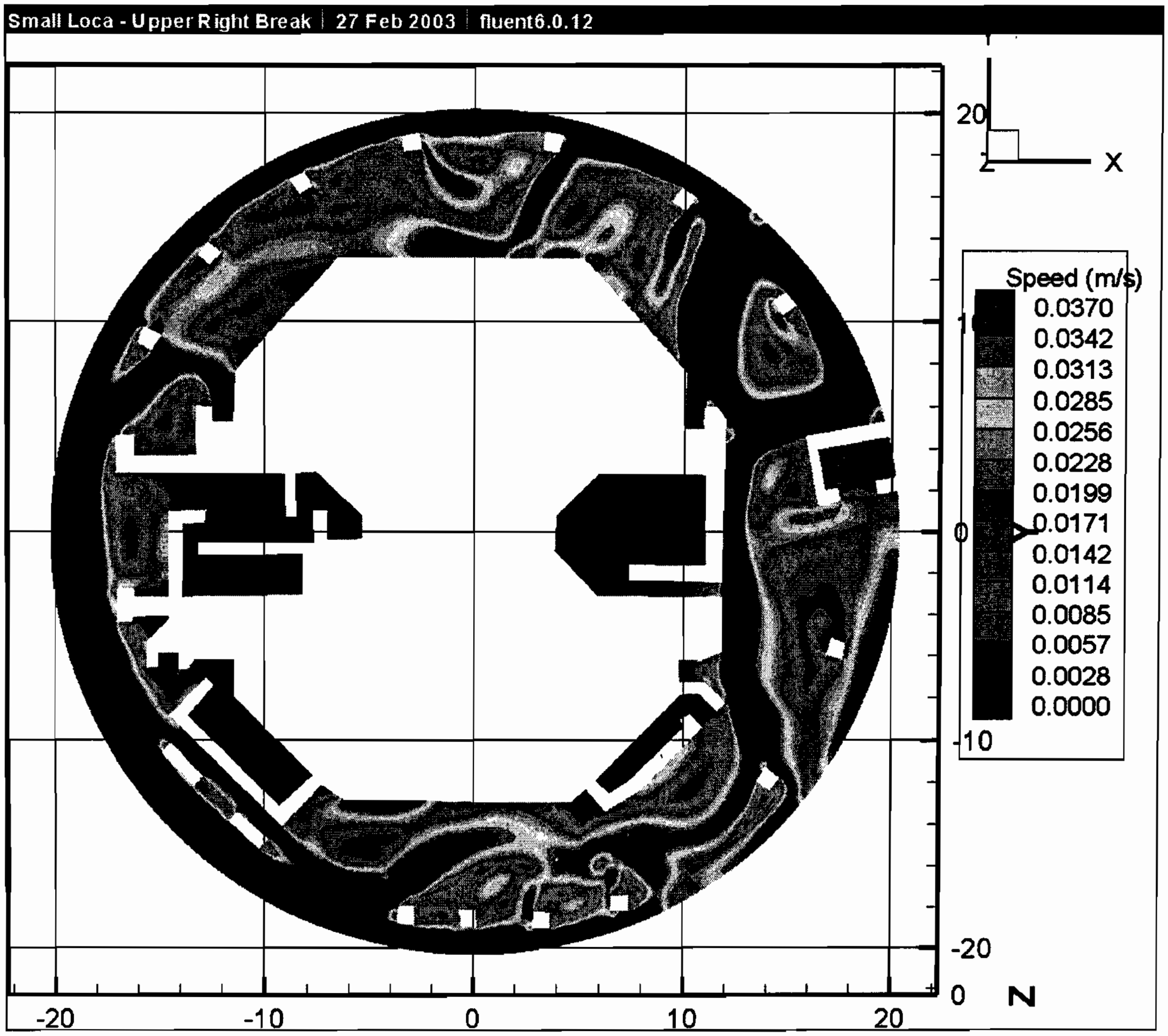


Figure III.2-24. Small LOCA Break Located in the Upper-Right Quadrant. Speeds Greater Than or Equal to the Fiber Threshold (0.037 m/s) Are Colored RED.

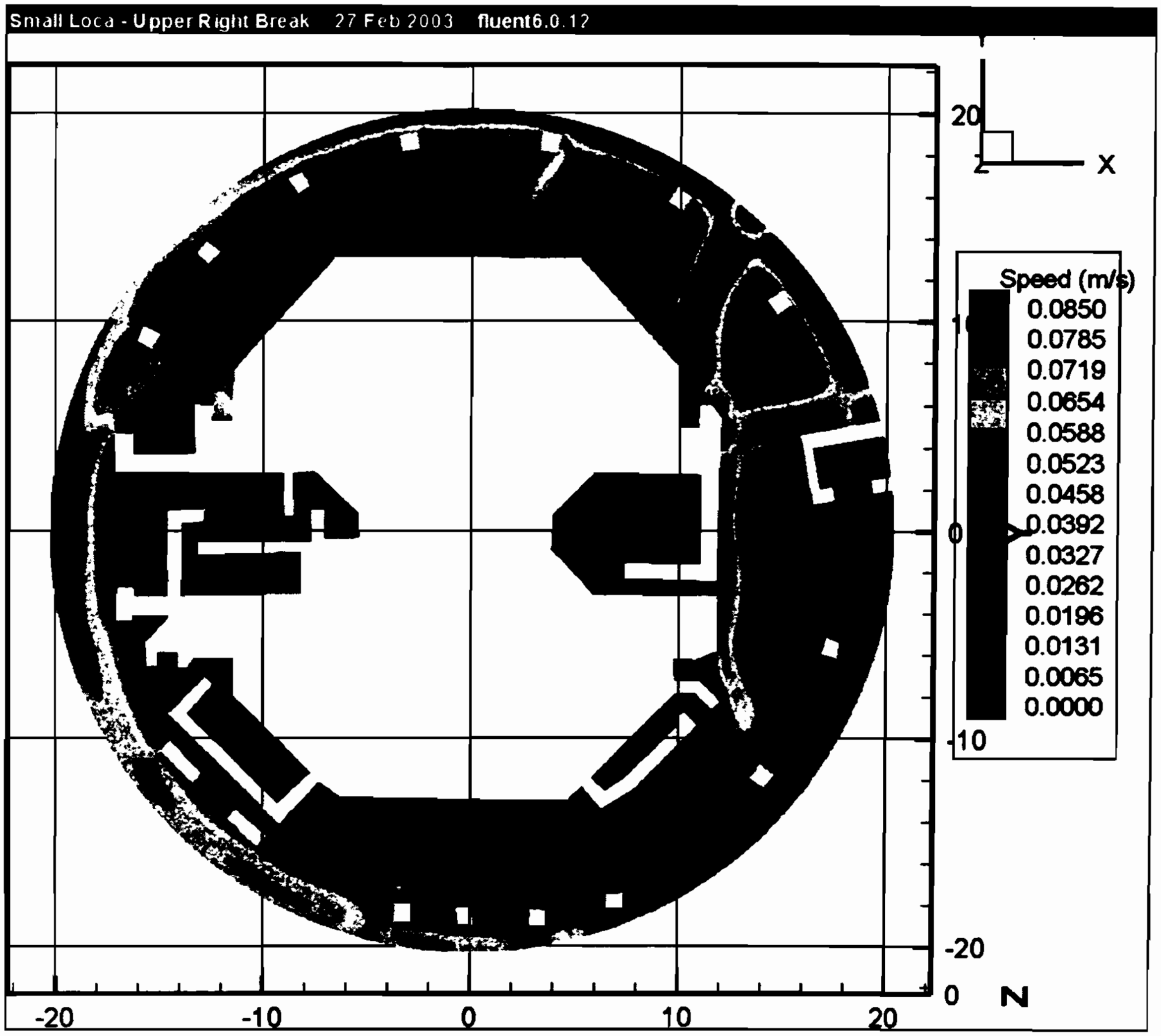


Figure III.2-25. Small LOCA Break Located in the Upper-Right Quadrant. Speeds Greater Than or Equal to the RMI Threshold (0.085 m/s) Are Colored RED.

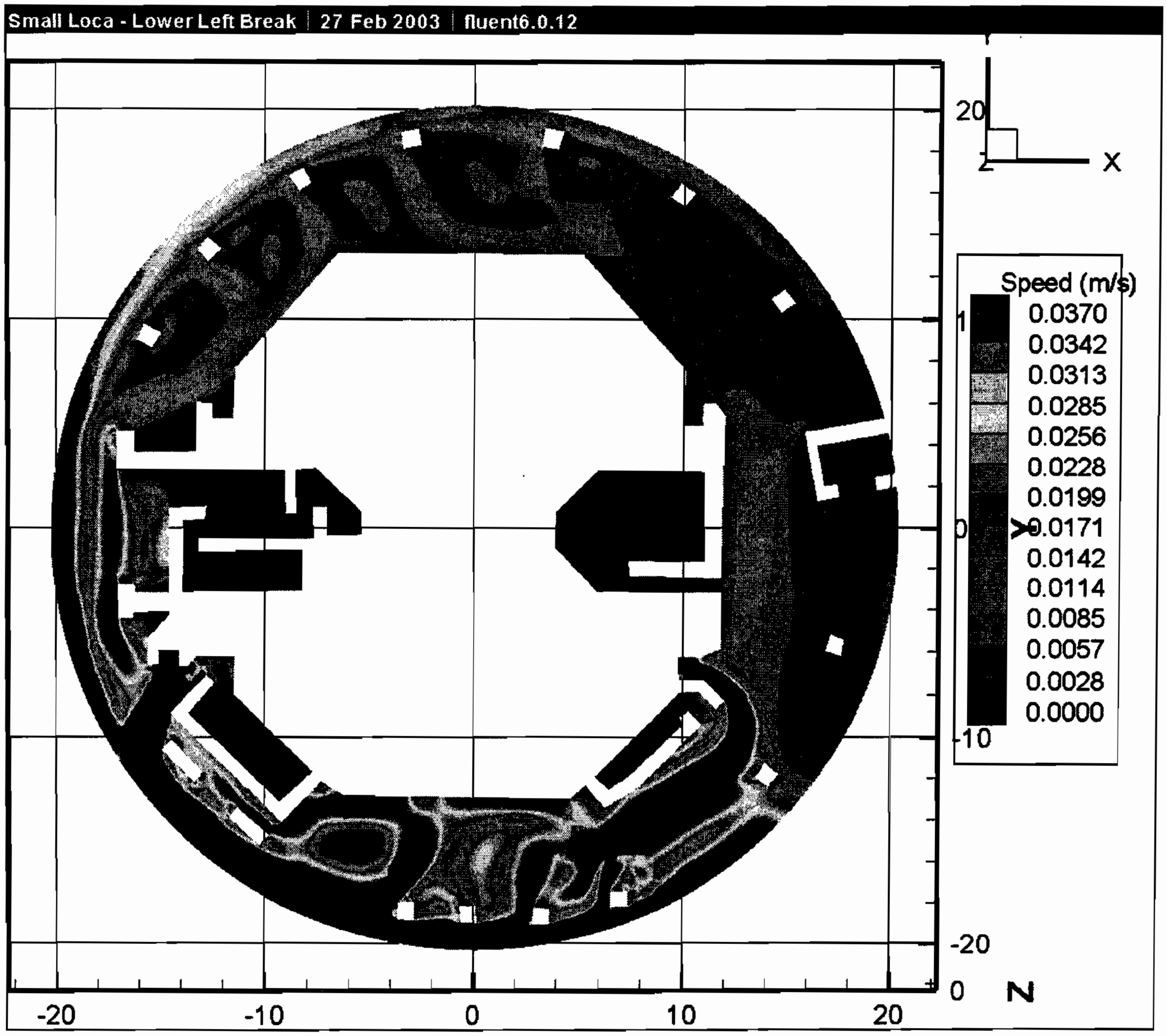


Figure III.2-26. Small LOCA Break Located in the Lower-Left Quadrant. Speeds Greater Than or Equal to the Fiber Threshold (0.037 m/s) Are Colored RED.

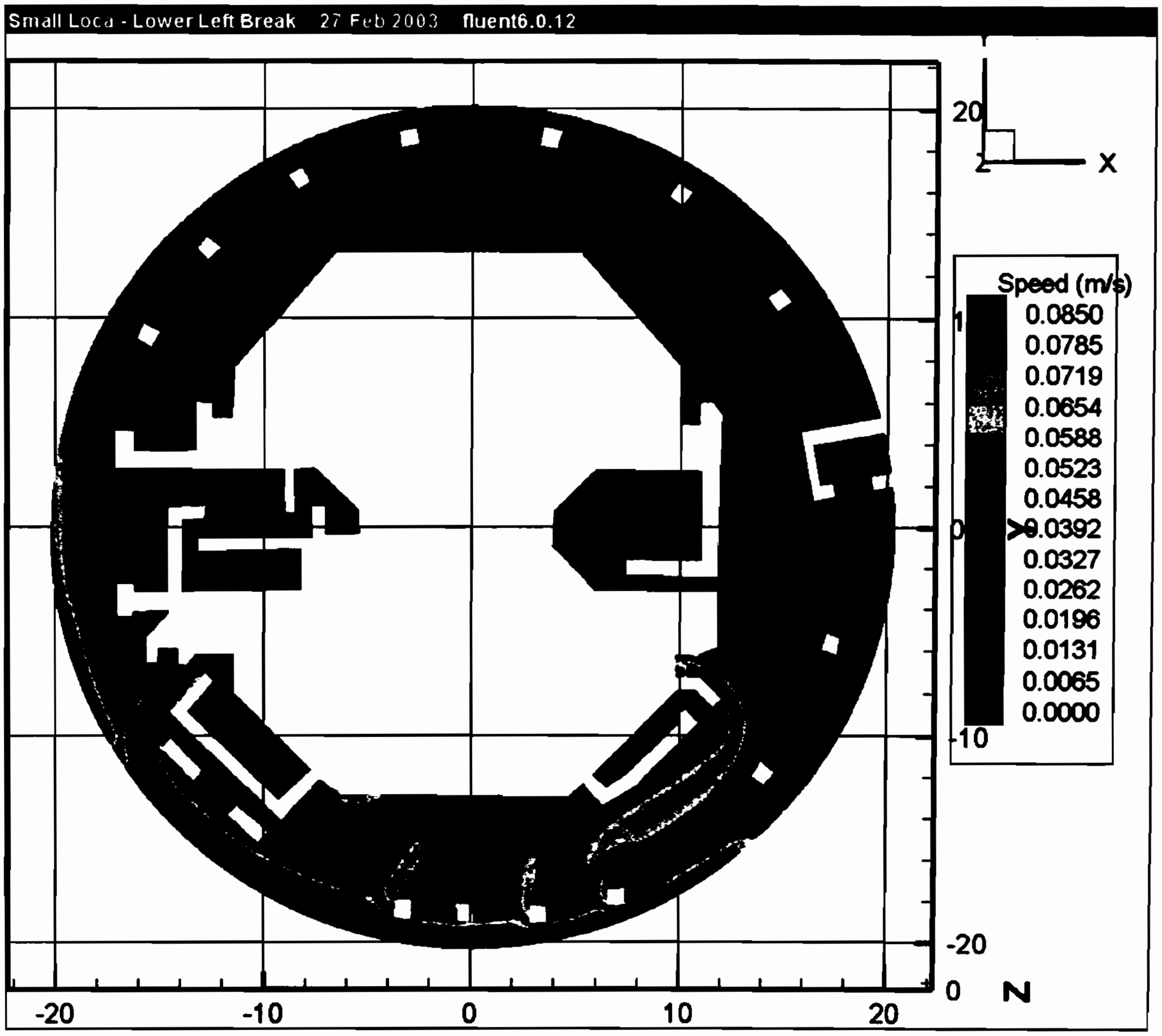


Figure III.2-27. Small LOCA Break Located in the Lower-Left Quadrant. Speeds Greater Than or Equal to the RMI Threshold (0.085 m/s) Are Colored RED.

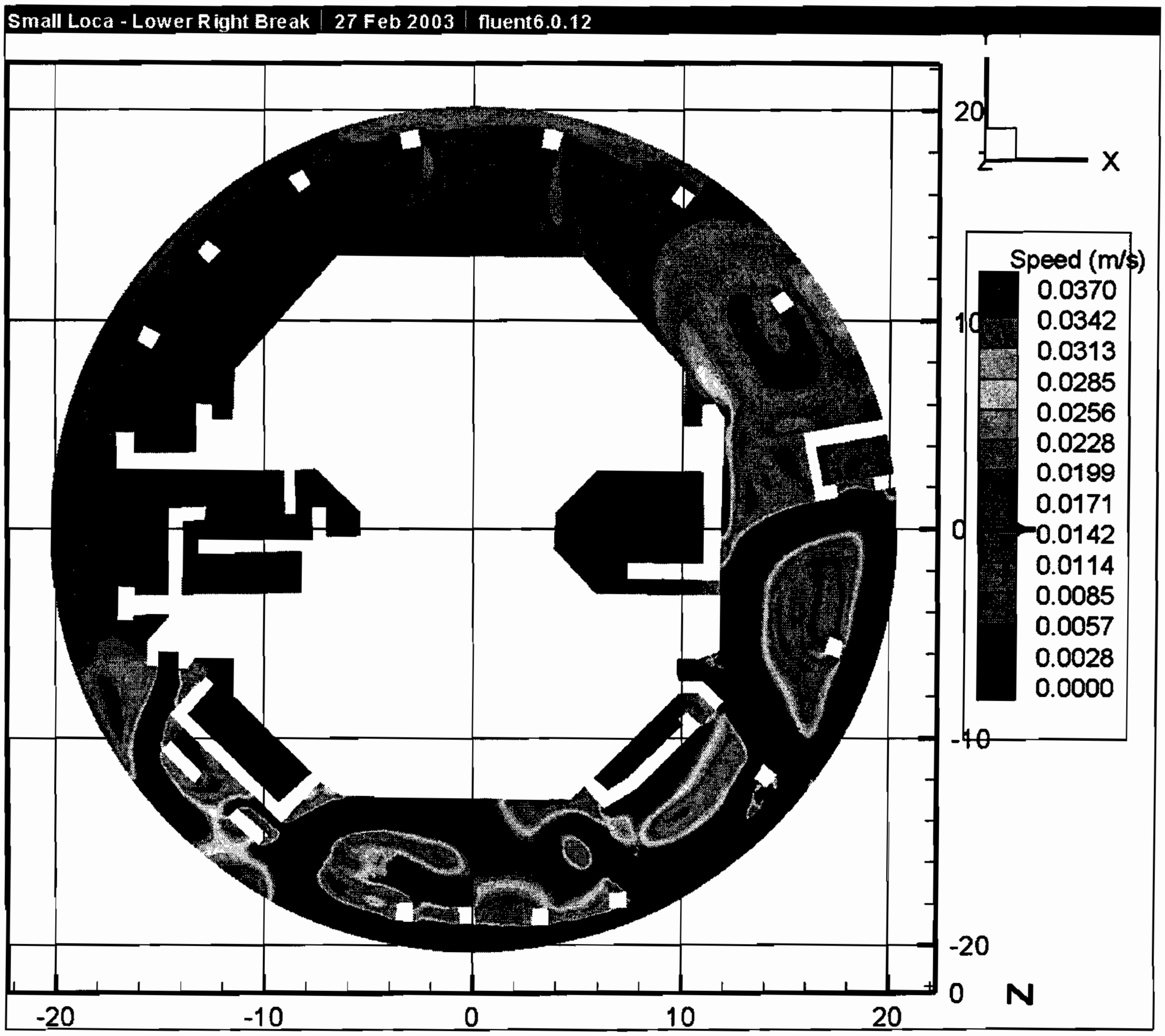


Figure III.2-28. Small LOCA Break Located in the Lower-Right Quadrant. Speeds Greater Than or Equal to the Fiber Threshold (0.037 m/s) Are Colored RED.

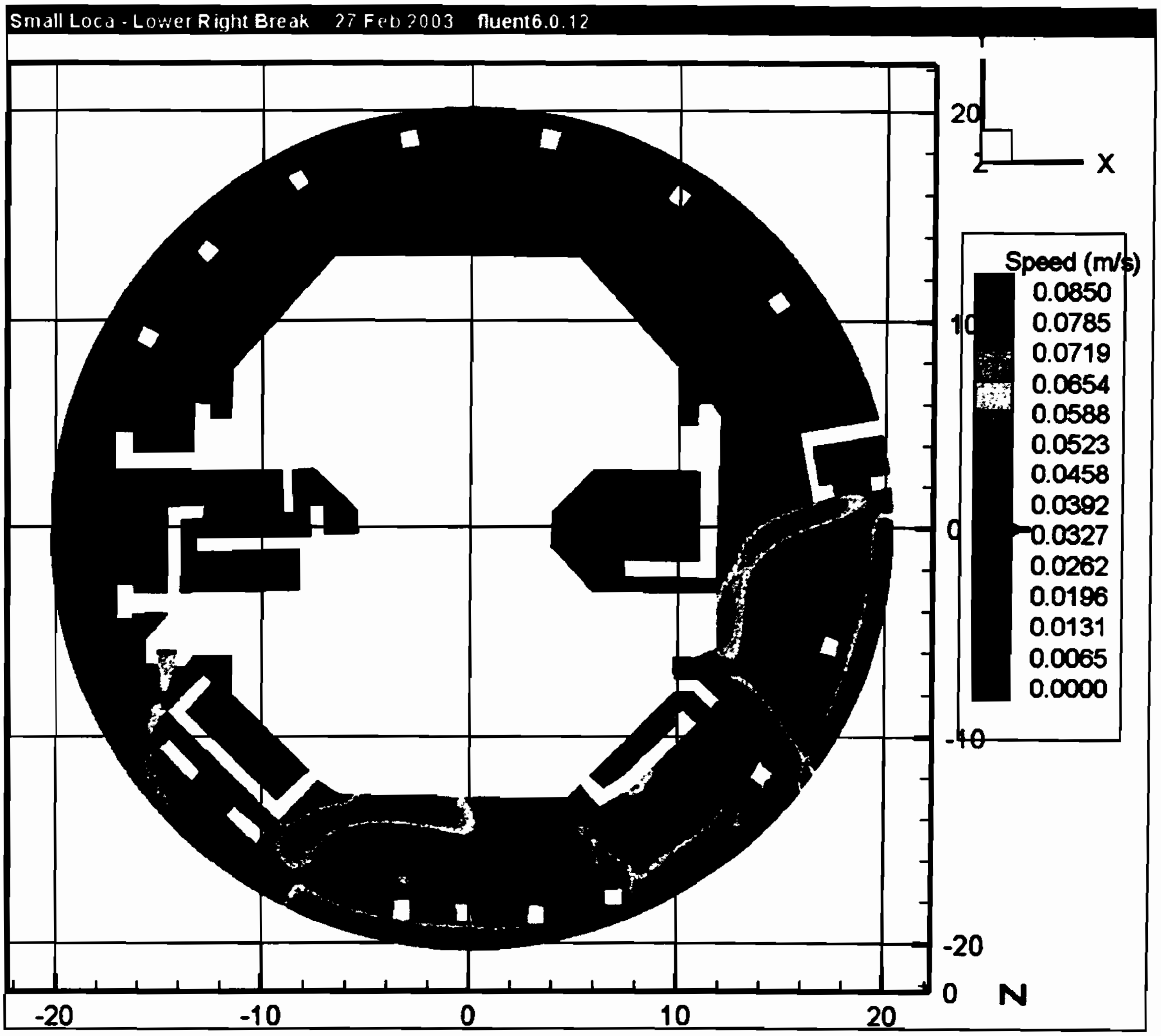


Figure III.2-29. Small LOCA Break Located in the Lower-Right Quadrant. Speeds Greater Than or Equal to the RMI Threshold (0.085 m/s) Are Colored RED.

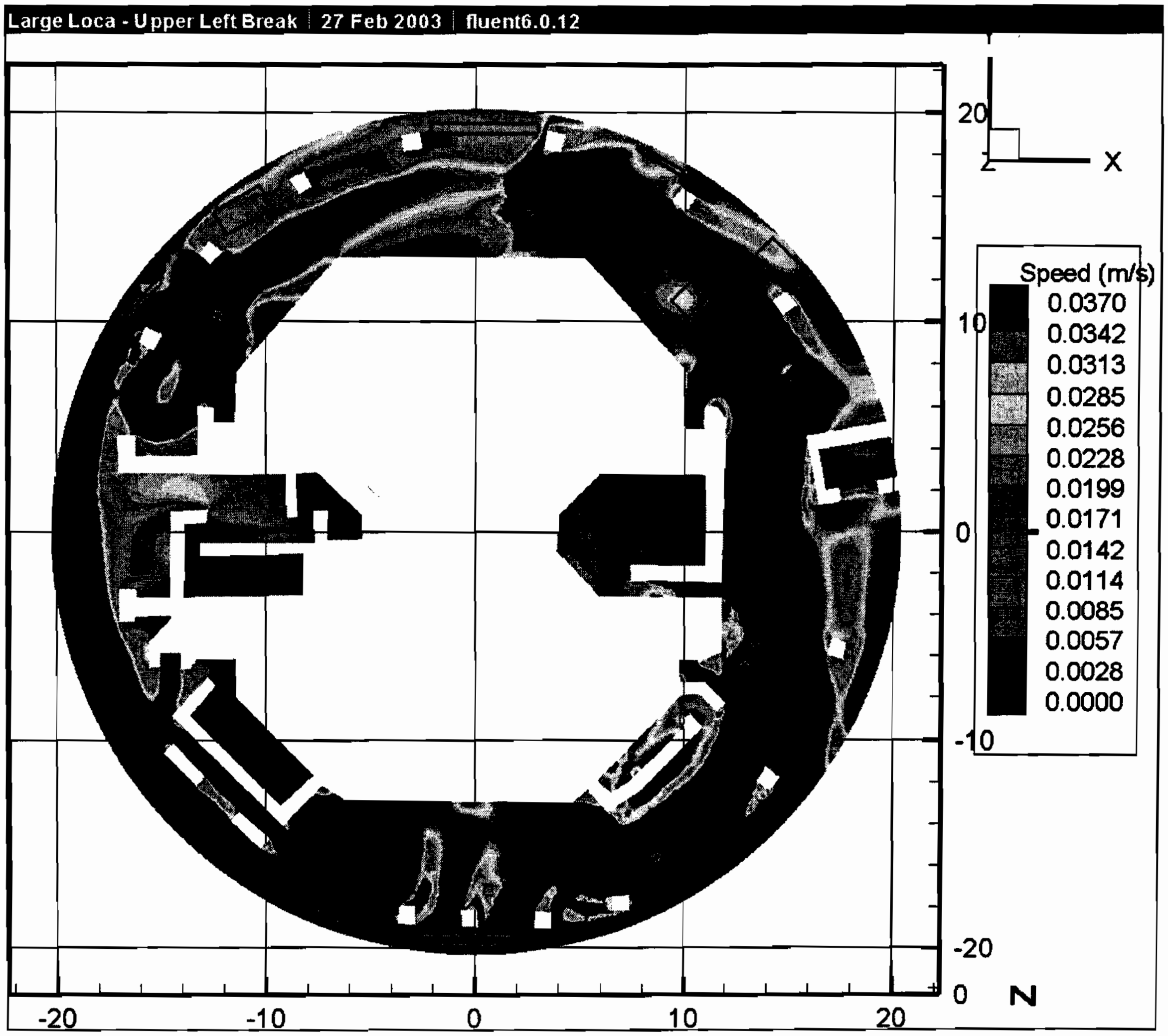


Figure III.2-30. Large LOCA Break Located in the Upper-Left Quadrant. Speeds Greater Than or Equal to the Fiber Threshold (0.037 m/s) Are Colored RED.

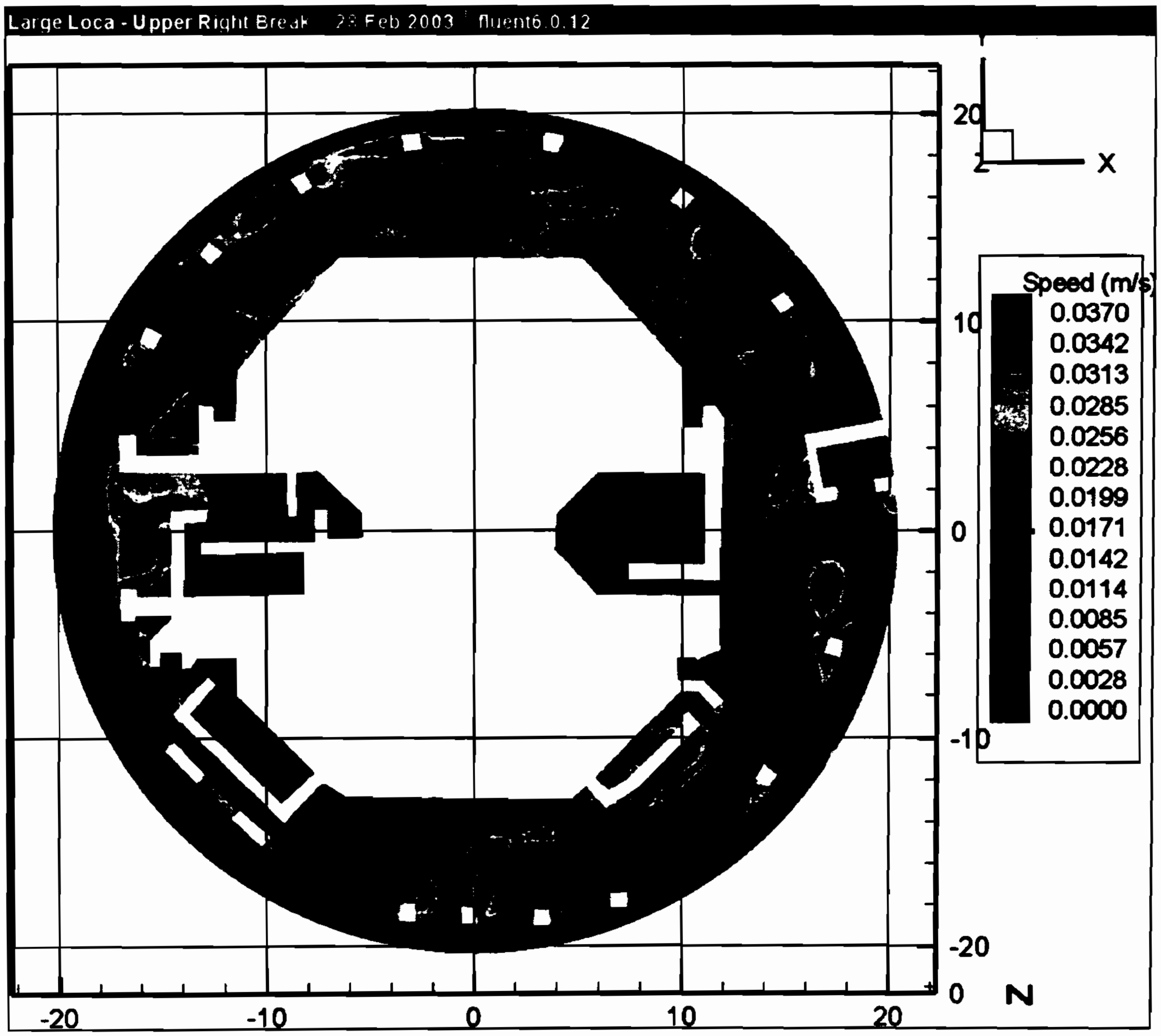


Figure III.2-31. Large LOCA Break Located in the Upper-Right Quadrant. Speeds Greater Than or Equal to the Fiber Threshold (0.037 m/s) Are Colored RED.

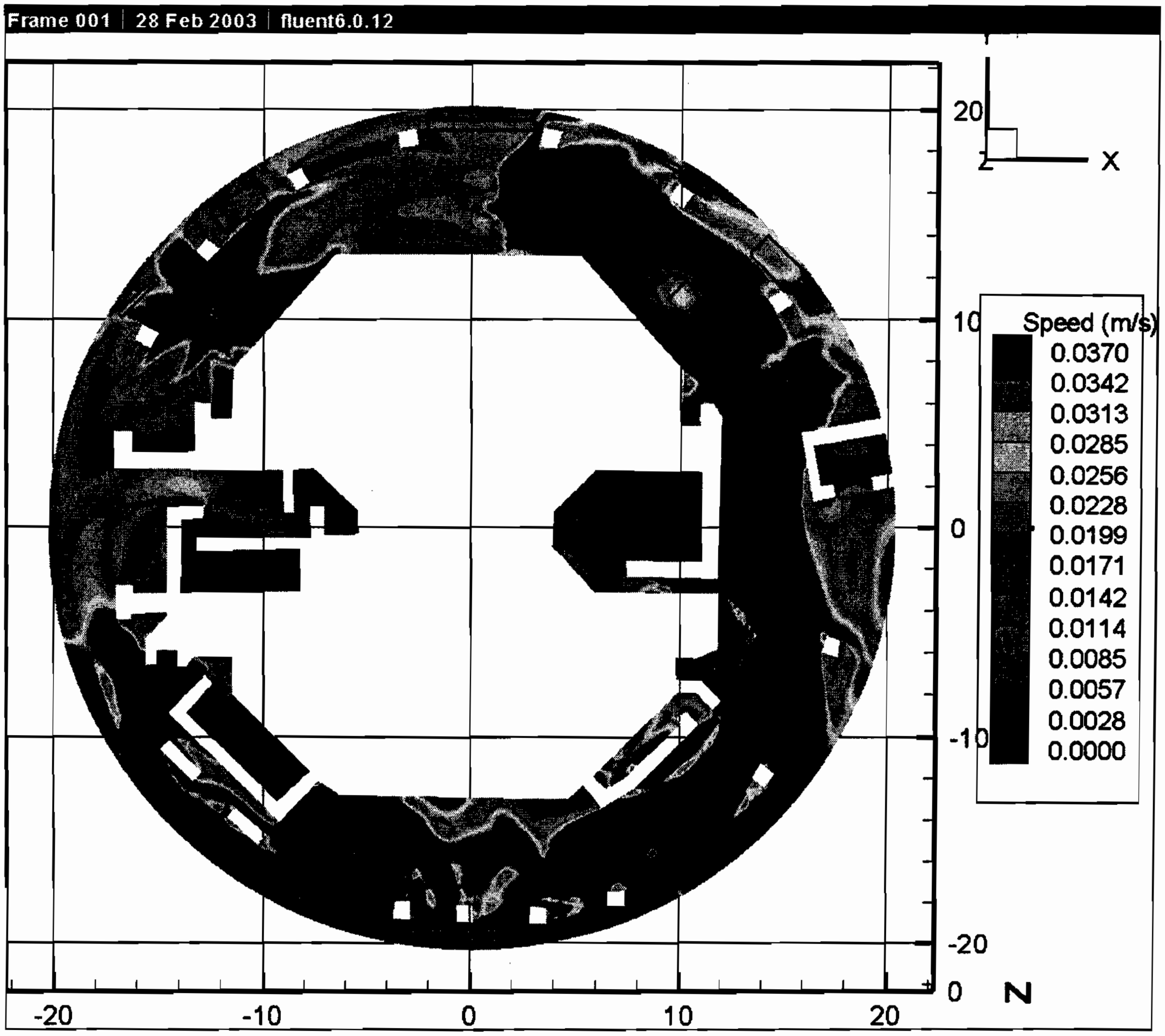


Figure III.2-32. Large LOCA Break Located in the Lower-Left Quadrant. Speeds Greater Than or Equal to the Fiber Threshold (0.037 m/s) Are Colored RED.

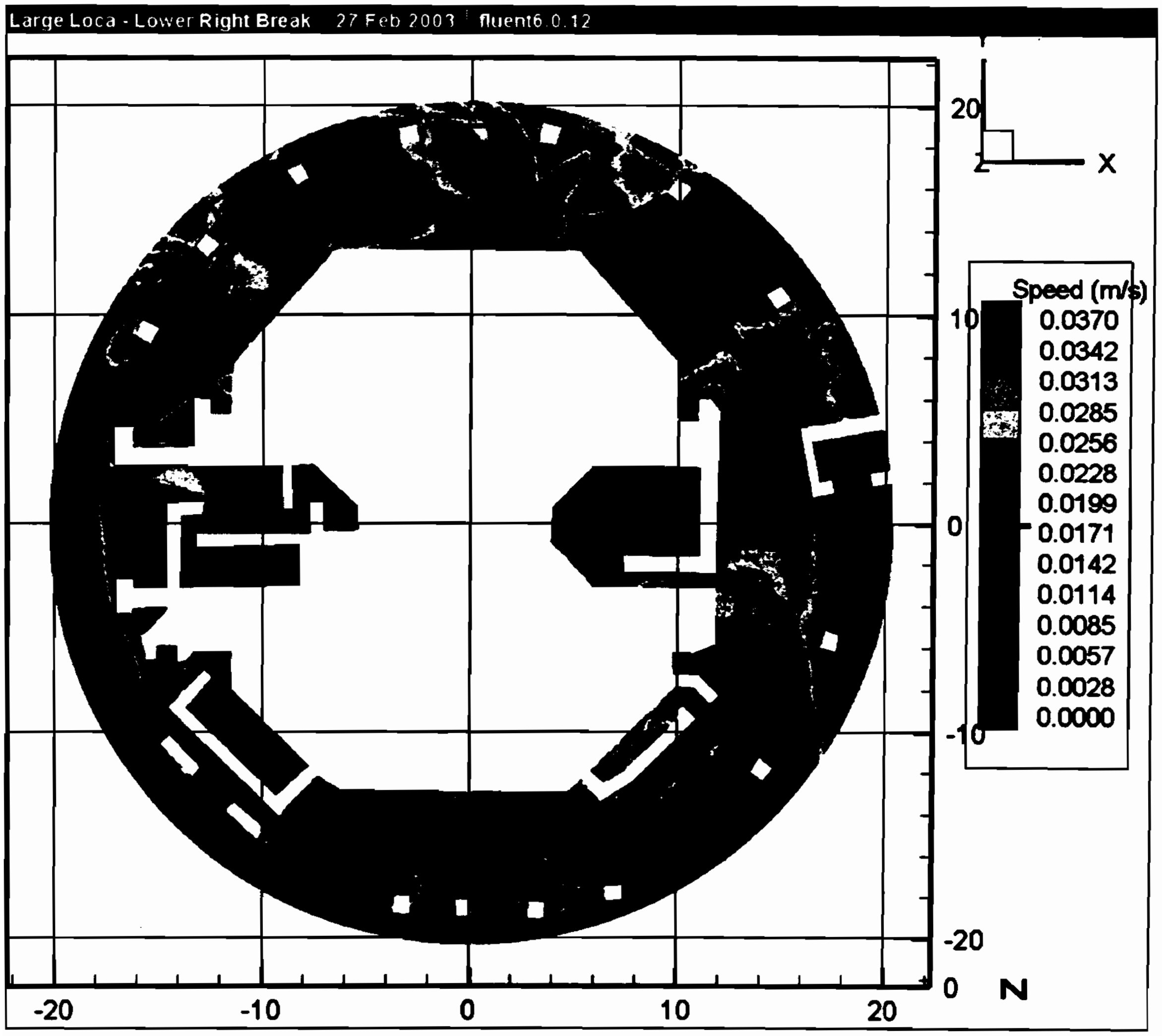


Figure III.2-33. Large LOCA Break Located in the Lower-Right Quadrant. Speeds Greater Than or Equal to the Fiber Threshold (0.037 m/s) Are Colored RED.

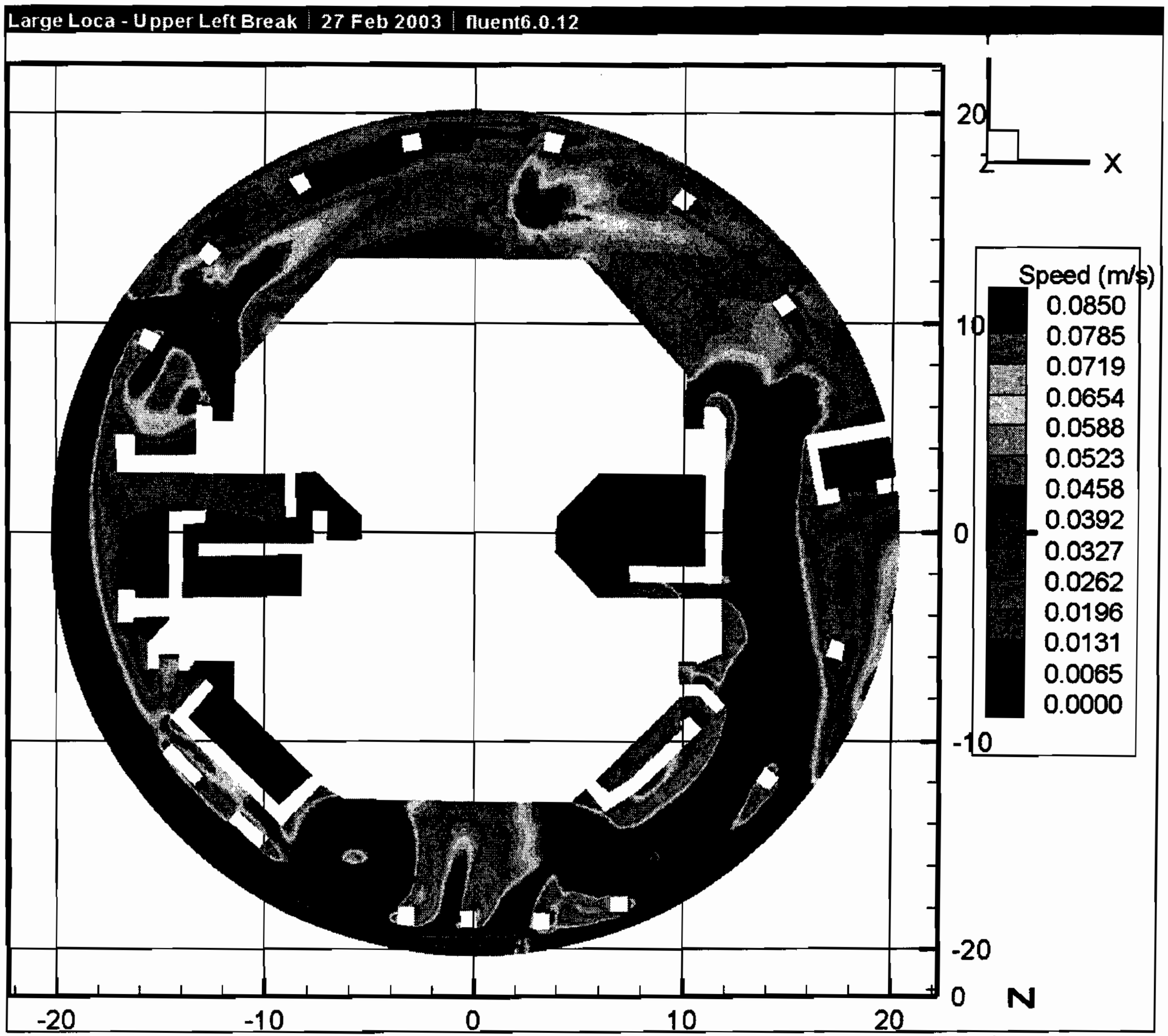


Figure III.2-34. Large LOCA Break Located in the Upper-Left Quadrant. Speeds Greater Than or Equal to the RMI Threshold (0.085 m/s) Are Colored RED.

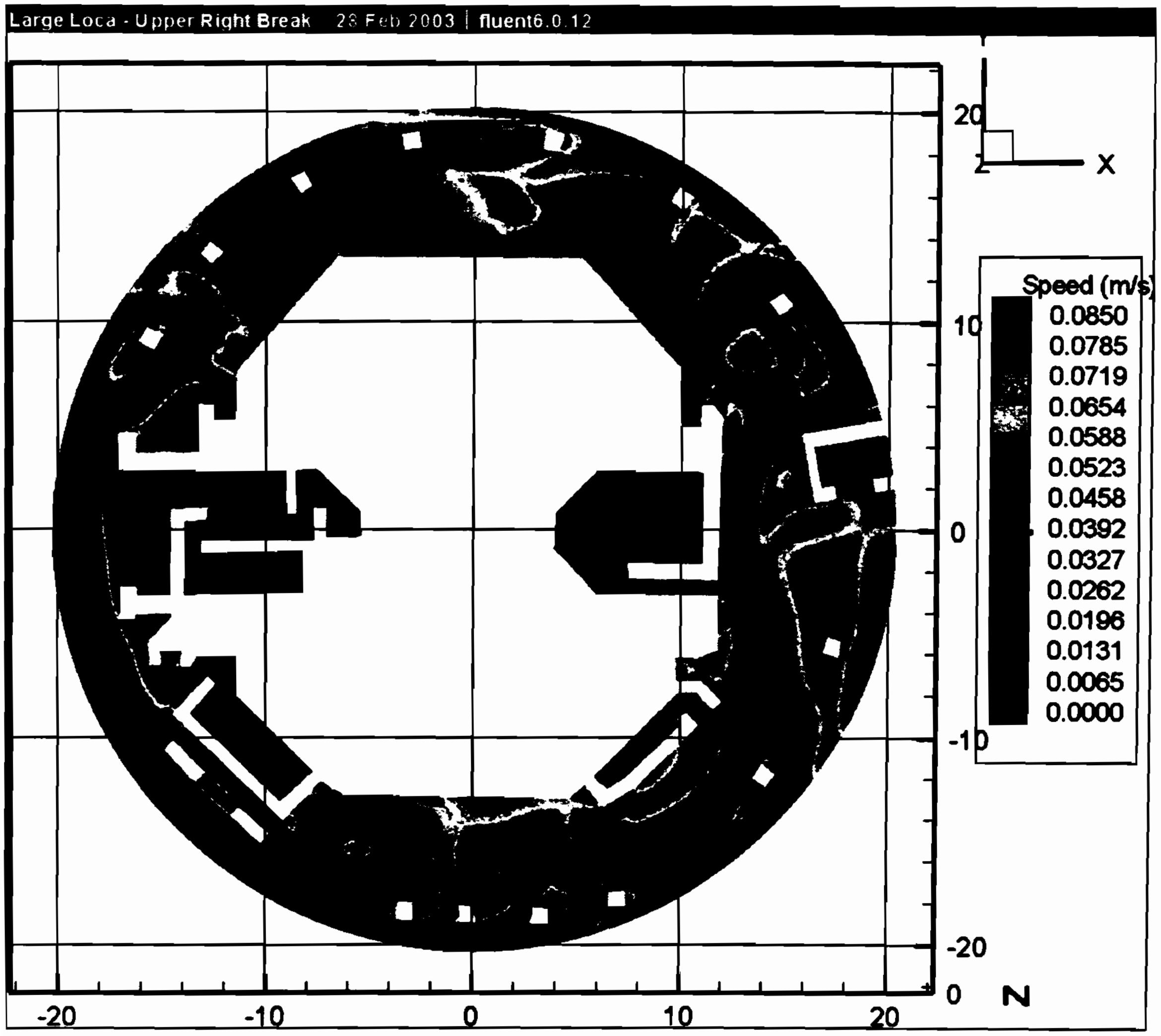


Figure III.2-35. Large LOCA Break Located in the Upper-Right Quadrant. Speeds Greater Than or Equal to the RMI Threshold (0.085 m/s) Are Colored RED.

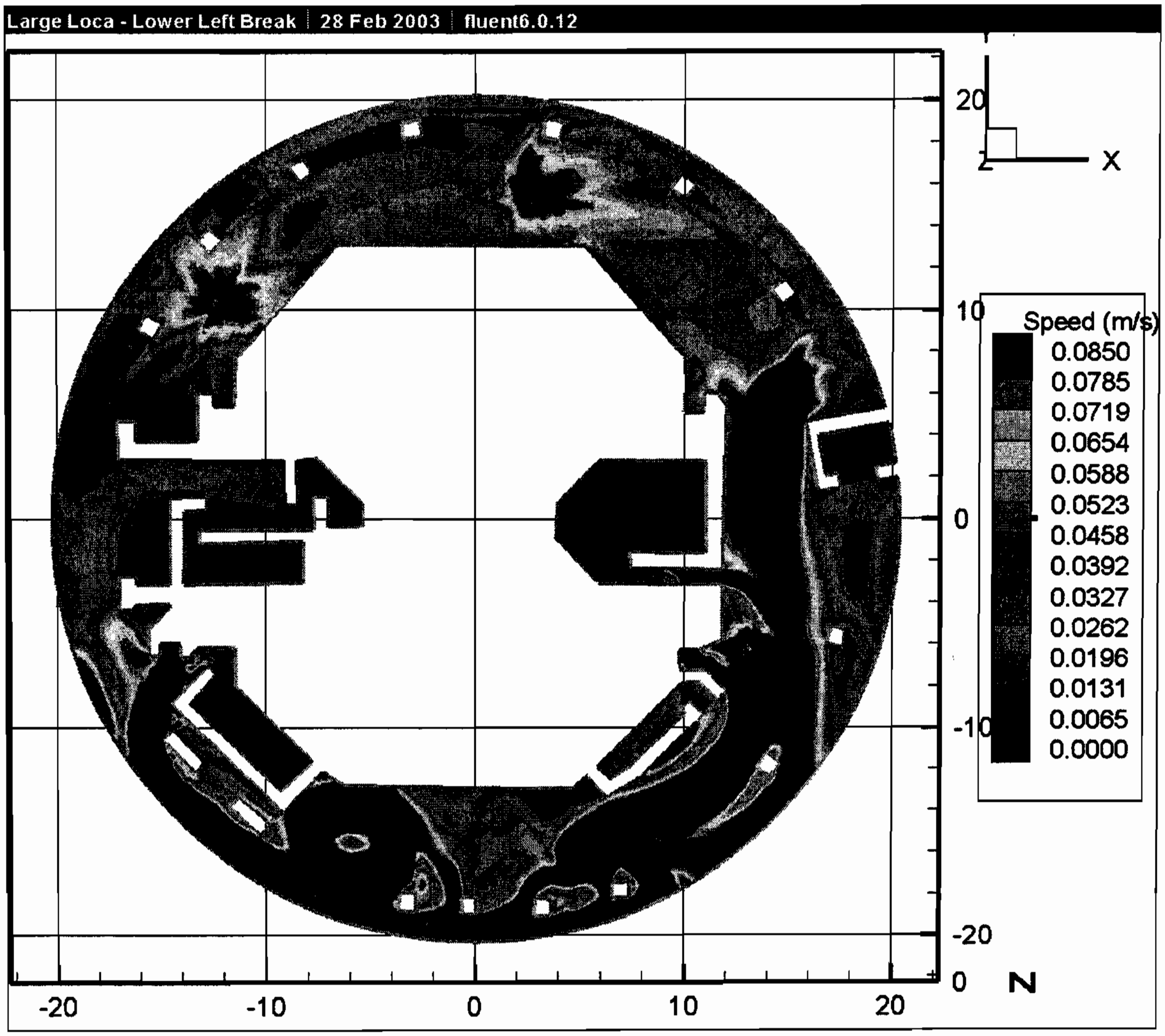


Figure III.2-36. Large LOCA Break Located in the Lower-Left Quadrant. Speeds Greater Than or Equal to the RMI Threshold (0.085 m/s) Are Colored RED.

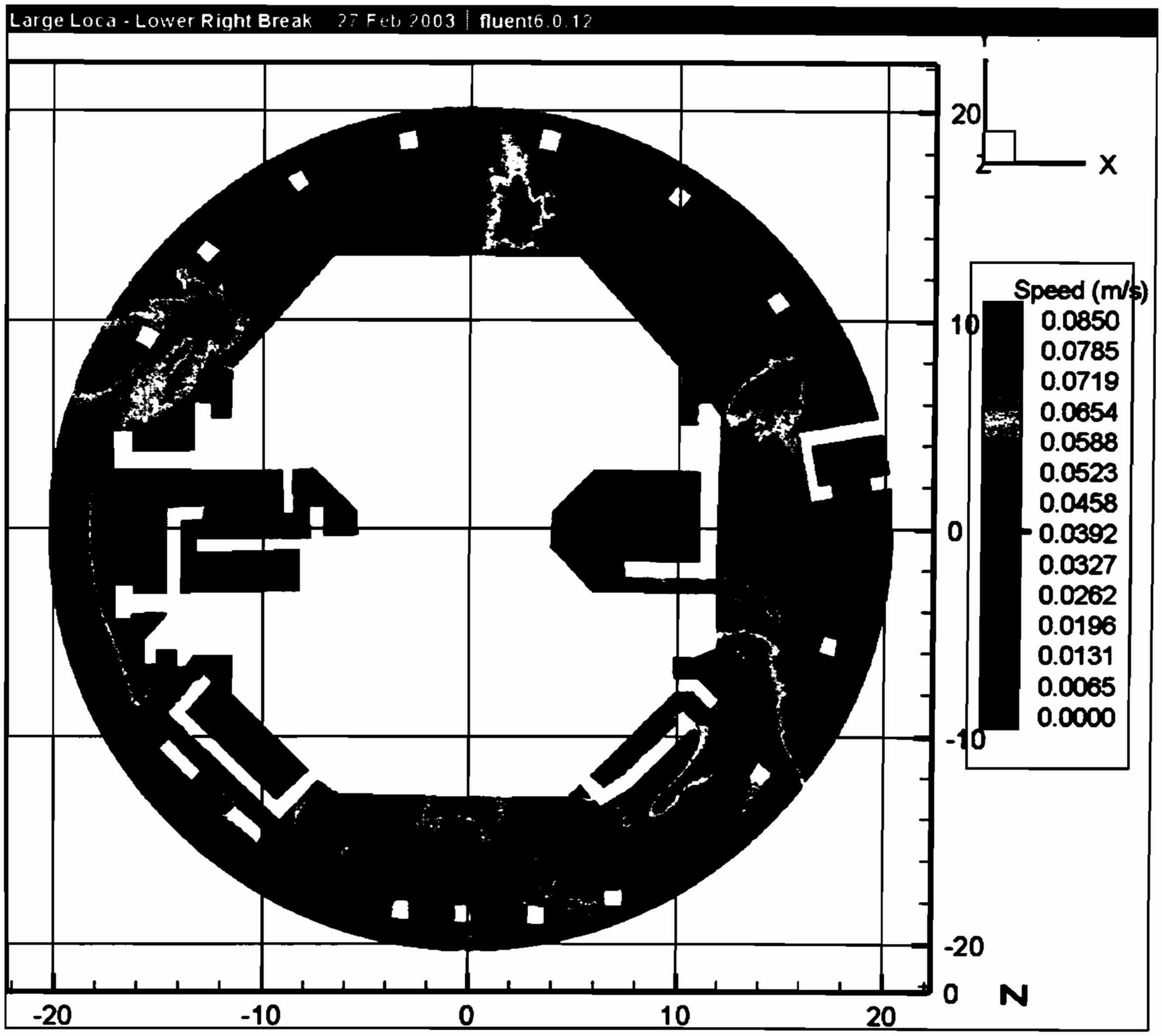


Figure III.2-37. Large LOCA Break Located in the Lower-Right Quadrant. Speeds Greater Than or Equal to the RMI Threshold (0.085 m/s) Are Colored RED.

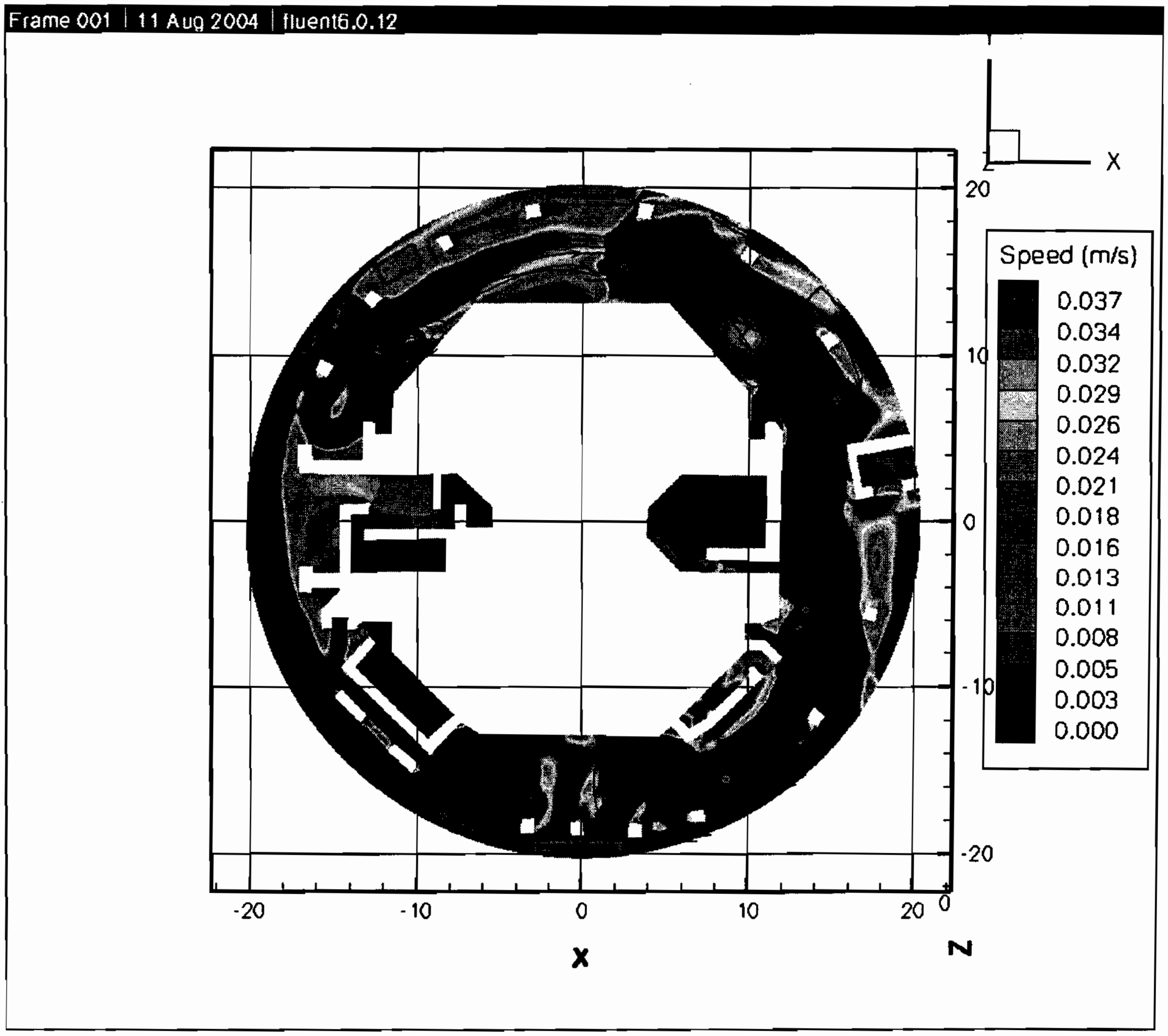


Figure III.2-38. Streamtraces across Two Splash Locations, Coordinates (-12,10) and (5,15), as Shown in the Figure, for a Large LOCA Break Located in the Upper-Left Quadrant. Speeds Greater Than or Equal to the Fiber Threshold (0.037 m/s) Are Colored RED.

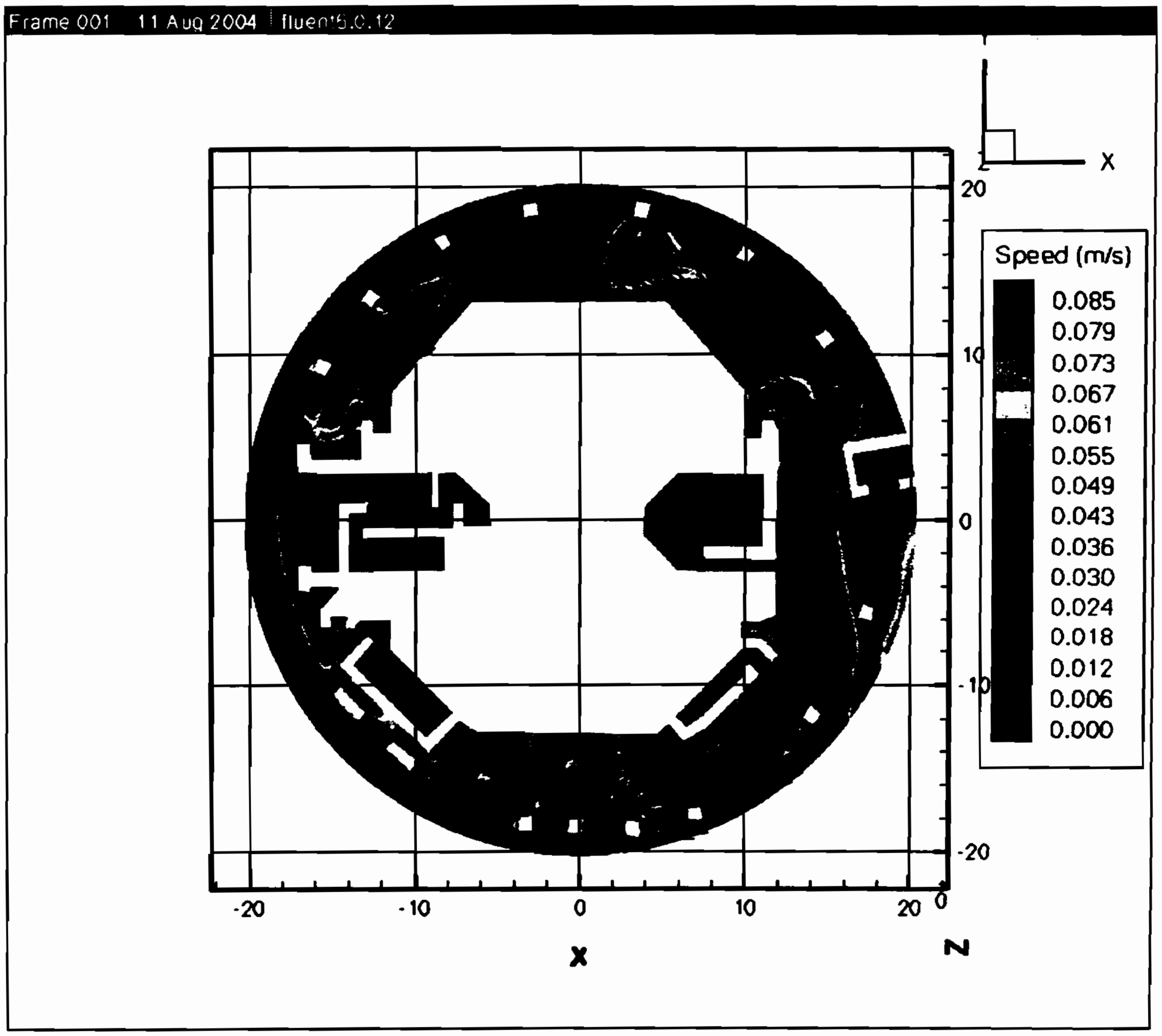


Figure III.2-39. Streamtraces across Two Splash Locations, Coordinates (-12,10) and (5,15), as Shown in the Figure, for a Large LOCA Break Located in the Upper-Left Quadrant. Speeds Greater Than or Equal to the RMI Threshold (0.085 m/s) Are Colored RED.

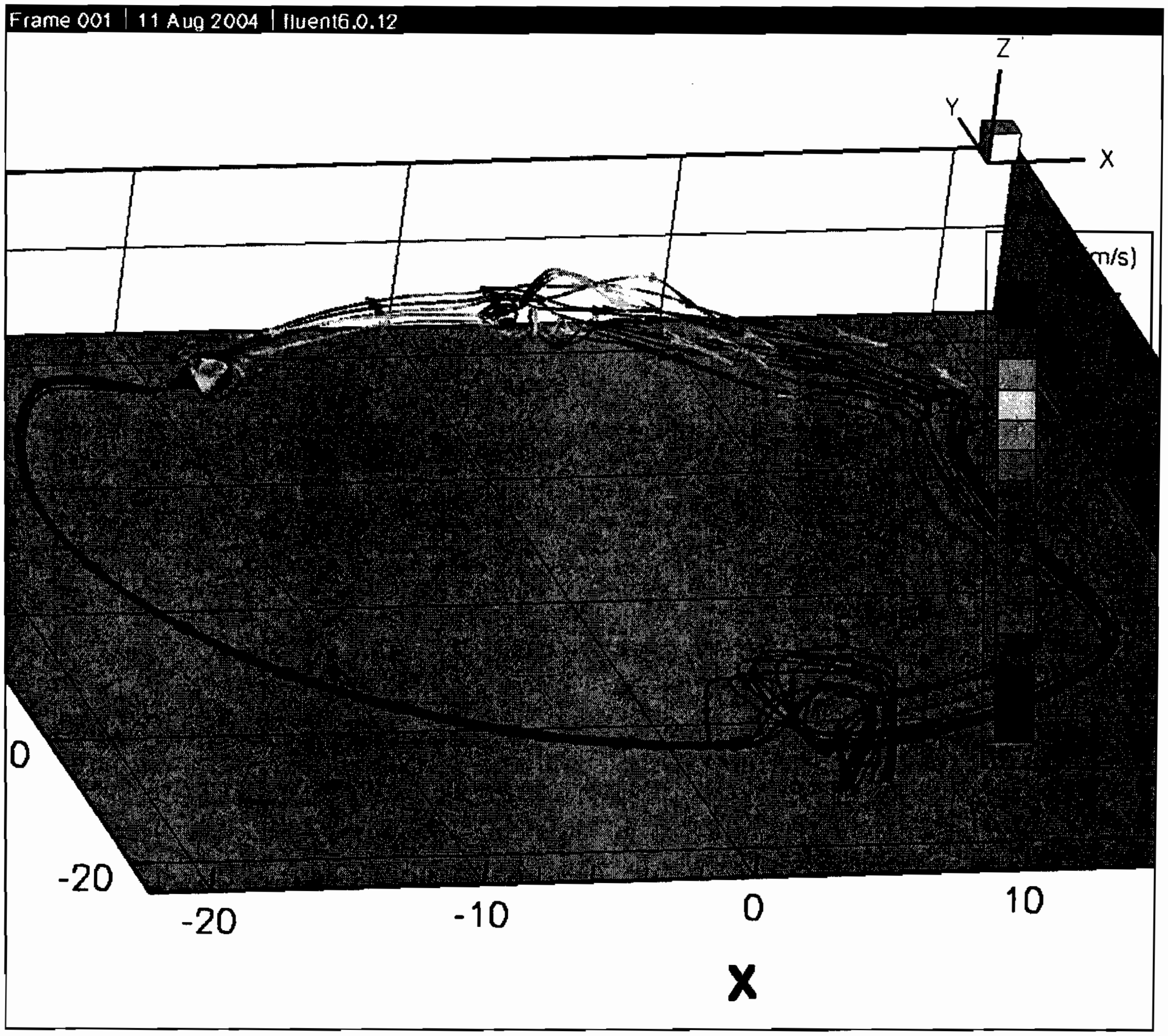


Figure III.2-40. Oblique View of the Streamtraces, as Shown in Figure III.2-38 for the Fiber Threshold Velocity. Traces Are Color Coded to the Local Fluid Velocity. Speeds Greater Than or Equal to the Fiber Threshold (0.037 m/s) Are Colored RED.

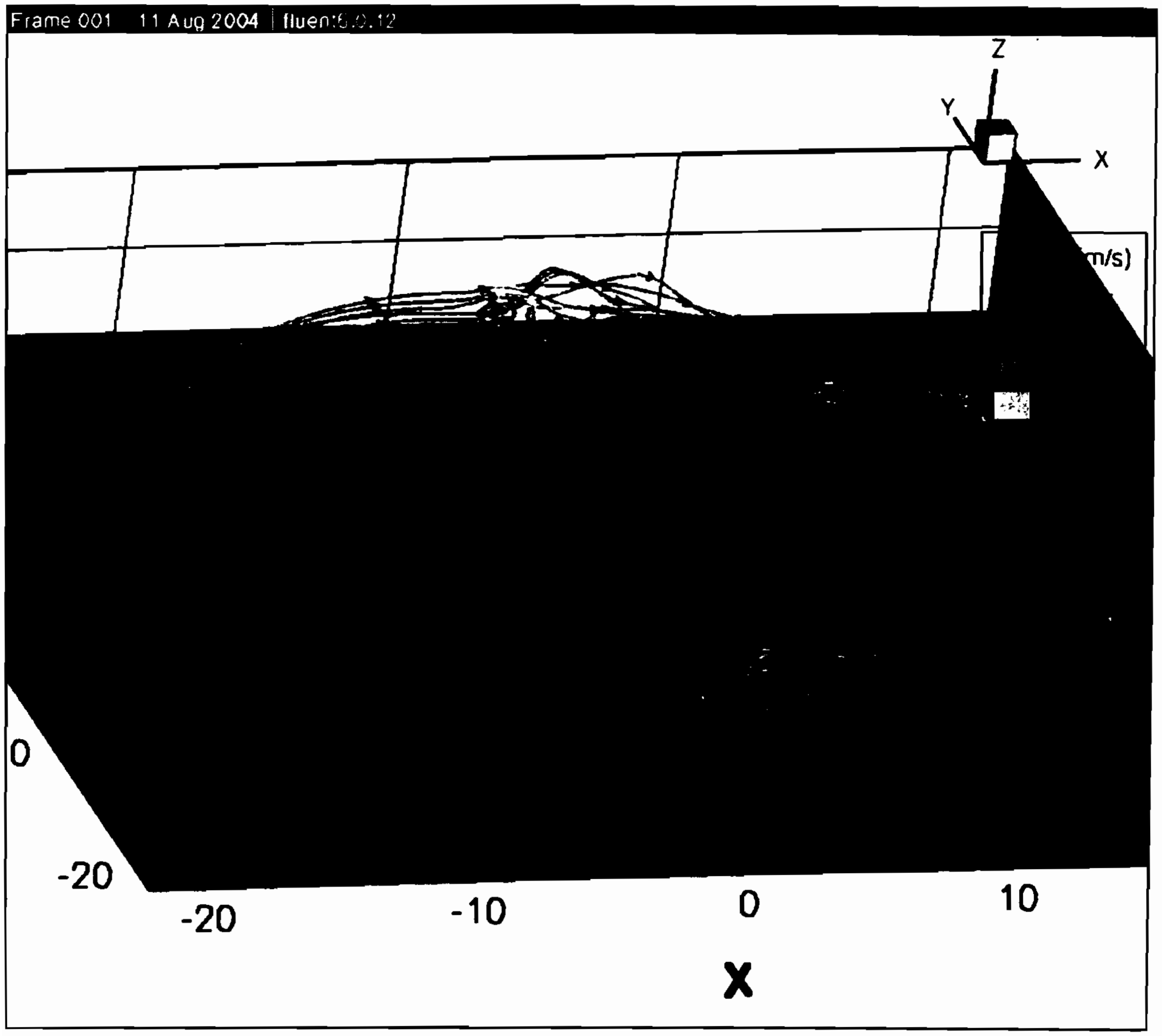


Figure III.2-41. Oblique View of the Streamtraces Shown in Figure III.2-39 for the RMI Threshold Velocity. Traces are Color Coded to the Local Fluid Velocity. Speeds Greater Than or Equal to the RMI Threshold (0.085 m/s) Are Colored RED.

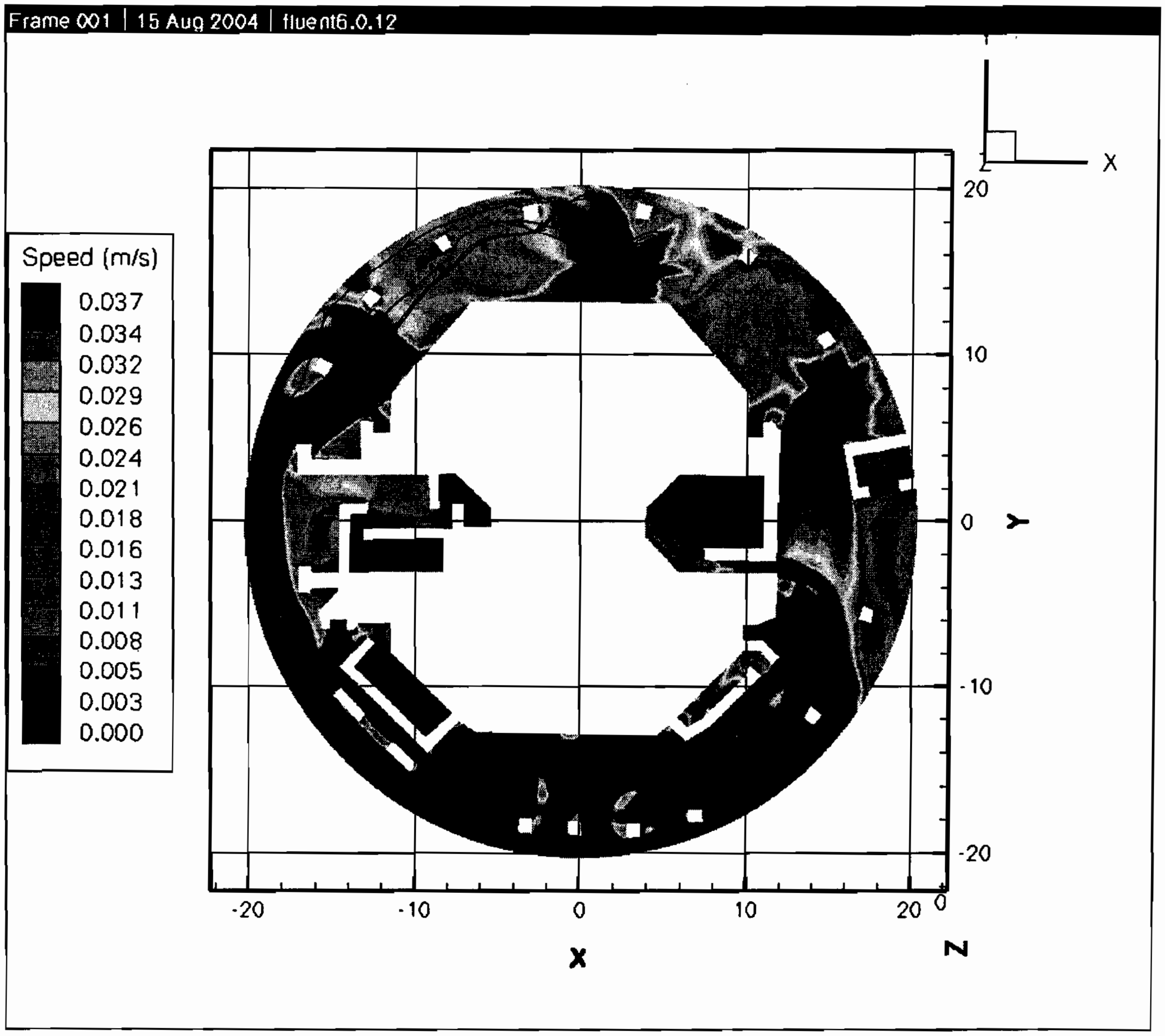


Figure III.2-42. Streamtraces across Two Splash Locations, Coordinates (-12,10) and (5,15) as Shown in the Figure, for a Large LOCA Break Located in the Lower-Right Quadrant. Speeds Greater Than or Equal to the Fiber Threshold (0.037 m/s) Are Colored RED.

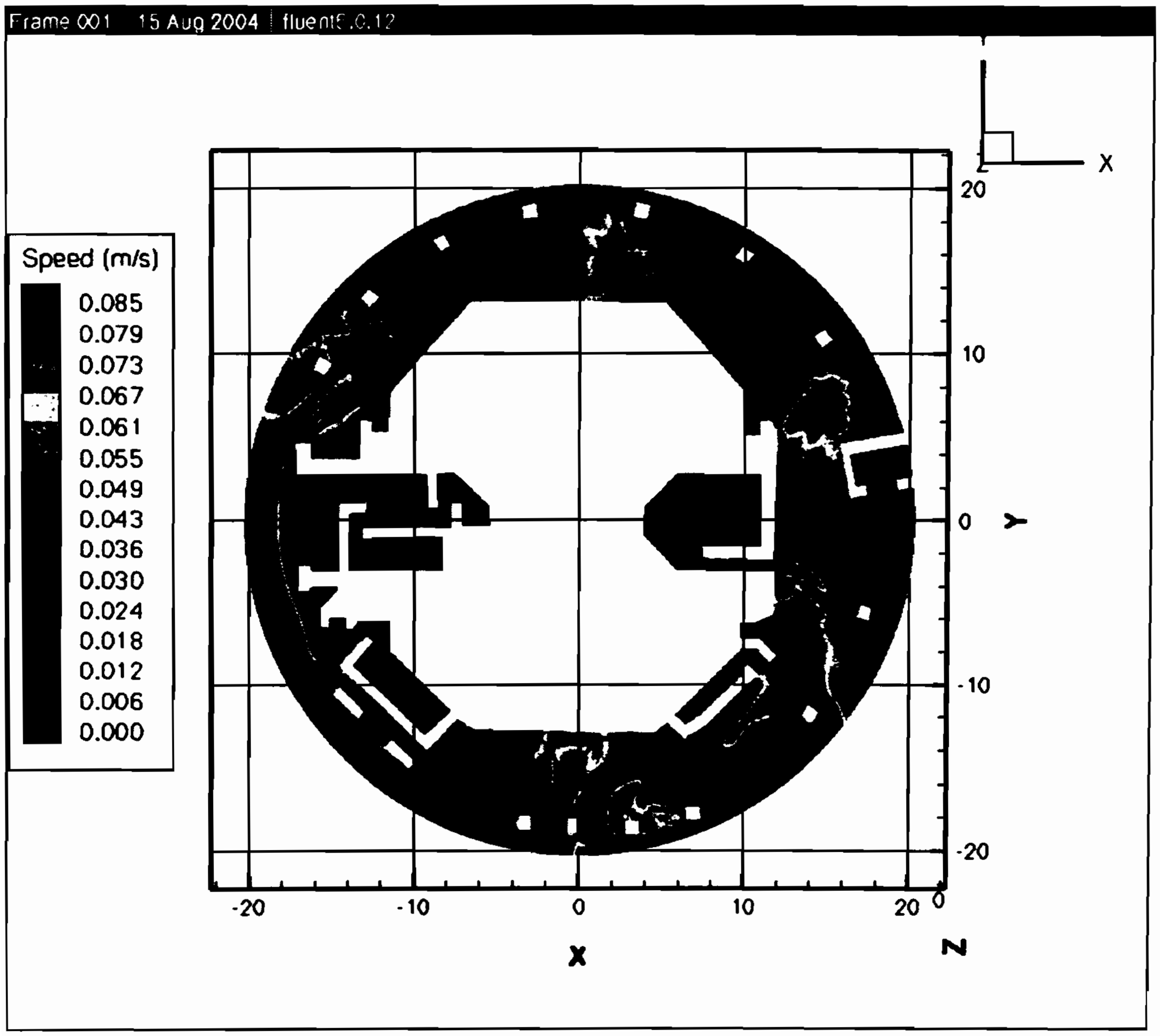


Figure III.2-43. Streamtraces across Two Splash Locations, Coordinates (-12,10) and (5,15), as Shown in the Figure, for a Large LOCA Break Located in the Lower-Right Quadrant. Speeds Greater Than or Equal to the RMI Threshold (0.085 m/s) Are Colored RED.

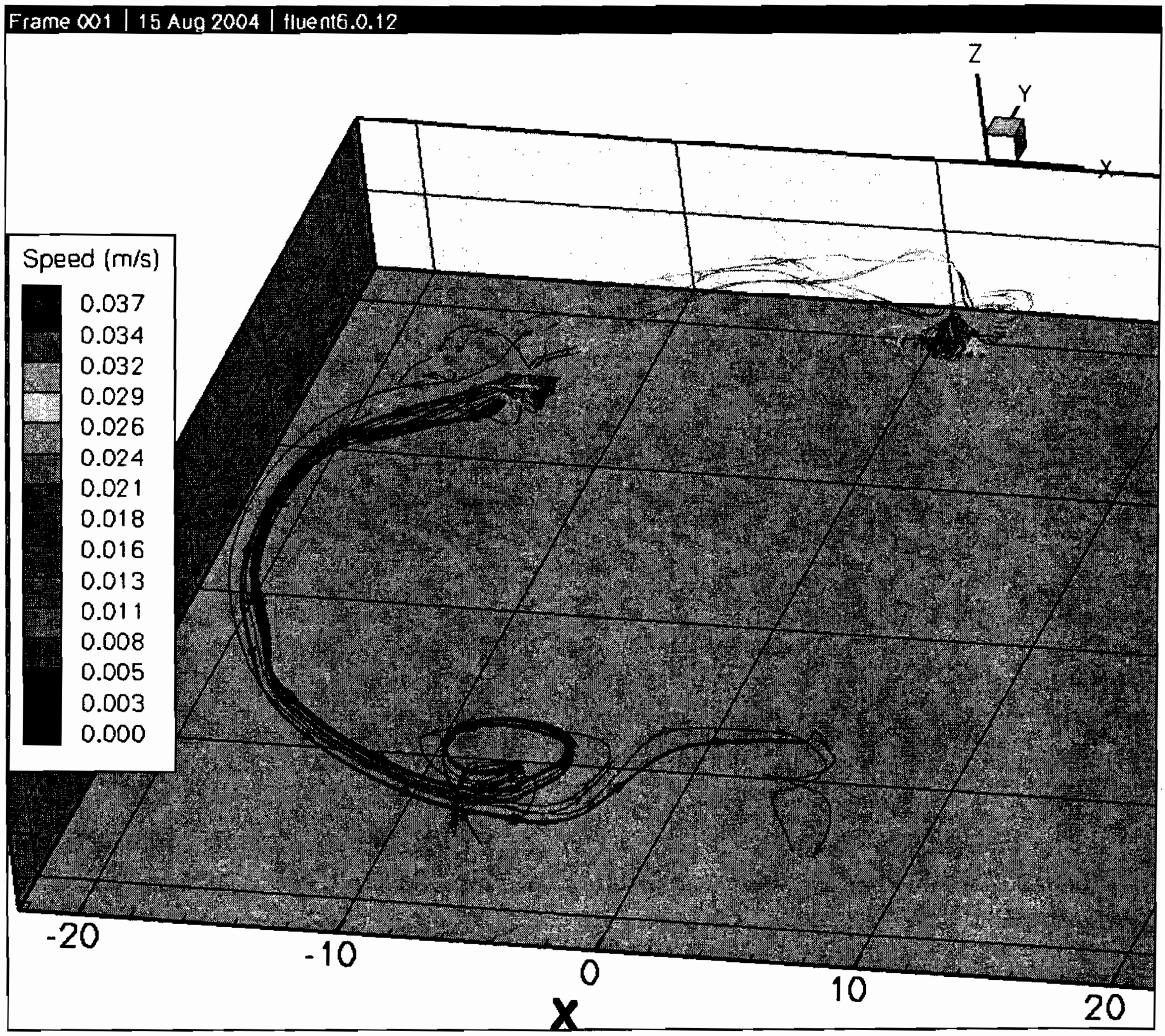


Figure III.2-44. Oblique View of the Streamtraces Shown in Figure III.2-42 for the Fiber Threshold Velocity. Traces Are Color Coded to the Local Fluid Velocity. Speeds Greater Than or Equal to the Fiber Threshold (0.037 m/s) Are Colored RED.

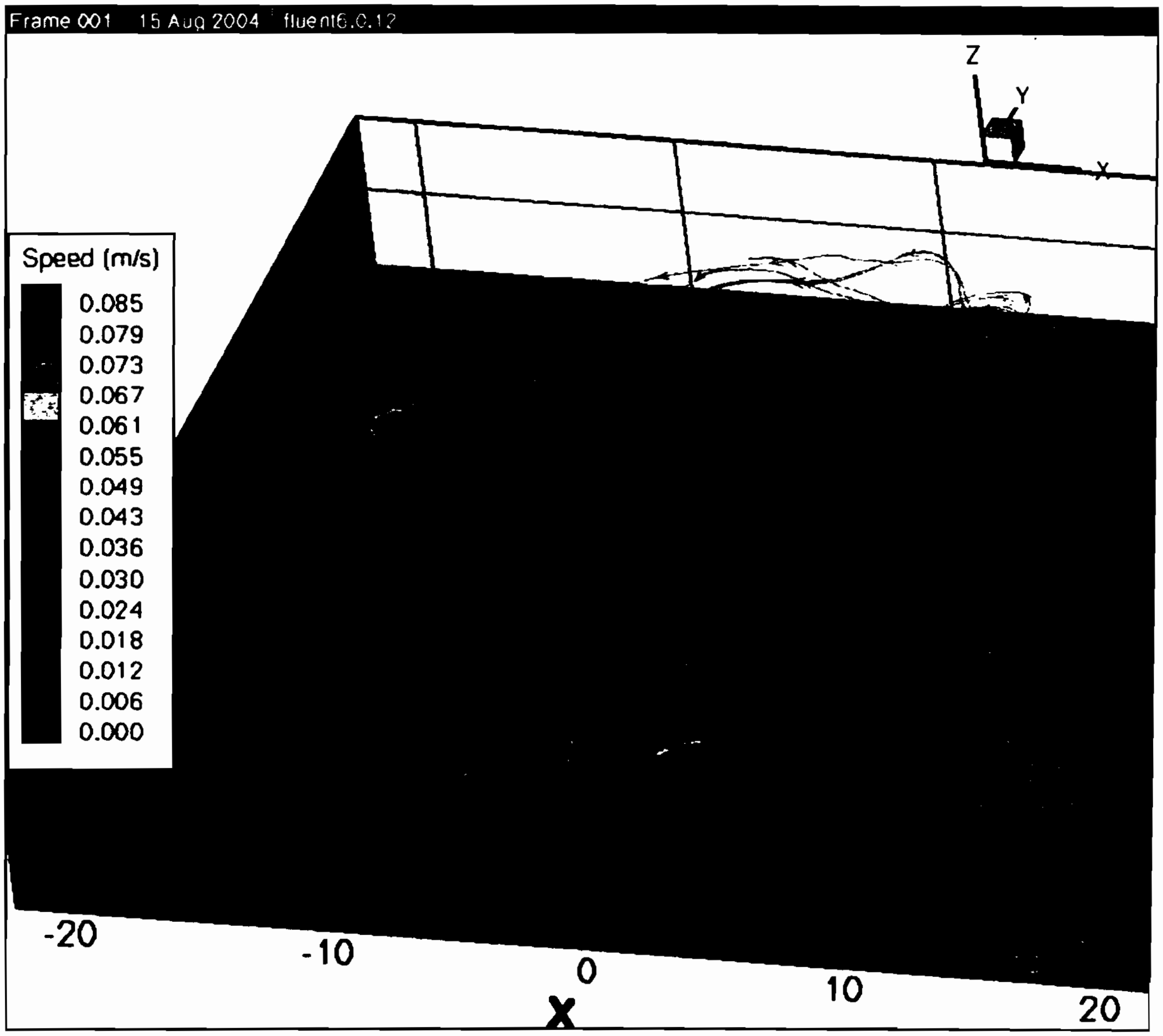


Figure III.2-45. Oblique View of the Streamtraces Shown in Figure III.2-43 for the Fiber Threshold Velocity. Traces Are Color Coded to the Local Fluid Velocity. Speeds Greater Than or Equal to the RMI Threshold (0.085 m/s) Are Colored RED.

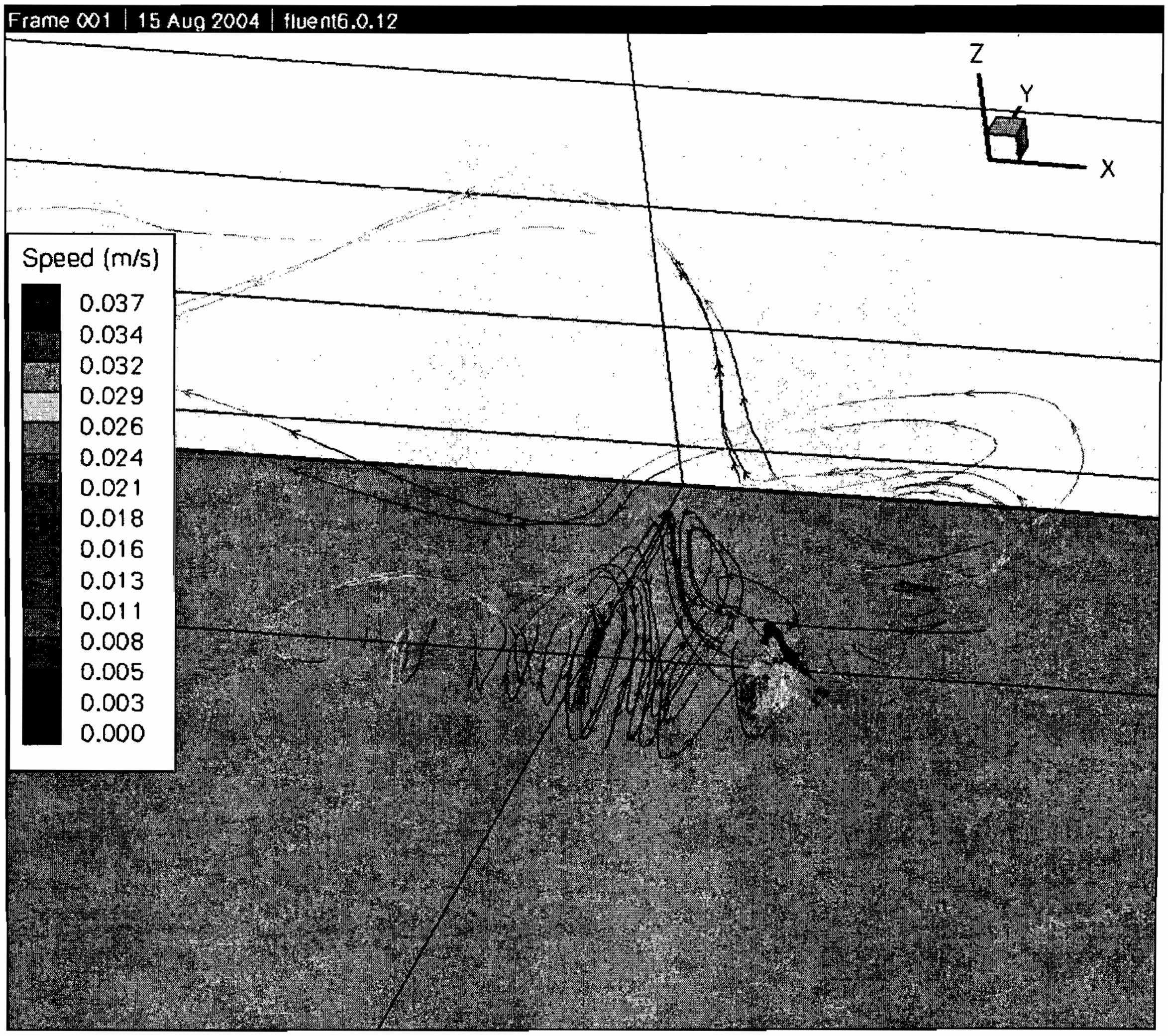


Figure III.2-46. Large LOCA Lower-Right Break, Zoom in at Upper-Right Splash Location Shown in Figures III.2-42 and III.2-43. Traces Are Color Coded to the Local Fluid Velocity. Speeds Greater Than or Equal to the Fiber Threshold (0.037 m/s) Are Colored RED.

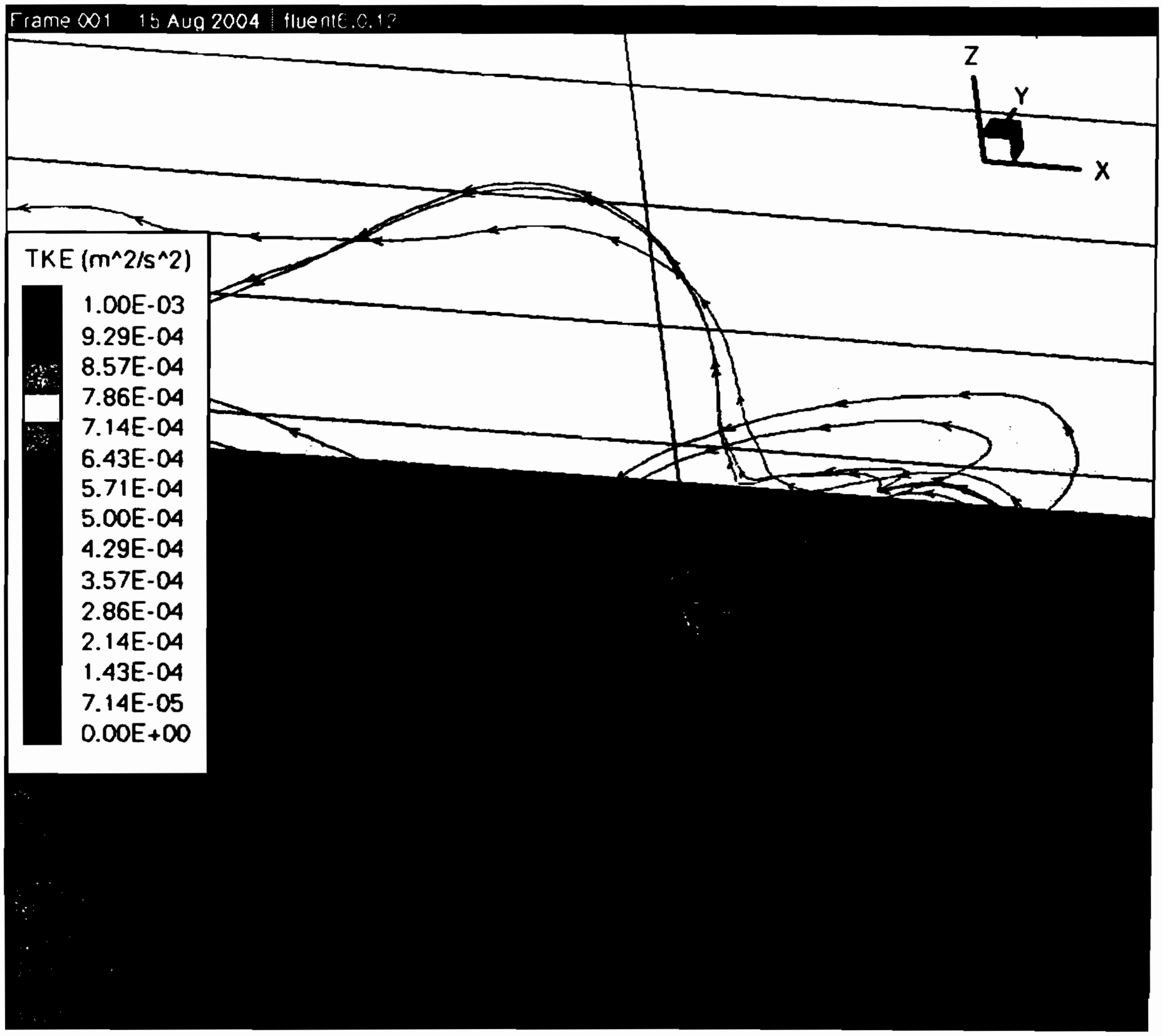


Figure III.2-47. Same as Figure III.2-46, with Streamlines Color Coded by TKE.

III.3 SUMP POOL DEBRIS TRANSPORT

The CFD analyses characterized the flow conditions in the sump for a selection of LOCA accident scenarios. These conditions include flow velocity patterns, pool turbulence, and flow streamlines. The pool velocity and turbulence characteristics determine areas of the pool where debris entrapment may occur. The flow streamlines can be used to determine whether debris entering the pool at a discrete location would be likely to pass through one of the potential entrapment locations. The debris transport process was broken down using a logic chart approach to facilitate the individual transport steps—steps that could be determined analytically, experimentally, or simply judged. The subsequent quantification of the chart then provided an estimate of the overall sump pool debris transport.

III.3.1 Debris Transport Logic Chart Methodology

Key to the evaluation of sump pool debris transport is "when" and "where" the debris enters the pool. The question of when debris enters the pool is basically separated into whether the debris was directly deposited onto the sump floor during the blowdown phase or entered the pool with the subsequent drainage of the containment sprays. To put the timing in perspective, the reactor cavity would likely fill in less than 12 minutes (e.g., a large LOCA break flow rate of 7400 GPM would fill the reactor cavity volume, estimated by the plant to be less than 12,000 ft^3, in less than 12 minutes neglecting the contribution from the containment sprays), and the sump pool should reach a reasonable steady state in ~30 minutes. The entrance location for blowdown-deposited debris is a debris distribution on the floor that likely favors deposition nearer the location of the break. The question of where the debris enters the pool is decomposed into whether the debris is blown onto the break room floor (SG compartment housing the break) or the remainder of the sump floor, which is the lower-level annulus floor. Debris transport into the pool via the spray drainage would enter at the primary drainage locations. The debris transport analysis requires a distribution for where the washdown debris enters the sump pool. The spray drainage analysis in Appendix VI provides a distribution for drainage flows entering the sump pool. The assumption used in these analyses is that the distribution of washdown debris entering the pool mimics that of the spray water distribution for debris deposited outside the break compartment. Note that the blowdown deposition analyses determined substantial debris deposition within the break compartment that would subsequently wash directly to the break compartment floor; this deposition was considered in the debris introduction to the pool. The drainage from the containment sprays drained into the sump pool at many locations including floor drains, stairwells, an equipment hatch, the containment liner, overflow from upper levels into the annular gap, refueling pool drains, spray falling directly into the steam generator compartments and the containment spray trains location at the sump level. To simplify the analysis, the multiple drainage entrance locations into the sump pool were grouped into seven groups around the sump annulus. Figure III.3-1 shows this distribution in an event chart format. One of these charts is applied to each size category of each type of insulation. The distributions in the chart (moving from left to right) are the following:

1. the blowdown transport deposition distribution that splits the total debris among debris deposited in the upper level floors, the break compartment floor, and the remainder of the lower level (sump) floor;
2. the washdown transport distributions of whether the debris deposited in the upper levels would likely transport to the sump pool or remain in the upper levels;
3. the distribution of the locations where debris entrained in the containment spray drainage would enter the sump pool;
4. the distributions associated with sump pool formation debris transport;
5. the distributions associated with pool recirculation debris transport; and
6. the distributions associated with potential debris erosion.

Each transport path is assumed to transport debris to one of three destinations, which include (1) accumulation on the sump screens, (2) debris entrapped within the inactive pools, and (3) debris otherwise entrapped at locations along the transport pathways. The fraction of the debris predicted to accumulate on the screens is then the transport fraction for the size and type of debris. The overall transport fraction by insulation type is obtained by applying the debris-size distributions to the size-specific transport fractions.

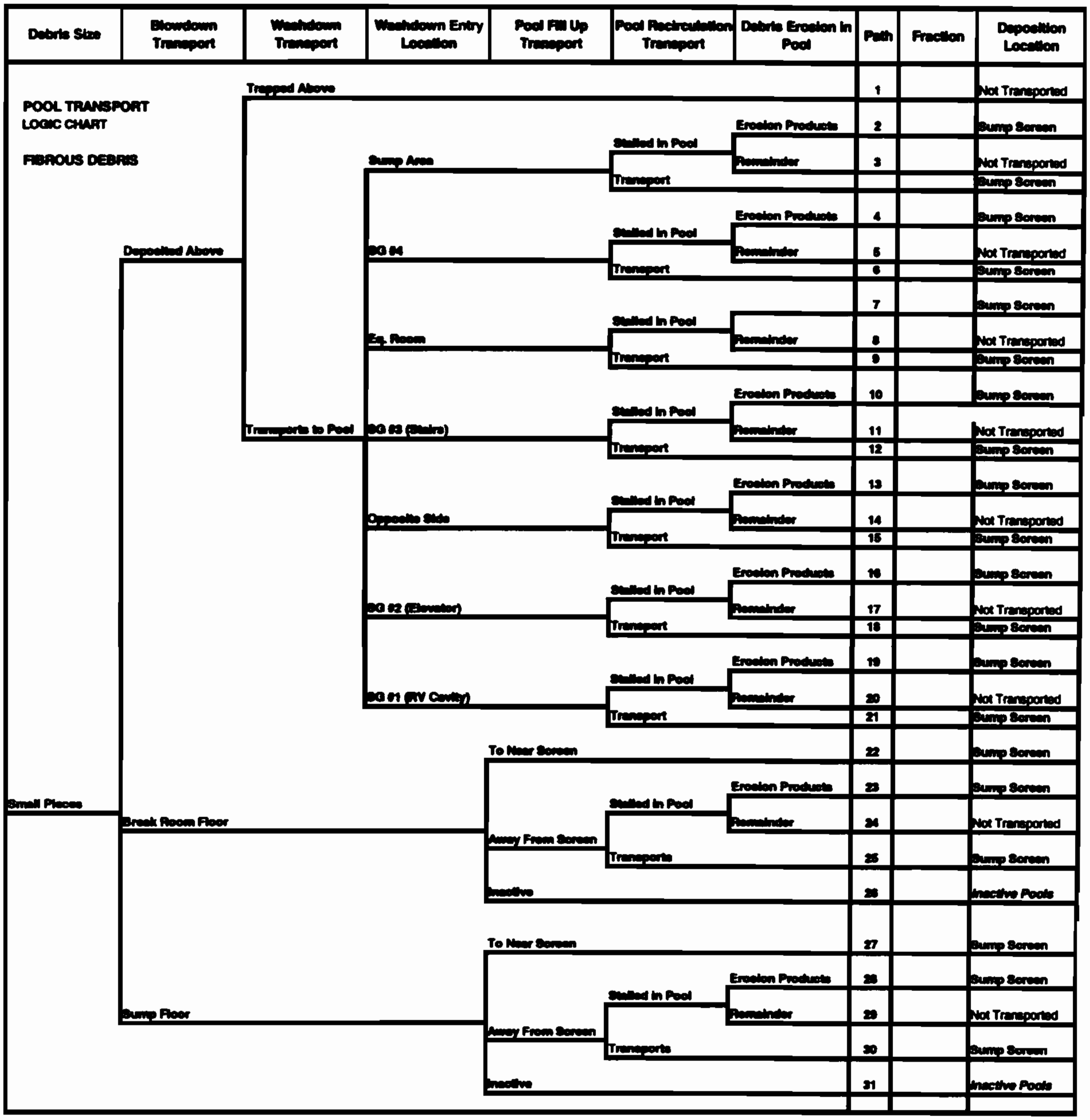


Figure III.3-1. Sump Pool Debris Transport Chart

III.3.2 Blowdown/Washdown Debris Entry into the Sump Pool

The distributions for the blowdown and washdown phases of the transport analysis were obtained from the details of the volunteer blowdown/washdown debris transport analyses documented in Appendix VI; these distributions are shown in Table III.3-1.

The volunteer-plant fibrous debris was categorized as (1) fines, (2) small pieces, (3) large pieces, and (4) intact pieces. The fines and small pieces represent debris capable of passing through a typical grating during blowdown. The fines are generally the individual fibers that remain suspended in the sump pool, whereas the small-piece fibrous debris typically would readily sink to the pool floor in hot water. Thus, the fines and small pieces must be evaluated differently. The large-piece and intact-piece debris represents debris too large to pass through a grating, which is a process fundamental to blowdown debris transport evaluations. The difference between the large and intact piece debris is whether the fibrous insulation continues to be protected by covering material. With large-piece debris, the fibrous insulation is subject to erosion, whereas the intact-piece debris insulation is not. Another distinguishing difference is that the covering materials on the intact debris, which include nearly intact blankets, are more likely to snag onto structures, including gratings during blowdown transport such that it is less likely to fall back to a floor or wash off with the sprays. The guidance-report (GR) baseline small-fines category corresponds to the combination of the fines and small-piece debris in the volunteer-plant analyses, and the GR large-piece debris corresponds to the large- and intact-piece debris in the volunteer-plant analyses.

Table III.3-1. Blowdown/Washdown Debris Transport Fractions

Debris Size and Type	Debris Transport Fractions				
	Blowdown Transport			Washdown Transport	
	Deposited in Upper Levels	Deposited on Break Room Floor	Deposited on Sump Floor	Remains Trapped Above	Transports to Sump Pool
Fibrous Debris					
Fines	0.92	0.05	0.03	0.07	0.93
Small Pieces	0.92	0.05	0.03	0.37	0.63
Large Pieces	0.57	0.39	0.04	0.81	0.19
Intact Pieces	0.69	0.30	0.01	0.78	0.22
RMI Debris					
< 2-in.	0.47	0.50	0.03	0.38	0.62
2 to 6-in.	0.35	0.61	0.04	0.69	0.31
> 6-in.	0.22	0.77	0.01	0.68	0.32

The volunteer-plant RMI debris was categorized as (1) debris pieces less than 2-inches, (2) pieces between 2- and 6 -inch in size, and (3) pieces greater than 6 inches in size. The GR RMI size groups were subdivided at 4-inch rather than the 2- and 6-inch used for the volunteer plant analysis. However, the combination of the volunteer-plant analysis categories less than 6-inch is a reasonable representation of the GR small-fines category, leaving the pieces larger than 6-inch to represent the large-piece debris.

The debris washing down from the upper levels was assumed to enter the sump pool with the same distribution as the spray drainage. However, blowdown debris that was preferentially deposited in the steam generator (SG) compartment where the break (SG1) occurred and its adjacent SG compartment (SG4) would wash directly to the floors of these compartments, regardless of the spray drainage fractions. For the volunteer plant, the spray drainage distribution, as shown in Table III.3-2, was obtained from the spray drainage analysis documented in Appendix VI. The location distributions for debris washing down from the upper levels are provided by debris size category in Table III.3.3 and III.3.4 for fibrous and RMI debris, respectively. Because the larger debris was preferentially trapped in SG1 and SG4, these washdown location fractions are larger.

Table III.3-2. Spray Drainage Distribution into the Sump Pool

No.	Location in Annular Sump	Spray Drainage Water Sources	Drainage Fraction
1	Annulus Section Containing Recirculation Sumps	Floor drains and annular gap sources	0.14
2	Vicinity of SG4 Access (Steam Generator Adjoining Break Room)	SG4 personnel access doorway and liner flow. Includes flow from a 6-in. refueling pool drain.	0.08
3	Vicinity of Interior Equipment Room Access (~90° from Sumps)	Refueling pool water drains into equipment room below refueling pools, then exits doorway into sump and liner flow.	0.06
4	Vicinity of SG3 Access	SG3 personnel access doorway, annular gap sources, and stairwell. Includes flow from a 6-in. refueling pool drain.	0.18
5	Annulus Section Directly Opposite Recirculation Sumps	Floor drains and annular gap sources.	0.09
6	Vicinity of SG2 Access	SG2 personnel access doorway, floor drains, upper-level equipment hatch, annular gap sources, and stairwell. Includes flow from a 6-in. refueling pool drain.	0.25
7	Vicinity of SG1 Access (Compartment with Break)	SG1 personnel access doorway, floor drains, and annular gap sources. Includes flow from a 6-in. refueling pool drain.	0.20

Table III.3-3. Fibrous Debris Entrance Distributions to Sump Pool

No.	Location in Annular Sump	Drainage Fraction	Fines Debris	Small-Piece Debris	Large-Piece Debris	Intact-Piece Debris
1	Sumps	0.14	0.09	0.09	0.01	0.01
2	SG4	0.08	0.17	0.17	0.28	0.22
3	Eq. Room	0.06	0.04	0.04	0	0
4	SG3	0.18	0.12	0.12	0.07	0.07
5	Opposite	0.09	0.06	0.06	0.01	0.01
6	SG2	0.25	0.16	0.16	0.07	0.07
7	SG1	0.20	0.36	0.36	0.56	0.62

Table III.3-4. RMI Debris Entrance Distributions to Sump Pool

No.	Location in Annular Sump	Drainage Fraction	<2-in. Debris	2- to 6-in. Debris	>6-in. Debris
1	Sumps	0.14	0.06	0.01	0.01
2	SG4	0.08	0.24	0.28	0.22
3	Eq. Room	0.06	0.02	0	0
4	SG3	0.18	0.06	0.07	0.07
5	Opposite	0.09	0.04	0.01	0.01
6	SG2	0.25	0.09	0.07	0.07
7	SG1	0.20	0.49	0.56	0.62

III.3.3 Sump Pool Debris Transport Estimates

Debris transport in the sump pool was separated into the following three phases: (1) transport of floor deposited debris during the formation (fill-up) of the sump pool, (2) debris transport in an established sump during recirculation mode, and (3) long-term erosion of exposed fibrous debris in the sump pool.

III.3.3.1 Pool Formation Debris Transport

Based on observations taken during the integrated debris transport tests [NUREG/CR-6773], the primary driver for moving debris during pool formation, especially for the large debris, is the sheeting flow as the initial water from the break spreads across the sump floor. Debris initially deposited on the floor is pushed along with the wave front. As such, the movement of the debris has significant momentum that can carry the debris past the openings into interior spaces. Once the water depth becomes significant, further transport occurs due to the drag forces of the flow of water, and for larger debris that transport becomes substantially less dynamic than the sheeting flow transport. Individual fibers will move as suspended debris following the water flow.

In the volunteer plant, a majority of the debris initially deposited on the floor of the compartment containing the break (SG compartment 1 in this evaluation) would likely transport from that compartment onto the annular sump floor through either of the personnel access door for SG 1 or the door for SG 4. Because the break is in SG 1, considerably more flow would exit the door to SG 1 than to SG 4. In the scenario evaluated herein, the larger portion of the break room flow and therefore the debris (perhaps 75%) would flow through the personnel access door into the annulus on the side nearer the access for the reactor cavity (the flow distribution assumption was discussed in Section III.2.1). A smaller portion of the debris would exit the SG compartment through the access door into SG compartment 2. In the volunteer plant, nearly all of the essentially inactive pool is the water below the sump floor in the reactor cavity. All other quiescent regions would have sufficient water circulation that suspended fibers over time would circulate from those regions. When debris exits a SG compartment through a personnel access door due to the initial sheeting flow, the flow splits, with part going toward the recirculation sumps and part going in the opposite direction. In the scenario analyzed, the part going away from the sump screens flowed past the narrow passageway into the room leading to the reactor cavity access hatch. For debris to follow water into this passageway, it must essentially make a 90° bend in a short distance. Therefore, it must be concluded that only a small fraction of debris moving with the dynamic wave front, especially larger debris, will make the 90° turn into the reactor cavity passageway.

With these concepts in mind, the pool transport distributions were judged as shown in Table III.3-4. Starting with the fines, it is assumed that 75% of the flow exits the SG1 compartment on the reactor cavity side, then that 60% of that flows in the direction of the reactor cavity, then that 50% of the flow makes the turn into reactor cavity passageway, which indicates that perhaps 25% of the fines initially on the break room floor goes into the reactor cavity on initial formation of the pool. Because these fibers are suspended, the remaining pool formation could increase this number to, for example, a conservative 40%. Then, the remaining amount is split 50%–50%, as toward the recirculation sump and away from the sump. With each fibrous debris category of increasing size, the fraction into the reactor cavity is decreased somewhat, with the even split maintained between the "toward" and "away" from the screen. With the heavier metallic debris, even the smaller pieces would transport less readily than the fiber pieces.

For debris initially deposited on the annular sump floor, a significant fraction of this debris could be located such that it would not be greatly affected by flow from the break compartment to the reactor cavity because the exit from the break compartment is near the entrance to the reactor cavity. However, larger debris deposition would also likely be preferential near the break compartment door. For lack of better justifications, the same distributions judged for debris initially deposited on the break room floor are assumed for debris initially deposited on the annular sump floor. In any case, only a few percent of the total debris is estimated to be deposited on the annular sump floor due to the relatively small doorway areas as compared with the upward area of the SG compartments.

Table III.3-4. Pool Formation Debris Transport Distributions

Debris Size and Type	**Pool Formation Debris Transport Distributions**					
	Floor of Break Room			**Floor of Sump Pool**		
	Toward Screen	**Away from Screen**	**Into Inactive Pools**	**Toward Screen**	**Away from Screen**	**Into Inactive Pools**
Fibrous Debris						
Fines	0.30	0.30	0.40	0.30	0.30	0.40
Small Pieces	0.35	0.35	0.30	0.35	0.35	0.30
Large Pieces	0.40	0.40	0.20	0.40	0.40	0.20
Intact Pieces	0.40	0.40	0.20	0.40	0.40	0.20
RMI Debris						
<2 in.	0.35	0.35	0.30	0.35	0.35	0.30
2 to 6 in.	0.40	0.40	0.20	0.40	0.40	0.20
>6 in.	0.50	0.50	0.00	0.50	0.50	0.00

III.3.3.2 Recirculation Pool Debris Transport

Important aspects of sump pool debris transport were observed during the integrated debris transport tests [NUREG/CR-6773]. For low-density fiberglass debris, the fines (e.g., individual fibers) remain suspended and move with the flow of water, whereas the debris pieces of significant size readily saturate with water at the water temperatures typical of LOCA accidents and then sink to the pool floor, where further transport depends on the flow velocity and turbulence near the floor. For RMI debris, all debris sinks to the floor of the pool, with the occasional exception of a piece of debris that encapsulates an air pocket, keeping that piece buoyant.

The CFD analyses provide realistic descriptions of the floor-level flow conditions, which were described in Section III.2 as contours established so that the velocities higher than the experimental measured threshold are clearly indicated. The velocity contours illustrate the portion of the pool where debris would most likely readily move with the flow. In addition to velocity contours, the streamline plots provide a reasonable connecting pathway whereby a piece of debris would likely travel from its original location in the pool to the recirculation sumps. If a transport pathway passes through a slower portion of the pool, then debris moving along that pathway could stall and not transport to the recirculation sump. Otherwise, the transport is very likely.

The effects of pool turbulence are more difficult to quantify. Test observations have shown the occasional reentrainment of debris once stalled in relatively quiescent water. Water within quiescent regions typically tends to rotate, sending debris into the center of the vortex, where it becomes semi-trapped. However, that occasional pulsation can kick a piece of debris out of the vortex and back into the main stream. Although this behavior cannot be reasonably quantified, transport estimates should be enhanced to consider these effects.

A detailed transport analysis using the CFD predicted flow contours and flow streamlines would subdivide the sump pool floor into relatively fine subdivisions, where each subdivision would have a source term for debris depositing onto the pool floor at that location. Then the transport of the debris from each specific subdivision would be independently evaluated using a streamline generated from that subdivision to the recirculation sumps to illustrate where that debris would likely reside after movement ceases. Quantification of all the subdivision transport results would provide an overall sump pool transport fraction for each debris category. The transport results should then be adjusted to account for pool turbulence effects on debris, i.e., the threshold transport tumbling velocities reported in NUREG/CR-6772 were measured in very uniform and turbulence-dampened flows but turbulence is capable of moving debris where bulk flow will not. One method of accounting for turbulence effects would be to decrease the threshold velocities for transport.

In this analysis, the above detailed model description was simplified to only seven subdivisions for the sump floor. Even then, the available CFD streamlines did not form a complete set. Therefore, the individual pool transport fractions used to populate the transport charts were basically engineering judgments made while viewing the velocity profiles. The individual transport estimates are provided in Table III.3-5. The CFD flow velocity contours maps used to make these judgments are shown in Figures III.2-33 and III.2-37 for fibrous and RMI debris, respectively. A sampling of corresponding flow streamline plots are shown in Figures III.2-42 and III.2-43, for fibrous and RMI debris, respectively. The transport fractions range from 100% transport for the suspended fibers and debris located nearer the recirculation sumps to 0% transport for the largest debris located on the opposite side of the containment.

III.3.3.3 Sump Pool Debris Erosion

The only source of data for the erosion of fibrous debris in a sump pool was the integrated debris transport tests documented in NUREG/CR-6773. Four longer-term tests (3- to 5-hour durations) were conducted in this test program where debris accumulation on the simulated sump screen was collected after every 30 minutes.

Three sources of fibrous debris contributed to this accumulation: (1) small-piece debris tumbling or sliding along the floor, (2) suspended fibers initially introduced into the tank, and (3) fibers that had eroded from the small-piece debris residing on the floor of the tank. Late into these tests, most of the small-piece debris had already either transported to the screen or had come to relative rest in some quiescent location on the tank floor; therefore, its contribution should have been minimal near the end of the tests. Also, late in the tests, water recirculation should have substantially reduced the initially suspended fibers so that continued accumulation would fall off quite noticeably. Note that sufficient time had elapsed in each test for the water in the tank to be replaced (tank water volume divided by the simulated break flow) from 19 to 46 times during the course of the test. Because the continued accumulation tended to hold at a somewhat sustainable rate, it is likely that continued erosion was supporting the continued debris accumulation.

Table III.3-5. Recirculation Pool (Steady-State) Debris Transport Fractions

Location Where Debris Enters Sump Pool	**Fraction of Debris Transported to Sump Screen**						
	Fibrous Debris				**RMI Debris**		
	Fines	**Small Pieces**	**Large Pieces**	**Intact Pieces**	**<2 in.**	**2 to 6 in.**	**>6 in.**
Debris Entering with Annular Sump Pool by Containment Spray Drainage (Debris Assumed to Enter Established Sump Pool)							
Annulus Section Containing Recirculation Sumps	1	1	1	1	1	1	1
Vicinity of SG4 Access (SG Adjoining Break Room)	1	1	1	1	1	1	1
Vicinity of Interior Equipment Room Access (~90° from Sumps)	1	1	1	1	1	1	1
Vicinity of SG3 Access (Includes Inter-Level Stairwell)	1	0.5	0.4	0.3	0.3	0.2	0.1
Annulus Section Directly Opposite Recirculation Sumps	1	0.2	0.1	0	0.1	0	0

Vicinity of SG2 Access (Includes Inter-Level Stairwell and Hatch)	1	0.5	0.4	0.3		0.3	0.2	0.1
Vicinity of SG1 Access (Compartment with Break, Includes Multiple Floor Drains)	1	0.7	0.6	0.5		0.5	0.4	0.3
Debris Directly Blowdown Deposited onto Sump Floor but Subsequently Relocated Away from Recirculation Sumps during Pool Formation (Section III.3.3.1)								
Initially on Break Room Floor, Relocated Away from Recirculation Sumps	1	0.3	0.2	0.1		0.2	0.1	0
Initially Spread Around Annular Sump Floor, Relocated Away from Recirculation Sumps	1	0.3	0.2	0.1		0.2	0.1	0

Table III.3-6 shows the end of test debris accumulation rates for these longer-term tests. Although these tests were run several hours, as indicated in the table, the tests were of short duration compared with LOCA long-term recirculation times. One of the four tests was conducted with a shallower pool of 9-in. depth compared with the usual depth of 16 in. Note that the accumulation was about eight times more rapid for the shallow pool test than for the deeper tests. Also note that during the shallow pool test, the water recirculation in terms of water replacements (46) was significantly more frequent for the 9-in. test than for the 16-in. tests; thus, the initial suspended debris would have been more readily filtered from the tank. Therefore, most of the longer-term debris accumulation should have been due to the continued erosion of fibrous debris in the tank. Further, the erosion rate was greater in the shallow depth pool, which can most likely be attributed to the greater turbulence in the shallow pool relative to the deeper pools.

Table III.3-6. Late-Term Debris Accumulation in Integrated Debris Transport Tests

Test ID	Pool Depth (in.)	Test Duration (Hours)	Accumulation Rate near the End of the Test (Percent of Debris in Tank/hr)	Approximate Number of Water Replacements During the Test
LT1	16	4	0.4	26
LT2	9	4	2	46
LT3	16	3	0.3	19
LT4	16	5	0.3	32

In conclusion, the only applicable test data for long-term debris erosion in a sump pool strongly indicate a sustainable rate of erosion that is affected by the relative turbulence in the pool. It should also be noted that the small-piece debris residing on the floor of the pool, late term, was generally found in quiescent locations, not necessarily directly under the simulated break flow. It might also be noted that the turbulence associated with the spray drainage was not simulated. Because the 16-in. depth more closely resembles the fully established volunteer-plant pool, the erosion rate of 0.3 percent of the current tank debris/hour is adapted for this analysis.

In the debris transport charts, the overall fraction of debris on the sump floor that erodes into fines in required. Using the long-term recirculation mission time of 30 days, analysis indicates that nearly 90% of the initial debris mass would become eroded if this erosion rate remained constant throughout the 30 days. This calculation took into account the steadily decreasing mass of debris the pool using the following equation.

$$f_{eroded} = 1 - (1 - rate)^{Number\ of Hours}$$

Therefore, in the debris transport charts, 90% of the small- and large-piece debris predicted to reside on the sump floor is assumed to erode into suspended fibers unless the debris is still enclosed in a protective cover.

There are substantial sources of uncertainty with this calculation:

1. The durations of the integral debris transport tests were 3 to 5 hours. This leaves the question "Does the erosion rate taper off with time?" In addition, it is not certain that all of the end-of-test debris accumulation was due to erosion products.

2. The test results include the usual variances in test data, such as flow and depth control, and debris collection.

3. Although the test series was designed to approximate the flow and turbulence characteristics of the volunteer plant sump pool, the tank characteristics may have been significantly different than what might occur at the plant. The difference in the erosion rates between the 9 and 16-inch pool depths in the integrated tests clearly illustrate the effect of pool turbulence onf fibrous debris erosion.

4. The geometry of the volunteer plant sump pool is larger and more complex than the test tank used in the integrated tests.

5. Large piece debris was not tested in the long term tests.

The 90% debris eroded value is used for both the small and large piece debris, despite the uncertainties. With such limited data, the use of 90% is necessary to ensure conservatism in the overall transport results. It may be that this number can be relaxed once better erosion data is available.

III.3.4 Quantification Results

The blowdown/washdown/pool transport estimates presented in Sections III.3.2 and III.3.3 were entered into debris transport charts shown generically in Figure III.3-1 and quantified to obtain overall transport fractions. A separate chart was created for each size category and for each type of debris. Figures III.3-2, III.3-3, III.3-4, and III.3-5 illustrate the transport processes for fibrous debris categories of fines, small pieces, large pieces, and intact pieces, respectively. Figures III.3-6, III.3-7, and III.3-8 illustrate the transport processes for RMI debris categories of pieces <2 in., 2 to 6 in., and >6 in., respectively.

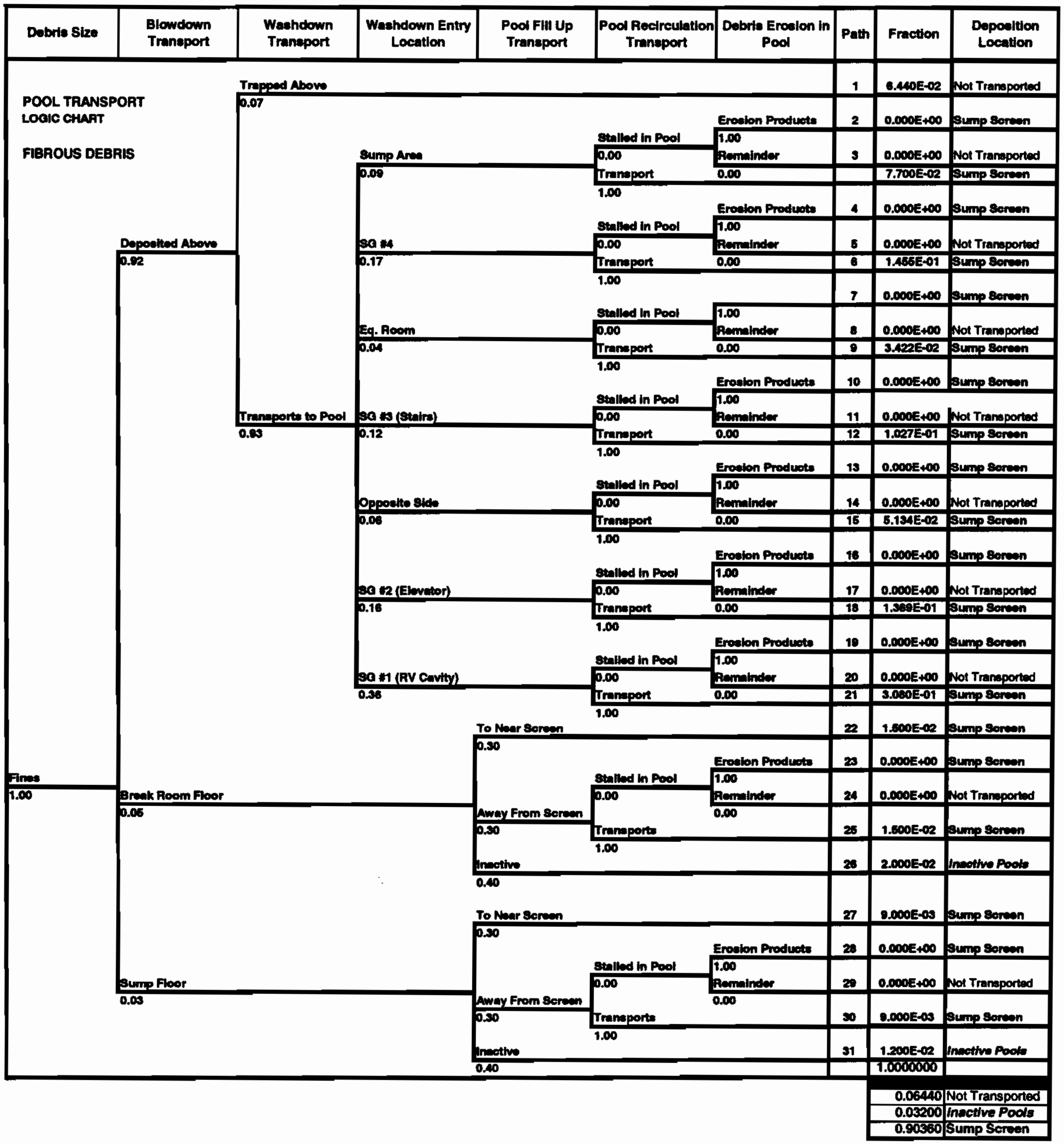


Figure III.3-2. Sump Pool Debris Transport Chart for Fine Fibrous Debris.

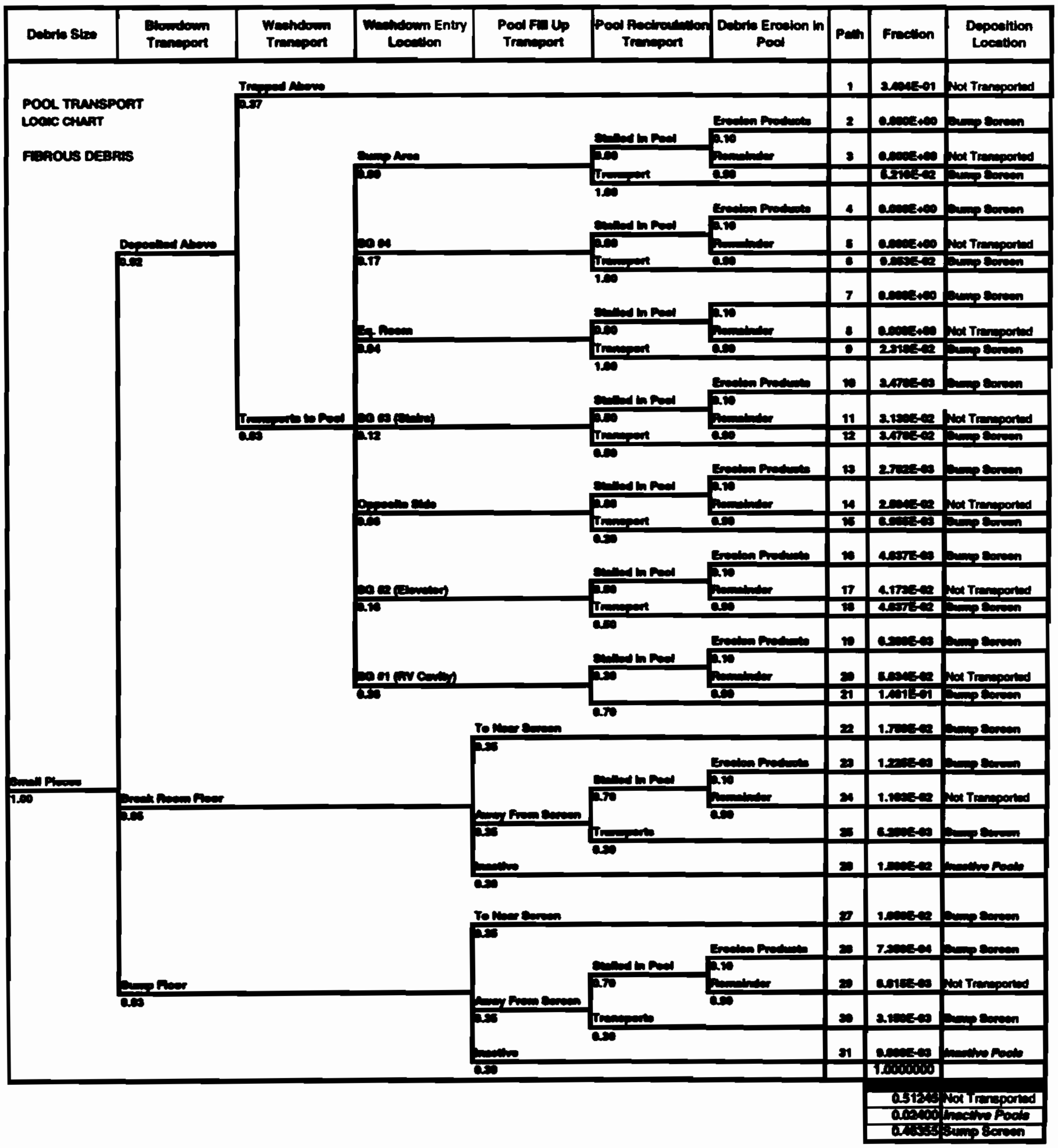


Figure III.3-3. Sump Pool Debris Transport Chart for Small-Piece Fibrous Debris.

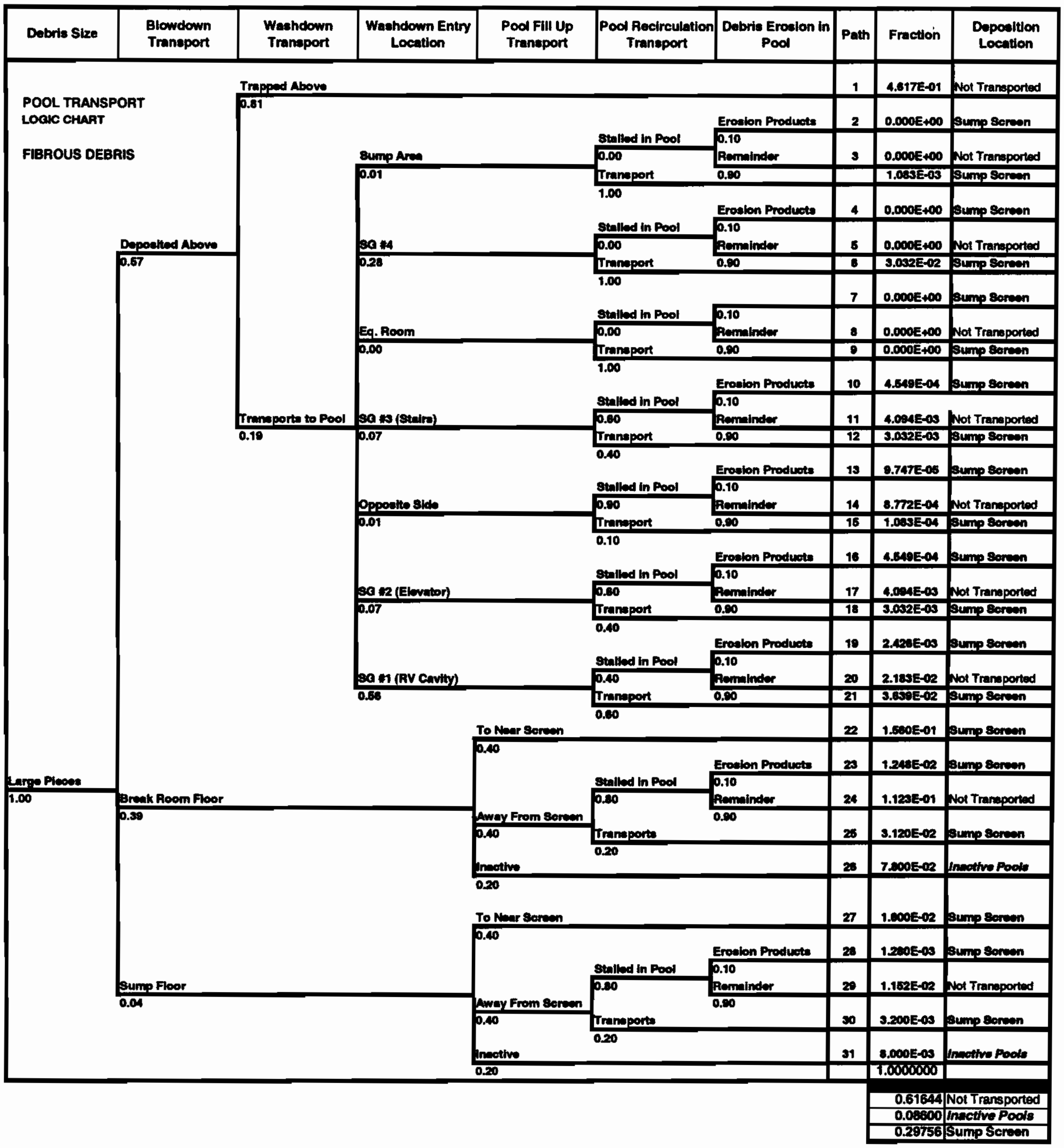


Figure III.3-4. Sump-Pool-Debris Transport Chart for Large-Piece Fibrous Debris.

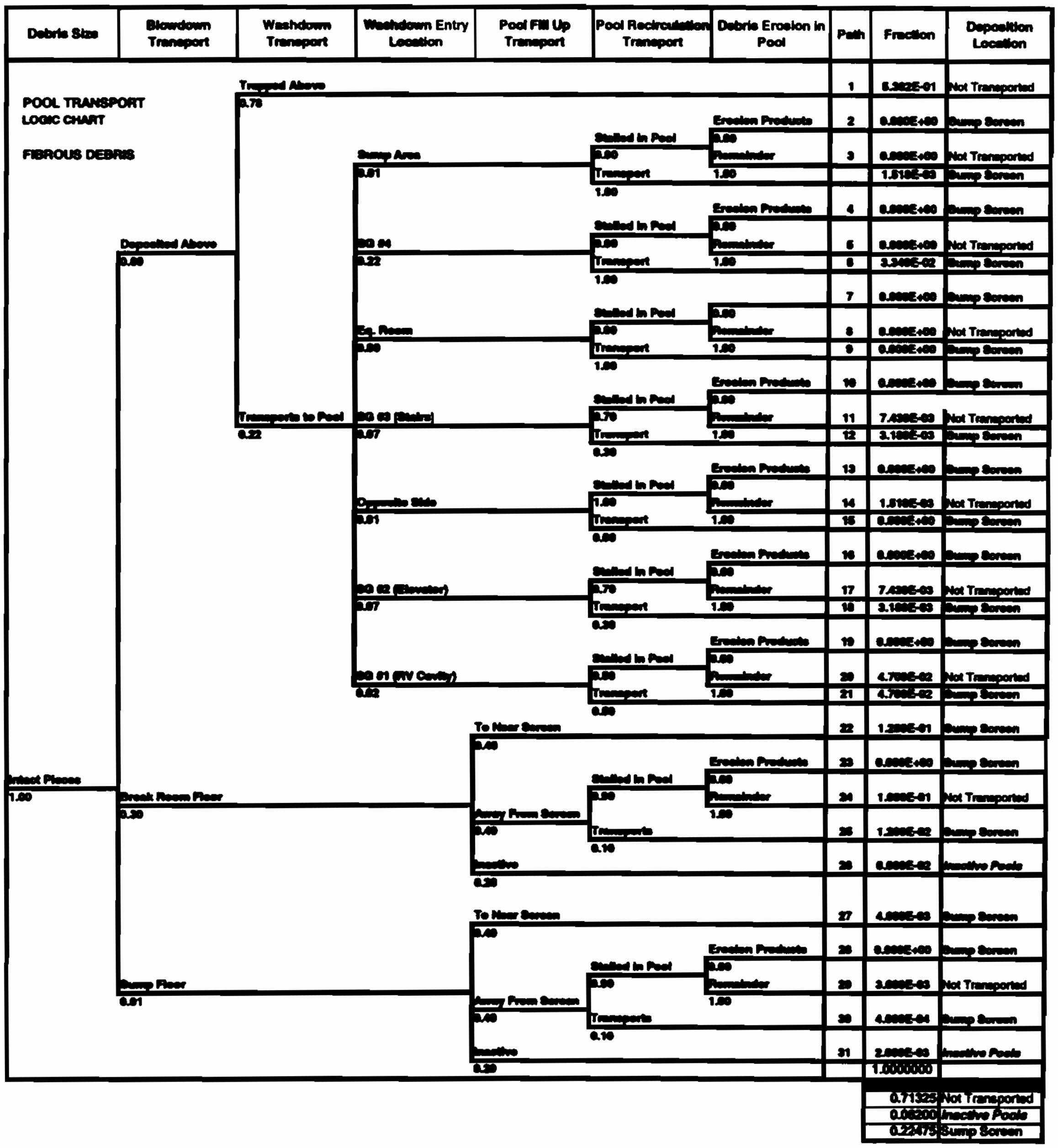
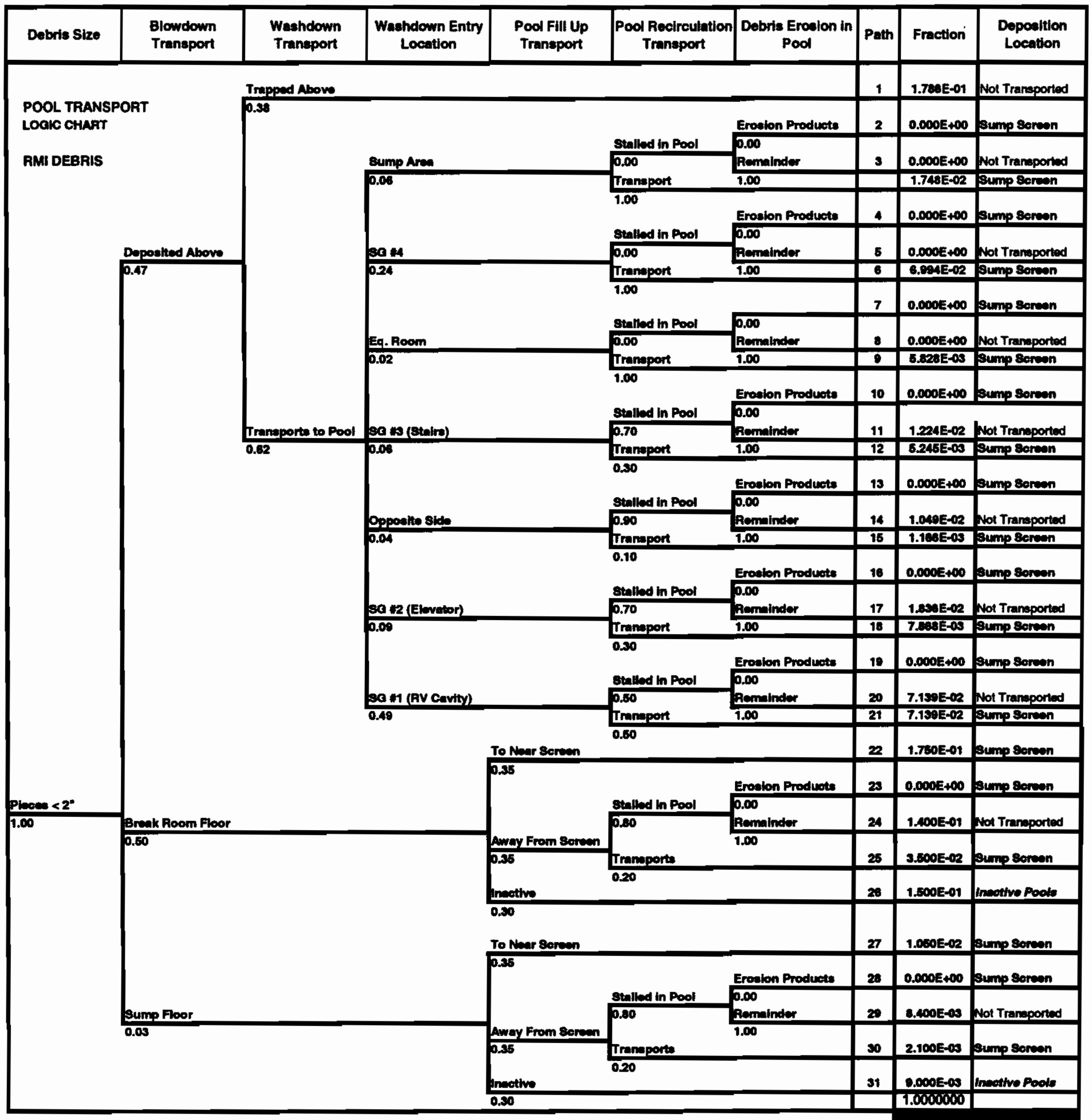


Figure III.3-5. Sump-Pool-Debris Transport Chart for Intact-Piece Fibrous Debris.



0.43948	Not Transported
0.15900	*Inactive Pools*
0.40152	Sump Screen

Figure III.3-6. Sump-Pool-Debris Transport Chart for <2-in. RMI Debris.

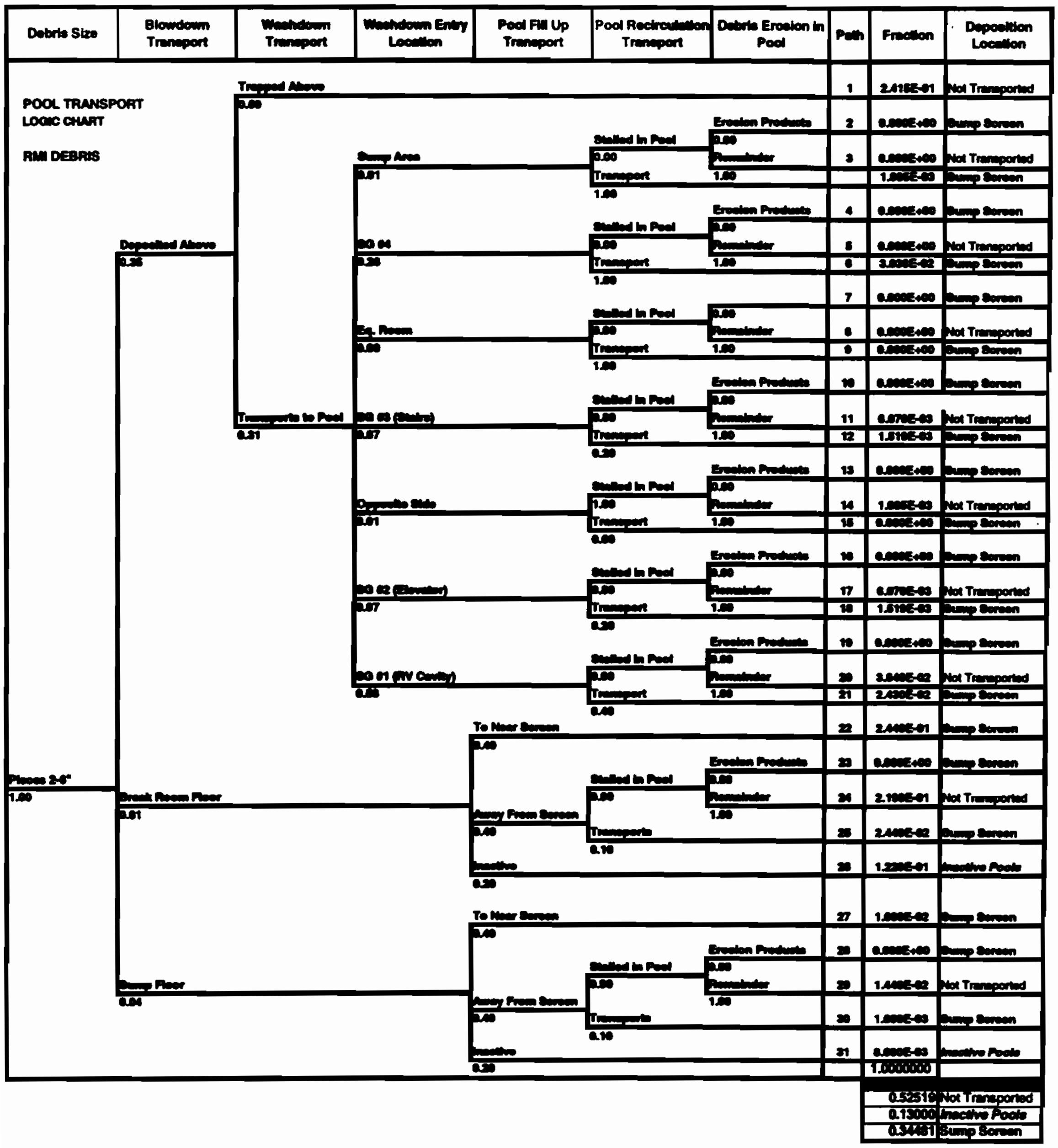


Figure III.3-7. Sump-Pool-Debris Transport Chart for 2- to 6-in. RMI Debris.

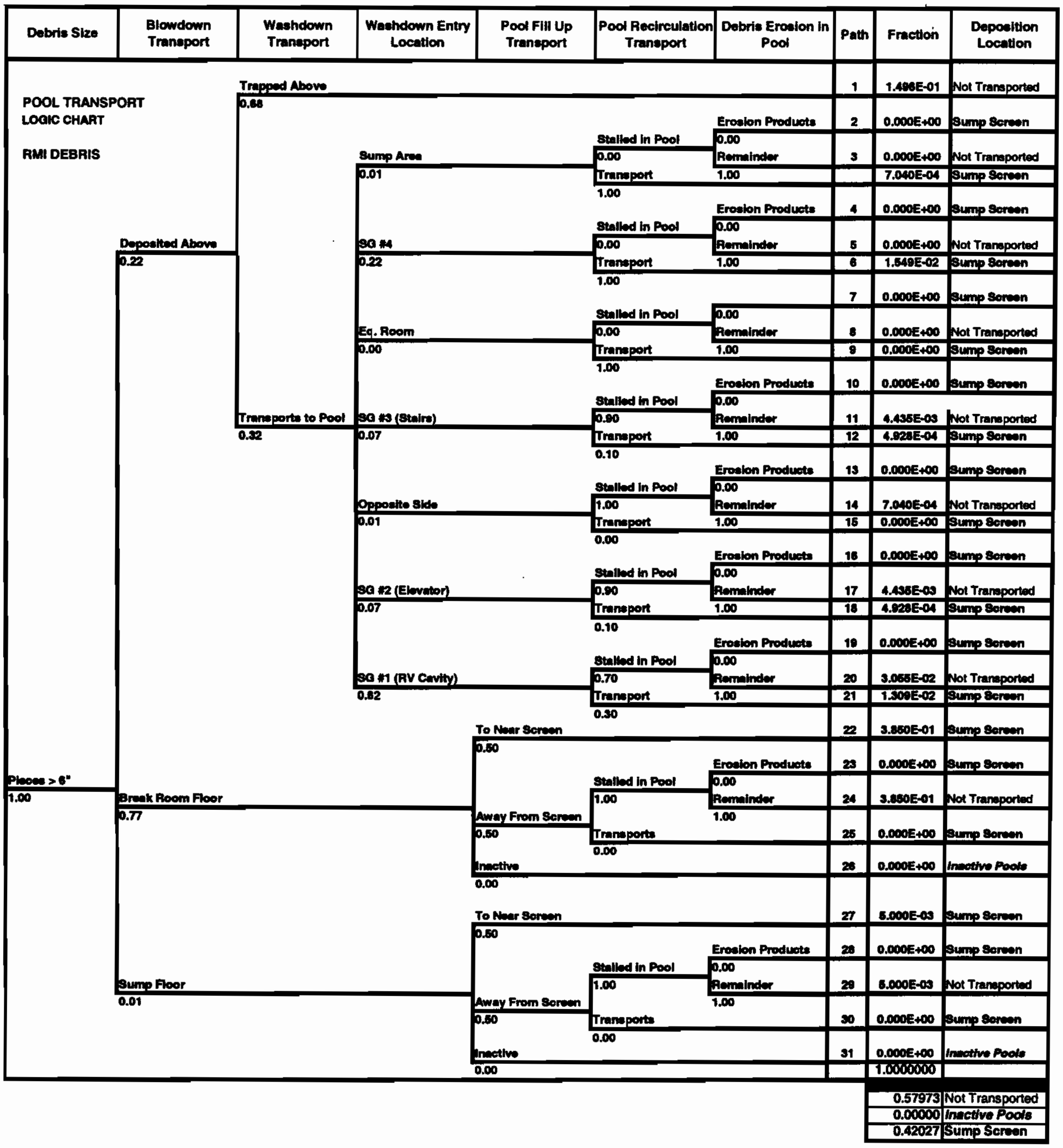


Figure III.3-8. Sump-Pool-Debris Transport Chart for >6-in. RMI Debris.

The quantified results by debris category and insulation type are shown in Table III.3-7, and the same results combined for each insulation type are shown in Table III.3-8. The analysis indicated that ~52% of the fibrous and ~42% of the RMI debris would accumulate on the recirculation screens for a large LOCA in steam-generator compartment 1 (SG1). The sump pool transport fractions for the small and large piece debris are quite high, 97 and 96%, respectively. The high fraction for debris eroded made a substantial contribution to these numbers. However, to put this assumption into perspective, if only 10% had been assumed for the erosion, the pool transport fractions would still have been 73 and 66%, respectively.

The RMI debris transport fractions were dominated by the large (> 6-inches) debris since 98.4% of the RMI was predicted to be in this category. It should be pointed out that this category includes quite large pieces including intact or nearly intact cassettes, which would required a faster flow to move the debris than 0.28 ft/s implemented into the CFD analyses.

Table III.3-7. Quantified Category-Specific Sump-Pool-Debris Transport Results

Debris Category	Category-Specific Debris Transport Fractions				
	Size Distribution	Entering Pool	Into Inactive Pools	Sump Pool Transport	Overall Transport
Fibrous Debris					
Fines	0.133	0.90	0.032	1	0.90
Small Pieces	0.397	0.64	0.024	0.97	0.62
Large Pieces	0.235	0.45	0.086	0.96	0.44
Intact Pieces	0.235	0.40	0.062	0.56	0.23
RMI Debris					
<2 in.	0.011	0.66	0.15	0.61	0.40
2 to 6 in.	0.005	0.63	0.13	0.55	0.35
>6 in.	0.984	0.85	0	0.49	0.42

Table III.3-8. Quantified Insulation-Specific Sump-Pool-Debris Transport Results

Debris Category	Insulation Specific Debris Transport Fractions			
	Entering Pool	Into Inactive Pools	Sump Pool Transport	Overall Transport
Fibrous	0.57	0.05	0.88	0.52
RMI	0.85	0.0024	0.50	0.42

The fractions of the sump pool floor where the floor level flow velocity was slower than the threshold velocities for debris (0.12 and 0.28 ft/s for fibrous and RMI debris, respectively) were calculated from the CFD results presented in Section III.2. The floor fractions corresponding to a large break in SG1 (lower-right quadrant in the CFD results) are 0.41 and 0.22 for fibrous and RMI debris, respectively. These floor area fractions are compared in Figure III.3-9 with the sump pool transport fractions by insulation type and size categories. In this scenario, if the debris was uniformly introduced into the pool across the pool cross sectional area and erosion was not significant, then the area fractions might be a reasonable indicator of the pool debris transport fractions. However, as shown, the area fractions are a poor indicator of debris transport when the debris is introduced into the pool in a more realistic and nonuniform manner and erosion is substantial. A uniform area fraction model can easily underpredict the pool debris transport by a factor of two or more.

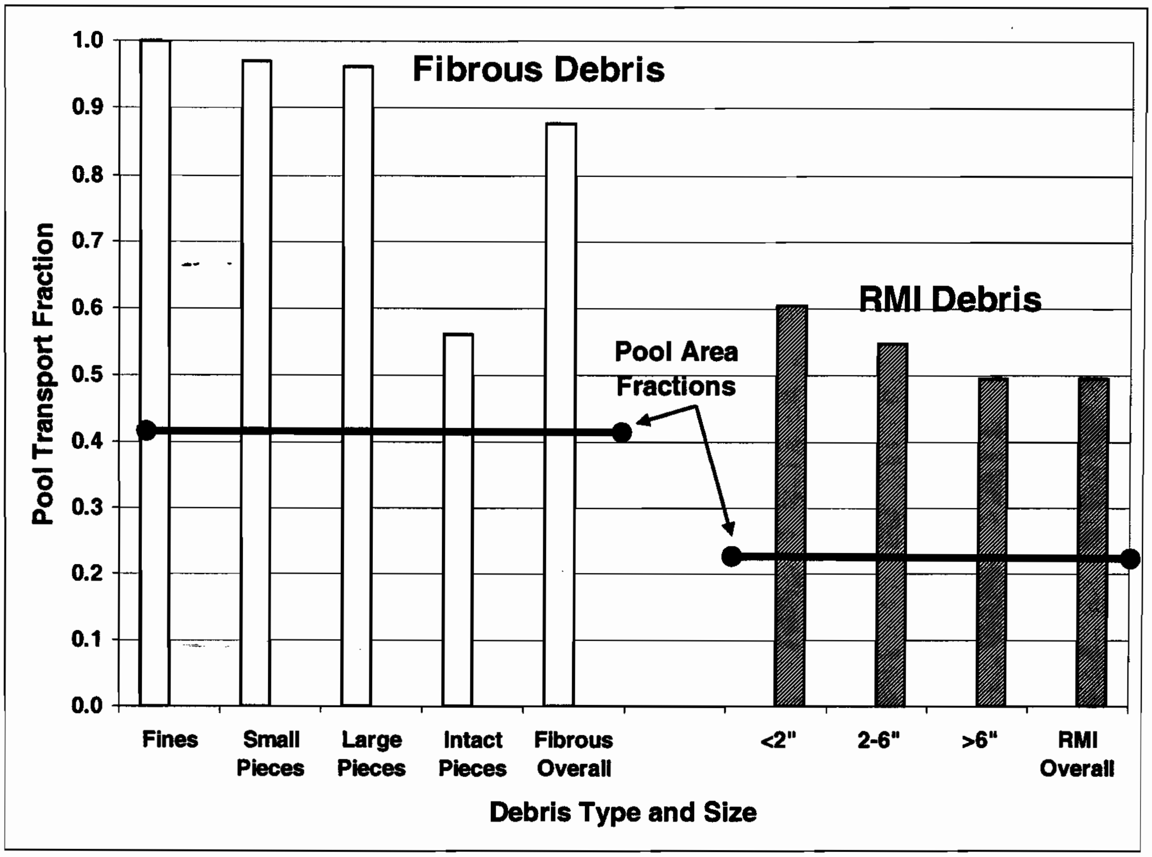


Figure III.3-9. Comparison of Sump Pool Transport Fraction with Velocity Area Fractions.

The transport of debris from its generation in the ZOI throughout the containment during the RCS depressurization phase, then the washdown transport by the containment sprays, and then its transport through the sump pool to the recirculation sump screens is a rather intractable problem. A logic chart method was used to decompose the overall transport problem into many smaller problems that were subsequently evaluated by either analysis or simply conservatively judged. As such, the results of the volunteer analyses contain many sources of uncertainties; however, these uncertain results are plausible results and show insight into the many aspects of debris transport that should be useful to subsequent evaluations. These sources of uncertainty regarding sump pool transport include (1) the timing and locations where debris enters the pool; (2) concerns regarding the effects of local pool turbulence that can move debris even when the bulk flow does not; (3) lack of data regarding erosion rates for debris that can decompose within the pool (e.g., fibrous debris); (4) the simplification of the analysis; and (5) the limited scenario space that can be realistically evaluated.

The debris transport results in this section pertain to a large LOCA in SG1. The same LOCA in another compartment could easily result in different transport results, which could be higher or lower than the scenario evaluated herein. In addition, the sump pool debris transport was evaluated herein using simplified nodalization, as discussed above.

A more detailed evaluation would likely refine these transport results significantly; however, the transport methodology has been demonstrated.

III.4 CONCLUSIONS AND RECOMMENDATIONS

Section III.2 outlined a method for performing reactor containment pool flow dynamic analysis. A commercial CFD code was used to perform the simulations and assess the flow properties relevant to debris transport. From the simulations, flow area fractions in excess of debris transport threshold velocities were obtained. Transient containment pool fill-up simulations were performed that could potentially be used to design debris diversion systems to sequester debris into zones that do not participate in the flow when sump pumps are engaged.

Recommendations for future simulations include performing grid-mesh convergence studies, further analysis of debris degradation mechanisms, and flow diversion. The grid-mesh convergence studies are required to have a defensible CFD analysis. Additional constraints on the grid mesh, not used or presented in this document, should include clustering grid points near the mass flow injection locations (break and splash locations) and development of a proper boundary layer grid near the no-slip walls, particularly on the containment floor. With additional grid points near the floor, a near-wall velocity profile will be established. This near-wall velocity gradient and drag forces could have an impact on debris transport and should be thoroughly investigated as part of the grid refinement study. The debris degradation mechanisms should also be the subject of further study. Examples of degradation have been shown in this document, but no attempt to quantify the dynamics has been made at this time.

The transport of debris from its generation in the ZOI throughout the containment during the RCS depressurization phase, then the washdown transport by the containment sprays, and then its transport through the sump pool to the recirculation sump screens is a rather intractable problem. A logic chart method was used to separate the overall transport problem into many smaller problems that were subsequently evaluated by either analysis or engineering judgment. As such, the results of the volunteer analyses contain many source of uncertainties; however, these uncertain results are plausible results and show insight into the many aspects of debris transport that should be useful to subsequent evaluations. These sources of uncertainty regarding sump pool transport include (1) the timing and locations where debris enters the pool; (2) concerns regarding the effects of local pool turbulence that can move debris even when the bulk flow does not; (3) lack of data regarding erosion rates for debris that can decompose within the pool (e.g., fibrous debris); (4) the simplification of the analysis; and (5) the limited scenario space that can be realistically evaluated.

The debris transport results in this appendix pertain to one LOCA scenario: a large LOCA in SG1. The same LOCA in another compartment could easily result in different transport results that could be higher or lower than the scenario evaluated herein. In addition, the sump pool debris transport was evaluated herein using simplified nodalization, as discussed above. A more detailed evaluation would likely refine these transport results significantly; however, the transport methodology has been demonstrated.

III.5 REFERENCES

[NUREG/CR-6772, 2001] D. V. Rao, B. C. Letellier, A. K. Maji, and B. Marshall, 2002, “GSI-191: Separate-Effects Characterization of Debris Transport in Water.“ NUREG/CR-6772, LA-UR-01-6882.

[NUREG/CR-6773, 2002] D. V. Rao, C. Shaffer, B. C. Letellier, A. K. Maji, and L. Bartlein, “GSI-191: Integrated Debris-Transport Tests in Water Using Simulated Containment Floor Geometries,“ NUREG/CR-6773, LA-UR-02-6786, December 2002.

APPENDIX IV: DEBRIS TRANSPORT COMPARISON

The NEI GR baseline debris transport recommendations contain both conservative and nonconservative assumptions which were used to simplify the transport evaluation. To assess the effect of the nonconservative assumptions used in the baseline model, the baseline model was applied to the pressurized-water-reactor (PWR) volunteer plant, whereby those baseline results could be compared with the detailed debris transport evaluation performed for the volunteer plant. The comparison supported the review and acceptance of the NEI baseline evaluation methodology by illustrating that the baseline predicted conservative debris transport results for the volunteer plant. Insights gained from this comparison regarding debris entrapment in the inactive pool and the transport of large debris support staff imposed limitations on the acceptance of the baseline methodology.

Because the volunteer plant contains substantial quantities of both fibrous and reflective metal insulation (RMI), the baseline model was applied to both types of insulation debris. Detailed sump pool debris transport analyses were performed for the volunteer plant containment as documented in Appendix III. Detailed blowdown and washdown debris transport analyses were performed for the volunteer-plant containment documented in Appendix VI. Appendix IV (this appendix) compares the GR baseline analysis to the detailed analyses for the volunteer plant as documented in Appendices III and VI.

The comparison is based on the GR baseline two-group debris-size distributions, i.e., small fines and large-piece debris. The detailed analyses used a four-group distribution of fines, small pieces, large pieces, and intact pieces. The detailed four-group results were reduced to two groups by combining the fines and small-piece debris into the NEI small-fines group and combining the large-piece and the intact-piece groups into the NEI large-piece group. This approach enabled a direct comparison.

The size distributions for both the NEI baseline results and the detailed analyses results were based on destruction pressures of 10 psi for the fibrous debris and 4 psi for the RMI debris. The respective size distributions were obtained from the research documented in Appendix II. The radii of the fibrous and RMI zone of influence (ZOI) for these pressures are 11.9D and 21.6D, where D is the diameter of the pipe which breaks (see Appendix I). In applying the baseline model to the volunteer plant, it was assumed that the containment was highly compartmentalized.

The baseline and detailed analyses results are compared by debris size in Tables IV.1 and IV.2 for fibrous and RMI debris, respectively. Table IV.3 compares the overall transport fractions, which combine the small fine debris and the large debris to obtain the total estimated screen accumulation. The respective debris-size distributions shown in Table IV.1 were used to calculate the overall transport results shown in Table IV.3. Note that the transport fractions in Tables IV.1 and IV.2 are pertinent only to the respective size categories.

Table IV.1. Baseline Comparison with Detailed Volunteer-Plant Fibrous Transport Results

Transport Phase	Debris Transport Fractions			
	Fine/Small Debris		Large-Piece Debris	
	Baseline	Detailed	Baseline	Detailed
Fraction of Debris Generated	0.60	0.53	0.40	0.47
Fraction of Debris Generated That Transports into Upward Levels by Blowdown	0.25	0.92	0	0.63
Fraction of Debris Generated That Transports Directly to Sump Pool Floor by Blowdown	0.75	0.08	1	0.37
Fraction of Debris Generated That Blows into Upper Levels and Washes down into Sump Pool	1	0.71	0	0.21
Fraction of Debris Generated That Enters Sump Pool	1	0.73	1	0.50
Fraction of Debris Generated That Enters Inactive Sump Pool	0.14	0.03	N/A	0.07
Fraction of Debris Generated That Enters Active Sump Pool	0.86	0.70	1	0.43
Fraction of Debris Enters Sump Pool That Transports to Sump Screens	1	0.98	0	0.76
Fraction of Debris Generated That Accumulates on Sump Screens	0.86	0.69	0	0.33

Delete

Delete Upwar

Delete Directly

Delete Upper

Delete

Delete

Delete Sump

Delete Screen

Table IV.2. Baseline Comparison with Detailed Volunteer-Plant RMI Transport Results

Transport Phase	Debris Transport Fractions			
	Fine/Small Debris		Large-Piece Debris	
	Baseline	Detailed	Baseline	Detailed
Fraction of Debris Generated	~~0.75~~	~~0.02~~	~~0.25~~	~~0.98~~
Fraction of Debris Generated that Transports into Upward Levels by Blowdown	0.25	0.44	0	0.22
Fraction of Debris Generated that Transports Directly to Sump Pool Floor by Blowdown	0.75	0.56	1	0.78
Fraction of Debris Generated that Blows into Upper Levels and Washes Down into Sump Pool	0	0.55	0	0.32
Fraction of Debris Generated that Enters Sump Pool	0.75	0.80	1	0.85
Fraction of Debris Generated that Enters Inactive Pools	~~0.11~~	~~0.15~~	~~N/A~~	~~0~~
Fraction of Debris Generated that Enters Active Sump Pool	~~0.64~~	~~0.65~~	~~1~~	~~0.85~~
Fraction of Debris that Enters Sump Pool that Transports to Sump Screens	1	0.59	0	0.49
Fraction of Debris Generated that Accumulates on Sump Screens	0.64	0.39	0	0.42

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Table IV.3. Comparison of Overall Baseline and Detailed Analysis Transport Fractions

Debris Type	Fraction of ZOI Insulation Debris Accumulated on Sump Screens	
	Baseline	Detailed
Fibrous Debris	0.52	0.52

RMI Debris	0.48	0.42

Substantial uncertainty exists in various aspects of the volunteer plant analyses that affect this comparison, includes the following:

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- Uncertainties in determining the debris generation size distributions.
- Uncertainties in specifying various aspects of the blowdown and washdown debris transport and deposition processes.
- Uncertainties in estimating the locations where debris enters the sump pool and when the debris enters with respect to the formation of the pool.
- Uncertainties in estimating the quantities of debris transported into the inactive pool regions.
- Uncertainties in estimating debris transport within an established sump pool.

The following points apply to the comparison of the fibrous debris transport:

1. The baseline recommendation for the debris-size distribution assumed 60% for the small fine debris, which is higher than the 53% determined from the integration of the air jet debris generation data and used for the detailed analysis (Appendix II).
2. In the detailed analysis, most of the smaller fibrous debris was predicted to be deposited in the upper levels during blowdown debris transport, rather than directly on the sump floor as proposed in the baseline model. Because the transport of this upper-level debris to the sump pool by containment spray drainage (washdown) is delayed by a variable and indeterminate period of time, it must be postulated that relatively little of the debris reaches the sump floor in time to be entrained in the water flow filling the inactive pools (primarily the reactor cavity in the volunteer plant), which occurs relatively early in the accident sequence (<12 minutes). The detailed analyses predicted that at the end of the blowdown/washdown transport a significantly less amount of debris, compared to the baseline analyses, would enter the active sump pool
3. The baseline model sump pool transport on to the sump screen was 100% of debris entering the sump pool for small fines and 0% for large-piece debris. The baseline model predicted more small fine debris accumulation on the sump screens than did the detailed analyses. However, the detailed analyses predicted substantial accumulation of large-piece debris on the screens, whereas the baseline predicted none.
4. The baseline and detailed analyses both predicted that approximately 52% of the fibrous debris generated within the ZOI would accumulate on the sump screens.

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The following points apply to the comparison of the RMI debris transport:

1. The baseline recommends using more small fine RMI debris (75% pf debris generated) than that was determined from the integration of the air jet debris generation data and

used for the detailed analysis (2%) (Appendix II). The primary reason for the large difference is the large increase in ZOI volume predicted by the ANSI/ANS-58.2-1988 standard when that standard is applied to jet impingement pressures as low as 4 psi. That is, although damage extends to 4 psi, only a small amount of small fine debris is generated over much of the ZOI volume. Most of the ZOI debris is large-piece debris.

2. The detailed analyses predicted lesser quantities of small fines RMI debris than fibrous debris would deposit in the upper levels of the containment (44% versus 92% of debris generated), although it was substantially more than the baseline model recommendation of 25%. A primary reason for this result was that so little blowdown debris transport data exist for RMI debris and thus the blowdown analyses conservatively assumed a large fraction of debris depositing directly on the sump floor. Both the detailed and baseline analyses predicted that approximately the same amount of debris would enter the active sump pool at the end of the blowdown/washdown transport (65% versus 64% of debris generated).
3. The baseline model sump pool transport was 100% for small fines and 0% for large-piece debris. The baseline model predicted more small fine debris accumulation on the sump screens than did the detailed analyses (64% versus 39% of debris generated). However, the detailed analyses predicted substantial accumulation of large-piece debris on the screens (42% of debris generated), whereas the baseline predicted none.
4. The baseline method predicted slightly more RMI debris accumulation on the sump screens than did the detailed analyses, i.e., 48% as compared with 42% of the debris generated.

CONCLUSIONS NUMBER AS HEADER?

The application of the baseline methodology to the volunteer plant predicted approximately the same accumulation of fibrous debris and conservatively more RMI on the sump screen than did the detailed transport analyses. Although this comparison does not explicitly demonstrate that the baseline methodology is conservative relative to fibrous debris transport in the detailed volunteer plant evaluation, detail-specific conservatisms built into various aspects of the blowdown/washdown and pool debris transport analyses still support the overall conclusion that the baseline methodology is conservative with respect to its application to the volunteer plant. Even though the baseline and detailed evaluation arrived at the same fractions for sump screen debris accumulation, the intermediate steps disagreed. Due to the diversity among the PWR containment designs, this analysis does not conclusively demonstrate that the baseline methodology will be conservative for debris transport in all of the PWRs. In addition, substantial sources of uncertainty were noted in the detailed volunteer plant analyses.

Insights gained from this comparison regarding debris entrapment in the inactive pool and the transport of large debris support staff imposed limitations on the acceptance of the baseline methodology to prevent an outlier plant from demonstrating adequate NPSH margin using the baseline methodology where adequate NPSH margin might not exist in reality. The limitations resulted from the following two concerns that should be addressed before accepting baseline method results for plant-specific analyses.

First, if a plant baseline analysis estimates a relatively large fraction of the debris trapped in the inactive pools, as could be the case with a large reactor cavity volume and a shallow sump pool, then the baseline inactive pool fraction should be more limited than the current baseline model.

Note that the detailed analyses reported herein predicted only approximately 3% of the small fibrous debris generated would trap in the inactive pool as compared with 14% that was predicted using the baseline model. Based on this comparison the staff limits the fraction of debris assumed to be trapped in the inactive pool should be limited to no more than ~15% unless a higher fraction is adequately supported by analyses or experimental data.

Second, if the characteristic sump pool transport velocities are relatively high, such that large transport fractions for large debris are indicated, then the baseline method should be modified to include the transport of large debris. In the volunteer plant, for example, the detailed analysis predicted approximately 98% of the RMI debris generated in the ZOI (based on a destruction pressure of 4 psi) was large pieces with size greater than 6 in., of which about 42% would be transported to the sump screens. The characteristic transport velocities must be compared with typical debris transport velocities to determine whether the baseline method should be modified to include the transport of large debris. Characteristic transport velocities can be sufficiently estimated using recirculation flow rates and nominal sump dimensions to determine if a potential exists that substantial portions of the large debris will transport. If substantial transport of large debris is reasonably possible and if such transport can alter the outcome of the sump performance evaluation, the licensees should evaluate large debris transport.

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APPENDIX V: CONFIRMATORY HEAD-LOSS ANALYSES

Confirmatory research was performed to determine whether specific parameter assumptions made in the Nuclear Energy Institute (NEI) guidance report are conservative with respect to more realistic parameters. This research also provided additional insights into the estimation of head-loss parameters for the NUREG/CR-6224 head-loss correlation. Additional guidance is also provided for determining appropriate parameters for a mix of multiple fiber and particulate components.

V.1 FIBROUS DEBRIS HEAD-LOSS PARAMETERS

A comparison of specific surface areas (*Sv*) deduced from head-loss test data and the simple geometric correlation of four divided by the characteristic fiber diameter (4/d) is presented for NUKON™ and Kaowool™ insulation debris. The test data used in both of these deductions are available in the BWROG head-loss tests documented in Volume 1 of the BWROG Utility Resolution Guidance (URG).

V.1.1 NUKON™ Fibrous Debris

The URG has three head-loss tests that used only NUKON™ insulation debris and used a type of strainer that behaved similarly to that of a flat-plate screen (i.e., a truncated cone strainer). These tests were numbered 2, 4, and 5 and used 8, 8, and 16 lb of NUKON™, respectively, and no particulate. The flow velocities through the bed varied from ~0.15 to 0.75 ft/s, resulting in a total of 15 head-loss data points. A specific surface area was deduced for each data point using the NUREG/CR-6224 head-loss correlation and using an as-manufactured density of 2.4 lb/ft^3 and a fiberglass material density of 175 lb/ft^3 (NUREG/CR-6224 study recommendations). The resultant *Sv* values are compared in Figure V-1.

The comparison was based on the debris bed compression as determined by the NUREG/CR-6224 correlation (the ratio of the compressed thickness divided by the uncompressed thickness), which is directly affected by the flow pressure (i.e., flow velocity). The average value for *Sv* was ~170,600 ft^{-1}. The nominal diameter for NUKON™ fibers has been specified as 7.1 µm, which translates into an *Sv* of 171,710 ft^{-1}. The NUREG/CR-6224 study recommended an *Sv* of 171,420 ft^{-1}. For NUKON™ insulation debris, the *Sv* determined using four divided by the fiber diameter is in excellent agreement with the experimentally deduced value.

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The NEI guidance has recommended using a material density of 159 lb/ft^3 rather the NUREG/CR-6224 study value of 175 lb/ft^3. Confirmatory analysis using the NUREG/CR-6224 correlation confirmed that it is conservative to use 159 lb/ft^3 rather than 175 lb/ft^3, provided that the remaining head-loss parameters of 2.4 lb/ft^3 for the as-manufactured density and 171,000 ft^{-1} for the specific surface area are maintained. The lower value for the material density estimates a slightly higher head loss than does the larger value.

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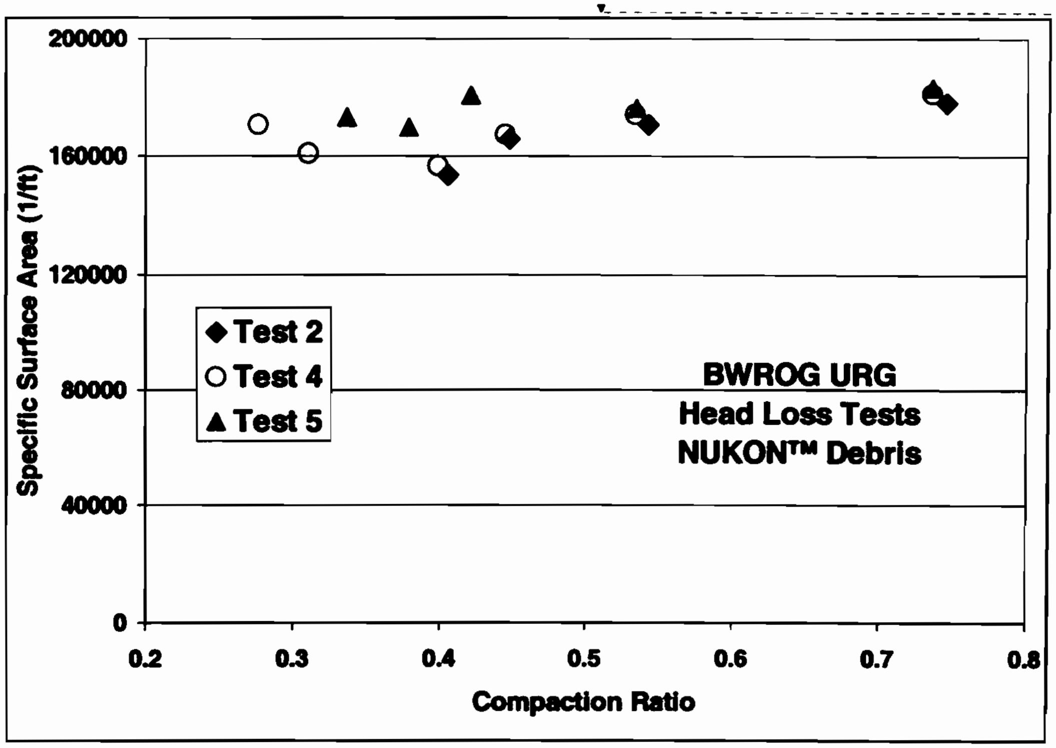

Figure 0V-1. NUKON™ Specific Surface Area.

Similarly, the NEI guidance recommended using 62.4-lb/ft^3 (1.0 g/cm^3) for material density of latent fibers to enhance transport (neutral buoyancy). The latent debris characteristics test results [LA-UR-04-3970, 2004a] that analyzed latent debris collected in the containments of several volunteer plants show that the latent debris fibers had material densities ranging from 1.3 to 1.9 g/cm^3. Again, confirmatory analyses verified that it is conservative from a head-loss prediction perspective to assume that the latent fiber material density is 1.0 g/cm^3 rather than 1.3 to 1.9 g/cm^3, provided that the remaining head-loss parameters are appropriately specified.

V.1.2 Kaowool™ Fibrous Debris

The URG has one valid head-loss test[*] that used Kaowool™ insulation debris and used a type of strainer that behaved similarly to that of a flat-plate screen (i.e., a truncated cone strainer). Test J13 initially had added 12 lb of Kaowool™, then later added 5 lb of iron oxide corrosion products (CPs), and then subsequently added another 5 lb of CP. The flow velocities through the bed varied from ~0.31 to 0.62 ft/s, resulting in a total of nine head-loss data points (three data points without particulate). A specific surface area was deduced for each data point using the NUREG/CR-6224 head-loss correlation, with the NUREG/CR-6224 study recommended parameters for the corrosion products used as input.[†] The recommended fiber material density for Kaowool™ is 160 lb/ft^3.

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[*] Test J12 also used Kaowool, but the quantities of corrosion products overwhelmed the debris bed that if all of the corrosion products had filtered from the flow, the granular bed, not counting the Kaowool, would have been nearly 2 inches thick. The head-loss contribution due to Kaowool was overshadowed by the corrosion products that the test was not valid for determining the specific surface area for Kaowool.

[†] The NUREG/CR-6224-recommended parameters are 183,000 ft^{-1} for the specific surface area, 324 lb/ft^3 for the particulate material density, and 65 lb/ft^3 for the granular packing-limit density.

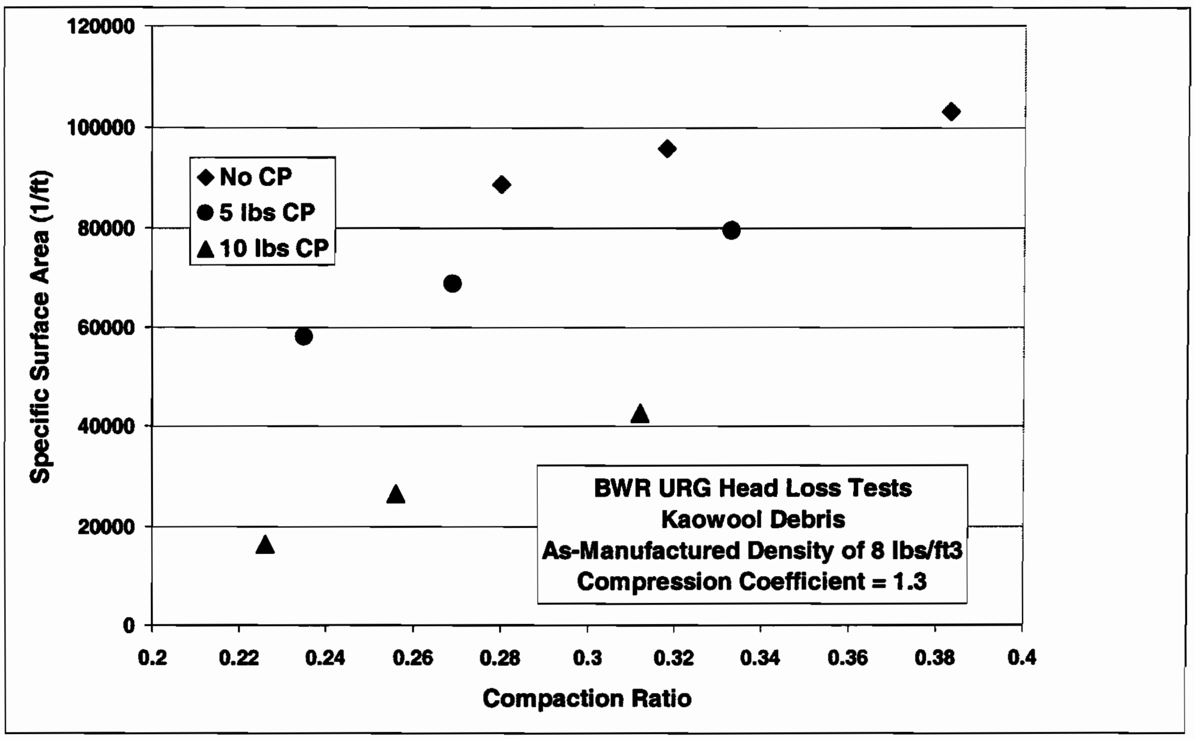

Figure V-2. Kaowool Specific Surface Area Assuming Base Parameters

The NEI guidance recommends an as-manufactured density of Kaowool™ ranging from 3 to 12 lb/ft^3, whereas the URG recommended a value of 8 lb/ft^3, apparently a midrange value. First, the *Sv* values were deduced from Test J13 data by assuming an as-manufactured density of 8 lb/ft^3 and the same bed compression correlation that was so successful for NUKON™. These resultant *Sv* values are compared in Figure V-2. The values of *Sv*, as shown, are very scattered, ranging from 16,000 to 103,000 ft^{-1}. All in all, the NUREG/CR-6224 correlation does not work well with these input parameters. Noting that the as-manufactured density cited in the guidance report (GR) ranged from 3 to 12 lb/ft^3, it was subsequently determined that a smaller value of the density would reduce the scatter in the resultant *Sv* values. Further, it was discovered that stiffening the compression function also reduced the scatter. A second comparison of the deduced *Sv* values was developed assuming an as-manufactured density of 4 lb/ft^3 and a leading compression coefficient of 0.5 (rather than the standard 1.3). The results are shown in Figure V-3. The comparison in Figure V-3 has the deduced values in good agreement, with an average value of 165,500 ft^{-1}.

The nominal diameter for Kaowool™ fibers has been specified as 2.7 to 3.0 μm in the NEI guidance, which translates into an *Sv* of 406,400 to 451,500 ft^{-1} using the four-divided-by-the-diameter formula. Although using such high values for *Sv* is conservative, the simple formula is not even close to the experimentally deduced value of 165,500. The application of an *Sv* of 406,400 ft^{-1} would substantially overpredict the results of Test J13.

The coefficient of the NUREG/CR-6224 compression correlation is an important issue. The standard coefficient of 1.3 was developed and validated essentially using NUKON™; therefore, the validation of other fibrous insulation must assess the validity of this value for the insulation under consideration. It is noted that the baseline guidance in the GR considers this point by including the constant K (Equation 3.7.2-4 in Section 3.7.2.3.1.1 of the baseline guidance with a default value of 1 for K). For Kaowool™, a K = 0.385 and a Sv of 165,500 ft^{-1} in the NUREG/CR-6224 correlation predicts URG Test J13 results reasonably well.

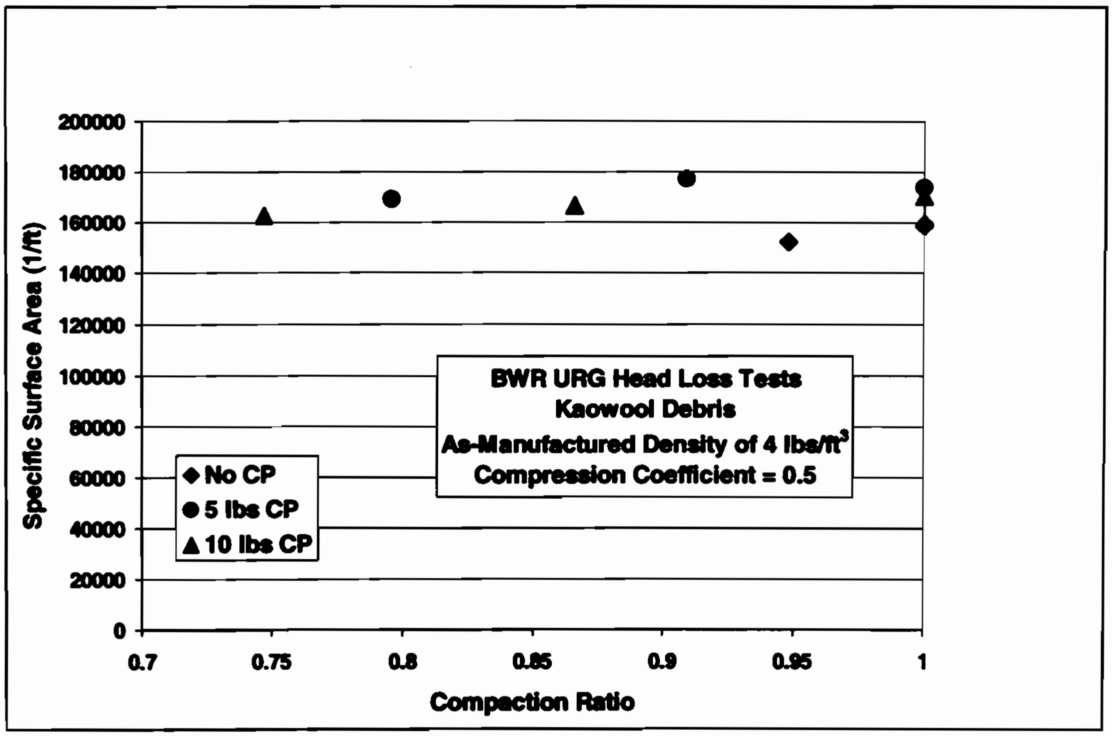

Figure V-3. Kaowool™ Specific Surface Area Using Modified Parameters.

V.1.3 Comparison of Fibrous Debris

The specific surface areas are compared in Figure V-4 for areas determined using the four-divided-by-the-fiber-diameter formula and the two experimentally deduced values presented herein for NUKON™ and Kaowool™. The following points are made:

1. The coefficient(s) for the compression correlation also have a role in the application of the NUREG/CR-6224 correlation to the various types of fibrous debris.
2. The 4/d formula was formerly validated using NUKON™, but not necessarily for other types of fibrous insulations.
3. The 4/d formula is not reliable and should not be applied indiscriminately. It should not be assumed that because this formula overpredicts Kaowool™ head losses that it will be conservative for untested types of fibrous debris. The only reliable method of determining the specific surface area of a particular insulation material is deduction from applicable test data.

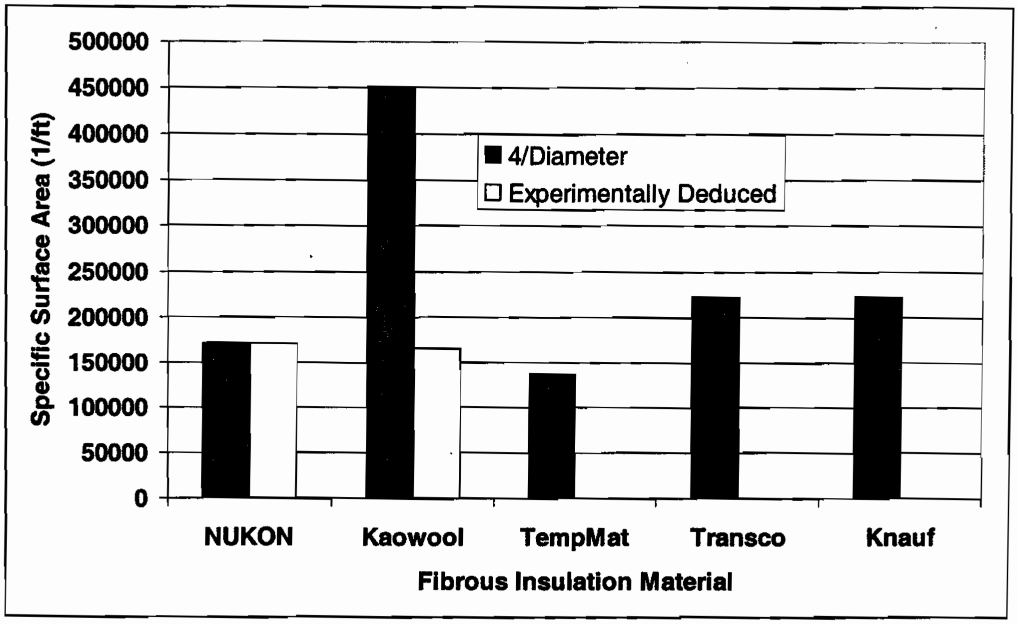

Figure 0-4. Comparison of Fibrous Insulation Specific Surface Areas.

V.2 PARTICULATE DEBRIS HEAD-LOSS PARAMETERS

In Section 3.7.2.3.1.1 of the GR, the NEI guidance recommends using the simple formula of six divided by the characteristic particle diameter to determine the specific surface areas for particulate debris. The following confirmatory analyses provide insights into this relationship and experimentally deduced values for particulate *Sv.*

V.2.1 Iron Oxide Corrosion Products

During the resolution of the BWR strainer blockage issue, the iron oxide CPs that accumulate in a boiling-water-reactor (BWR) suppression pool were the primary particulate in the head-loss calculations. The BWR sludge (CP) is characterized by the size distribution shown in Table V-1.

The NUREG/CR-6224 correlation recommends a specific surface area of 183,000 ft^{-1} for head-loss estimates with CP, which has been validated by comparison with test data. Using the midrange diameters from Table V-1 to estimate the *Sv* for the CP distribution using the 6/d formula, the *Sv* estimate becomes 48,400 ft^{-1} (almost a factor of four less than the NUREG/CR-6224 recommendation). Note that a factor-of-4 error in the *Sv* can result in a factor as large as 16 in error in the head loss at low flow velocities.

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If the minimum value of the range is used (assuming a minimum particle size of 2 μm for the 0- to 5-μm size group), then an *Sv* of ~290,000 ft^{-1} is calculated (~58% higher than the recommended validated area). The smaller particles have more effect on the particulate *Sv* than do the larger particles, which is why the midrange diameters are not a valid representation of the distribution. Using the smallest diameters of each group is conservative but can result in

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large estimates of *Sv*. Further, these examples illustrate that it is difficult to determine where in a size range is an appropriate diameter for the *Sv* determination using 6/d.[*]

Table V-1. Size Distribution of BWR Suppression Pool Iron Oxide Corrosion Products[†]

Size Range (μm)	Percent by Number of Particles	Percent by Weight
0–5	81%	0.3%
5–10	14%	1.5%
10–75	5%	98.2%

An example of how the 6/d formula works over a particle-size grouping is illustrated in Figure V-5, where 6/d is plotted for particle diameters ranging from 5 to 75 μm (typical distribution grouping). If it is assumed that particles are uniformly distributed (by weight) across this size range (which is not necessarily a valid assumption), then the average 6/d corresponds to a diameter of 25.8 μm, whereas the midrange diameter is 40 μm. Because this simple arithmetic relationship arrives at differing conclusions, depending on the range specification, this method cannot be used reliably in a general sense, even if the uniform distribution assumption is valid.

In summary, the only reliable method of determining the *Sv* for a particulate, unless the particulate-size distribution is known in much greater detail than has been typically specified to date, is to deduce *Sv* from valid head-loss test data. It is conservative to use the lower diameter of each size group but this can lead to large estimates of the *Sv*. However, this method is valid when applicable head-loss data are lacking. Another difficulty is the determination of the smallest particles in the distribution. Although most particulates will have sub-micron particles in the distribution, fiber debris beds may not filter such small particles or certainly the efficiency of filtration could be rather low and is difficult to determine.

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[*] Similar results were obtained when 6/d was applied to concrete dust head-loss test data during NRC-sponsored tests documented in LA-UR-04-1227.

[†] The NEI guidance (Table 4-3) and NUREG/CR-6224 (Table E-2) both have the percentages in the center column of this table listed as percentages by weight. However, the BWROG URG (Table 7) lists this column as percentages by the number of particles, as shown here. Because the data originated from the BWROG and the numbers only seem to make sense as the number of particles, it is assumed here that the URG is the correct source. Therefore, it is believed that the heading was mislabeled in NUREG/CR-6224 from which the NEI adapted the data for the guidance. In any case, it is conservative to assume that 81% of the particulate by weight is <5 microns because this assumption leads to very high specific-surface-area estimates.

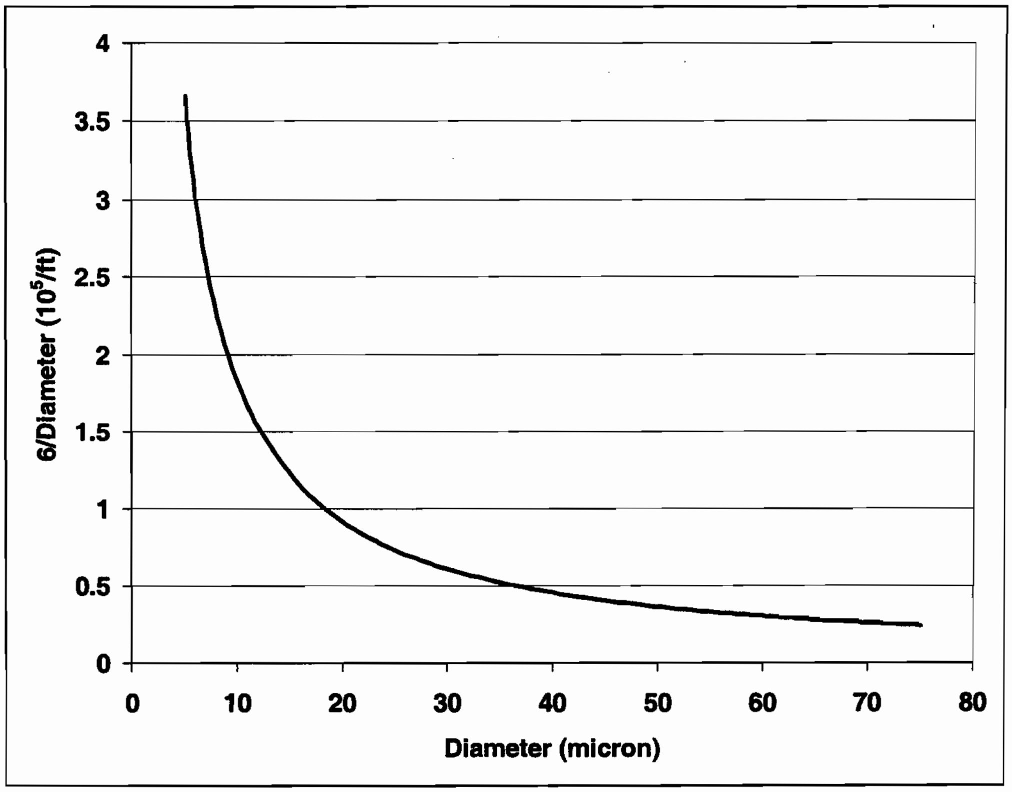

Figure V-5. Example of *Sv* Variation with Particle Diameter.

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V.2.2 Latent Debris

The characteristics of latent debris collected from inside containments of several nuclear plants have been determined by Los Alamos National Laboratory (LANL) [LA-UR-04-3970, 2004a]. These characteristics included properties of material composition and hydraulic flow properties (e.g., specific gravities and characteristic dimensions). Based on these characteristic properties, surrogate latent particulate debris[*] was formulated for testing in the closed-circulation head-loss simulation loop operated by the Civil Engineering Department at the University of New Mexico (UNM).[†] Applying the NUREG/CR-6224 head-loss correlation to the test data for the surrogate latent debris resulted in parameter recommendations for the application of the correlation to plant latent debris. Those recommendations are summarized below, together with insights gained from the surrogate latent debris data reduction. The test apparatus and base test procedures are described in detail in the calcium silicate debris test report [LA-UR-04-1227, 2004b].

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The plant debris characteristics pertinent to the specification of a recipe to create a suitable latent particulate surrogate include the particulate specific gravity and the particulate-size distribution. The particulate-size distribution, shown in Table V-2, was used as a recipe for the

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[*] A surrogate was required to provide the quantities of debris needed for head-loss testing. The latent debris collected in containment required the special handling associated with radioactive materials.

[†] NUKON™ insulation debris was selected to form the fiber bed to filter the surrogate particulate from the flow because of its well-established head-loss properties.

particulate. The surrogate particulate debris tested at UNM was constructed from common sand and soil (referred to as dirt) with the sand used for the two larger size groups and the dirt for the <75-µm-size group. The specific gravity of the latent debris characterized at LANL varied but is well represented as a specific gravity of 2.7, and both the sand and dirt used to formulate the surrogate were found to have a specific gravity near 2.7. The dirt had a clay component that tended to disintegrate, in part, in water, thereby adding substantial particulate <10 µm to accommodate the LANL finding that substantial very fine debris was collected in the filters. Both granular (thin-bed) and nongranular debris beds were tested.

Table V-2. Surrogate Particulate Size Distribution

Size Range (µm)	Fraction
500 to 2000	0.277
75 to 500	0.352
< 75	0.371

Tests were conducted using the individual size groupings for the 75- to 500-µm sand and the <75-µm dirt (without the other groups present) to determine specifically the head-loss characteristics of these individual size groupings; then the latent debris recipe was tested with all three size groups represented according to the recipe. The largest size group (500 µm to 2 mm) was not individually tested because of its relatively minor impact on the recipe head loss; its small specific surface area was estimated using the 6/d equation. For the other two size groups, the specific surface area was deduced from the head-loss data. The bulk densities of the three components were estimated by measuring the bulk volume in a calibrated beaker for a weighted mass of particulate. Given the particle specific gravity and the bulk densities, the granular debris bed porosities were estimated. The test results for the surrogate latent particulate debris are summarized in Table V-3.

Table V-3. Summary of Test Results

Particulate (µm)	Bulk Density (lbm/ft^3)	Limiting Granular Porosity	Limiting Granular Solidity	Specific Surface Area (ft^{-1})
500 to 2000 (Sand)	104	0.38	0.62	2000
75 to 500 (Sand)	99	0.41	0.59	10,800
<75 (Dirt)	39	0.77	0.23	285,000
Recipe	63 to 75	0.62 to 0.55	0.38 to 0.45	106,000

A range of numbers is shown for the bulk density and limiting granular porosity and solidity due to the uncertainty associated with filtration of the very fine dirt from the water flow, i.e., how much of the dirt introduced into the test loop actually resided in the debris bed. Test-loop water turbidity measurements clearly showed that significant, sometimes substantial quantities of the fine dirt were not filtered from the flow by the fibrous bed. If there is a minimum particle size for effective filtration, it is most certainly significantly <10 µm and likely less than a few microns. Table V-3 presents nominal estimates for the specific surface area for each component; however, there is significant uncertainty in determining these numbers. The primary uncertainty associates with the less than 75 microns particulate was the filtration efficiency of the finer particles. Assessing the uncertainties in the turbidity resulted in the conclusion that between 30% and 45% of the particulate remained in solution, which corresponded to a range of about 250,000 ft^{-1} to 340,000 ft^{-1} in the specific surface area when the correlation was applied. For the two larger particulate size groups (75 to 500 microns and 500 to 2000 microns), the uncertainties were analytically estimated using the 6/diameter formula where the diameter was ranged from the smallest diameter particles up to 25% of the range. These estimated uncertainties are compared in Figure V-6.

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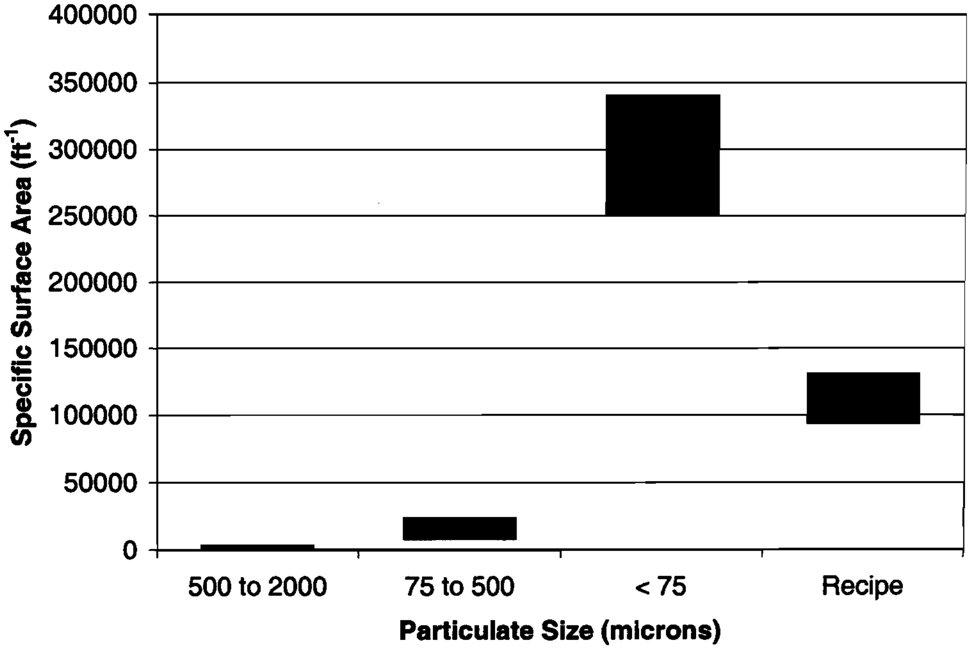

Figure V-6. Comparison of Component and Recipe Specific-Surface-Area Ranges.

Key points that can be deduced from the foregoing discussions relative to latent debris are the following:

1. The head loss through granular (thin-bed) debris is controlled by the limiting porosity (solidity), which depends on the composition of the debris. Solidity certainly is not a fixed number, as is indicated in the presentation of the NEI guidance as a solidity of 0.2. Handbooks on soils show many materials with limiting porosity <0.8, e.g., common sand is ~0.40 to 0.43 and was experimentally verified in the LANL tests.

2. The major contributors to the head loss are the increasingly smaller particles (<75 μm), as illustrated by the 6/d formula, until the particles become too small for filtration. However, it is difficult to determine some limiting particle diameter that will not filter.

3. It is difficult to formulate specific recommendations for the appropriate parameters to use in the NUREG/CR-6224 correlation for pressurized-water-reactor (PWR)-containment latent particulate because the latent debris composition will vary from plant to plant and because the latent debris transported to the sump screen will also be plant specific because of such differences as flow velocities. In addition, the uncertainties associated with whether the surrogate recipe suitably represents actual containment latent debris further compound the problem of developing recommended characteristics for latent debris. More important than specific recommendations are the methods for ascertaining appropriate head-loss parameters once the plant has assessed latent debris accumulation on the sump screen.

4. The surrogate latent particulate debris head-loss tests effectively demonstrate the necessity of characterizing the latent particulate so that appropriate parameters can be estimated. For example, if the entire mass of the latent debris is assumed to be deposited onto the sump screen, then a lower specific surface area, such as the recipe in these tests, can be applied. However, if transport analyses are used to limit the transport of latent particulate to only the fine particulate, then the appropriate specific surface area would be more like that of the fine dirt in these tests. The same consideration also applies to the limiting packing density.

5. It is recommended that plant latent debris estimates be separated into as many particle size groupings as reasonably possible and then that subsequent transport analysis be applied to each group to determine the particulate makeup on the sump screen.

6. Wherever possible, specific surface areas should be determined for each size group based on test data. When the areas must be estimated from the particle diameters, the appropriate diameter is clearly not the mean or average diameter of the size group but a diameter closer to the minimum diameter of the group. The minimum diameter should normally result in a conservative specific surface area.

7. The use of the simple geometric relationship of 6/d to estimate the specific surface areas for particulate is not reliable because the appropriate diameter within the range is not known. This point is illustrated in Table V-4, where values for *Sv* are estimated using both the mid-range and minimum diameters for each size group in the surrogate latent particulate recipe; these values are compared to the *Sv* deduced from the experimental head loss and the particle diameters that correspond to the experimental *Sv*. This minimum diameter in the size range estimates a conservative *Sv*; however, that number could be unacceptably large if the minimum size for the smallest particles is not well known. The use of mid-range diameters is unacceptable because this

approach excessively underpredicts *Sv* values for plant-specific evaluations. . If the specific surface areas corresponding to the minimum particle diameters in each size grouping range are unacceptable, then head loss test data is required to determine a specific surface area for the particulate size distribution in question.

8. The NEI guidance recommends the use of 100 lb/ft^3 for the material density of latent particulate, whereas LA-UR-04-3970 indicates a density of ~168 lb/ft^3 (specific gravity of ~2.7). The use of the lighter density of 100 lb/ft^3 is conservative relative to a heavier density of 168 lb/ft^3, for example, if the other head-loss parameters are appropriately specified.

Table V-4. Comparison of Specific Surface Area Estimation Methods

Particulate Size (µm)	Analysis			Experimental Sv	
	Mid-Range Diameter (µm)	Sv = 6/d Mid-Range Sv (ft^{-1})	Sv = 6/d Mid-Range Sv (ft^{-1})	Sv Deduced from Experimental Head-Loss Data (ft^{-1})	6/Sv Experiment (µm)
500 to 2000 (Sand)	1250	1460	3660	2000	914
75 to 500 (Sand)	287.5	6360	24,380	10,800	169
<75 (Dirt)	37.5	48,770	914,000*	285,000	6.4
Recipe	88.2	20,740	349,000	106,000	17.3

* Assuming a 2-µm minimum particle size.

V.3 FORMULAS FOR MIXING MULTIPLE FIBER AND PARTICULATE COMPONENTS

Most head-loss testing has been performed with a single type of fibrous debris, e.g., NUKON™, and particulates such as CPs. However, plant-specific analyses may well postulate debris beds containing more than one type of fiber and several types of particulate. The application of the NUREG/CR-6224 correlation requires the head-loss properties for the mixture to be estimated from the individual species properties.

V.3.1 Mixture of Specific Surface Areas

The equation for the mixture of the specific surface areas simply multiplies each area by the species volume and sums these products to get the total surface area, which is then divided by the total volume to get the mixture-average specific surface area. Such an equation was recommended in NUREG/CR-6371. Section 3.7.2.3.1.1 of the NEI guidance on the mixing equation* recommends using the square of the specific surface area rather than the linear

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*The NEI guidance refers to NUREG/CR-6371 as the source of their recommendation; however, NUREG/CR-6371 recommends using the linear, not the square of the area in the mixing. The NEI source for the squaring equation has not been provided for review.

relationship The following equation for the mixing is set up to accommodate the linear (n = 1), the square (n = 2), or any other exponent. Performing example mixing evaluations demonstrated that using the square results in larger values for the mixture of specific surface areas than does using the linear relationship; therefore, it is conservative to use the square of the specific surface area in the mixing rather than the linear.

$$Sv_{Mixture} = \left[\frac{\sum_i \frac{m_i}{\rho_i} Sv_i^n}{\sum_i \frac{m_i}{\rho_i}} \right]^{\frac{1}{n}} ,$$

where

Sv = the specific surface area for component *i* or for the mixture,

m_i = the mass of component *i*,

ρ_i = the material (solid) density of the particles in component *i*, and

n = the weighting exponent.

For the surrogate latent particulate debris, mixing the three constituents to get the recipe test result seemed to work best using an n = 4/3 (assuming that ~40% of the fine dirt did not filter from the flow). Because of the substantial uncertainties associated with head-loss predictions, it is prudent to include a safety factor; therefore, the NEI recommendation of using the square of the specific surface area in the mixing equation is a good recommendation.

V.3.2 Mixture Densities

The equation for the mixture of densities (bulk, material, or granular) simply adds all of the species masses and then divides by the total of the species volumes as

$$\rho_{Mixture} = \frac{\sum_i m_i}{\sum_i \frac{m_i}{\rho_i}} ,$$

where

ρ_i = the density of the particles in component *i* and

m_i = the mass of component *i*.

This density mixing equation can be reduced to the following, even simpler form:

$$\frac{1}{\rho_{Mixture}} = \sum_i \frac{f_i}{\rho_i} ,$$

where

f_i = the mass fraction of component i.

V.4 PROCEDURES FOR APPLYING THE NUREG/CR-6224 CORRELATION

The application of the NUREG/CR-6224 head loss correlation requires several input parameters that must be conservatively specified to ensure bounding head loss predictions. The most reliable method of determining these input parameters is the application of the correlation to appropriate head loss test data. Analytical determinations are suitable under some conditions if sufficient conservatism is used throughout the determination. Although the correlation was developed for flat screen geometries, the correlation has been successfully applied to other strainer geometries such as the stacked-disk strainers.

V.4.1 Experimental Determination of Correlation Parameters

The proper application of the NUREG/CR-6224 correlation to applicable head loss test data leads to input parameters that ensure bounding head loss prediction when the correlation is applied to postulated plant conditions that would form debris beds similar to those in the tests. The closer the test data is to the postulated debris beds the more certain the determination of the input parameters. Appropriate conservatism is required whenever the test data is dissimilar to the postulated conditions.

V.4.1.1 Success Criteria for Applicable Test Data

The assumptions associated with the development of the NUREG/CR-6224 correlation included:

1. The debris bed consists of fibrous debris with or without particulate debris.
2. The debris bed has a uniform thickness.
3. The debris bed is homogeneous.
4. The flow approach velocity is perpendicular to the debris bed.
5. The flow and debris accumulation on the screen are relatively quasi-steady-state.

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Therefore, the success criterion for applicable test data to determine applicable correlation input parameters includes:

1. The test debris bed consists of some mixture of fibrous debris with or without particulate debris.
2. The debris bed must be relatively uniform.
3. The debris bed must be relatively homogenous.
4. The approach velocity must be perpendicular to the flow.
5. The debris accumulation, flow rate through the debris bed, the temperature, and the measured pressure differential across the bed must be relatively steady.
6. The quantities of debris in the bed must be known.

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When tests are conducted, care must be taken to minimize edge effects where a portion of the flow can leak through an edge gap between the debris bed and the test chamber. Flow that bypasses the debris bed, such as edge leakage or holes penetrating the debris bed reduce the debris bed flow velocity below that deduced from the flow instrumentation. A non-uniform bed can have shallow locations where water preferentially flows through the bed thereby reducing the measured head loss.

Typically, in head loss testing, debris is introduced into a closed loop test apparatus and then allowed to settle onto the test screen. Some debris, particularly the particulate can penetrate the screen subsequently returning to the screen after transiting the flow loop. The gradual filtration process of the finer particulate causes the pressure differential measurements to be initially transient. Therefore, sufficient time must be allowed to let the filtration become relatively steady-state before recording the test data point. The filtration process is such that the finest of the particulate is the most difficult to get to filter completely out of the flow but the finer particulate has a greater impact on head loss on per mass basic than does the coarser particulate. The finest of the particulate might not be filterable under some conditions. All these considerations are taken into account when assessing the quality of the head loss test data for application to determining correlation input parameters.

V.4.1.2 Parameter Deduction

The input parameters required by the NUREG/CR-6224 correlation include:

1. Debris quantities
    a. Quantity of fibrous debris expressed as the thickness of the debris on the screen assuming its nominal density before destruction (referred to as as-manufactured density). Note that this is equivalent to specifying its mass.
    b. Mass of particulate debris.
2. Flow approach velocity
3. Temperature-dependent water properties
    a. Viscosity
    b. Density
4. Material specific surface areas
    a. Fibrous debris
    b. Particulate debris
5. Densities
    a. Material density of fibers in the fibrous debris
    b. Material density of the particulate
    c. As-manufactured density of the fibrous insulation.
    d. Sludge density of the particulate (also referred to as the granular density or packing limit density)
6. Compression function coefficients (e.g., 1.3 and 0.38 for NUKON).

The experimental determination of a set of parameters for a specific debris bed would be performed along the following steps.

1. Select the appropriate head loss test for each particulate parameter determination. When applying the correlation to the data from a particular test, the test parameters specify the approach velocity, the quantities of debris, and the temperature.

2. Determine a set of densities for the debris bed test data. Manufacturer's data can often supply the densities but if that data is not readily available, volume displacements for measured masses of debris can determine densities for typical debris. Bulk densities are determined from bulk displacements and material densities from water displacement.

3. If possible, experimentally evaluate the compression function coefficients from test data where the particular fibrous debris is the only debris in the bed and the bed is thick

enough to allow reasonable thickness measurements as a variety of velocities and bed thicknesses. The coefficients can be determined from statistical analysis of the thickness data. If applicable thickness test data is not available, initially assume the coefficients validated for NUKON™ (i.e., $\alpha=1.3$ and $\gamma=0.38$).

4. With these other parameters determined, as discussed, the remaining parameters are the specific surface areas for the fibrous and particulate debris. Starting with a fibrous debris bed without any particulate, adjust the specific surface area until the correlation reasonably bounds the data. The resultant specific surface area then applies to that particular fibrous debris. Note that other uncertainties are subsumed into the specific surface area.

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5. With the specific surface area for the fibrous debris determined, another test(s) is selected that uses that fibrous debris but also has the particulate under study. The specific surface area of particulate is adjusted until the correlation reasonably bounds the data. This specific surface then represents the specific surface area for the particulate.

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The above procedure has developed a set of parameters from a set of tests. The quality of the recommended parameters is greatly improved by simulating as many tests as reasonably possible because the resultant parameters will vary somewhat from test to test. If the NUKON™ compression coefficients were initially assumed and do not reasonably apply to the fibrous debris in question, then the data analysis may need to vary these parameters. An example is the lead alpha coefficient (as proposed in GR Section 3.7.2.3.1.1) in steps 3, 4, and 5 in an attempt to get the parameter deductions from the specific tests to align into coherent set of parameters.

The evaluation should include thin-bed head loss tests, as well as, mixed bed tests (i.e., the granular packing limit compression was not reached). The filtration efficiency can increase substantially when the flow must pass through a granular bed as opposed to a fibrous bed due to the reduced porosity. It is desirable to ensure that the worst case particulate filtration is considered in the determination of the particulate specific surface area.

V.4.1.3 Parameter Recommendations (bounding)

The recommended correlation input parameters should ensure that the most severe head losses associated with a particulate type of debris bed are conservative enough to ensure a bound prediction of the head loss associated with a particular postulated bed of debris. The recommendation should consider the uncertainties associated with head loss testing, e.g., non-uniformities in the test debris bed could have reduced the measured head loss below that which would have been measured if the bed had been truly uniform. Other considerations include the potential variability in manufacturing processes of the debris. If for example, parameters are recommended for calcium silicate debris, then it can be expected that those parameters will likely be universally used for any calcium silicate debris calculation. However, the recommendations should include a built in safety factor because the manufacturing of calcium silicate varies with manufacturer and even by a single manufacturer from one batch to another.

V.4.1.4 Ranges of Validated Parameters

The NUREG/CR-6224 correlation was developed and initially validated to support the resolution of the BWR strainer blockage issue. This development focused and validation on NUKON™

fibrous debris and iron oxide corrosion products. The insulation in the volunteer plant was NUKON™, hence the focus was on NUKON™. Form all BWRs, the dominant form of particulate was the corrosion products that formed and collected in the suppression pools. Therefore, the baseline validation compared the correlation results to head loss tests using these two types of debris. In addition to the baseline, other validations were performed using other types of materials. A lesser amount of corrosion products is expected in a PWR containment than in a BWR containment. Therefore, the more likely particulates in a PWR containment will be latent particulate, coatings debris, and particulate insulation debris (e.g., calcium silicate).

Over the years, many analyses have applied the correlation to head loss test data over various ranges of test data. These test programs typically explored the head loss until a judgment was made that the test encompassed the parameter space needed for a particular application. The maximum head loss tested were typically not larger than approximately 25 ft-water primarily due to the limits of the test apparatus, which is generally sufficient more most applications.

Because most test apparatus were constructed of materials that were not able to reliably withstand the higher temperatures expected in a post-LOCA sump pool, the available test data does not extend the range of postulated sump temperatures. However, because the data on the effect of temperature-dependent water viscosity and density are available it has been deemed acceptable to test at lower temperatures and then analytically extend calculations into the higher temperatures. However, this recommendation does not necessarily include the potential for debris decomposition at higher temperatures, which in some tests was factored into the tests by pre-aging the debris using techniques such as boiling the debris for a period to break down the binder. For other parameters, the correlation is not validated beyond the ranges of the test parameters tested. Care must be taken, in reviewing the data to ensure that a significant gap in data does not exist within the validation range at a parameter that significantly affects the current application.

Specific validations for screens that function effectively as flat plates are listed in Table V-5 and V-6 for fibrous and particulate debris, respectively. Validations that involved special geometries are discuss in Section V.4-2.

Table V-5. Validation Ranges for Fibrous Insulation Debris

Debris Type	**Velocity (fps)**	**Temperature (°F)**	**Debris Bed Thickness (in)**	**Comments**	**References**
NUKON™	0.15 to 1.5	60 to 125	1/8 to 4.5		NUREG/CR-6224 NUREG/CR-6367 NEA/CSNI/R (95)11 LA-UR-04-1227 SER Appendix V
Kaowool	0.3 to 0.62	~85			SER Appendix V

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Transco Thermal-Wrap	0 to 0.5	129			NEA/CSNI/R (95)11
Mineral Wool	0 to 0.23	55 to 131			

Table V-6. Validation Ranges for Particulate Insulation Debris

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Debris Type	**Velocity (fps)**	**Temperature (°F)**	**Particulate to Fiber Mass Ratio**	**Comments**	**References**
Iron Oxide Corrosion Products	0.15 to 1.5	60 to 125	0 to 30	With 0 to 2-in bed of NUKON™ or Kaowool	NUREG/CR-6224 NUREG/CR-6367 NEA/CSNI/R (95)11 SER Appendix V
Calcium Silicate	0.1 to 0.8	70 to 140	0.5	With NUKON™ 1/10 to 1.6 in Few test to mass ratio up to 2	LA-UR-04-1227
Latent Particulate (surrogate)		70 to 140	1 to 40	With NUKON™ 0.2 to 2.3 inch	LA-UR-04-3970 SER Appendix V
Min-K			< 0.2		NEI Guidance

V.4.2 Analytical Determination of Correlation Parameters

When test data is not available, the specific surface area may be calculated for some materials. For fibrous debris of low-density fiberglass in which the fibers have relatively uniform cross sections, e.g., NUKON™, the specific surface area can be reasonably estimated using 4/d (where d is the diameter of fibers). The extension of this relationship to fibrous insulations with fine fibers such as mineral wool has not been documented. Some evaluation is needed to generally accept the specific surface area of fiber is equivalent to 4/d. Until such a demonstration is documented, a significant safety factor should be factored into the specific surface area estimate to compensate the uncertainties.

The specific surface area particulate can be calculated using 6/d. For the NEI guidance assumption that coatings debris forms 10-micron particulate, the use of 6/d is appropriate because the particle all have the same diameter. However, for a realistic particulate, the particle sizes vary over a wide range, typically from sub-micron to a few millimeters. If the size distribution is known in fine detail, a reasonable specific surface area can be estimated but typically, the size distribution is specified by mass fractions associated with only 3 or 4 size groups. The latent debris discussed in Section V.2.2 of this report is an example of coarse size distributions. Because the smaller particles contribute substantially more to the specific surface area than the coarser particles, using the mid-range size of each grouping results in estimates of specific area which is significantly smaller than the actual. Using the smaller diameter of each size group to calculate the specific surface area would result in a conservative estimate. However, the smallest diameter of the smallest size group, which corresponds to the finest particulate that can be filtered by the debris bed, depends upon several other factors. If the smallest particle size is estimated too small, the resultant specific surface area can become significantly larger than actual leading to overly conservative head loss estimates.

When applying the specific surface areas for the latent debris specific surface areas given in Section V.2.2, the value of 106,000 ft^{-1} applies to the entire recipe. If analytical refinement in a plant-specific analysis seeks to reduce the transport such that the larger particulate is assumed to not transport to the sump screens, thereby reducing the particulate mass in the debris bed, the 106,000 ft^{-1} specific area no longer apples. If for example, only the particulate of diameter less than 75 micron is assumed to reach the screens, the appropriate specific surface area would be 285,000 ft^{-1}.

The above discussion on using 6/d to calculate the specific surface area of particulate applies to hardened particulate that does not change shape under debris bed pressures. For particulate consisting of materials that can deform (e.g., calcium silicate) special care must be taken because the 6/d specific area may not adequately represent the particulate behavior that has been demonstrated that causes the high head loss associated with calcium silicate.

V.4.3 Application to Special Strainer Geometries

The application of the NUREG/CR-6224 correlation to special strainer geometries for which the stacked disk strainer is given in Section 7.3.2.2 of NUREG/CR-6808. Several full scale or prototype scale test program have been performed where the application has been validated.

V.4.3.1 Beginning and Ending Strainer Conditions

The correlation can be applied to the initial debris loading on these strainer designs by using the total screen area and the appropriate input parameter determined from flat screen head loss testing or other means as discussed above. Then the correlation can be applied to the fully engulfed debris loading by assuming what has been referred to as the circumscribed screen area, which neglects the screen area within the gaps that has been completely filled with debris. In between, many analyses have assumed a linear extrapolation between the end conditions. Another alternative is discussed next.

V.4.3.2 Experimentally Determined Effective Strainer Areas

The NUREG/CR-6224 correlation is applied to the head loss test data using appropriate input parameter determined from flat screen head loss testing or other means as discussed above. Developing the correlation to fit the data, which has a range of debris loading (ranging from a clean screen to a fully engulfed screen) involves making a plot of the effective screen area versus debris loading. This plot can then be used to determine head losses for the design as a function of debris loading.

V.4 REFERENCES FOR APPENDIX V

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[LA-UR-04-3970, 2004a] Mei Ding, Amr Abdel-Fattah, Stewart Fischer, Bruce Letellier, Kerry Howe, Janet Garcia, and Clint Shaffer, "Characterization of Latent Debris from Pressurized-Water-Reactor Containment Buildings," Los Alamos National Laboratory report LA-UR-04-3970 (June 2004).

[LA-UR-04-1227, 2004b] C. J. Shaffer, M. T. Leonard, B. C. Letellier, J. D. Madrid, A. K. Maji, K. Howe, A. Ghosh, and J. Garcia, "GSI-191: Experimental Studies of Loss-of-Coolant-Accident-Generated Debris Accumulation and Head Loss with Emphasis on the Effects of Calcium Silicate Insulation," Los Alamos National Laboratory report LA-UR-04-1227 (April 2004).

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APPENDIX VI: DETAILED BLOWDOWN/WASHDOWN TRANSPORT ANALYSIS FOR PRESSURIZED-WATER-REACTOR VOLUNTEER PLANT

VI.1 INTRODUCTION

In the event of a loss-of-coolant accident (LOCA) within the containment of a pressurized-water reactor (PWR), piping thermal insulation and other materials in the vicinity of the break will be dislodged by break-jet impingement. A fraction of this fragmented and dislodged insulation and other materials, such as chips of paint, paint particulates, and concrete dust, will be transported to the containment floor by the steam/water flows induced by the break and by the containment sprays (CSs). Some of this debris eventually will be transported to and will accumulate on the recirculation sump suction screens. Debris accumulation on the sump screen may challenge the sump's ability to provide adequate, long-term cooling water to the emergency core-cooling system (ECCS) and to the CS pumps. The Generic Safety Issue (GSI)-191 study titled "Assessment of Debris Accumulation on PWR Sump Performance" addresses the issue of debris generation, transport, and accumulation on the PWR sump screen and its subsequent impact on ECCS performance. The purpose of the GSI-191 study is to determine whether debris accumulation in containment following a postulated LOCA would prevent or impede the performance of the ECCS. Los Alamos National Laboratory (LANL) has been supporting the United States (US) Nuclear Regulatory Commission (NRC) in the resolution of GSI-191.

Analytical studies were performed and small-scale experimental programs [NUREG/CR-6772, 2002, NUREG/CR-6773, 2002] were conducted to support the resolution of GSI-191. A parametric evaluation of the US PWR plants demonstrated that potential sump-screen blockage was a plausible concern for operating PWRs [NUREG/CR-6762, Vol. 2, 2002]. As part of the GSI-191 study, a US PWR plant was volunteered and selected for a detailed analysis to develop and demonstrate a methodology for estimating the debris-transport fractions within PWR containments using plant-specific data. This report documents the blowdown and washdown transport portion of the study, describes the methodology, and provides an estimate for the transport of debris from its points of origin to the sump pool. The transport analysis consisted of (1) blowdown debris transport, where the effluences from a high-energy pipe break would destroy insulation near the break and then transport that debris throughout the containment; and (2) washdown debris transport caused by the operation of the CSs. Along the debris-transport pathways, substantial quantities of debris came into contact with containment structures and equipment where that debris could be retained, thereby preventing further transport. The blowdown/washdown debris-transport analysis provides the source term for the subsequent sump-pool debris-transport analysis.

The volunteer plant has a large, dry, cylindrical containment with a hemispherical dome constructed of steel-lined reinforced concrete and having a free volume of ~3 million ft^3. The nuclear steam supply system is a Westinghouse reactor with four steam generators (SGs). Each of the SGs is housed in a separate compartment that vents upward into the dome. Approximately two-thirds of the free space within the containment is located in the upper dome region, which is relatively free of equipment. The lower part of the containment is compartmentalized. The internal structures are supported independently so that a circumferential gap exists between the internal structures and the steel containment liner. Numerous pathways, including the circumferential gap, interconnect the lower compartments. The CS system has spray train headers at four different levels; however, ~70% of the spray nozzles are located in the upper dome. Some spaces in the lower levels are not sprayed by the spray system; therefore, areas of significant size exist where debris washdown by the sprays would not occur. The sprays activate when the containment pressure exceeds 18.2 psig. If the sprays do not activate, debris washdown likely would be minimal. The insulation composition for the volunteer plant is ~13% fiberglass, 86% reflective metal insulation (RMI), and 1% Min-K insulation. For the purposes of this study, it was assumed that the fiberglass insulation was one of the low-density fiberglass (LDFG) types. For plant-specific analyses, these transport results for fibrous debris may have to be adjusted to compensate if the fiberglass insulation makeup is determined to be significantly different.

The effluences from a high-energy pipe break not only would destroy insulation near the break but also would transport that debris throughout the containment (i.e., blowdown debris transport). Substantial amounts of this airborne* debris would come into contact with containment structures and equipment and would be deposited onto these surfaces. As depressurization flows slow, debris would settle gravitationally onto equipment and floors. If pressurization of the containment were to occur, the CSs would activate to suppress pressurization. These sprays would tend to wash out remaining airborne debris (except in areas not covered by the sprays), and the impact of these sprays onto surfaces and the subsequent drainage of the accumulated water would wash deposited debris downward toward the sump pool (i.e., washdown debris transport). In addition, CSs could degrade certain types of insulation debris further through the process of erosion, thereby creating even more of the fine transportable debris.

An assessment of the likelihood of blocking the recirculation sump screens requires an estimate of the debris transport from the containment to the sump pool.† The debris transport within the sump pool is analyzed separately from this analysis, but the sump pool analysis requires the quantities of debris and the entry locations and timing as input to that analysis. An objective of this analysis was to develop and demonstrate an effective methodology for estimating the transport of debris from the debris point of origin in the containment down to the sump pool, thereby providing the source term to the sump-pool debris-transport analysis. Applying the methodology to the volunteer plant generated plausible debris-transport fractions for that plant.

The analyses herein considered only one break location: a LOCA located in one of the SG compartments, which is a probable location for that plant because most of the primary system piping is located in these compartments.

Neither the debris-size distributions nor the overall transport fractions in this report are valid for plant-specific evaluations because these fractions were calculated using LOCA-generated debris-size distributions that did not account properly for PWR jet characteristics. Boiling-water-reactor (BWR) jet characteristics were substituted for PWR jet characteristics because the PWR jet analyses had not yet been performed. When the PWR jet characteristics do become available, the overall transport fractions can be recalculated easily using PWR LOCA-generated debris-size characteristics.

The basic concepts of this methodology are applicable to the assessment of the debris transport within other PWR plants, as well; however, that application depends on the plant-specific aspects of each plant. The complexity of a plant-specific methodology could vary significantly from one plant to the next.

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[*] The terms "airborne" and "airflow" are used loosely with regard to gas flows, which actually consist of both air and steam.

[†] The simplest and most conservative assessment would be to assume 100% transport to the sump pool.

VI.2 DEBRIS-TRANSPORT PHENOMENOLOGY

The transport of debris within a PWR would be influenced both by the spectrum of physical processes and phenomena and by the features of a particular containment design. Because of the violent nature of flows following a LOCA, insulation destruction and subsequent debris transport are rather chaotic processes. For example, a piece of debris could be deposited directly near the sump screen or it could take a much more tortuous path, first going to the dome and then being washed back down to the sump by the sprays. Conversely, a piece of debris could be trapped in any number of locations. Aspects of debris-transport analysis include the characterization of the accident, the design and configuration of the plant, the generation of debris by the break flows, and both air- and water-borne debris dynamics.

Long-term recirculation cooling must operate according to the range of possible accident scenarios. A comprehensive debris-transport study should consider an appropriate selection of these scenarios, as well as all engineered safety features and plant-operating procedures. The maximum debris transport to the screen likely will be determined by a small subset of accident scenarios, but this scenario subset should be determined systematically. Many important debris-transport parameters will be dependent on the accident scenarios. These parameters include the timing of specific phases of the accident (i.e., blowdown, injection, and recirculation phases) and pumping flow rates. The blowdown phase refers to primary-system depressurization. The injection phase corresponds to ECCS injection into the primary system, a process that subsequently establishes the sump pool. The recirculation phase refers to long-term ECCS recirculation.

Many features in nuclear-power-plant containments significantly affect the transport of insulation debris. The dominant break flows will move from the break location toward the pressure suppression system (i.e., the suppression pool in BWR plants and the upper regions of the compartment in PWR plants). Structures such as gratings are placed in the paths of these dominant flows and likely would capture substantial quantities of debris. The lower-compartment geometry—such as the open floor area, ledges, structures, and obstacles—defines the shape and depth of the sump pool and is important in determining the potential for debris to settle in the pool. Furthermore, the relative locations of the sump, LOCA break, and drainage paths from the upper regions to the sump pool are important in determining pool turbulence, which in turn determines whether debris can settle in the pool.

Transport of debris is strongly dependent on the characteristics of the debris that has formed. These characteristics include the types of debris (insulation type, coatings, dust, etc.) and the size distribution and form of the debris. Each type of debris has its own set of physical properties, such as densities, specific surface areas, buoyancy (including dry, wet, or partially wet), and settling velocities in water. Several distinct types of insulation are used in PWR plants [NUREG/CR-6762, Vol. 2, 2002]. The size and form of the debris, in turn, depends on the method of debris formation (e.g., jet impingement, erosion, aging, and latent accumulation). The size and form of the debris affect whether the debris passes through a screen, as well as the transport of the debris to the screen. For example, fibrous debris may consist of individual fibers or of large sections of an insulation blanket and all sizes within these two extremes.

The complete range of thermal-hydraulic processes affects the transport of insulation debris, and the containment thermal-hydraulic response to a LOCA includes most forms of thermal-hydraulic processes. Debris transport is affected by a full spectrum of physical processes, including particle deposition and resuspension for airborne transport and both settling and resuspension within calm and turbulent water pools for both buoyant and nonbuoyant debris.

The dominant debris-capture mechanism in a rapidly moving flow likely would be inertial capture; however, in slower flows, the dominant process likely would be gravitational settling. Much of the debris deposited onto structures likely would be washed off by the CSs or possibly by condensate drainage. Other debris on structures could be subject to erosion.

A panel of experts was convened to identify and rank the important phenomena, processes, and systems in regard to PWR debris transport [LA-UR-99-3371, 1999]. The insights gained from the work of this panel were factored into the analysis methodology. Additionally, all of the experimental and analytical research performed to resolve the BWR strainer-blockage issue was accessed for this analysis [LA-UR-01-1595, 2001; NUREG/CR-6369-1, 1999; NUREG/CR-6369-2, 1999; NUREG/CR-6369-3, 1999]. A summary was published on the base of knowledge for the effect of debris on PWR ECC sump performance [NUREG/CR-6808, 2003].

VI.3 METHODOLOGY

VI.3.1 Overall Description

Transport of LOCA-generated debris from its point of origin to the PWR sump pool is a complex process involving many physical processes and complex plant-specific geometry. To evaluate the blowdown and washdown debris transport within the drywell of a BWR plant, the NRC developed a methodology that accomplished the objectives of the drywell-debris-transport study (DDTS) [NUREG/CR-6369-1, 1999; NUREG/CR-6369-2, 1999; NUREG/CR-6369-3, 1999]. The methodology used herein was based on the BWR methodology.

The BWR methodology separated the overall transport problem into many smaller problems that were either amenable to the solution or that could be judged conservatively. The breakdown of the problem was organized using logic charts that were similar to well-known event-tree analyses. For some solution steps, sufficient data were available to solve that step reasonably. For other steps, insufficient data were available; therefore, the solution had to be found using engineering judgment that was applied after the available knowledge base was reviewed. Judgments were tempered to the desired level of conservatism called for in that particular analysis (sometimes assuming the worst case for a particular step). The result of each specific analysis was a transport fraction, defined as the fraction of insulation contained within the pipe-break destructive zone of influence (ZOI) that subsequently was damaged or destroyed by a LOCA and was eventually transported to the suppression pool. Certainly, the degree of refinement that is feasible depends on available resources and time restraints. Also, the conservatism in the estimates for each step in the divided problem may be compounded when the final transport fraction is quantified.

The PWR debris-transport methodology necessarily will differ from the BWR transport methodology because of differences in plant designs. These differences include the basic transport pathways, dominant capture mechanisms, and the timing of the accident sequence events. The dominant transport pathway for a PWR is different from the dominant pathway for a BWR. In a BWR, where pressure suppression would be due to steam condensation in the suppression pool, the debris initially would be transported directly to the suppression pool, where the ECCS strainers operate. In PWR containments, which are designed to suppress pressurization by channeling break effluences* to the relatively large free volume of PWR containments, debris likely would be blown away from the sump area initially. Because one-half

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*In an ice-condenser plant, the break effluences would be channeled through the ice banks to condense steam.

to three-quarters of the containment free volume typically is located in the upper regions of the containment that includes the dome, it is justified to assume that a significant fraction of the small debris is blown directly into the upper regions, where the debris will settle onto floor surfaces or structures. Although debris blown into the upper regions then could be washed back down to the compartment sump area by the CSs, the washdown pathway can be tortuous and could certainly result in substantial debris entrapment.

The dominant debris-capture locations are different in a PWR than in a BWR. In many typical PWRs, the likely dominant locations are the upper regions of the containment, the ice condensers in an ice-condenser plant, the refueling pool, an outer annulus pool, and the sump pool. In the volunteer-plant containments, dominant locations for debris capture may not exist; rather, the debris likely would be blown throughout the entire containment. Gratings in a PWR could play a substantially different role versus the gratings played in the BWR methodology because the debris likely would be blown up through a grating as opposed to down through a grating. Debris trapped underneath a grating would be less likely to remain there than debris trapped on top of a grating.

Debris transport during the washdown phase would be caused by the water drainages of break recirculation overflow, the CSs, and condensate. The most important of these drainages would be the drainage of the activated CSs because the sprays usually cover a majority of the containment free volume, whereas the break overflow would wash only surfaces directly below the break. In a PWR, the break overflow could impinge on piping and equipment before reaching the containment sump floor, thereby washing debris from these surfaces, as well as potentially dispersing the flow. In a BWR, the break overflow for a majority of postulated breaks would pass down through at least one grating, where the flows would erode larger debris trapped on the gratings directly below the break—a situation less likely in a PWR. Although condensate drainage could transport debris from surfaces, the quantities of debris transported would likely be much less than the quantities transported by spray drainage.

The following methodology was designed specifically to analyze debris transport within the volunteer-plant containments; however, it is also directly applicable to several other containment designs, and it can be modified to tailor the methodology to any other PWR design. The best method for a particular plant will depend on the complexity of the containment design. If the containment has definitive upper and lower compartments that are separated by relatively few and narrow pathways, the analysis may be used to track debris transports in a manner similar to the DDTS analysis. Using an ice-condenser plant as an example, the containments were designed specifically to channel break flow through the ice banks to the dome region. This generally means that the connecting flow pathways between the lower and upper containments include the ice banks, small air-circulation return pathways, (needed to establish post-blowdown air circulation through the ice banks), and refueling-pool drains. Debris capture through the ice banks could be substantial. In addition, a large fraction of the small and fine debris would be blown into the dome region, where substantial quantities could be retained, even with the CSs operating.

The analysis here would focus on debris capture in the ice banks during blowdown and on debris retention in the upper compartment during the spray washdown process to identify debris transported from the lower containment and not likely to return there. Some plants would have a flooded outer annulus in which debris deposited in that pool would be less likely to transport from that pool to the sump pool. A conservative estimate of the maximum debris quantities that would be expected to transport to the sump pool can be made by subtracting masses of debris retained at various locations from the generation totals.

The design of the volunteer-plant containments is more complex than an ice-condenser design, from a debris-transport point of view; that is, the lower and upper regions of the containment are less well defined and are connected by several different pathways, thereby making it difficult to determine the motion of air and steam flows and the transport of debris. Certainly, system-level codes such as MELCOR can model the progression of break flows throughout the containment; however, the input model for the volunteer plant would have to be rather detailed to follow the flows through all of the lower levels in the containment. The modeling detail must include all of the levels and rooms and separate sprayed areas from non-sprayed areas. The model would need to simulate all of the connecting flow pathways, such as stairwells, equipment hatches, and doorways. A detailed thermal-hydraulics analysis was not performed for the volunteer-plant analysis.

The transport and deposition of insulation debris cannot be simulated realistically using a thermal-hydraulics computer code that incorporates aerosol transport models. The primary mode of debris capture during the violent primary-system depressurization is inertial capture. The available models for inertial capture are based on data taken for rather simple geometries (e.g., a bend in a pipe). Inertial capture in the complex geometry of containments cannot be modeled reasonably using current codes. However, inertial capture can be determined in specific parts of the containment. For example, at the volunteer plant, the personnel access doors between an SG compartment and the sump annulus have at least one 90° bend. A LOCA, particularly a large LOCA, in an SG compartment would result in depressurization flows that would carry insulation debris through these doors with the flow. As the flow underwent the sharp bend, some types of debris would be deposited by inertia on the wall at the bend. The tests conducted at the Colorado Engineering Experiment Station, Inc. (CEESI) demonstrated an average inertial capture fraction for fibrous debris of 17% at such a bend if the surface were wetted, and analysis has shown that surfaces within the containment likely would build a filmy layer of condensation rapidly. Because the CSs do not impinge on these wall surfaces, the debris would remain attached to those surfaces. In this situation, small amounts of debris can be removed from the equation, thereby lowering the transport fraction. Perhaps many of these types of definable captures can add up to a significant reduction in the transport fraction. Again, the size of that reduction would depend somewhat on both the geometry/conditions and the depth of the analysis.

The basic idea of the mechanics of this methodology is to look for such reductions systematically. The demonstration of this methodology in this volunteer-plant analysis assumed a large LOCA occurred inside SG compartment number 1 (SG1) of the containment. Figure IX-1 illustrates this methodology in the general sense. The idea is first to estimate the blowdown dispersal of the debris until all of the debris is associated with some surface area. Then the likelihood of debris remaining on each of these surfaces during washdown is estimated or judged. For example, debris deposited onto a surface that has been impacted by the CSs is much more likely to transport than debris deposited onto surfaces that have been wetted only by condensate.

As with the DDTS, the debris for transport must first be categorized according to type and size according to transport properties so that the transport of each type of debris can be analyzed independently. All insulation located within the break-region ZOI is assumed to be damaged to some extent. These categories and their properties are the subject of Section VI.3.2.

The containment free volume in the volunteer plant was subdivided into many regions based on geometry and the locations of the CSs. The volume region containing the postulated LOCA was analyzed separately and first. For SG1, a MELCOR simulation of only the break compartment was used to determine the distribution of flows exiting that compartment (i.e., the fraction of flow going upward into the dome as opposed to the fraction entering the lower levels through personnel access doors). Debris capture within SG1 was based on such considerations as flows through gratings and flows making sharp bends (see Section VI.3.3.1). In each region, debris capture would deposit debris onto the "floor" or "other" surfaces, based on surface areas and judgment regarding whether debris was deposited by settling or by another mechanism. Floor surfaces were treated separately because these surfaces would collect and drain spray water differently from vertical surfaces, for example, and because debris that gravitationally settles would deposit onto horizontal surfaces. These surfaces were divided further according to their exposure to spray and condensate moisture. All surfaces would collect condensate. The sprays would impact some surfaces directly, and others simply would be washed by the process of spray drainage. Debris entrained by spray-drainage water could become captured a second time as the drainage fell from one level to another.

Because the chart illustrated in Figure IX-1 would become unreasonably large if it were developed for the entire volunteer-plant containment, another approach was used. The process was handled using an equation-format model (described in Figure IX-1), with the input entered into data arrays.

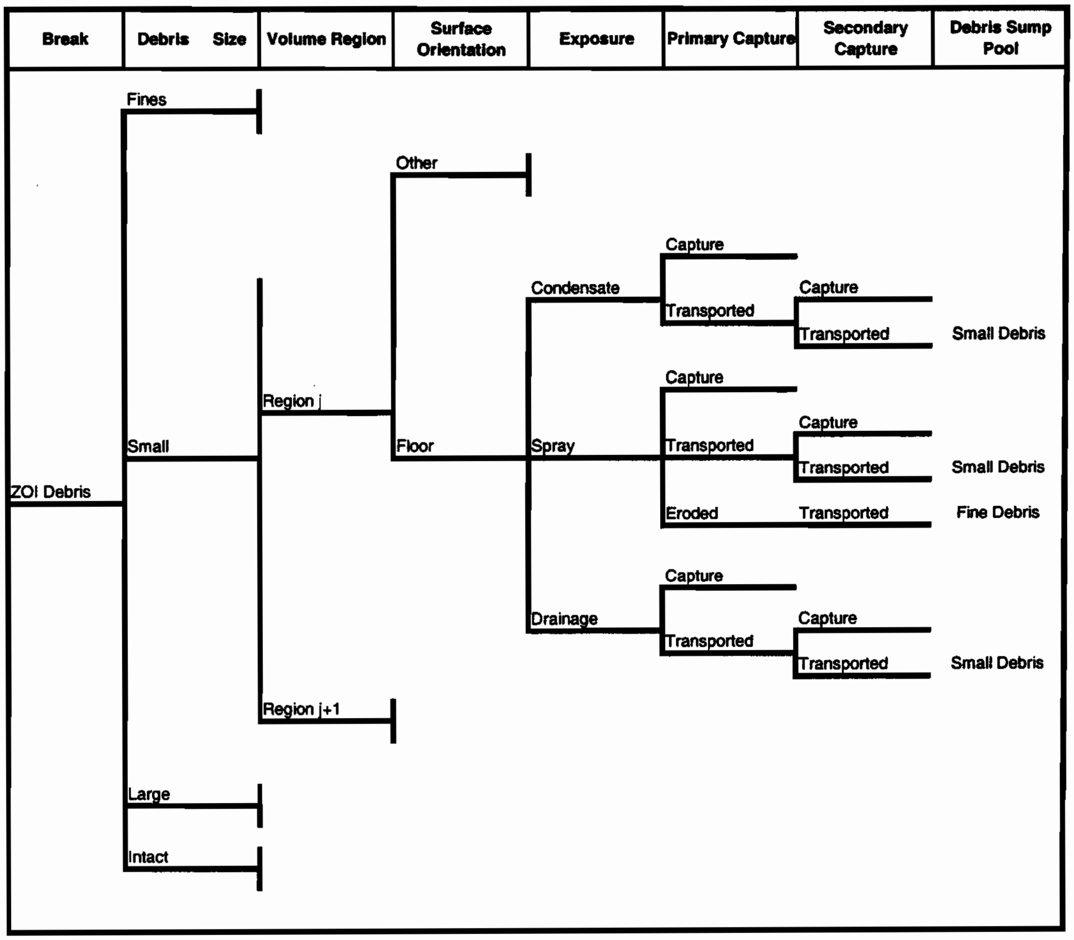

Figure IX-1. Example of a Section of a Debris-Transport Chart.

VI.3.2 Debris-Size Categorization

The types of insulation used inside the volunteer-plant containments include fiberglass insulation,* RMI, and stainless-steel-encapsulated Min-K insulation at ~13.4%, 85.7%, and 0.9%, respectively [NUREG/CR-6762, Vol. 2, 2002]. Although a majority of the insulation within these containments is RMI, the fibrous insulation more likely would cause blockage of the sump. First of all, the RMI debris would transport less easily than the fibrous debris (i.e., it takes a faster flow of water to move RMI debris than it does for fibrous debris). In addition, it takes substantially more RMI debris on the sump screens to block the flow effectively through the screens than it does for fibrous debris. Although the Min-K debris, in combination with the fibrous debris, could create substantial head losses on the screen, the inventory of the Min-K in the containments is relatively low. Therefore, the primary focus in this analysis was on the transport of fibrous debris, with the transport of RMI and Min-K estimated more crudely.

The difficulties associated with determining debris-size distributions to represent the LOCA-generated debris are (1) the limited debris-generation data and (2) the need to determine the characteristics of the LOCA jet (i.e., the size of the ZOI and volumes within specific pressure isobars). The limitations in the debris-generation data must be handled by skewing the integration of size fractions conservatively over the ZOI toward the smaller debris sizes; the more limited the data, the more conservative the integration. The determination of the jet characteristics for a PWR jet is a relatively straightforward analysis; but those characteristics unfortunately were not yet available for use in this report. Because, debris-size distributions are necessary to determine estimates for the overall transport of debris to the sump pool, assumptions were made to provide distributions that were suitable to illustrate the transport methodology. Therefore:

Neither the debris-size distributions nor the overall transport fractions in this report are valid for plant-specific evaluations.

However, the transport fractions for specific debris-size classes are considered to be valid for the volunteer plant.

VI.3.2.1 Fibrous Insulation Debris-Size Categorization

All insulation located within the break-region ZOI is assumed to be damaged to some extent. The damage could range from slight damage (insulation erosion occurring through a rip in the blanket cover) that leaves the blanket attached to its piping to the total destruction of a blanket (with its insulation reduced to small or very fine debris). For the purposes of this analysis, all of the insulation within the ZOI was considered to be debris. The fibrous debris was categorized into one of four categories based on transport properties so that the transport of each type of debris could be analyzed independently. These categories and their properties are shown in Table IX-1.

The primary difference between the two smaller and two larger categories was whether the debris was likely to pass through a grating that is typical of those found in nuclear power plants. This criterion also was used in the DDTS analysis. Thus, fines and small pieces pass through gratings but large and intact pieces do not. The fines and small pieces are much more

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*The type (or types) of fiberglass insulation used in the volunteer-plant containments has yet to be determined. This analysis assumes that the fiberglass is LDFG.

transportable than the large debris. The fines were then distinguished from the small pieces because the fines would tend to remain in suspension in the sump pool, even under relatively quiescent conditions, whereas the small pieces would tend to sink. Furthermore, the fines tended to transport slightly more as an aerosol in the containment-air/steam flows and were slower to settle than the small pieces when airflow turbulence decreased. The CEESI tests illustrated that when an LDFG blanket was completely destroyed, 15% to 25% of the insulation was in the form of very fine debris (i.e., debris too fine to collect readily by hand).

The distinguishing difference between the large and intact debris was whether the blanket covering was still protecting the fibrous insulation. The primary reason for this distinction was whether the CSs could further erode the insulation material.

Table IX-1. Debris-Size Categories and Their Capture and Retention Properties

Fraction Variable	Size	Description	Airborne Behavior	Waterborne Behavior	Debris-Capture Mechanisms	Requirements for Crediting Retention
D_F	Fines	Individual fibers or small groups of fibers.	Readily moves with airflows and slow to settle out of air, even after completion of blowdown.	Easily remains suspended in water, even relatively quiescent water.	Inertial impaction Diffusiophoresis Diffusion Gravitational settling Spray washout	Must be deposited onto surface that is not subsequently subjected to CSs or to spray drainage. Natural-circulation airflow likely will transport residual airborne debris into a sprayed region. Retention in quiescent pools without significant flow through the pool may be possible.
D_S	Small Pieces	Pieces of debris that easily pass through gratings.	Readily moves with depressurization airflows and tends to settle out when airflows slow.	Readily sinks in hot water, then transports along the floor when flow velocities and pool turbulence are sufficient. Subject to subsequent erosion by flow water and by turbulent pool agitation.	Inertial impaction Gravitational settling Spray washout	Must be deposited onto surface that is not subsequently subjected to high rates of CSs or to substantial drainage of spray water. Retention in quiescent pools (e.g., reactor cavity). Subject to subsequent erosion.
D_L	Large Pieces	Pieces of debris that do not easily pass through gratings.	Transports with dynamic depressurization flows but generally is stopped by gratings.	Readily sinks in hot water and can transport along the floor at faster flow velocities. Subject to subsequent erosion by flow water and by turbulent pool agitation.	Trapped by structures (e.g., gratings) Gravitational settling	Must be either firmly captured by structure or on a floor where spray drainage and/or pool flow velocities are not sufficient to move the object. Subject to subsequent erosion.

Fraction Variable	Size	Description	Airborne Behavior	Waterborne Behavior	Debris-Capture Mechanisms	Requirements for Crediting Retention
D_I	Intact	Damaged but relatively intact pillows.	Transports with dynamic depressurization flows, stopped by a grating, or may even remain attached to its piping.	Readily sinks in hot water and can transport along the floor at faster flow velocities. Assumed to be still encased in its cover, thereby not subject to significant subsequent erosion by flow water and by turbulent pool agitation.	Trapped by structures (e.g., gratings) Gravitational settling Not detached from piping	Must be either firmly captured by structure or on a floor where spray drainage and/or pool flow velocities are not sufficient to move the object. Intact debris subsequently would not erode because of its encasement.

The volume (or mass) distribution, D_i, of the four categories of insulation debris was estimated first. This estimate assumed that the fibrous insulation within the ZOI was uniformly distributed and that the distribution must add up to one, as

$$\sum_{i=1}^{N_{types}} D_i = 1 \quad ,$$

where

D_i = the fraction of total debris that is type *i*.

The volume of each category of debris is simply the distribution fraction multiplied by the total volume of insulation within the ZOI. In debris-transport analysis, volumes of fibrous debris have been used interchangeably with mass on the basis that the density is that of the undamaged (as fabricated) insulation. Certainly the density would be altered by the destruction of the insulation and again when the debris became water saturated. For example, the physical volume of debris on the screen must include the actual density of the debris on the screen as

$$V_i = D_i V_{ZOI} \quad ,$$

where

V_i = the volume of debris of type *i* and
V_{ZOI} = the total volume of insulation contained within the ZOI.

The estimation of the debris-size distribution must be based on experimental data. When sufficient data are available, the following analytical model illustrates how the fraction of fine and small debris can be estimated from that data. Using the spherical ZOI destruction model, the fraction of the ZOI insulation that becomes type-*i* debris is given by the following integration:

$$F_i = \frac{3}{r_{ZOI}^3} \int_0^{r_{ZOI}} g_i(r)\, r^2 \, dr \quad ,$$

where

F_i = the fraction of debris of type *I*;
$g_i(r)$ = the radial destruction distribution for debris of type *I*;
r = the radius from the break in the spherical ZOI model; and
r_{ZOI} = the outer radius of the ZOI.

Typical test data provide an estimate of the damage to insulation samples at selected distances from the test jet nozzle (i.e., the size distribution of the resultant debris). The jet pressure at the target is determined from test pressure measurements, suitable analytical models, or both. Thus, the size distribution as a function of the jet pressure is obtained. The volume associated with a particular level of destruction is determined by estimating the volume within a particular pressure isobar within the jet [i.e., any insulation located within this pressure isobar would be damaged to the extent (or greater) associated with that pressure]. The isobar volumes then are converted to the equivalent spherical volumes; thus, the debris-size distribution is associated with the spherical radius (i.e., $g_i(r)$). The distribution would be specific to a particular kind of insulation, jacketing, jacketing seam orientation, and banding.

To demonstrate the transport methodology completely, it was assumed that the fibrous insulation used in the volunteer-plant containments was LDFG insulation, for which significant data are available to predict the LOCA-generated size distribution. The most extensive debris-generation data for LDFG insulation are the data from the BWR Owners' Group (BWROG) air-jet impact tests (AJITs) [NEDO-32686, 1996]. These data, combined with the jet characteristics of a PWR LOCA, could result in a realistic LOCA size distribution; however, the PWR jet characteristics were not available at the time of this writing.

The development of a suitable size distribution for the purposes of demonstrating this methodology follows. For fibrous debris, the BWROG correlated the fraction of the original insulation that became fine debris with the distance from the jet nozzle and then crudely estimated the ZOI destruction fractions for specific types of insulation. The fine debris in the BWROG analysis correlates with the combined fine and small debris of Table IX-1.

For the NUKON™ insulation debris—both jacketed and unjacketed insulation—the BWROG recommended in its Utility Resolution Guidance (URG) the assumption that 23% of the insulation within the ZOI be considered in the strainer head-loss evaluations during the resolution of the BWR strainer blockage issue. Applying this recommendation to this analysis means that 23% of the ZOI would be distributed between the fine and small debris and that the remaining 77% would be distributed between the large and intact debris. The NRC reviewed the BWROG recommendations and documented its findings in a safety evaluation report (SER) [NRC-SER-URG, 1998]. Although the NRC had some reservations regarding the BWROG's method for determining the debris fractions, the NRC believed the debris fractions to be conservative primarily because the blanket seams were arranged in the AJITs to maximize the destruction of the blankets.

Whereas the BWROG's recommendations were based on AJITs, more recent testing using two-phase jet impact testing indicated the need for somewhat higher small-debris fractions than did the AJIT data (refer to the staff evaluation of GR, Section 3.4.2.2 in this report, for the evaluation of the two-phase jet concern). Ontario Power Generation (OPG) of Canada conducted these debris-generation tests [OPG, 2001]. A comparison of the AJIT and the OPG tests was discussed in a report [NUREG/CR-6762, Vol. 3, 2002] that was supporting the PWR parametric evaluation [NUREG/CR-6762, Vol. 1, 2002]. This comparison illustrated the potential for more small debris to be generated by a two-phase jet than was supported by the AJIT data. For the parametric evaluation, the qualitative conclusion of comparing these two sets of test data was that the small debris fraction should be increased from the BWROG's recommendation. An engineering judgment was used to increase the recommended destruction fraction for small debris from 23% to 33%. The remaining 67% of the insulation would be assumed to be large debris either exposed or enclosed in its covering material.

For this analysis, the small-debris fraction of 33% that was used in the parametric evaluation was split to accommodate the fine- and small-debris categories of this analysis. The analysis of the AJIT testing performed at CEESI to support the DDTS determined that whenever entire blankets were completely destroyed, 15% to 25% of the insulation was too fine to collect by hand.* Complete destruction here means that nearly all of the insulation was either fine or small pieces. In any case, the 15% to 25% of the blanket (an average of 20%) can be considered fine debris for the purposes of this analysis. For this analysis, it was assumed that 20% of the 33%-small-debris fraction was fine debris (i.e., 0.2 × 0.33 = 0.066). Therefore, 7% of the ZOI insulation was estimated to be destroyed into fine debris, leaving 26% for the small-piece debris.

In a similar manner, the parametric evaluation of the 67%-large-debris fraction was split in this analysis to accommodate the large and intact debris categories. In DDTS analysis, based on the AJIT data, 40% of the blanket insulation was assumed to remain covered. The DDTS assumption of 40% was accepted for the covered (intact) debris fraction for this analysis. However, that number had to be adjusted downward to account for the increase in the small-debris fraction from 23% to 33% (i.e., 0.67/0.77 × 0.4 = 0.35). Therefore, 35% of the ZOI insulation was considered to be intact debris, leaving 32% for the exposed large-piece debris. The debris category distribution for fibrous debris assumed in this analysis is summarized in Table IX-2.

Table IX-2. Fibrous-Debris-Category Distribution

[illegible]	[illegible]
Fines	7%
Small Pieces	26%
Large Pieces	32%
Intact	35%

VI.3.2.2 RMI Insulation Debris-Size Categorization

In the volunteer-plant containments, the RMI insulation is made of stainless steel. TPI manufactured the insulation around the reactor vessel. Diamond Power Specialty Company (DPSC) manufactured all of the other RMI inside the containments and marketed it as DPSC MIRROR™ insulation. Furthermore, the insulation panels generally are held in place simply by buckling the panels together (i.e., an absence of bands on most panels). Because the reactor vessel insulation is shielded from a postulated jet impingement for the most part, LOCA-generated RMI debris would consist primarily of the DPSC type. The threshold jet-impingement pressure required to damage DPSC MIRROR™ insulation with standard bands was estimated by the BWROG [NEDO-32686, 1996] and accepted by the NRC [NRC-SER-URG, 1998] as 4 psi; these data should be applicable to the volunteer-plant RMI. Therefore, some debris could be formed from any insulation subjected to a differential of 4 psi or greater, but the extent of damage would depend on the magnitude of the pressure. Insulation that is closer to the break would be destroyed completely and form small pieces of debris, whereas insulation farther from the break may remain nearly intact. A size distribution is needed for the transport analysis. Data from two experimental programs provide limited information on the extent of destruction that

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* This debris either was blown through the fine-mesh screen at the end of the test chamber and lost from the facility or was deposited onto surfaces inside the chamber in such a dispersed manner that it could be collected only by hosing down the walls and structures.

would occur in this type of RMI insulation. These programs were (1) the Siemens Karlstein tests [SEA-95-970-01-A:2, 1996], and (2) the BWROG AJIT [NEDO-32686, 1996].

Swedish Nuclear Utilities conducted metallic insulation jet impact tests at the Siemens AG Power Generation Group (KWU) test facility in Karlstein am Main, Germany (1994 thru 1996 can't find these in references). During this test program, the US NRC conducted a single RMI debris-generation test to obtain debris-generation data and debris samples that are representative of RMI used in US plants. The NRC test sample was provided by the DPSC. The NRC-sponsored test was performed with a high-pressure blast of two-phase water/steam flow from a pressurized vessel connected to a target mount by a blowdown line with a double-rupture disk. The target was mounted directly on a device designed to simulate a double-ended guillotine break (DEGB) such that the discharge impinged the inner surface of the RMI target as it would an insulation cassette surrounding a postulated pipe break. Most of the RMI debris was recovered and analyzed with respect to size distribution. The overall size distribution for the total recovered debris mass is shown in Figure IX-2, and a photograph of the recovered RMI debris is shown in Figure IX-3. This debris sample is likely typical of debris formed from the RMI cassettes nearest the break.

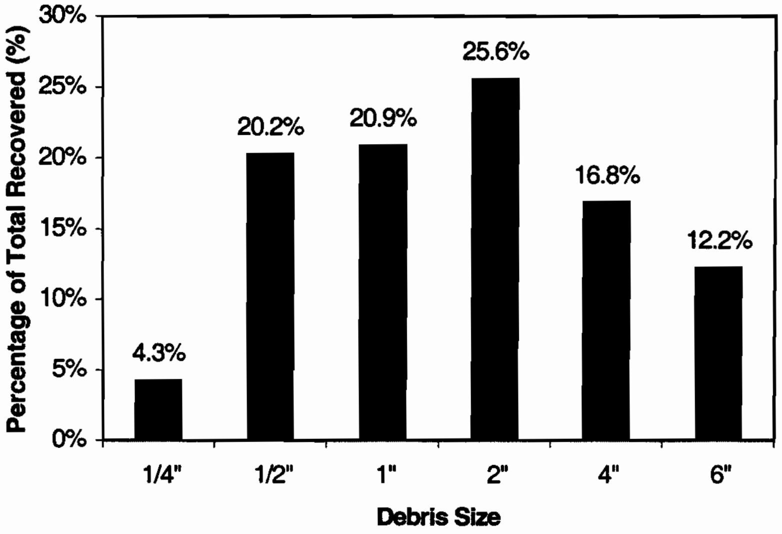

Figure IX-2. Size Distribution of Recovered RMI Debris.

Figure IX-3. RMI Debris Observed in Siemens Steam-Jet Impact Tests.

The BWROG-sponsored tests conducted at CEESI examined the failure characteristics of various types of insulation materials when subjected to jet impingement forces. CEESI has compressed-air facilities that provided choked nozzle airflow. This airflow was directed at insulation samples mounted inside a test chamber that did not pressurize significantly but retained most of the insulation debris for subsequent analysis. The variety of insulation materials tested included samples of the stainless-steel DPSC MIRROR™ insulation. The test samples were mounted at various distances from the nozzle, thereby subjecting similar samples to varying damage pressures. In this manner, the test data were used to estimate the threshold pressure required to damage this type of insulation. The data also provided information regarding the size distribution of the resulting debris. The formation of debris was dependent on the separation of the outer sheath, which in turn depended on the type, number, and placement of the supporting bands. The data used herein were for stainless-steel DPSC MIRROR™ cassettes mounted either with standard bands or without bands; therefore, these data are conservative with respect to data for cassettes mounted with even stronger banding. The recorded debris-generation data separated the quantities of debris into several distinct size groupings. For this transport analysis, the debris was grouped into three size groups: (1) debris generally smaller than 2 in. in size, (2) debris larger than 2 in. but smaller than 6 in., and (3) all RMI pieces larger than 6 in. (including both debris and relatively intact insulation cassettes). Figure IX-4 shows the fractions of the collected debris for the two finer groups as a function of the damage pressure on the cassette; all other insulation either remained relatively intact or formed debris larger than ~6 in.

The BWROG data describe the damage to stainless-steel DPSC MIRROR™ insulation (standard banding) when subjected to jet pressures of up to 120 psi. The NRC-sponsored Siemens test demonstrates the complete destruction of stainless-steel DPSC MIRROR™ insulation when impacted by the highest jet pressure near the break. A gap exists in the data between 120 psi and the higher pressure near the jet. The damage to the RMI within the ZOI was estimated using the spherical equivalent volume method in conjunction with BWR-specific data (i.e., volumes with specific pressure isobars). The BWROG analysis that was provided to the utilities [NEDO-32686, 1996] was used to convert jet isobar volumes to equivalent spherical volumes. Furthermore, the outer radius of the equivalent sphere was assumed to be 12D (i.e., 12 times the diameter of the pipe break), which corresponds to an insulation destruction pressure of 4 psi for a BWR radial offset DEGB. **The resultant size distribution can demonstrate the overall transport methodology fully but is not suitable for PWR plant-**

specific analyses. The BWROG data were applied when the impact pressure was <120 psi; the Siemens data were conservatively applied when the impact pressure was >130 psi (insulation totally destroyed), and a linear extrapolation was applied between 120 and 130 psi. The data shown in Figure IX-2 indicates that when the insulation is totally destroyed, ~70% of the debris would be <~2 in. in size and the remaining 30% would be between 2 and 6 in. in size.

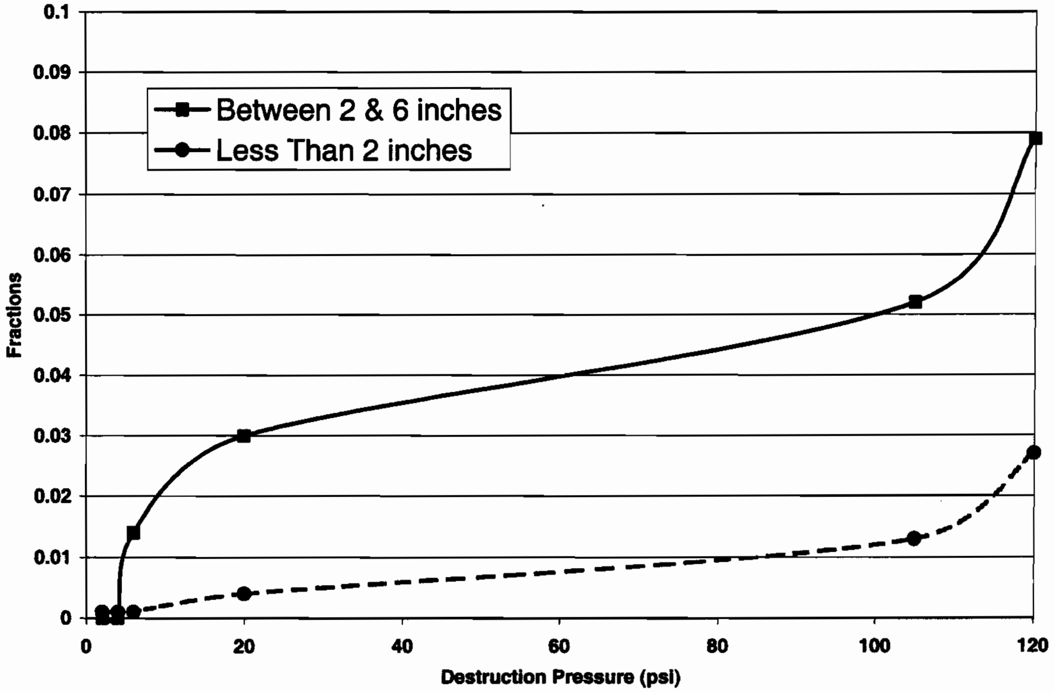

Figure IX-4. Relative Damage of Stainless-Steel DPSC MIRROR™ Insulation.

Because of variability and uncertainty in debris-generation estimates, as well as the use of BWR-specific jet characteristics, it is prudent to enhance the fractions for the finer groups of debris, noting that the smaller debris would transport more easily than would the larger debris. One uncertainty is the fact that the BWROG data were generated using an air jet, whereas the postulated accident would involve a two-phase steam/water jet; the comparison of two-phase and air test data has indicated that a two-phase jet could generate finer debris than could an air jet. To make the debris-generation estimates more conservative to compensate for variability and uncertainty in the estimates, the fractions for the two fines size groups were increased by 50%. The spherical volume damage estimates with and without the 50% increase are shown in Table IX-3.

Table IX-3. RMI Debris Category Distribution

Category	Category Percentage	
	Integration Result	Conservative Estimate
<2 in.	14%	21%
Between 2 and 6 in.	8%	12%
>6 in.	78%	67%

VI.3.2.3 Min-K Insulation Debris-Size Categorization

In locations where insulation thickness was a specific concern, such as pipe-whip-restraint locations, fully encapsulated Min-K insulation was used instead of the usual RMI insulation. Containment-wide, ~0.9% of the insulation is Min-K. Although the potential quantities of Min-K debris would be substantially smaller than corresponding quantities of fibrous or RMI debris, a small amount of Min-K particulate debris could contribute more significantly than RMI debris to sump-screen head loss. In particular, Min-K debris dust would contribute to the particulate load in the debris bed when combined with the fibrous debris on the screens. Min-K is a thermo-ceramic insulation (also referred to as a particulate insulation) that is made of microporous material. The particulate insulations include calcium silicate, asbestos, Unibestos, Microtherm, and gypsum board. Test data have demonstrated that microporous particulate, combined with fibrous debris, creates a debris bed that can cause relatively high head losses across that bed. This head loss is over and above the corresponding head loss associated with more ordinary particulate, such as corrosion products. The most notable of the particulate insulation types has been calcium silicate.

Limited debris-generation data exist for the microporous insulations, and most of the available data were obtained for calcium silicate. No debris-generation data were available for Min-K insulation. The primary source of calcium silicate debris-generation data are test data from tests conducted by the OPG [NUREG/CR-6762, Vol. 3, 2002]. These tests involved impacting aluminum-jacketed calcium silicate insulation targets with a two-phase water/steam jet. The size distribution data are shown in Figure IX-5.

Even if it is assumed that Min-K behaves similarly to calcium silicate with regard to debris generation, the OPG data cover only a limited range of damage pressures. Integrating the damage over the spherical ZOI requires a conservative extrapolation to a full range of pressures. The ZOI for Min-K corresponds to a destruction pressure of 4 psi, based on the BWROG guidance to utilities. At high pressures, the conservative extrapolation should assume that complete destruction of the insulation occurs (i.e., all of the insulation is pulverized to dust). At lower pressures, the damage fractions of the lowest pressures tested would be extended out to the ZOI boundary. This crude conservative extrapolation indicates that about half of the insulation should be considered dust. In addition to the conservative extrapolation, the debris-generation fraction is conservative with respect to the jacket seam angle relative to the jet. The seams in the test data shown in Figure IX-5 were oriented toward maximum damage. In reality, the seams within the ZOI likely would be distributed more randomly with respect to the jet; therefore, many of the jackets would provide more protection for the Min-K than is indicated by the OPG data. On the other hand, applying data for calcium silicate to Min-K insulation introduces substantial uncertainty.

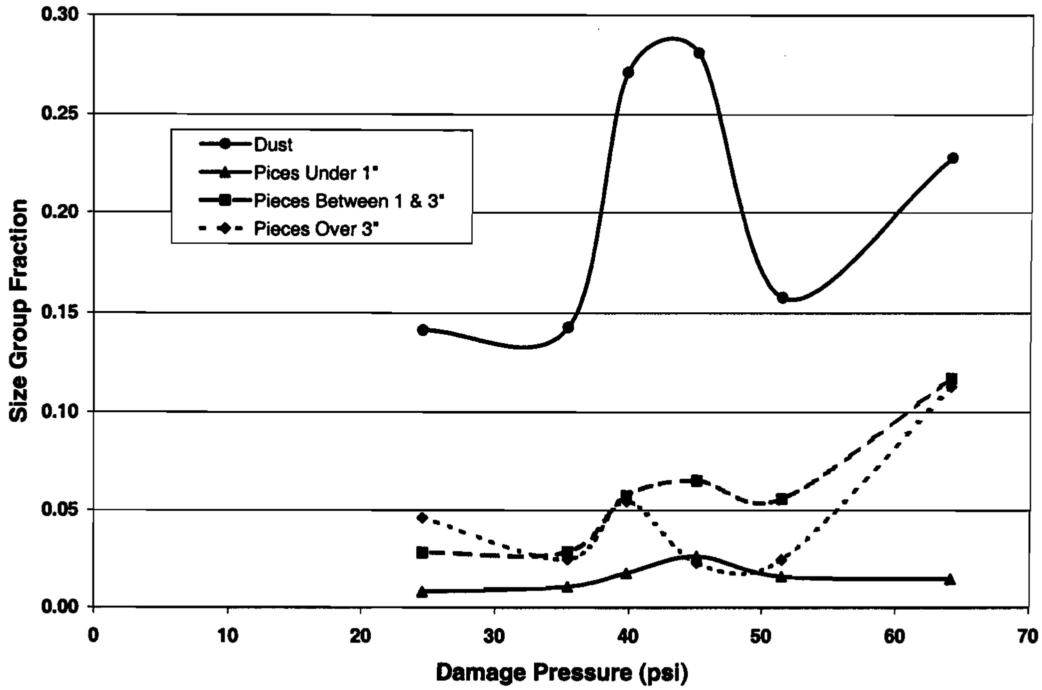

Figure IX-5. Debris-Size Distributions for OPG Calcium Silicate Tests.

Another source of uncertainty is the location of the minimal quantities of Min-K insulation with respect to the break. A key assumption of the ZOI integration is a uniform distribution of insulation within the ZOI. However, with so little Min-K insulation inside the volunteer-plant containments, all damaged Min-K insulation could be located preferentially near or far from the break. Therefore, all Min-K insulation could be destroyed totally or only slightly damaged. Another source of uncertainty that has not been assessed experimentally is the subsequent erosion of the Min-K debris by the CSs. In light of these uncertainties, it is conservative and prudent to assume that all of the Min-K insulation inside a ZOI would be pulverized to dust.

VI.3.3 Blowdown Debris Transport

The break region, SG1, would be the source of all insulation debris and would be subject to the most violent of the containment flows, and the primary debris capture mechanism in this region would be inertial capture. For these reasons, the transport of debris within the region of the pipe break likely should be solved separately from that of the rest of the containment. The methodology is described for fibrous-debris transport but also was applied to RMI debris in a similar manner.

VI.3.3.1 Break-Region Dispersion and Capture

The first step in determining the dispersal of debris near the debris-generation source was to determine the distribution of the break flow from the region—specifically, the fractions of the flow directed to the dome versus other locations. This determination was accomplished using the containment thermal-hydraulics code MELCOR. The containment was designed to force reactor-coolant-system (RCS) break effluents upward through the open tops of the SG compartments and into the dome. Figure IX-6 shows the nodalization diagram for the break-region MELCOR calculation.

The LOCA-generated debris that was not captured within the region of the break would be carried away from the break region by the break flows. The primary capture mechanism near the break would be inertial capture or entrapment by a structure such as a grating. The break-region flow that occurred immediately after the initiation of the break would be much too violent to allow debris simply to settle to the floor of the region.

The inertial capture of fine and small debris occurs when a flow changes directions, such as flows through the doorways from the SG compartments into the sump-level annular space. These flows must make at least one 90° bend through these doorways, and these surfaces would be wetted by steam condensation as well as by the liquid portion of the break effluence. Debris-transport experiments conducted at CEESI [NUREG/CR-6369-2, 1999] demonstrated an average capture fraction of 17% for fine debris and small debris that make a 90° bend at a wetted surface. Other bends in the flow would occur as the break effluents interacted with equipment and walls.

The platform gratings within the SG compartments would capture substantial debris, even though the gratings do not extend across the entire compartment. The CEESI debris-transport tests demonstrated that an average of 28% of the fine and small debris was captured when the airflow passed through the first wetted grating that it encountered and that an average of 24% was captured at the second grating. By definition, the large and intact debris would be trapped completely by a grating. In addition, equipment such as beams and pipes was shown to capture fine and small debris. In the CEESI tests, the structural maze in the test section captured an average of 9% of the debris passing through the maze.

To evaluate the transport and capture within the break region, the evaluation must be separated into many smaller problems that are amenable to resolution. This separation can be accomplished using a logic-chart approach that is similar to the approach developed for the resolution of the BWR-strainer-blockage issue [NUREG/CR-6369-1, 1999]. The chart for a LOCA in the volunteer-plant SG1 is shown in Figure IX-7 and is based on the MELCOR nodalization diagram in Figure IX-6. This chart tracks the progress of small debris from the pipe break (Volume V12) until the debris is assumed to be captured or is transported beyond the compartment. Because SGs 1 and 4 are joined at two locations, the compartments were combined into one model (i.e., a LOCA in SG1 will discharge to the containment through SG4 as well).

| The questions across the top of the chart, shown in Figure IX-7, alternate among volume capture, flow split, and junction capture as the debris-transport process progresses through the nodalization scheme. The nodalization scheme was constructed to place the gratings at junction boundaries. The first chart question (header) after the initiator asks how much debris would be captured in Volume V12, where the LOCA was postulated to occur. The evaluation of this question involves simply estimating the fraction of small debris that was deposited by inertia near the pipe break; the remainder of the debris would be assumed to transport beyond this volume. The next question in the chart concerns a flow split (i.e., the distribution of the break flow going upward or downward from the break). The flow split is actually a debris split (i.e., how much debris goes in each direction). For fine- and small-piece debris, it is reasonable to assume that the debris split is approximated by the flow split. For large and intact-piece debris, the debris split may differ from the flow split, depending on the geometry. The third question concerns the amount of the debris captured at the flow junction between two volumes. The two

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junctions in the third question represent gratings that extend partly across the compartment at two levels. The fourth question starts the cycle over again for the next set of volumes in the sequence.

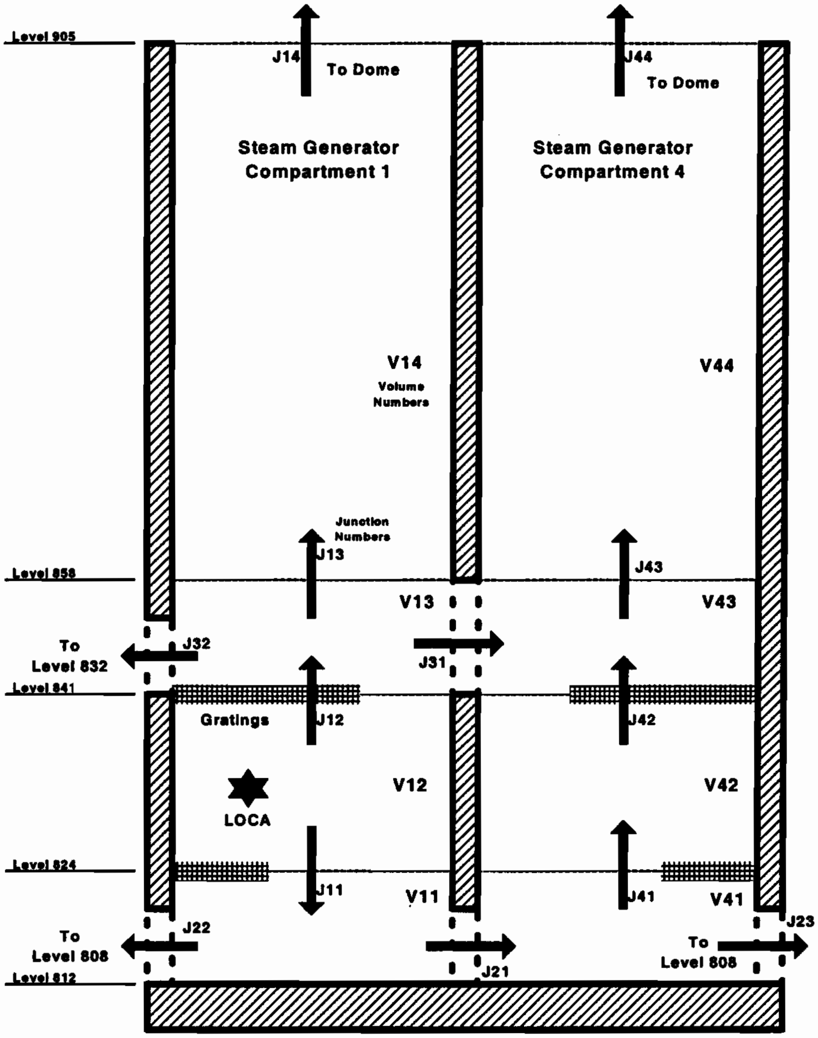

Figure IX-6. Break-Region Nodalization.

Once the distributions are inserted into the chart and the results are quantified, the results will indicate the distribution of captured debris within the compartments, as well as the debris transport from the compartments. The chart also will indicate where the debris that is transported from the SG compartments goes (e.g., to the dome or to the lower levels through access doorways).

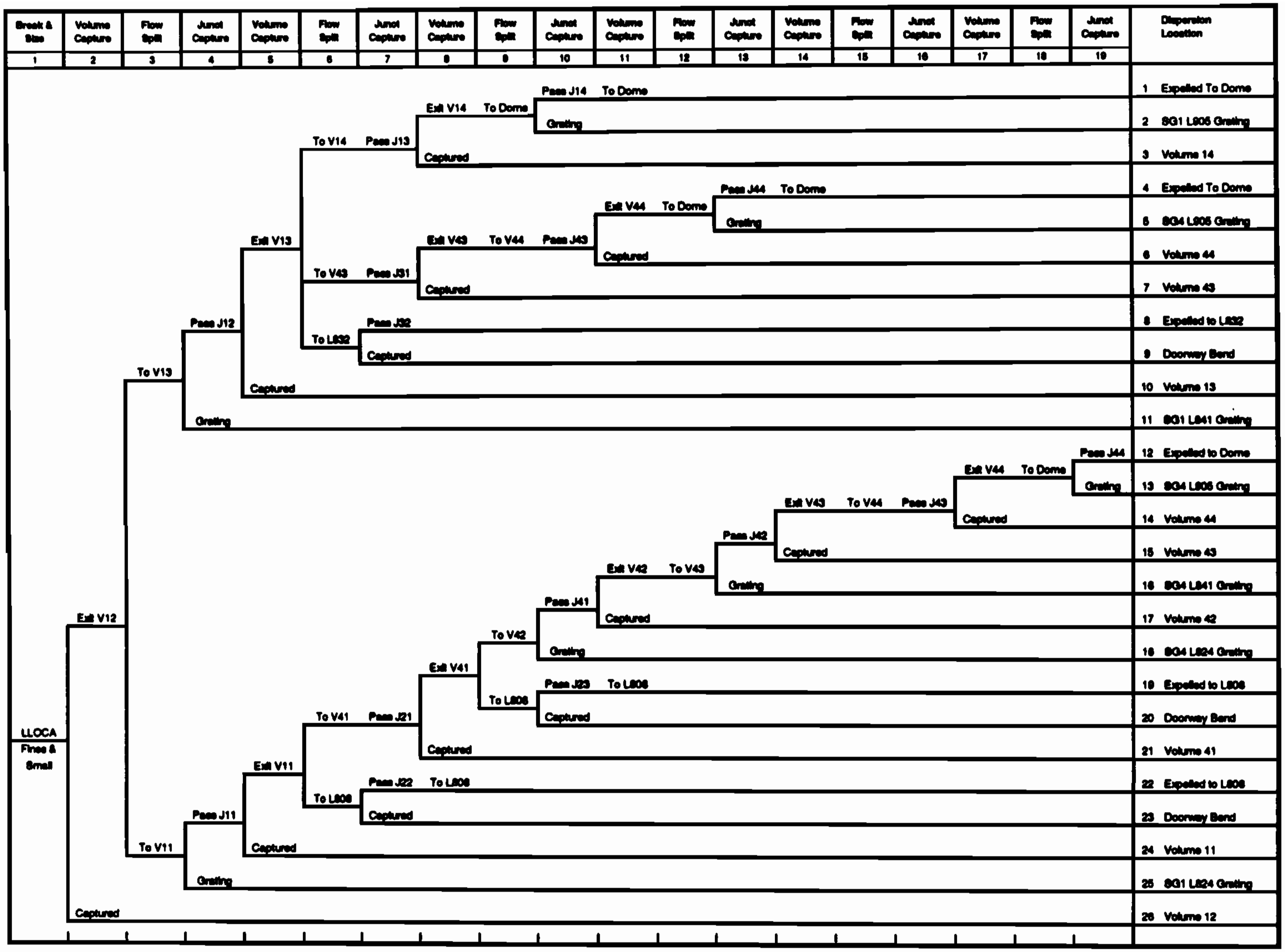

Figure IX-7. Chart for the Structure of Break-Region Debris-Transport.

VI.3.3.2 Dispersion and Capture throughout the Containment

The debris dispersion model used to evaluate debris transport within the volunteer-plant containments estimated dispersion throughout the containment first by free volume and then by surface orientation within a volume region. Dispersion distributions were based first on actual volumes and areas and then were adjusted using weighting factors that were based on engineering judgment.

VI.3.3.2.1 Dispersion by Region

As the containment pressurizes following a LOCA, break flows carrying debris would enter all free volume within the containment. Larger debris would tend to settle out of the break flows as the flow slowed down after leaving the break region. However, the fine and smaller debris more likely would remain entrained so that fine and small debris would be distributed more uniformly throughout the containment. Certainly, the distribution would not be completely uniform because of debris being captured along the way, which is the reason for the weighting factors.

First, the containment free volume was subdivided into volume regions. This subdivision was based on geometry (i.e., floor levels and walls) and on the location of CSs. Specifically, areas where deposited debris likely would not be entrained by the CSs were separated from areas that were impacted by the sprays. Some areas that were not actually sprayed still could be washed by the drainage of spray water as the water worked its way down through the containment structures. Areas where debris could be deposited without subsequently being washed downward by the sprays and the spray drainage could reduce the estimated transport fractions.

The total free volume of the containment is the sum of the free volumes for all of the volume regions. The volunteer-plant containment free volume was subdivided into a total of 24 volume regions ($J = 24$) as

$$V_{cont} = \sum_{j=1}^{J} Vc_j \quad ,$$

where

V_{cont} = the total free volume of the containment;
Vc_j = the free volume in containment region *j*; and
J = the number of volume regions.

The following equations define the dispersion model;

$$V_{i,j} = F_{i,j}\, D_i\, V_{ZOI} \quad ,$$

where

$V_{i,j}$ = the volume of debris-type *i* located in region *j*;
$F_{i,j}$ = the fraction of debris-type *i* deposited in region *j* during blowdown;

D_i = the fraction of total debris-type *i*; and

V_{ZOI} = the total volume of insulation contained within the ZOI.

For fibrous debris, the numbering system is *i* = 1, 2, 3, and 4 for fines, small pieces, large pieces, and intact debris, respectively.

The volume dispersion distribution must add up to one, as

$$\sum_{j=1}^{J} F_{i,j} = 1 \text{ (for each } i\text{)} \quad .$$

The break region was designated as Region 1 (i.e., *j* = 1 and $F_{i,1} = F_{i,break}$), and the methodology for the break-region dispersion fraction was provided in Section VI.3.3.1. The remaining distribution fractions were estimated using the following volume and engineering judgment weighted distribution:

$$F'_{i,j(j\neq 1)} = (1 - F'_{i,break}) \frac{wc_{i,j}\ Vc_j}{\sum_{j=2}^{J} wc_{i,j}\ Vc_j} \quad ,$$

where

$wc_{i,j}$ = the weighting factor based on engineering judgment.

If all of the $wc_{i,j}$ were set to one, then the distribution would be simply a volume-weighted distribution.

For large and intact pieces, many of these weighting values $wc_{i,j}$ were set to zero to reflect the fact that large and intact debris likely would not transport into many of the lower-level volume regions. It is anticipated that most of the large and intact debris would reside in the break-region volume, sump-pool volume, containment-dome volume, or refueling area.

The substantial quantities of debris transported into the dome subsequently would tend to either fall out of the atmosphere or be washed out by the CSs. About half of this debris would be deposited onto the Level 905 floors that are associated with the dome. However, the other half would fall below this level, thereby entering other volume regions. The volume distribution function $F'_{i,j}$ is modified as follows to account for debris fallout between regions:

$$F_{i,j} = F'_{i,j} + T_j\ F'_{i,2} \quad ,$$

where

T_j = the fraction of debris (type independent) located in the dome that subsequently falls or washes to region *j*.

The values of T_j are based on the opening areas into regions below the dome (e.g., the cross-sectional area of the SG compartments divided by the total cross-sectional area of the containment provides the values for debris that is falling into an SG compartment). The value for a region receiving no debris from dome fallout would be zero. Note that the dome volume region was designated Region 2; therefore, the value for region 2 (i.e., T_2) must be negative to remove debris from Region 2:

$$T_2 = -\sum_{j=1}^{J} T_{j(j\neq 2)} \quad .$$

VI.3.3.2.2 Dispersion by Surface Orientation and Exposure

Once the debris was dispersed to a volume region, it was assumed to have been deposited within that region. Some residual fine debris could remain airborne in regions that are not impacted by the sprays; however, the total quantity of this residual airborne debris was not expected to be significant.

The surface area within each volume region was subdivided into six subsections. These subsections reflect both the differing surface orientations and their exposure to moisture. The floors were separated from all of the other surfaces because the floors would receive the gravitationally settled debris and the other surfaces could be flooded partially by spray drainage. The spray water would not accumulate on the other surfaces, which include the walls, ceilings, and equipment.

Three surface exposures or moisture conditions were considered in the analysis: surfaces wetted directly by the CSs, surfaces not directly sprayed but washed by spray drainage (most likely floor surfaces), and surfaces wetted only by steam condensation. All surfaces likely would be wetted by condensation. The surface exposure determined how likely debris that was deposited onto that particular surface subsequently would be transported by the flow of water.

These areas were described by the following three-dimensional array:

$A_{j,k,l}$ = area for volume region *j*, orientation *k*, and exposure *l*.

All of the area within a particular volume region then would be

$$A_j = \sum_{k=1}^{2} \sum_{l=1}^{3} A_{j,k,l} \quad .$$

The numbering system is $k = 1$ and 2 for "floor" and "other" surfaces, respectively, and $l = 1$, 2, and 3, for condensate, spray, and drainage exposures, respectively.

The surface-area distribution fractions were estimated using the following area and engineering judgment weighted distribution:

$$f_{i,j,k,l} = \frac{w_{i,j,k,l}\ A_{j,k,l}}{\sum_{k=1}^{2}\sum_{l=1}^{3} w_{i,j,k,l}\ A_{j,k,l}} \quad ,$$

where

$f_{i,j,k,l}$ = the fraction of debris-type *i* deposited within volume region *j* that was deposited onto surface *k*, *l*; and

$w_{i,j,k,l}$ = the weighting factor based on engineering judgment for debris-type *i* deposited within volume region *j* that was deposited onto surface *k*, *l*.

An equivalent expression for $f_{i,j,k,l}$ is

$$f_{i,j,k,l} = \frac{w_{i,j,k,l}\ g_{j,k,l}}{\sum_{k=1}^{2}\sum_{l=1}^{3} w_{i,j,k,l}\ g_{j,k,l}} \quad ,$$

where

$$g_{j,k,l} = \frac{A_{j,k,l}}{A_j} \quad .$$

The fractions summed within a particular volume region and for a particular debris type must add up to one:

$$\sum_{k=1}^{2}\sum_{l=1}^{3} f_{i,j,k,l} = 1 \quad .$$

If all of the $w_{i,j,k,l}$ were set to one, then the distribution would be simply an area-weighted distribution. If all the $w_{i,j,k,l}$ were set to zero for $k = 2$ ("other" surfaces), then all of the debris would be deposited on the floor, as likely would be the case for the large and intact debris. It is anticipated that most of the large and intact debris would reside on the floors in the break-region volume, sump pool volume, containment dome volume, or refueling area. In the SG compartment, much of the large debris stopped on the underside of a grating could fall back down after the depressurization flows subsided.

The volume of debris on a particular surface is expressed by

$$V_{i,j,k,l} = f_{i,j,k,l}\ F_{i,j}\ D_i\ V_{ZOI} \quad .$$

VI.3.4 Washdown Debris Transport

Debris that is deposited throughout the containment subsequently would be subject to potential washdown by the CSs, the drainage of the spray water to the sump pool, and (to a lesser extent) the drainage of condensate. Debris on surfaces that would be hit directly by CS would be much more likely to transport with the flow of water than would debris on a surface that is wetted merely by condensation. The transport of debris entrained in spray water drainage is less easy to characterize. If the drainage flows were substantial and rapidly moving, the debris likely would transport with the water. However, at some locations, the drainage flow could slow and be shallow enough for the debris to remain in place. As drainage water dropped from one level to another, as it would through the floor drains, the impact of the water on the next lower level could splatter sufficiently to transport debris beyond the main flow of the drainage, thereby essentially capturing the debris a second time. In addition, the flow of water could erode the debris further, generating more of the very fine debris. These considerations must be factored into the analysis. The washdown processes are illustrated schematically in Figure IX-8.

The drainage of spray water from the location of the spray heads down to the sump pool was evaluated. This evaluation, reported in Appendix A **Appendix I?**, provided insights for the transport analysis, such as identifying areas that were not impacted by the CSs, the water drainage pathways, likely locations for drainage water to pool, and locations where drainage water plummets from one level to the next.

VI.3.4.1 Debris Erosion during Washdown

Experiments conducted in support of the DDTS analysis demonstrated that insulation debris could be eroded further by the flow of water. The primary concern of the DDTS analysis was LDFG debris that was deposited directly below the pipe break and therefore was inundated by the break overflow. Debris erosion in this case was substantial (i.e., ~9%/h at full flow). Debris erosion due to the impact of the sprays and spray drainage flows was certainly possible but was found to be much less significant. The DDTS study concluded that <1% of the LDFG was eroded because of the CSs. Debris erosion occurring because of condensation and condensate flow was neglected. Debris with its insulation still in its cover was not expected to erode further. For RMI debris, erosion was not a consideration. However, for a microporous insulation such as calcium silicate or Min-K, the washdown erosion has not been determined; it would be expected to be substantial and could potentially erode this type of debris completely into fine silt.

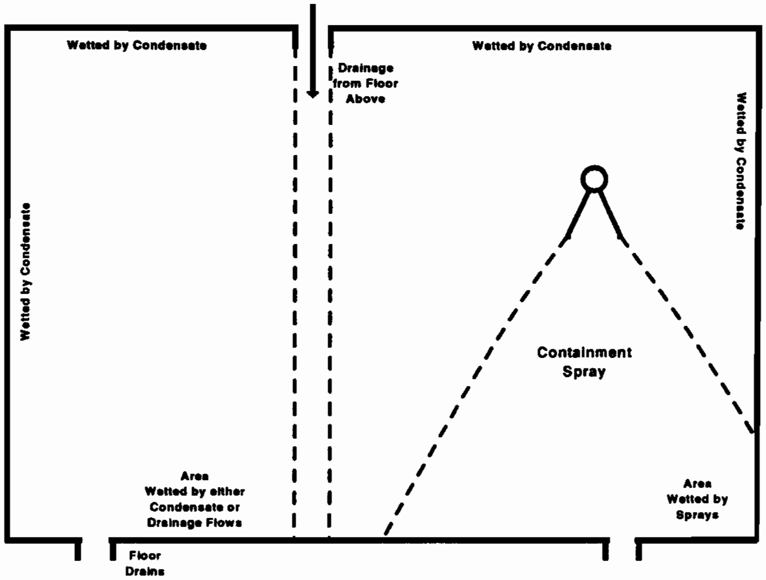

Figure IX-8. Schematic of Debris-Washdown Processes.

Because the byproduct of the erosion process is more of the very fine and easily transportable debris, the process must be evaluated. All erosion products were assumed to transport to the sump pool. Recall that this debris would remain suspended in the sump pool until filtered from the flow at the sump screens. Therefore, even a small amount of erosion could contribute significantly toward the likelihood of screen blockage.

The only erosion process evaluated herein was the erosion of debris that was impacted directly by the CSs. Erosion caused by break overflow was deferred to the degeneration of debris due to sump pool turbulence associated with the plummeting of the break flow into the pool. This assumption neglects the erosion of any large debris that is deposited on top of the lower grating in SG1 and impacted directly by the break overflow; however, this quantity of debris was not considered to be substantial. Most of the debris that is located directly below the break likely would be pushed away from the break and into the sump pool. Note that the floors of the SG compartments are 4 ft above the floor of the sump pool. At switchover, the SG floor would not be flooded but that at the maximum pool height, that pool would have a depth of 0.7 ft in the SG compartment.

The assumed fractions of fibrous debris that were eroded are summarized in Table IX-4. It was assumed that condensate drainage would not cause further erosion of debris and that intact or covered debris would not erode further. Erosion does not apply to fine debris because that debris is already fine. About 1% of the small- and large-piece debris that was directly impacted by the sprays was considered to have eroded. This amount of erosion was considered to be conservative because the DDTS concluded that the erosion was <1%. No erosion of the intact debris was assumed because the canvas cover likely would protect the insulation.

Table IX-4. Total Erosion Fractions for Fibrous Debris

[illegible]	[illegible]	[illegible]	Large	[illegible]
Condensate	N/A	0	0	0
Sprays	N/A	1%	1%	0

To estimate the volume of debris that was eroded, the volume of debris that was impacted by the sprays first must be estimated. The latter estimate can be made using the data arrays that were already established in this methodology. These volumes for small and large debris, respectively, are estimated using the following two equations:

$$Vspr_2 = \sum_{j=1}^{J} \sum_{k=1}^{2} f_{2,j,k,2} \, F_{2,j} \, D_2 \, V_{ZOI}$$

and

$$Vspr_3 = \sum_{j=1}^{J} \sum_{k=1}^{2} f_{3,j,k,2} \, F_{3,j} \, D_3 \, V_{ZOI} \quad .$$

The volumes that are eroded (*E_2* and *E_3* for small and large debris, respectively) are simply 1% of the debris volumes impacted by the sprays, given as

$$E_2 = e_{spr} Vspr_2$$

and

$$E_3 = e_{spr} Vspr_3 \quad ,$$

where the spray erosion fraction e_{spr} is 0.01.

VI.3.4.2 Capture Retention during Washdown

The retention of debris during washdown must be estimated for the debris deposited on each surface (i.e., the fraction of debris that remains on each surface). These estimates, based on experimental data and engineering judgment, were assigned somewhat generically. For surfaces that would be washed only by condensate drainage, nearly all deposited fine and small debris likely would remain there. The DDTS assumed that only 1% of the fibrous debris would be washed away in the more realistic central estimate of that study (a value of 10% was assumed for the upper-bound estimate). When the 1% assumption was applied, all of the surfaces that drained only condensate would have a retention fraction of 0.99 with respect to fibrous debris.

For surfaces that were hit directly by sprays, the DDTS assumed 50% and 100% for the central- and upper-bound estimates for small fibrous debris. Large and intact debris likely would not be washed down to the sump pool (retention fractions of 1). For surfaces that were not sprayed directly but subsequently drain accumulated spray water, such as floors close to spray areas, the retention fractions were much less clear. These fractions likely would vary with location and drainage flow rates and therefore must be area location specific, with more retention for small pieces than for fine debris.

The retention fraction for a specific volume region is expressed as

$$R_{i,j} = \sum_{k=1}^{2} \sum_{l=1}^{3} f_{i,j,k,l} \; r_{i,j,k,l} \quad ,$$

where

$R_{i,j}$ = the fraction of debris-type *i* retained in region *j*; and

$r_{i,j,k,l}$ = the fraction of debris-type *i* retained, on surface *k*, *l*, in region *j*.

These volume region retention fractions $R_{i,j}$ do not account for the quantities that are eroded from the captured pieces of debris. To complete the erosion model, the volumes of eroded debris that came from debris that remained captured versus debris that transported to the sump pool were estimated. Therefore, the debris that remained captured during the washdown process is estimated using the following two equations for small- and large-piece debris, respectively:

$$Rspr_2 = \sum_{j=1}^{J} \sum_{k=1}^{2} r_{2,j,k,2}\ f_{2,j,k,2}\ F_{2,j}\ D_2\ V_{ZOI}$$

and

$$Rspr_3 = \sum_{j=1}^{J} \sum_{k=1}^{2} r_{3,j,k,2}\ f_{3,j,k,2}\ F_{3,j}\ D_3\ V_{ZOI} \quad .$$

Therefore, the volumes of eroded debris associated with the debris that remained captured are expressed as

$$ER_2 = e_{spr} Rspr_2$$

and

$$ER_3 = e_{spr} Rspr_3 \quad .$$

Debris transported from its original volume region still could be captured at a lower elevation. This secondary capture was neglected in this analysis.

VI.3.5 Debris Volumes Introduced to the Sump Pool

The primary result of the blowdown/washdown transport analysis is the volume that is transported to the sump pool by debris category. The volumes of debris transported to the pool are given by

$$V_{i,pool} = \left[1 - \sum_{j=1}^{J} R_{i,j} F_{i,j} \right] D_i\ V_{ZOI} + Ve_i \quad ,$$

where

$V_{i,pool}$ = the volume of debris-type *i* transported to the sump pool and

Ve_i = the volumes of eroded debris transferring from small- and large-debris categories to the fine-debris category.

The erosion translation array is given by

$$Ve_i = \begin{bmatrix} +(E_2 + E_3) \\ -(E_2 - ER_2) \\ -(E_3 - ER_3) \\ 0 \end{bmatrix} \quad .$$

This array adds the eroded product ($E_2 + E_3$) to the fine-debris category and subtracts the eroded volume from the noncaptured small- and large-debris categories ($E_i - ER_i$).

The total debris that transports to the pool is

$$V_{pool} = \sum_{i=1}^{4} V_{i,pool} \quad .$$

This model does not track debris transport in sufficient detail to determine where the debris would enter the sump pool. It was assumed simply that the debris would be mixed uniformly with flows entering the pool.

VI.3.6 Transport Fractions

The overall debris-transport fraction now can be estimated as

$$TF_{ZOI} = \frac{V_{pool}}{V_{ZOI}} \quad ,$$

where

TF_{ZOI} = the fraction of insulation that is located in the ZOI and subsequently is transported to the sump pool.

The transport fractions for each individual debris category can be estimated as

$$TF_i = \frac{V_{i,pool}}{D_i \, V_{ZOI}} \quad ,$$

where

TF_i = the fraction of debris-type *i* that is generated within the ZOI and subsequently is transported to the sump pool.

Note that the translation of erosion products from the small- and large-debris categories to the fine-debris category has been incorporated into the transport fractions.

VI.4 DEBRIS-TRANSPORT ANALYSIS

When the methodology presented in Section VI.3 was used, plausible estimates were developed for the transport of insulation debris within the volunteer-plant containments. Because of the complexity of the analysis and the limited available data, substantial uncertainty exists in these estimates. Engineering judgment that was used to fill gaps in the data was tempered conservatively. Despite the uncertainty, the transport analysis illustrated trends, as well as plausible estimates of the fractions of the debris that was generated and subsequently could transport to the sump pool.

VI.4.1 Fibrous Insulation Debris Transport

As discussed in Section VI.3.2, the insulation that is used in the volunteer-plant containments consists of fibrous, RMI, and Min-K insulation at ~13.4%, 85.7%, and 0.9%, respectively. The majority of the available debris-transport data was obtained for LDFG insulation debris, specifically experimental data taken for the DDTS [NUREG/CR-6369-2, 1999]. Although a majority of the insulation within these containments is RMI, the fibrous insulation debris, in combination with particulate, is expected to be a larger challenge to the operation of the recirculation sump screens. Therefore, the debris transport for the fibrous debris was analyzed first. Even with the available transport data for LDFG debris, the transport analysis required the application of conservatively tempered engineering judgment.

VI.4.1.1 Fibrous Blowdown Debris Transport

The first consideration in performing the dispersion estimate for the fibrous blowdown insulation debris was the dispersion and deposition within the break region (assumed to be a break in SG1), where deposition likely resulted from inertial impaction. The dispersion through the remainder of the containment was subsequently estimated.

VI.4.1.1.1 Break-Region Blowdown Debris Deposition

The effluences from the break would carry insulation debris with the flows into the upper-containment dome through the large opening at the top of the SG compartment and into lower compartments through the compartment access doorways. Along the way, substantial portions of that debris likely would be inertially deposited or otherwise entrapped onto structures. In general, the break-region flow immediately after the initiation of the break would be much too violent to allow debris simply to settle to the floor of the region.

VI.4.1.1.1.1 Characterize Break Flows within Break Region

The thermal-hydraulic MELCOR code was used to determine the distribution of the break effluents from the SG compartment. When a break in SG1 was postulated, it was determined that most of the break effluent would be directed upward toward the large upper dome. Because of the large openings connecting SG1 to SG4, the venting to the dome would occur through both SG compartments. Effluents venting into lower-level compartments (surrounding the two SGs) by way of open access doorways would flow at much lower rates than the upward flows to the dome. The nodalization of the two SG compartments is shown in Figure IX-6, where the break was postulated to occur in Volume V12. Break effluents that are typical of three break sizes were assumed: large-break (LB) LOCA, medium-break (MB) LOCA, and small-break (SB) LOCA. The results of the MELCOR simulations are summarized in Table IX-5, where the distributions from a particular control volume are shown by the connecting junction. For example, given an LB LOCA scenario, ~80% of the flow from Volume V12, where the break was postulated, went upward through Junction J12, with the remainder going downward through Junction J11. Note that the flow splits were somewhat transient and that the results in Table IX-5 are reasonable approximations of the transients over the time where most debris transport would occur. LB LOCA and MB LOCA flows were reasonably steady over the transport period, but SB LOCA flows were not steady because of transition into natural circulation after ~6 s.

Inertial debris deposition is dependent on the flow velocities transporting the debris. The MELCOR calculations predicted transient flow velocities for each flow junction and each size of break. The general ranges of these velocities are provided in Table IX-6. The velocities are in the general range as the test velocities for which the debris-capture data were measured in the DDTS.

Table IX-5. Break Effluent Flow Splits

Break Size	Flows Exiting Volume V_i through Junction J_i								
	V12		V11		V41		V13		
	J11	J12	J21	J22	J23	J41	J13	J31	J32
LB LOCA	20%	80%	70%	30%	5%	95%	62%	33%	5%
MB LOCA	20%	80%	70%	30%	14%	86%	62%	33%	5%
SB LOCA	15%	85%	80%	20%	30%	70%	66%	28%	6%

Table IX-6. Characteristic Velocities in SG1

Postulated Break Size	Characteristic Velocities	
	m/s	ft/s
LB LOCA	25–200	80–660
MB LOCA	5–45	15–150
SB LOCA	1–8	5–25

VI.4.1.1.1.2 Debris-Transport Distributions from Volumes

The very fine debris would transport more like an aerosol in that the particles would disperse within the flow and follow the flow. Portions of this debris would be deposited onto structures along the transport pathways, primarily because of inertial deposition at bends in the flow. However, with larger debris, the tendency would be greater for the debris not to follow the flow through sharp bends in the flow and larger debris would more likely be trapped by a structure such as a grating. In addition, gravitational settling as the flow velocities slow would be more effective for larger debris than smaller debris. For example, following an LB LOCA in an SG compartment, a large, nearly intact insulation pillow could travel upward with the main flow to the containment dome unless an obstacle, such as a grating, impeded that pillow. However, this pillow would be much less likely to follow the flow through a connecting doorway to the next SG compartment.

Assumptions based on engineering judgments that were tempered by experimental observations were required to reach a solution. The assumptions provide a reasonable crude approximation of debris transport from a volume when there is a split in the flow. These assumptions are the following.

- The fine and small fibrous debris would be well dispersed within the flow and would transport uniformly with the flow; therefore, the debris-transport junction distributions for fines and small debris are the same as the junction flow distributions in Table IX-5.

- Large and intact debris would not make the turn to exit SG1 at Level 832 (Junctions J31 and J32). In addition to the turn, most of this debris that was moving toward these exits would be stopped by the gratings that cover ~45% of the cross-sectional area of the compartment that is nearest those exits.

- Large and intact debris entering SG4 at the floor level (Level 812) would be much less likely to follow the flow through the 90° bend and subsequently transport upward through SG4. Debris entering Volume V41 that is not captured in Volume V41 would exit by either Junction V23 or V41. For large and intact debris, the flow fractions for Junction V41 were reduced by one-half and two-thirds, respectively (engineering judgment).

Applying these assumptions to the transport of the large and intact debris through the node junctions resulted in the junction transport distributions that are shown in Table IX-7 and Table IX-8.

Table IX-7. Large-Debris-Transport Junction Distributions

Break	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
LB LOCA	20%	80%	70%	30%	52%	48%	100%	0%	0%
MB LOCA	20%	80%	70%	30%	57%	43%	100%	0%	0%
SB LOCA	15%	85%	80%	20%	65%	35%	100%	0%	0%

Table IX-8. Intact-Debris-Transport Junction Distributions

Break	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
LB LOCA	20%	80%	70%	30%	68%	32%	100%	0%	0%
MB LOCA	20%	80%	70%	30%	71%	29%	100%	0%	0%
SB LOCA	15%	85%	80%	20%	77%	23%	100%	0%	0%

VI.4.1.1.1.3 Capture Fractions at Junctions

Debris-transport data from the Army Research Laboratory (ARL) and the CEESI tests that were conducted to support the DDTS [NUREG/CR-6369-2, 1999] provide average capture fractions for LDFG debris that is passing though typical gratings and around typical structures, such as piping and beams, and for debris making a 90° bend. These structures and the bend were wetted during the tests; the data do not apply to dry structures. These data are assumed to apply in general to the volunteer-plant containments because it is expected that the containment surface would be wetted rapidly by steam condensation,* as well as liquid break effluent, and because the range of predicted flow velocities (Table IX-6) are in general agreement with the flow velocities of the tests. The flow velocities ranged from 25 to 150 ft/s for the ARL tests and from 35 to 60 ft/s for the CEESI tests. The debris capture was most applicable to MB LOCAs and perhaps least applicable to SB LOCAs.

Fine and small fibrous debris could be captured inertially onto wetted surfaces whenever the break flow changed direction, such as flows through the doorways from the SG compartments into the sump-level annular space. These flows must make at least one 90° bend through those entrances. Debris-transport experiments that were conducted at CEESI demonstrated an average capture fraction of 17% for fine and small debris that were making a 90° bend. These surfaces would be wetted because of steam condensation and the liquid portion of the break

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*Based on analyses performed for the DDTS [NUREG/CR-6369-3, 1999].

effluence. Other flow bends likely would occur within the violent three-dimensional flows near the break. The platform gratings within the SG compartments would capture substantial amounts of debris, even though the gratings do not extend across the entire compartment. The CEESI debris-transport tests demonstrated that an average of 28% of the fine and small LDFG debris was captured when the airflow passed through the first wetted grating encountered and that an average of 24% was captured at the second grating. The large and intact debris, by definition, would be trapped completely by a grating. In addition, equipment (such as beams and pipes) was shown to capture fine and small debris. In the CEESI tests, the structural maze in the test section captured an average of 9% of the debris passing through the maze.

Grating Capture: In the volunteer plant, partial gratings exist at three levels in each of the SG compartments. The gratings extend out over ~22%, 45%, and 15% of the SG cross-sectional area at plant elevations 824, 841, and 905 ft, respectively.* If it is assumed that 28% of small and fine fibrous debris and 100% of the large and intact debris are captured from the flow by a grating as the flow passes through the grating, the capture fractions for model junctions that contain a grating are provided in Table IX-9.

Table IX-9. Grating Capture Fractions at Model Junctions

Grating Level	Model Junctions	Fine and Small Debris		Large and Intact Debris	
		Unit Area Capture Fraction	Junction Capture Fraction	Unit Area Capture Fraction	Junction Capture Fraction
Level 905	J14 and J44	0.28	0.04	1.0	0.15
Level 841	J12 and J42	0.28	0.13	1.0	0.45
Level 824	J11 and J 41	0.28	0.06	1.0	0.22

Doorway Capture: Depressurization flows also would exit the SGs by way of the SG access doorways at Levels 808 and 832. Flows traveling through these pathways would carry debris directly into the lower levels of the containment; in fact, some of the debris likely would be deposited near the recirculation sumps. Because these doorways were designed with at least one 90° bend, debris would be deposited inertially onto wetted surfaces at each bend in the flow. Furthermore, because the CSs would not impact these vertical surfaces, the debris likely would remain on the surfaces once it was captured there. The CEESI data showed an average of 17% debris capture at its 90° bend for debris that was small enough to already have passed through a grating (i.e., fines and small debris). It was assumed that 17% of fine and small debris that was transported from the SG break region through the Level 808 and Level 832 doorways to the bulk containment would be captured at a bend (one bend assumed). No comparable data exist for the large and intact debris; however, the larger debris would be much less likely to stick to a wall once it impacted inertially against the wall. Because of a lack of appropriate data, it was assumed conservatively that no large or intact debris would be captured at these doorways.

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*These fractions were estimated from plant drawings.

VI.4.1.1.1.4 Capture Fractions within Volumes

As illustrated in Figure IX-7, debris would be captured on structures within the model nodes, as well as the node junctions. As the break effluents flowed around and through the structural and equipment congestion within the SG compartment, debris would be driven inertially onto surfaces where some portion of it would remain captured. The structures include the pumps; SGs; and associated piping, beams, equipment stands, cabling, etc. The chaotic nature of the flows as the break jet is deflected off structures and wall surfaces could create a multitude of bends in the flow that could deposit debris inertially onto wall surfaces and irregular wall features. In the CEESI tests, ~9% of the fine and small debris was deposited onto wetted structures as the debris passed through a test structural assembly and 17% was captured onto a wetted surface at a sharp 90° bend in flow. Estimates of the amounts of debris captured within a node volume were based on this CEESI test data and on conservatively tempered engineering judgment. It is likely conservative to capture more debris within the SG than to transport the debris throughout the containment because washdown within the SG should be relatively greater than some other areas of the containment and because debris washed off the SG structures can go directly to the sump pool.

Applying a number of engineering judgments in conjunction with the CEESI data resulted in estimates for the capture of debris within each volume of the break-region debris-transport model. These estimates, along with the associated assumptions, are provided in Table IX-10.

Table IX-10. Fractions of Debris Captured within Each Volume

SG [illegible]					SG [illegible]			
Volume	Fines and Small Pieces	[illegible]	[illegible]		Volume	Fines and Small Pieces	Large Pieces	[illegible]
V14	1% (A)	2% (A)	5% (A)		V44	1% (A)	2% (A)	5% (A)
V13	1% (A)	2% (A)	5% (A)		V43	1% (A)	2% (A)	5% (A)
V12	14% (C)	30% (E)	50% (F)		V42	9% (B)	15% (E)	30% (G)
V11	26% (D)	40% (E)	80% (H)		V41	14% (C)	25% (E)	80% (H)

Assumptions
A. Volumes contain minimal structures and no significant flow bends; therefore, a minimal amount of capture occurs. It is somewhat more likely that large debris would be captured than small debris and more likely that intact debris would be captured than large debris.
B. Structures are equivalent to one CEESI structural test assembly (9%), and no significant flow bends exist.
C. Structures are equivalent to one CEESI structural test assembly (9%), and significant flow bending that is less than a sharp 90° bend exists (5%).
D. Structures are equivalent to one CEESI structural test assembly (9%), and significant flow bending that is equivalent to a sharp 90° bend exists (17%).
E. Large debris is more likely to be captured than small debris, and 50% more large debris is captured than small debris.

F. Intact debris is much more likely to snag on equipment than the large debris. In addition, some insulation within the ZOI likely could remain attached to piping.

G. Intact debris is much more likely to snag on equipment than the large debris.

H. The congestion of equipment and cables near the floor is expected to trap most of the intact debris as the flow makes a 90° bend near the floor. Intact debris is less likely to follow the distribution of flow than is smaller debris.

VI.4.1.1.1.5 Break-Region Debris-Transport Quantification

The logic chart shown in Figure IX-7 and discussed in Section VI.3.3.1 was used to quantify the various flow splits and capture and to estimate the debris deposition within and from SG1. These charts divide the evaluation into many smaller problems that are amenable to resolution—an approach that was adapted from the resolution of the BWR strainer-blockage issue [NUREG/CR-6369-1, 1999]. This chart tracks the progress either of small debris from the pipe break (Volume V12) until the debris is assumed to be captured or until the debris is transported beyond the compartment. Charts were quantified for each of the three LOCA sizes (i.e., small, medium, and large) and for three classifications of fibrous debris (i.e., fines and small pieces, large pieces, and intact pieces). Note that there was no basis to treat the fines and small pieces differently. The data that were used to quantify the charts are discussed in Sections VI.4.1.1.1.1 through VI.4.1.1.1.4. As an example, the chart for the transport of fines and small debris following an LB LOCA is shown in Figure IX-9.

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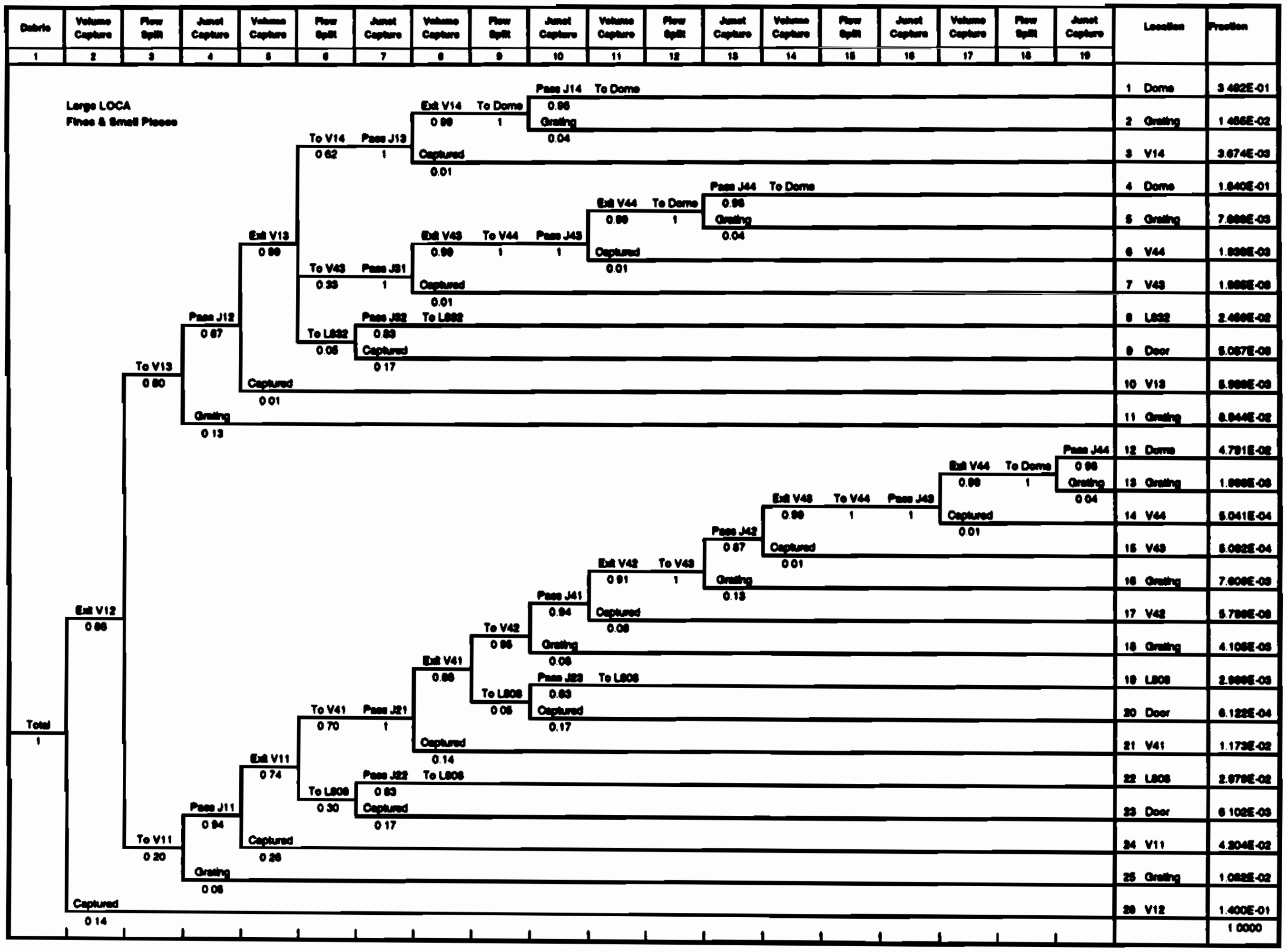
The overall results of the break-region quantification are shown in Table IX-11. The results for the three break sizes were averaged into a single set of results. This was done because the differences among the three size groups were substantially less than the substantial uncertainties associated with these analyses. The charts also provided information regarding the distribution of debris captured with the SGs, as well as the debris driven from the SGs.

Table IX-11. Distribution of Debris Captured and Exiting Break Region

Location	Debris Category		
	Fines and Small Pieces	Large Pieces	Intact Pieces
Captured within SGs 1 and 4	0.36	0.70	0.82
Expelled to Dome	0.58	0.26	0.17
Expelled to Level 832	0.03	0	0
Expelled to Level 808	0.03	0.04	0.01



Location		Fraction
1	Dome	3.492E-01
2	Grating	1.455E-02
3	V14	3.674E-03
4	Dome	1.840E-01
5	Grating	7.666E-03
6	V44	1.936E-03
7	V43	1.955E-03
8	L832	2.459E-02
9	Door	5.037E-03
10	V13	5.986E-03
11	Grating	8.944E-02
12	Dome	4.791E-02
13	Grating	1.996E-03
14	V44	5.041E-04
15	V43	5.092E-04
16	Grating	7.609E-03
17	V42	5.789E-03
18	Grating	4.105E-03
19	L806	2.989E-03
20	Door	6.122E-04
21	V41	1.173E-02
22	L806	2.979E-02
23	Door	6.102E-03
24	V11	4.204E-02
25	Grating	1.032E-02
26	V12	1.400E-01
		1.0000

Figure IX-9. Break-Region LB LOCA Transport Chart for Fines and Small Debris.

VI.4.1.1.2 Dispersion throughout Remainder of Containment

The debris dispersion model that was presented in Section VI.3.3.2 was used to evaluate debris transport within the volunteer-plant containments by estimating dispersion throughout the containment first by free volume and then by surface orientation within a volume region.

VI.4.1.1.2.1 Dispersion by Volume Region

The containment free volume was subdivided into volume regions that were based on geometry, such as floor levels and walls, and on the location of CSs. Specifically, areas where deposited debris likely would not be washed down by the CSs were separated from areas that were impacted by the sprays. The volunteer-plant free volume was subdivided into 24 distinct regions of free volume, as shown in Table IX-12. The volumes of each region were estimated from plant drawings.

Table IX-12. Subdivision of Containment Free Volume

No.	Volume Region	Volume (ft^3)	Volume Fraction Vc_i
1	SG1&4	76600	0.02570
2	Dome - Above 905.75-ft	1992060	0.66848
3	L873 - MS	39300	0.01319
4	Head Lay-Down - L871.5	17120	0.00574
5	Below Head Platform	5750	0.00193
6	Refueling A	45340	0.01521
7	Refueling B	53860	0.01807
8	Refueling C	48660	0.01633
9	Refueling D	47960	0.01609
10	SG2&3	76600	0.02570
11	Pressurizer	11250	0.00378
12	L860 Annulus - Section 1	34100	0.01144
13	L860 Annulus - Section 2	54580	0.01832
14	L860 Annulus - Section 3	94310	0.03165
15	L851 - FW	25800	0.00866
16	Accumulator Section	31500	0.01057
17	L832 Annulus - Section 1	37250	0.01250
18	L832 Annulus - Section 2	33940	0.01139
19	L832 Annulus - Section 3	69890	0.02345
20	L808 Annulus - Section 1	61650	0.02069
21	L808 Annulus - Section 2	30830	0.01035
22	L808 Annulus - Section 3	61650	0.02069
23	Reactor Cavity	25000	0.00839
24	Equipment Room L808	5000	0.00168
	Containment Total	2980000	1.00000

Key aspects of the region subdivision follow. The first region, designated SG1 and 4, is the SG compartment 1 where the break was postulated and its connected neighboring SG compartment, SG4. Debris dispersion and deposition in these SG compartments was predicted in Section VI.4.1.1.1. The second region represents the free volume above the highest floor (i.e., the dome region), which is approximately two-thirds of the entire containment free volume. As shown in Figure IX-10, the lower floor levels were subdivided azimuthally into three sectors to better distinguish the areas with CSs from areas without the sprays. The refueling pool area was subdivided into four regions to reflect the three different pools and the reactor-vessel (RV) head area [i.e., (A) storage pool for RV upper internals, (B) RV area, (C) storage pool for RV lower internals, and (D) pool for fuel transfer and storage].

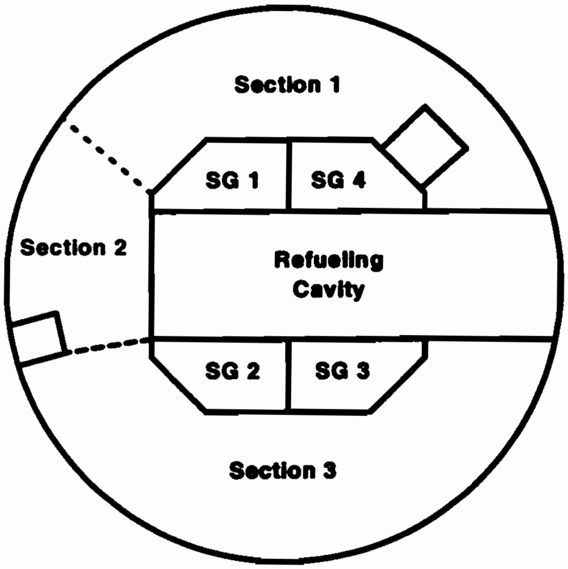


Figure IX-10. Volume Region Sector Model

Debris, particularly the larger debris, would not distribute uniformly throughout the free volume. The methodology presented in Section VI.3.3.2.1 applies weighting factors ($wc_{i,j}$) to the free-volume distribution to estimate the distribution of debris throughout the containment (i.e., the distribution of the debris among the 24 volume regions) by debris type. The very fine debris likely would transport somewhat uniformly with the depressurization flows, which would penetrate all free space within the containment as the containment pressurized. The transient nature of debris generation would also introduce nonuniformities into the dispersion of the fine debris. Because no rationale was found to weight the distribution of the fine and small debris away from that of a uniform free-volume distribution outside the break region, all weighting factors were assumed to be one for fine and small fibrous debris.

For the largest debris, specifically the large-piece and intact-piece classifications, the debris that is ejected from the SG compartments into the dome region likely would fall back to the floors and structures of the higher levels. The settling of debris that was ejected into the dome atmosphere was proportioned onto the upper floors according to the distribution of floor area (e.g., the cross-sectional area of a SG compartment divided by the cross-sectional area of the overall containment determined the fraction of settling debris that would fall into that compartment). The largest debris likely would not enter lower compartment volumes, except for debris ejected into the sump-level annulus via personnel access doorways. The assumed weighting factors for the large and intact debris were specified to preference the deposition of larger debris onto the uppermost floors and into the sump-level annulus. The large-piece debris was assumed to transport somewhat more easily than the intact-piece debris. The assumed weighting factors and the dome fallout fractions are shown in Table IX-13.

Table IX-13. Volume Region Weighting Factors

No.	Volume Region	Dome Fallout Fraction T_j	Volume Weighting Factors			
			Fines $wc_{1,j}$	Small Pieces $wc_{2,j}$	Large Pieces $wc_{3,j}$	Intact Pieces $wc_{4,j}$
1	SG1&4	0.0951	1	1	1	1
2	Dome - Above 905.75-ft	0	1	1	1	1
3	L873 - MS	0.0555	1	1	0.5	0.3
4	Head Lay-Down - L871.5	0.0349	1	1	0.8	0.5
5	Below Head Platform	0	1	1	0.3	0
6	Refueling A	0.0495	1	1	0.8	0.5
7	Refueling B	0.0579	1	1	0.8	0.5
8	Refueling C	0.0505	1	1	0.8	0.5
9	Refueling D	0.0596	1	1	0.8	0.5
10	SG2&3	0.0978	1	1	0.5	0.3
11	Pressurizer	0	1	1	0	0
12	L860 Annulus - Section 1	0.0092	1	1	0.3	0
13	L860 Annulus - Section 2	0.0052	1	1	0.3	0
14	L860 Annulus - Section 3	0.0241	1	1	0.3	0
15	L851 - FW	0	1	1	0	0
16	Accumulator Section	0.0060	1	1	0.8	0.5
17	L832 Annulus - Section 1	0	1	1	0	0
18	L832 Annulus - Section 2	0	1	1	0	0
19	L832 Annulus - Section 3	0	1	1	0	0
20	L808 Annulus - Section 1	0	1	1	1	1
21	L808 Annulus - Section 2	0	1	1	1	1
22	L808 Annulus - Section 3	0	1	1	0.3	0
23	Reactor Cavity	0	1	1	0	0
24	Equipment Room L808	0	1	1	0	0
	Total	0.5453				

The results of the blowdown distribution by groups of volume regions are illustrated in Figure IX-11. In this estimate, the largest portion of the debris was deposited inside the SG compartments, where the break was postulated because of inertial deposition that occurred as the fast-moving flows drove the debris into and through equipment and structures. This was particularly true for the larger debris, which could not pass through the gratings. The upper-level floors (871-, 873-, and 905-ft levels) received substantial debris falling or settling out of the dome atmosphere. The regions above the refueling pools received debris that was driven into those volumes, as well as debris falling or settling from the dome atmosphere; this comment also applies to the opposite SG compartments, SGs 2 and 3. The pressurizer compartment received only small amounts of fine and small debris and no larger debris because the compartment has a roof that prevents debris from falling into the compartment and is relatively small. The lower levels receive relatively small quantities of mostly large-piece debris because of their remoteness from the dome. Most of the debris entering Levels 832 and 808 was debris that was expelled from the SG compartments by way of the personnel access doorways; therefore, this debris would likely be located near those doors.

CSs would impact most of the deposited debris; these surface areas include the four SG compartments, the upper floor surfaces, and the refueling area. Regions that were not impacted by the sprays included the pressurizer compartment and certain portions of the lower levels. This observation suggests that a large fraction of the more transportable debris would transport to the sump pool.

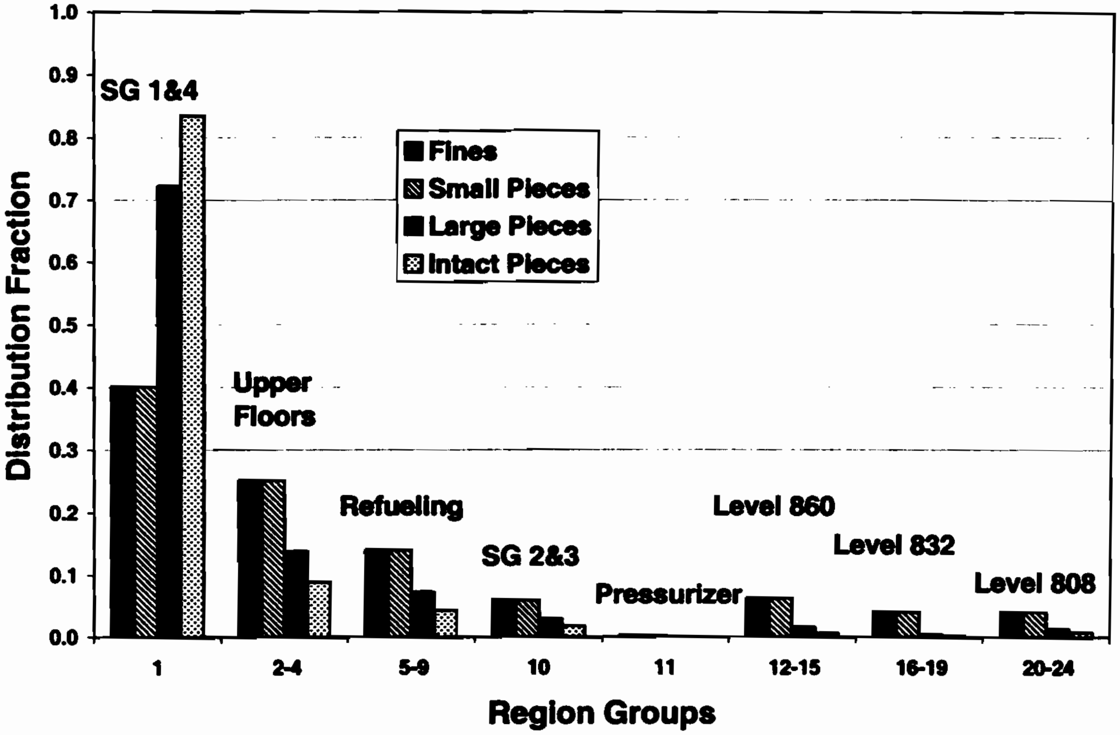


Figure IX-11. Blowdown Distribution by Region Groups.

VI.4.1.1.2.2 Dispersion by Surface Orientation and Surface Wetness

Once the debris dispersion prediction placed each type of debris within the 24 volume regions, the debris was dispersed further by surface area classification, i.e., orientation and exposure to moisture. The surface orientation was either "floor area" or "other" area; the distinction was that gravitational settling preferentially deposited debris onto the floor. The surface exposure to moisture included surfaces that were impacted directly by the CSs, surfaces subjected to spray drainage but not sprayed directly, and the remaining surfaces, which would be wetted by condensation. In this manner, the surface area within each volume region was subdivided into six surface groupings. This subdivision was based on both engineering drawings and engineering judgment. The drawings provided basic geometric information such as floor areas; however, engineering judgment, in addition to drawings, was required to estimate fractions of surfaces that were sprayed directly or covered by spray drainage. The estimated area distribution fractions are shown in Table IX-14.

The floor fraction is an estimate of the total surface area that would receive gravitationally settling debris. This estimate includes upward-facing equipment, as well as the floor (the equipment and piping was assumed to have the same floor fraction as the wall, floor, and ceiling surfaces). The condensate, spray, and drainage fractions represent the fraction of each orientation with this type of exposure. With these fractions, the surface areas and area ratios (i.e., $A_{j,k,l}$ and $g_{j,k,l}$) are determined. For example, the floor fraction for a given region multiplied by the spray $g_{j,k,l}$) are fractions for that region's floor multiplied by the total surface area of the region yields the floor surface area that was sprayed directly by the sprays.

Table IX-14. Regional Areas Fractions

No.	Volume Region	Floor Fraction	Floor Surface Area			Other Surface Area		
			Condensate Fraction	Spray Fraction	Drainage Fraction	Condensate Fraction	Spray Fraction	Drainage Fraction
1	SG1&4	0.07	0	1	0	0.1	0.5	0.4
2	Dome - Above 905.75-ft	0.09	0	1	0	0	1	0
3	L873 - MS	0.17	0.2	0.6	0.2	0.9	0.1	0
4	Head Lay-Down - L871.5	0.61	0	1	0	0	0	1
5	Below Head Platform	0.30	0.6	0.1	0.3	0	0	1
6	Refueling A	0.37	0	1	0	0	0	1
7	Refueling B	0.41	0	1	0	0	0	1
8	Refueling C	0.55	0	1	0	0	0	1
9	Refueling D	0.68	0	1	0	0	0	1
10	SG2&3	0.07	0	1	0	0.1	0.5	0.4
11	Pressurizer	0.04	1	0	0	1	0	0
12	L860 Annulus - Section 1	0.10	0.9	0.1	0	1	0	0
13	L860 Annulus - Section 2	0.19	0.1	0.6	0.3	0.6	0.1	0.3
14	L860 Annulus - Section 3	0.19	0.1	0.6	0.3	0.6	0.1	0.3
15	L851 - FW	0.19	0.8	0	0.2	1	0	0
16	Accumulator Section	0.13	0	0.5	0.5	0.5	0	0.5
17	L832 Annulus - Section 1	0.18	0.9	0	0.1	0.7	0	0.3
18	L832 Annulus - Section 2	0.15	0.4	0	0.6	0.6	0	0.4
19	L832 Annulus - Section 3	0.17	0.3	0.5	0.2	0.6	0	0.4
20	L808 Annulus - Section 1	0.18	0	0	1	0.7	0.3	0
21	L808 Annulus - Section 2	0.18	0	0	1	0.7	0.3	0
22	L808 Annulus - Section 3	0.19	0	0	1	0.7	0.3	0
23	Reactor Cavity	0.13	0	0	1	1	0	0
24	Equipment Room L808	0.21	0	0	1	1	0	0

Next, the area weighting factors ($w_{i,j,k,l}$) were estimated, which preference debris toward one surface over another. The dominant preferential debris deposition (and the only preference that can be estimated realistically) is gravitational debris that settles to the floor surfaces. The weighting factors for the non-floor surfaces ($k = 2$) were set first to 1 (i.e, $w_{i,j,2,l} = 1$), and then the weighting factors for the floor surfaces within each volume region were estimated for each debris type such that the weighting factors preferentially forced debris deposition onto the floor surfaces. The floor weighting factor estimates used the following equation, where the weighting factor is a function of two physical variables that can be estimated more readily. These variables are the fraction of the surface area that is floor area (a geometric determination) and the fraction of the debris that is deposited onto the floor (an engineering judgment and computational determination):

$$w_{floor} = \left(\frac{d_{floor}}{1-d_{floor}}\right)\left(\frac{1-g_{floor}}{g_{floor}}\right) ,$$

where

w_{floor} = the weighting factor for debris deposited onto the floor inside a volume;

d_{floor} = the fraction of the debris deposited within a volume that was on the floor; and

g_{floor} = the fraction of the volume surface area that is floor area.

The determination of the floor-area fraction (g_{floor}) is a straightforward estimate of the floor area divided by the total surface area in a volume region (listed in Table IX-14). In actuality, the surface-area estimate includes the areas associated with equipment and piping because debris can settle onto equipment and piping, as well as onto floors. To reduce the complexity of the area estimates, it was assumed that the area fractions for the equipment and piping were the same as the area fractions for the wall, ceiling, and floor surfaces. Because of this assumption and other geometrical assumptions, these area fractions have an inherent uncertainty associated with the estimates; however, this uncertainty should be significantly smaller than some of the other transport uncertainties.

Debris deposition processes other than gravitational settling, such as diffusiophoresis (condensation-driven deposition), do not depend on surface orientation for these processes; the weighting factors all would be set to 1. Driven debris could be deposited inertially onto any surface or could snag on an obstacle. Heavy, inertially deposited debris subsequently may fall to the floor, but substantially smaller debris likely would remain pasted onto the surface. Even heavy debris can remain on a nonhorizontal surface if the piece were physically snagged. Vertically moving debris eventually would settle onto a surface that is sufficiently horizontal to retain the debris. The fraction of debris deposition onto the floor is highly dependent on the size of the debris.

The estimate of the fraction of the debris that was deposited onto the floor depended greatly on conservative judgments; therefore, the fraction introduced substantial uncertainty into the transport estimates. The engineering judgments accounted for the geometry of the region under consideration, including the relative structural congestion. It was conservative to place the debris on the floor as opposed to other surfaces because more of the debris that was deposited on the floor would be subjected to spray washdown on the floor than on other surfaces. For the SG compartments where the pipe break was postulated (SGs 1 and 4), debris deposition data from the logic charts were used to estimate debris on the floor of these compartments. This estimate included larger debris that was trapped on the underside of gratings and that would likely fall back once the depressurization flow subsided. It was assumed that debris that fell or settled from the dome atmosphere into lower-level regions would fall or settle onto a floor surface.

A typical judgment estimate for fractions of debris that had been driven into an enclosure and that would subsequently settle to the floor was 0.4, 0.7, 0.99, and 0.99 of the fines, small pieces, large pieces, and intact pieces, respectively. For fine debris, the floor deposition fraction was two to three times the floor area fraction, thereby allowing a substantial settling of the very fine debris, even though diffusion processes would deposit the fine debris onto any surface. The floor fraction for small-piece debris was substantially higher than for the fine debris. Large and intact debris would fall to a horizontal surface unless it snagged on an obstacle. The floor fraction was set to 0.99 to place the large debris on the floor; however, some pieces could have snagged on an obstacle before reaching the floor.

For the far-side SG compartments (SGs 2 and 3) and the pressurizer compartment, the floor-debris deposition fractions acknowledged that the debris would have to travel downward in the compartment and through a variety of structures, including gratings, before reaching the floor; the fractions were reduced for these compartments. For instance, the gratings would catch much of the large debris before it could reach the floor. For open regions, such as the refueling pool regions, where a small amount of equipment and piping is located and the region is not enclosed completely by walls, the floor-debris fractions were increased substantially.

Once the weighting factors were estimated, the final deposition of the debris was determined both as a function of the region and by the surface orientation and its exposure to moisture. Figure IX-12 and Figure IX-13 illustrate the dispersion patterns in the containment according to surface orientation and surface wetness.

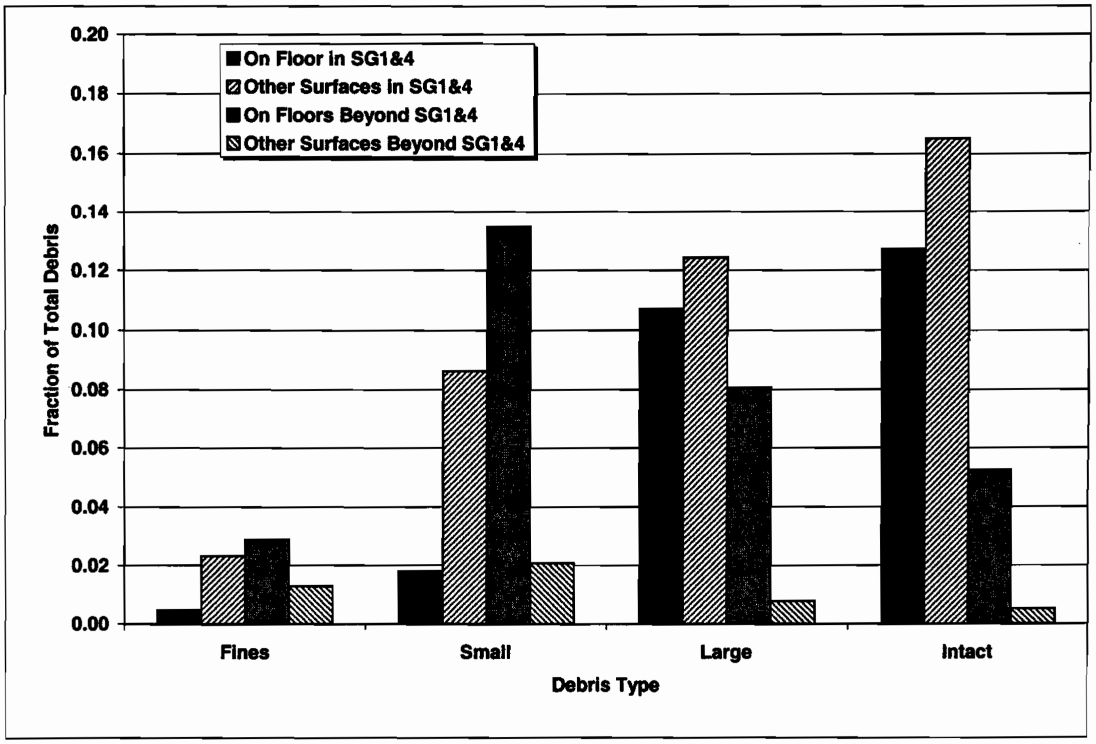


Figure IX-12. Blowdown Debris Dispersion by Surface Orientation.

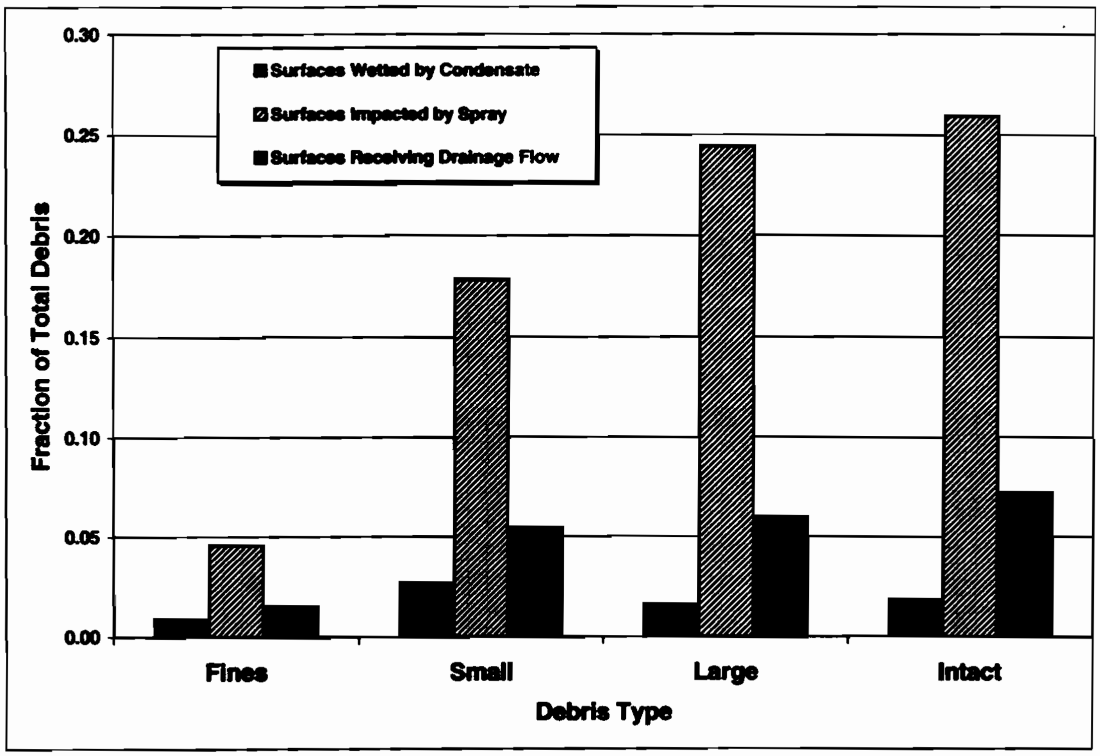


Figure IX-13. Blowdown Debris Dispersion by Surface Wetting.

In Figure IX-12, all of the LOCA-generated debris is distributed fractionally according to surface orientation (floor surfaces or other surfaces), whether the debris was captured within the break region (SGs 1 and 4), and debris type. This distribution reflects the debris-generation size distribution of Table IX-2 and the break-region capture fractions of Table IX-11. For the fines and small-piece debris, the largest fractions corresponded to floor surfaces outside or beyond the break region; debris preferentially settled onto the floors. Most of the debris that was captured within the break region was located on other structures that correspond to equipment, piping, and gratings within those SG compartments. For the larger debris, the majority of the debris was trapped within the break region by the congestion of structures. Nearly half of this debris either was deposited onto the floor of the break region or was assumed to fall to the floor after the break flows subsided. Most large debris that was ejected from the break region was predicted to fall out onto floor surfaces; therefore, small amounts of large debris were found on other structures outside of the break region.

In Figure IX-13, all of the LOCA-generated debris is distributed fractionally according to the surface wetting condition (condensate, sprayed, or spray drainage) and by debris type. Only relatively small quantities of debris were predicted to reside at locations where the debris would not be washed downward by the CSs or by the spray drainage. Conservatively speaking, the sprays falling from the upper dome would wash a majority of the surfaces within the SG compartments, as well as all of the upper floor surfaces and the refueling pool areas.

Although there is a relatively high degree of uncertainty with these blowdown transport results, the trends generally make sense. Because so little debris is protected from the CSs, these trends indicate a relatively high transport of debris to the sump pool.

VI.4.1.2 Fibrous Washdown Debris Transport

The CSs and condensation of steam throughout the containment and subsequent drainage to the sump pool would entrain substantial debris that was deposited onto the various surfaces and would transport the debris to the sump pool. In addition, these processes would degrade the fibrous insulation debris to some extent further, thereby creating more of the very fine, readily transportable debris.

VI.4.1.2.1 Surface Retention of Deposited Debris

The fraction of debris that stays on a specific surface, as opposed to being washed away, is referred to as the retention fraction. The fraction transported from a specific surface would then be 1 minus the retention fraction. Estimates of the retention fractions were essentially engineering judgments that were based on experience with small-scale testing during the DDTS. These experiments did not examine specifically the flow requirement needed to remove a piece of debris from a specified type of surface. Most of these tests dealt with either debris generation or airborne debris transport. One set of tests examined the erosion that was associated with fibrous debris inundated by water flow. During the conduct of these tests, experience with the handling of the debris provided some understanding regarding the ease or difficulty of forcing a piece of debris to move. These findings are summarized in Table IX-15. The estimated transport and corresponding retention fractions are shown in Table IX-16 and Table IX-17, respectively.

Debris transport due to condensate drainage would be expected to affect only the smaller debris. As condensation builds on a surface, it forms a thin film that subsequently drains and typically forms small rivulets of flow. This flow usually would move around significantly sized pieces of debris. Individual fibers could be entrained in the flow, or the fiber simply could be pushed to the sides of the rivulets. Some fine and small-piece debris certainly would transport, but the quantities of small debris transporting were estimated to be a small portion of the total. The DDTS's central estimate (realistic yet conservative) assumed that 1% of small debris transported (the extreme upper bound was 10%) but no large debris. The DDTS did not separate fines from small pieces. For this estimate, increasing the 1% to 2% for small-piece debris and increasing the 1% to 5% for the fines increased the level of conservatism. The larger debris was assumed not to transport because of condensate runoff.

Table IX-15. Fibrous-Debris Washdown Transport Trends

Debris Type	**Surfaces Wetted by Condensate**	**Surfaces Either Sprayed or Receiving Drainage Flow**	
		Without Intervening Floor Drains	**With Intervening Floor Drains**
Fines	Minority Transport	Nearly Complete Transport	
Small Pieces	Minority Transport	Majority Transport	
Large Pieces	No Significant Transport	Medium Transport	No Significant Transport
Intact Pieces	No Significant Transport	Minority Transport	No Significant Transport

Table IX-16. Estimated Fibrous-Debris Washdown Transport Percentages

Debris Type	Surfaces Wetted by Condensate	Surfaces Either Sprayed or Receiving Drainage Flow	
		Without Intervening Floor Drains	With Intervening Floor Drains
Fines	5%	99%	
Small Pieces	2%	70%	
Large Pieces	0%	50%	0%
Intact Pieces	0%	20%	0%

Table IX-17. Estimated Fibrous-Debris Washdown Retention Fractions

Debris Type	Surfaces Wetted by Condensate	Surfaces Either Sprayed or Receiving Drainage Flow	
		Without Intervening Floor Drains	With Intervening Floor Drains
Fines	0.95	0.01	
Small Pieces	0.98	0.3	
Large Pieces	1	0.5	1
Intact Pieces	1	0.8	1

Whenever fine and small-piece debris would be subjected to the substantial flows of the impacting CSs or the subsequent drainage of the sprays, the flow likely would entrain nearly all of the fine debris and a majority of the small debris. Test experience indicates that the CSs would wash fines from surfaces easily and carry those fines with the drainage to the sump pool. However, some of this fine debris would be pushed into relatively protected spots, corners, crevices, etc., where the debris would remain. Surfaces that were impacted directly by sprays and drained surfaces were grouped together for washdown transport because of the lack of information that was required to treat these two surface types differently. It was assumed that 99% of the fines would be transported from surfaces that were impacted by the sprays or drainage and that the other 1% experienced something less than total transport.

CSs also would wash substantial small-piece debris off structures, walls, and floors. The DDTS's central estimate was 50% (realistic yet conservative), with an extreme upper bound of 100%. Substantial quantities of debris likely would become trapped at locations that were protected from full spray flow due to the complex arrangements of containment equipment, piping, etc. It was assumed that 70% of the small debris would transport from surfaces that were impacted directly either by the CSs or by the subsequent drainage. This assumption adds additional conservatism to the DDTS's central estimate without becoming excessively conservative.

The 70% estimate was supported further by a simple floor-water drainage calculation, in which a uniform spray was applied to a floor area at a rate of flow corresponding to the containment-dome spray Trains A and B. A floor-area estimate indicates that ~800 ft^2 would be drained by each floor drain. A plant calculation estimated that the floor-water hold-up depth would be ~1.5 in. The separate-effect characterization of debris transport in water tests [NUREG/CR-6772, 2002] shows that a turbulent flow velocity as low as ~0.06 ft/s can cause a small piece of debris to tumble or slide along the floor. If circular drainage geometry is assumed, the transport estimate indicates that 30% to 40% of the floor area would not have sufficient flow velocity to transport small-piece debris. This calculation did not consider the effect of structures on the transport, which would create locations for debris entrapment. Therefore, the 70% estimate is a reasonable number for small-debris transport by the CSs.

For the large and intact pieces of debris, the surfaces were split into two additional categories based on whether the transport of the debris would encounter floor drain holes that would prevent further transport. A typical floor drain is ~6 1/2 in. in diameter and has a coarse grating that would stop any debris that is larger than ~3 in. square. A few floor drains have a relatively fine mesh screen over the hole. Floor surfaces are sloped to channel water to the drains. Large debris deposited onto the upper floors likely would have to pass through more than one of these floor drains to reach the sump. Large debris settling into the refueling pools would also have to pass through drains to reach the sump, some of which have a screen cover. The two largest of the refueling drains are nominal 6-in. drains without any cover or grating and are open during normal operation. Although a piece of large debris could pass through this 6-in. drain, the amount of debris would not be enough to treat these drains separately. It was assumed that these drains would stop further transport of large and intact debris.

Conversely, large and intact debris that is deposited at locations such as the SG compartments would not encounter any drain holes as the debris transports toward the sump pool. CSs would wash substantial quantities of large-piece debris off structures, walls, and floors. A portion of the large debris would be trapped on top of gratings and would not transport. Other large pieces would snag onto structures such that the sprays would not dislodge them. Substantial quantities of debris likely would become trapped at locations that are protected from full spray flow due to the complexities of containment equipment, piping, etc. Because large debris would transport less easily than small debris, it was assumed that 50% of the large debris was transported. The intact debris would be less likely to transport than the large-piece debris. Based on DDTS experience, the intact pieces of debris were significantly more likely to snag on structures than the large pieces, and substantial quantities of intact debris were likely to remain attached to the original piping. It was assumed that 20% of the intact debris would transport.

VI.4.1.2.2 Erosion of Debris by CSs

Experiments conducted in support of the DDTS analysis illustrated that insulation debris could be eroded further by the flow of water. Some debris erosion could occur because of the impact of the sprays and spray drainage flows, but the amount of erosion would not be great. The DDTS concluded that <1% of the fibrous debris eroded as a result of CS operation. Debris erosion caused by condensation and condensate flow was neglected. Debris containing insulation that is still in its cover would not be expected to erode further. The erosion of debris caused by the plummeting of the break flow into the sump pool is considered as part of the sump-pool transport analysis.

It was assumed that condensate drainage would not cause further erosion of fibrous debris and that intact or covered debris would not erode further. Erosion does not apply to fine debris because the debris is already fine. It was assumed that 1% of the small- and large-piece debris that was impacted directly by the sprays would erode. It was assumed that intact pieces of debris could not erode because its canvas cover would protect the fibrous materials.

VI.4.1.3 Quantification of Fibrous-Debris Transport

The transport of fibrous debris was quantified using the models presented in Section VI.3 and the input presented in Section VI.4.1. The quantified transport results are presented in Table IX-18. The table shows the transport fractions for each size category, as well as the overall transport fraction. It also shows the fractions of the total ZOI insulation that entered the pool, which were normalized to provide a size distribution for the debris entering the pool. About 57% of the ZOI fibrous insulation was predicted to transport to the sump pool, and nearly half of that would be the relatively transportable sizes. The transport fraction for the fines includes the erosion products from the predicted erosion of the small and large pieces of debris. The quantity of erosion products was approximately equal to 6% of the original generated fines.

Table IX-18. Fibrous-Debris-Transport Results

Debris Size Category	[illegible]	[illegible]	[illegible]	Distribution Entering Sump Pool
Fines	0.07	0.93	0.07	0.12
Small Pieces	0.26	0.66	0.17	0.30
Large Pieces	0.32	0.54	0.17	0.30
Intact Pieces	0.35	0.46	0.16	0.28
All Debris	1.00	0.57	0.57	1.00

VI.4.2 RMI Debris Transport

Roughly 85.7% of the insulation in the volunteer-plant containment is RMI. The debris-transport methodology discussed in Section VI.3 applies to RMI debris, as well as fibrous debris. Unfortunately, unlike the fibrous insulation, very little useful airborne transport data for RMI debris exist. Specifically, the capture fractions for the capture of RMI debris passing through structures such as gratings and of RMI debris inertially impacting surfaces have not been measured. Only secondary experience associated with RMI debris-generation experiments is applicable in this study. For RMI debris washdown, the pool transport velocities are available. Small-scale experiments suggest that RMI debris transports less easily than would the fibrous debris, primarily because the RMI debris is heavier. In addition, it would take substantially more RMI debris on the sump screen to block flow effectively through the screen than it would fibrous debris.

VI.4.2.1 RMI Blowdown Debris Transport

The capture fractions for RMI debris are likely much different from the corresponding fractions for fibrous debris. For fibrous debris, the capture fractions were very dependent on surface wetting; when the surfaces were dry, debris capture was minimal. For RMI, surface wetting may not be important. For instance, it seems likely that the capture of RMI on a grating depends on the foil folding over a bar in such a manner that it remains in place. Capture may depend on the debris remaining stuck on a structure. The amount of RMI debris that was captured by a grating could be significantly less than the amount of fibrous insulation; conversely, it could be substantially greater. Furthermore, the ability of flows to transport large cassette-like RMI debris is not known. Therefore, application of the Section VI.3 methodology required very conservative assumptions to compensate for the nearly complete lack of data.

VI.4.2.1.1 Break-Region Blowdown Debris Transport

It is conservative to overestimate the retention of debris within the SG compartments because subsequent debris washdown is more likely if the debris were in the SGs as opposed to being dispersed throughout the containment. Because the capture rates for RMI debris passing through a grating have not been determined, it was conservatively assumed that 100% of all RMI debris impacting a grating was stopped by that grating from further forward transport. Debris stopped on the underside of a grating likely could fall back once depressurization flows subside. Because the gratings do not extend completely across the SG compartments, substantial debris still could be propelled upward into the containment dome.

Likewise, the inertial capture of RMI debris by miscellaneous structures—such as pipes, beams, or vessels—or by inertial impaction whenever the flow makes a sharp bend—has not been determined. For instance, it would seem less likely that a piece of RMI debris would stick to a wall than would a small piece of fibrous debris. The fibrous-debris capture fractions for miscellaneous structures and sharp bends were applied to the RMI debris to conservatively overpredict the retention of RMI debris within the SG compartments. Applying these assumptions to the logic charts, which are similar to Figure IX-7, results in the conservative SG capture fractions shown in Table IX-19. The values for 2- to 6-in. and the larger-than-6-in. debris categories in Table IX-19 correspond to the values for the fibrous large- and intact-category values (shown in Table IX-11): a result of similar assumptions. The assumption that the gratings capture all of the RMI debris, even the smallest pieces, predicts substantially more RMI retention within the SG compartments than likely would occur in reality. The predicted over-conservative retention was necessitated by the lack of RMI transport data.

Table IX-19. Fractional Distribution of Debris Captured and Exiting Break Region

Location	RMI Debris Category		
	<2-In. Pieces	2- to 6-In. Pieces	>6-In. Pieces
Captured within SGs 1 and 4	0.64	0.70	0.82
Expelled to Dome	0.32	0.26	0.17
Expelled to Level 832	0.01	0	0
Expelled to Level 808	0.03	0.04	0.01

VI.4.2.1.2 Dispersion Throughout the Remainder of Containment

The 24-region subdivision of the containment free volume that was used in the fibrous-debris-transport estimate (Table IX-12) also was used for the RMI debris-transport estimate. The volume weighting factors that were estimated for fibrous-debris transport (Table IX-13) also were applied to the RMI debris because no rationale was found to weight the distributions otherwise. For RMI debris, no fine debris was postulated (i.e., even the smaller pieces of RMI debris should sink readily in water, as opposed to fibrous fines, which tend to remain in suspension). The predicted dispersion of RMI debris was judged to place more debris into locations where it subsequently would be predicted to transport with the CS drainage to the sump pool. The results of the blowdown dispersion by groups of volume regions are illustrated in Figure IX-14. As modeled, a majority of the debris was retained in the break region (SGs 1 and 4). In reality, it is likely that much more of the smaller debris would be blown free of the break region and into the upper dome region, where subsequent washdown to the sump pool would be substantially less than it would be if the debris were kept within the break region. However, the lack of RMI debris-transport data necessitated the conservative assumptions leading to these results.

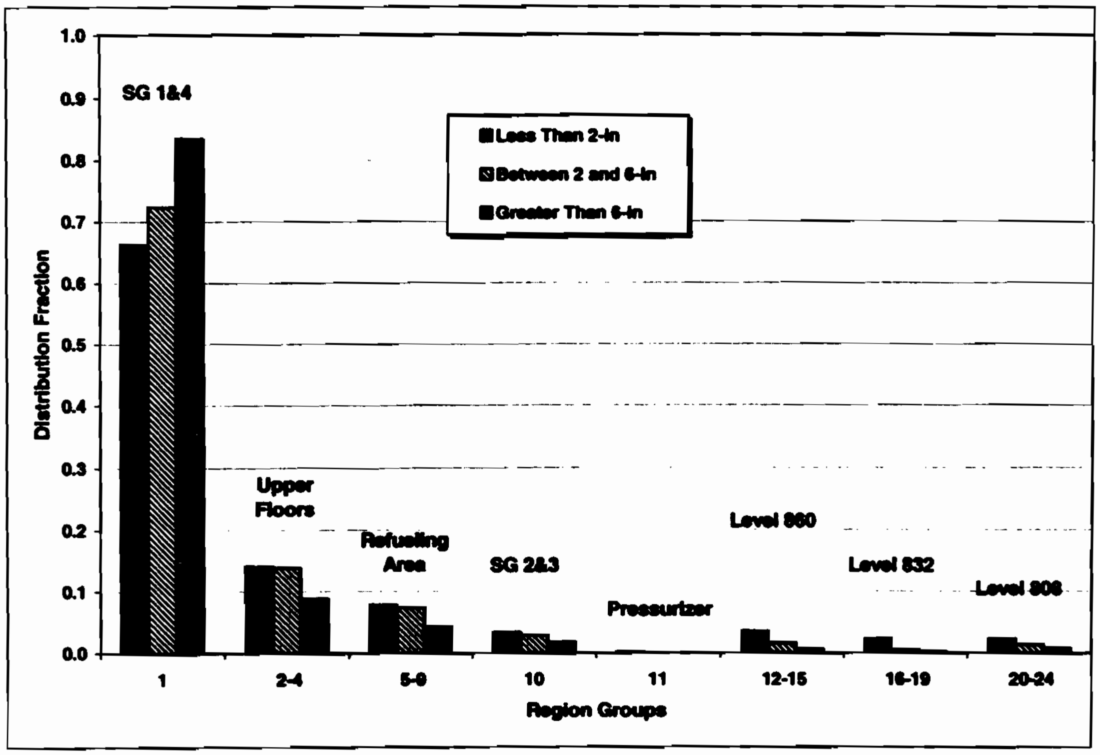


Figure IX-14. RMI Blowdown Distribution by Region Groups.

VI.4.2.1.3 Dispersion by Surface Orientation and Surface Wetness

A review of photos that were taken of RMI debris following RMI debris-generation tests indicates that RMI debris would reside preferentially on the floor surfaces [NEDO-32686, 1996, LA-UR-01-1595, 2001], although some RMI debris was caught on structures. However, the structures in these debris-generation tests were dry; therefore, it is not known if surface wetness would cause RMI to stick to wetted surfaces. Still, it is conservative to place the debris on the floors, where the subsequent washdown would be more effective. Therefore, the various surface-area-weighting factors were set to place most of the RMI debris on the volume region floors. It was assumed that 99% of the RMI debris would reside on the floor. The surface-area fractions shown in Table IX-14 apply to RMI debris as well as to fibrous debris. In these assumptions, ~99% of the RMI debris following blowdown was located where it either was impacted directly by the sprays or was located in the path of the spray drainage, leaving only 1% on surfaces that were wetted by condensation only.

VI.4.2.2 RMI Washdown Debris Transport

The RMI debris surface-retention fractions (i.e., the fraction that was not washed away) were estimated based primarily on engineering judgments and RMI pool debris-transport data. Small-scale testing of the transport of RMI debris in a pool of water demonstrated the ease or difficulty of forcing a piece of debris to move in a pool of water. Debris transport in a flowing layer of water that resides on a floor is similar to the transport of the debris in an established pool of water. Perceptions regarding the transport of RMI debris in nonpool situations are summarized in Table IX-20. The estimated transport and corresponding retention fractions are shown in Table IX-21 and Table IX-22, respectively.

Table IX-20. RMI-Debris-Washdown Transport Trends

Debris Type	Surfaces Wetted by Condensate	Surfaces Either Sprayed or Receiving Drainage Flow	
		Without Intervening Floor Drains	With Intervening Floor Drains
<2 in.	Minority Transport	Medium Transport	
2 to 6 in.	No Significant Transport	Medium Transport	No Significant Transport
>6 in.	No Significant Transport	Minority Transport	No Significant Transport

Table IX-21. Estimated RMI-Debris-Washdown Transport Percentages

Debris Type	Surfaces Wetted by Condensate	Surfaces Either Sprayed or Receiving Drainage Flow	
		Without Intervening Floor Drains	With Intervening Floor Drains
<2 in.	1%	40%	
2 to 6 in.	0%	30%	0%
>6 in.	0%	10%	0%

Table IX-22. Estimated RMI-Debris-Washdown Retention Percentages

Debris Type	Surfaces Wetted by Condensate	Surfaces Either Sprayed or Receiving Drainage Flow	
		Without Intervening Floor Drains	With Intervening Floor Drains
<2 in.	99%	60%	
2 to 6 in.	1%	70%	1%
>6 in.	1%	90%	1%

All debris that was deposited onto the SG compartment floors and the sump-level floors automatically was assumed to have entered the sump pool; this assumption was not indicated in the tables. This assumption primarily affected the debris that was deposited onto the break-region floor during either blowdown or washdown. The actual movement of this debris from the SG compartment floor into the outer annulus would be driven by the falling and spreading break flow; this would generally be expected to be a relatively high level of transport.

Debris transport resulting from condensate drainage would be expected to affect only the smaller debris. As condensation builds on a surface, it forms a thin film that subsequently drains and typically forms small rivulets of flow. This flow usually would not move around significantly sized pieces of debris. Significant transport of RMI debris does not seem likely; however, it is possible that some of the smaller debris could move with the condensate flow until the condensate flow linked up with more substantial water drainage. It was assumed that 1% of the debris that was <2 in. and subjected only to condensate drainage ultimately would transport to the sump pool. Furthermore, it was assumed that none of the debris that was >2 in. would transport to the sump pool.

Whenever pieces of debris <2 in. were subjected to substantial flows from impacting the CSs or from the subsequent drainage of the sprays, the flow likely would entrain a substantial portion of that debris. The evaluation of the transport of the smaller RMI debris that was exposed to sprays and/or spray drainage was based on a floor-pool drain velocity estimate and on the pool debris-transport threshold velocities. The drainage-flow velocity calculation assumed that a uniform spray was applied to an upper-level floor area corresponding to the containment-dome spray Trains A and B. A floor-area estimate indicated that ~800 ft^2 of floor area would be drained by each floor drain. A plant calculation estimated that the floor-water hold-up depth would be ~1.5 in. The separate-effect characterization of debris transport in water tests [NUREG/CR-6772, 2002] showed that a turbulent flow velocity of ~0.2 ft/s would be required to cause small stainless-steel RMI debris to tumble or slide along the floor. If it is assumed that circular drainage geometry exists, the transport estimate indicates that 60% to 80% of the floor area would not have sufficient flow velocity to transport small stainless-steel RMI debris, depending on the assumed thickness of the water layer. This conclusion resulted in the 40% transport estimate shown in Table IX-21. Because this calculation did not consider the effect of structures on the transport, which would create locations for debris entrapment, the 40% transport estimate is a reasonable number for the transport of RMI debris that is <2 in. by the CSs.

As was done for fibrous debris, pieces of RMI debris that were >2 in. were assumed not to pass through floor drains or refueling-pool drains. At locations where the larger debris would not encounter floor or refueling drains, 30% of the 2- to 6-in. debris and 10% of the >6-in. debris were assumed to transport. The corresponding fibrous-debris-transport number simply was reduced based on engineering judgment to account for the fact the RMI debris transports less easily than does fibrous debris. In any case, these two estimates affected only a relatively minor portion of the total debris.

Debris erosion of any significance would not happen to stainless-steel RMI debris; therefore, no erosion of the RMI debris by the CSs was considered in this study.

VI.4.2.3 Quantification of RMI Debris Transport

The transport of fibrous debris was quantified using the models presented in Section VI.3 and the input presented in Section VI.4.2. The quantified transport results are presented in Table IX-23. The table shows the transport fractions for each size category, as well as the overall transport fraction. It also shows the fractions of the total ZOI insulation that entered the pool. These fractions then were normalized to provide a size distribution for the debris entering the pool. Approximately 83% of the ZOI RMI was predicted to transport to the sump pool, but only ~20% of that amount was pieces <2 in.

Table IX-23. Fractional RMI Debris-Transport Results

Debris-Size Category	Category Generation Fraction	Size Category Transport Fraction	Fraction of ZOI Insulation	Distribution Entering Sump Pool
<2 in.	0.21	0.82	0.17	0.21
2 to 6 in.	0.12	0.76	0.09	0.11
>6 in.	0.67	0.85	0.57	0.68
All Debris	1.00	0.83	0.83	1.00

VI.4.3 Min-K Insulation Debris Transport

Less than 1% of the insulation in the volunteer-plant containment is Min-K insulation, a form of insulation referred to as microporous or particulate insulation. Although the transport methodology discussed in Section VI.3 also applies to Min-K insulation, a nearly complete lack of airborne transport data for this type of insulation exist, as well as debris-generation data, which were discussed in Section VI.3.2.3. Because of the lack of data for the generation of debris from Min-K insulation, the unknown erosion characteristics of this insulation, and the sparseness of the insulation within the containment (i.e., leads to a potential spatial nonuniform distribution), it was conservatively assumed that all Min-K located within the ZOI would be pulverized into a fine, highly transportable dust. If larger pieces of Min-K debris were inundated by the CSs, these pieces simply could dissolve into fine silt and transport with the spray drainage; however, this outcome is yet to be proven. Although <1% of the containment insulation is Min-K, this type of particulate debris could affect the sump-screen head losses significantly.

A conservative transport fraction for Min-K dust must be relatively high, and it seems likely that this fraction would be similar to the fraction for the transport of fibrous fines without the addition of erosion products, which was ~0.87. That is, the transport of fibrous fines generated from the ZOI to the sump pool was ~87%. (Note that the 93% value that was shown in Table IX-18 included erosion products.) Because the bulk of the 13% of fine fibers that did not transport was located on surfaces wetted only by condensate, it seems likely that a similar result would occur for the Min-K. For this study, it was assumed that 90% of the Min-K dust would transport to the sump pool.

VI.5 BLOWDOWN/WASHDOWN CONCLUSION

A methodology was developed that considers both transport phenomenology and plant features and that divides the overall complex transport problem into many smaller problems that either are amenable to solution by combining experimental data with analysis or that can be judged conservatively based on the foundation of debris-transport knowledge. The quantification of the methodology results in predicted transport fractions that are both conservative and plausible. The overall transport results are shown in Table IX-24. These transport fractions represent the fractions of the insulation by type that was initially located within the ZOI and that subsequently would transport to the sump pool. Detailed results, including size distribution information, are discussed in Sections VI.3 and VI.4.

Table IX-24. Overall Transport Results

Insulation Type	Overall Transport Fraction*	Debris-Size Distribution
Fibrous	57%	Table IX-18
RMI	83%	Table IX-23
Min-K	90%	All Dust

* Overall percentages are for demonstration only.

The overall transport fractions listed in Table IX-24 serve for demonstration purposes but are not valid for plant-specific evaluations because these fractions were calculated using LOCA-generated debris-size distributions that did not account properly for PWR jet characteristics. BWR jet characteristics were substituted for PWR jet characteristics because the PWR jet analyses had not been performed yet. When the PWR jet characteristics become available, it will be a simple matter to recalculate the overall transport fractions using PWR LOCA-generated debris-size characteristics.

Neither the debris-size distributions nor the overall transport fractions in this report are valid for plant-specific evaluations.

The transport fractions for each debris-size category are considered to be conservative for the LDFG insulation in the volunteer plant (but not necessarily for containments of other design). The fibrous-debris-transport analysis contained herein was based on LDFG insulation and may require adjusting for any high-density fiberglass insulation or mineral wool that may also be in the plant.

For the volunteer plant, a high percentage of the fine LOCA-generated debris most likely would transport to the sump pool via the spray drainage flows. The transport fractions tended to decrease as the debris size increased. A majority of the larger debris that was predicted to transport to the sump pool was stopped in the SG compartments that were associated with the break, where subsequent CS drainage was assumed to be readily capable of moving the debris downward to the pool.

The transport of the RMI and Min-K debris was skewed more conservatively toward larger transport fractions than was the fibrous debris because of the lack of transport data. Realistically speaking, the RMI might be expected to transport less readily than would the fibrous debris because it is heavier. However, a larger fraction of the RMI debris could be trapped in the break region (SG compartments), where it could be transported subsequently into the sump pool and thus the need to skew the transport fractions conservatively. A similar discussion applies to the Min-K because of the lack of LOCA debris-generation data, lack of erosion data, and the potential nonuniform placement of Min-K in the ZOI. Therefore, most of the Min-K must be conservatively assumed to transport to the sump pool as a fine dust or silt.

Conservative engineering judgments were made in this analysis at various steps along the way. The degree of conservatism that was associated with these judgments was intended to ensure conservative final results without straying too far from realistic behavior. The judgments were not intended to be upper bounding. For example, the erosion of LDFG by CSs was assessed in the DDTS as being <1%. In reality, the erosion may be significantly <1%. The 1% value was assumed to be conservative but not far from reality. In addition, many conservative judgments tend to compound as the analysis progresses.

The analyses herein considered only one break location (SG1), although they considered a range of break sizes at that location. Plant-specific analyses must consider a range of break locations. For the volunteer plant, LOCAs can occur within an SG compartment, which is likely the most probable location. A break in the same SG but at a different level likely would have a result similar to the one analyzed because most of the break effluent still would flow to the containment dome. A break in an SG compartment different from SG1 most likely would have a similar result, except that the debris would tend to enter the sump pool at different locations. A break outside the SG compartments, such as in a main steam line, would behave differently than a break inside an SG compartment and probably should be analyzed separately. A break in the pressurizer certainly would be different because that compartment does not vent directly to the containment dome as with the SG compartments (i.e., no major upper openings exist). Therefore, a larger fraction of the debris might be driven out of the pressurizer compartment directly into the sump area, but the total quantity of debris might be substantially less than a primary-loop piping break. Neither a pressurizer-line break nor a main steam-line break was analyzed herein.

In performing blowdown/washdown analyses, it is important that

- the debris-size categories match the characteristics of the debris-transport behavior;
- the break region is analyzed in substantial detail because so much of the debris capture is likely to occur in this region;
- the debris capture along the primary exits from the break region also should be analyzed in substantial detail;
- CS drainage patterns should be determined to support the washdown analysis and to indicate where the debris would enter the sump pool and how the spray drainage would impact sump pool turbulence; and
- vulnerable spray-drainage pathways, where potential debris blockage might occur, should be identified.

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Attachment 1 to APPENDIX VI:

VOLUNTEER-PLANT SPRAY-WATER DRAINAGE ANALYSIS

INTRODUCTION numbered headings throughout attachment?

A postulated LOCA in the volunteer plant would distribute insulation debris throughout the containment, whereby the subsequent drainage of spray water following the LOCA would transport portions of this insulation debris toward the recirculation sump screens. A best estimate of how the water would drain to the sump was performed to support subsequent debris-transport calculations. The analysis will help to identify spaces and surfaces where insulation debris likely would not be washed away by sprays or drainage flow (e.g., an area that was not impacted by sprays and has too little drainage flow to transport debris). The analysis will help to determine how the drainage water enters the sump pool, which in turn will affect debris transport within that pool.

SYSTEM DESCRIPTIONS

The CS systems in the volunteer plant consist of two independent trains (Trains A and B), with headers located in four containment regions. Spray nozzles are located in one of four regions of the containment:

- Region A—Containment dome spraying down toward Level 905;
- Region B—Below Level 905 spraying Level 860;
- Region C—Below Level 860 spraying Level 832; and
- Region D—Below Level 832 spraying Level 808.

The specifications are shown in Table 1 **consistently number by section? e.g., VI-1?**for both trains in Unit 1, combined. Spray Train B has one more nozzle in the dome than Train A; therefore, the flows that are associated with single train operations constitute essentially half of the flows shown for both trains. Unit 1 has seven more nozzles than Unit 2. The drainage estimate performed for Unit 1 is applicable also to Unit 2.

Table 1. Unit 1 Spray Nozzle Summary

Spray Region	Number of Nozzles	Nozzle Flow (gpm)	Region Flow (gpm)
A	545	20	10,900
B	134	20	2680
C	28	20	560
D	54	20	1080
Total	761	20	15,220

The containment was designed to drain the spray water down to the containment recirculation sumps. Furthermore, the containment apparently was designed to minimize water holdup, thereby maximizing the depth of the sump pool. Several features of the containment determine the primary drainage pathways in the containment. These features include the following.

<u>Floor Drains</u>—A primary means of draining spray water is the floor drains that drain water from one floor directly down to the next floor. A typical drain, which is ~6 1/2 in. in diameter, is shown in Figure 1. **consistently number by section? e.g., VI-1?**At the top of this figure, another type of drain is shown that drains directly to the containment sump. Floor surfaces are sloped to channel water into the drains.

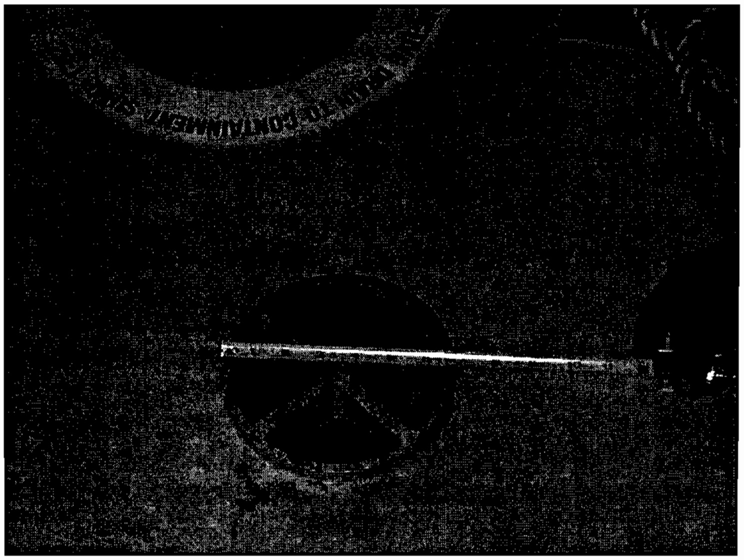

Figure 1. Typical Floor Drain.

<u>Water Barriers</u>—Water drainage is controlled by water barriers (curbs)—both concrete and metallic types—that are placed around floor-area perimeters to prevent water from draining from those perimeters. However, these barriers do not cover the entire perimeter of a floor. Gaps exist in the barriers at locations such as the areas around walkways and ladders. In many places, water can flow from a floor perimeter onto another floor, into the gap between the internal structures and the outer wall, into an SG compartment, into a stairwell, etc. A typical curb is shown in Figure 2. Figure 3 shows another curb next to an SG compartment that illustrates a discontinuity in a curb.

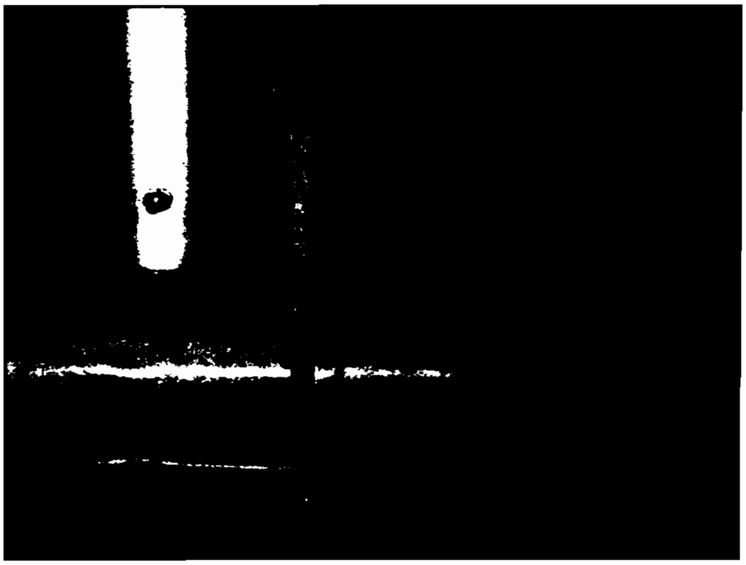

Figure 2. Typical Concrete Curb.

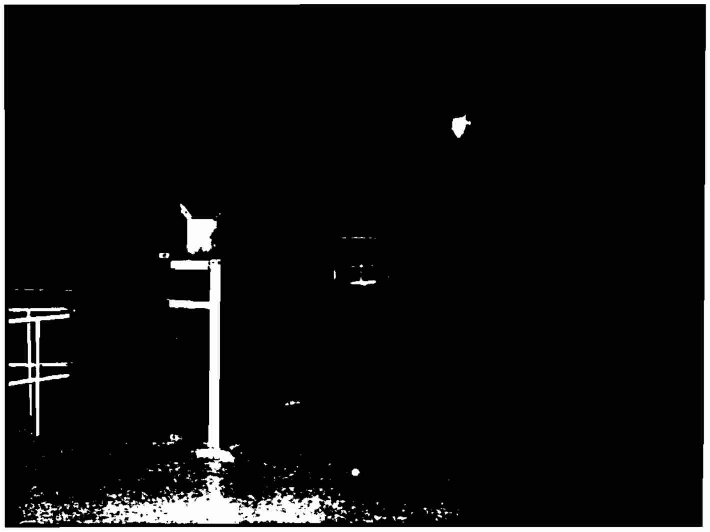

Figure 3. Gap in Concrete Curb Surrounding an SG Compartment.

<u>Refueling Pool Drains</u>—A substantial portion of the dome sprays will fall into the refueling cavity and accumulate in the three pool areas of the cavity. During normal operation, the pool drains are open, allowing spray water to drain down to the sump. The pool drains consist of 4-in. and 6-in. sizes. Figure 4 shows the drains in the pool that are used to store the reactor vessel lower internals during refueling. A 4-in. drain with a cover screen (with holes ~1/4 to 1/2 in. in diameter) is shown near the center of the photo. Two 6-in. drains also are shown in the upper-right (cover off) and lower-right corners (cover in place). These 6-in. drains are closed off with blind flanges during refueling and are uncovered during normal operations. The 4-in. drains drain down into the labyrinth of rooms on Level 808, which is located directly below the refueling pools. The two 6-in. drains drain to SGs 3 and 4. The pool that is used to store and transfer fuel is drained to Level 808 by a single 4-in. drain. The pool that is used to store reactor vessel

upper internals during refueling has a single 4-in. drain, which drains into the pool that stores the lower internals.

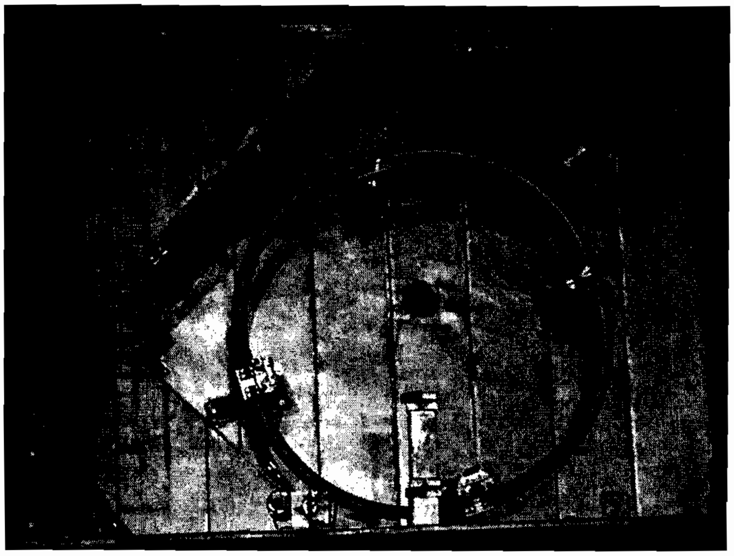

Figure 4. Refueling Pool Drains.

Floor Gratings—Water drainage between floors also occurs through the floor gratings that cover several open areas in the floors (e.g., the equipment-transfer floor hatches).

Stairwells—At several staircases, water can drain from one floor to the next. Two primary staircases extend all the way from sump Level 808 up to the top floor at Level 905.

APPROACH number as heading?

A review of containment drawings and plant documents led to many general observations.[*]

- Little, if any, water is expected to drain down the elevator shaft by way of the elevator doors. The elevator shaft was not treated as a wetted drain perimeter in the plant's minimum pool calculation, and the floors generally slope away from elevator. Furthermore, elevator doors may prevent water entry into the elevator shaft.

- The pressurizer compartment should remain essentially dry. A roof covers the compartment so that sprays do not enter this compartment. Drains and sloping floors generally prevent water flow into this compartment at other entrances.

- Water entering the SG compartments consists of dome-spray droplets falling directly into those compartments. Droplets falling onto the wall-tops and floor that are located between or near the SG compartments likely will flow into the SG compartments. Also, the two 6-in. refueling-pool drains flow directly into the SG compartments.

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[*] The most useful drawings were floor layouts that showed floor slopes, water barriers, and floor drains. The most useful document was a plant calculation of the minimum sump-pool height.

- Water entering the stairwells consists of spray droplets falling directly into stairwells and of some water overflowing a floor perimeter.

- Water entering the refueling cavity consists of spray droplets falling directly into the cavity. This water includes droplets that are falling onto walkways surrounding the refueling pool and that subsequently would flow into the pool.

- Water entering the gap between the inner containment structure and the containment outer wall consists of spray droplets impinging the outer containment wall and subsequently flowing down the liner and of water from gaps in the water barriers along the floor perimeters.

- Substantial quantities of water are intended to drain from one floor to the next by way of the floor drains between the floors.

Because of the complexity of the water drainage, many simplifying assumptions and engineering judgments were necessary. The primary assumptions include the following.

- All spray systems were active (only one possible spray scenario was evaluated).[*]

- No blockage of drain flows by debris was postulated.[†]

- Dome spray droplets fall vertically and distribute uniformly across the containment cross section before encountering any containment structure. Distribution was based on cross-sectional areas.

- Crosswalks on Level 905 that are directly between the refueling cavity and the SG compartments drain into those compartments.

- Refueling cavity walkways on Level 860 drain into pools.

- Levels 873 and 851 do not have floor drains (floor drains not shown in drawing).

- Water draining onto Level 849 from Level 860 subsequently drains to Level 832.

- Water drains that drain directly to a containment sump (e.g., the one shown in the upper portion of Figure 1) are neglected. These specialized drains were not delineated in the drawings and are assumed to be substantially fewer in number than the main floor drains.

Engineering judgments were necessary where insufficient data were available to estimate drainage accurately.

The calculational approach included the following steps:

- the locations of all spray nozzles were identified;

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[*] The scenario where one train operates and one train is inactive can be estimated by dividing all flows for both trains by a factor of 2.

[†] Insulation debris could block a floor drain or a refueling pool drain.

- the dome spray impacting and running down the containment liner was estimated;
- the main floor areas on Levels 808, 832, 860, and 905 were nodalized into three sections for each floor;
- the locations where the spray droplets would settle were identified; and
- the drainage process was tracked from the uppermost surfaces down to Level 808.

The dome spray nozzles, arranged around four rings for each of the two trains, are aimed in four different directions. Some of the nozzles apparently are aimed to spray the dome liner. A portion of this spray impacting the liner subsequently should drain down the liner itself. The number of nozzles aimed in each of the four directions was tabulated for each ring. Then the spray impact and runoff was judged for each ring location. Of the 10,900-gpm total dome spray flow, 700 gpm was estimated to flow down the liner.

Figure 5 illustrates the subdivision of the main floors. Section 1 includes the side of the containment where the main steam and feedwater lines penetrate the containment. Drainage on this side would be distinctly different from the remainder of the containment. Section 2 includes unique features such as Level 849 and Level 832; sprays do not extend into this section. Section 3 includes the remainder of the floors.

To estimate the distribution of settled dome spray water, the containment cross-sectional area was estimated for each section of floor, refueling cavity, SG compartment, open area, etc. It was assumed that the spray droplets would fall uniformly onto these areas. Once the settled flows were determined, the drainage from floor to floor was estimated, starting with the uppermost floor surface. For each floor section, a drainage distribution was estimated, based on floor sloping relative to drainage pathways.

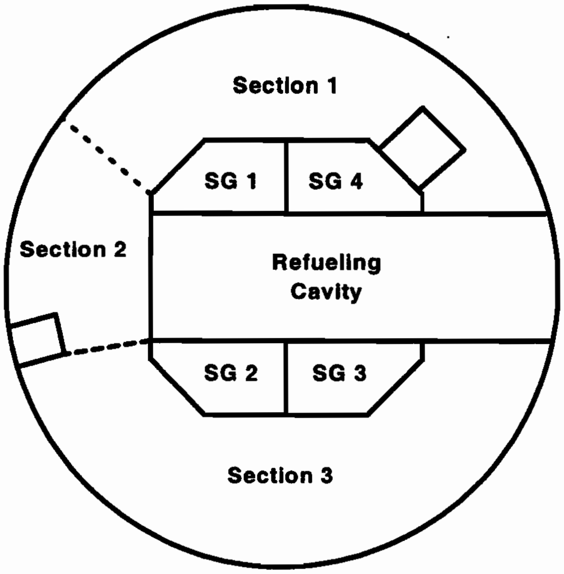

Figure 5. Schematic of Floor Sections.

The overall spray drainage is shown in Figure 6. The dashed lines represent spray droplets falling onto a surface* (the arrow head indicates the surface receiving the droplets). The numbers indicate flow rates in gallons per minute. The solid lines indicate water draining from one surface to another or water falling into and through a stairwell or the outer wall gap. A diagram illustrating where the water enters the Level 808 sump pool is shown in Figure 7.

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* The surfaces are not drawn to scale.

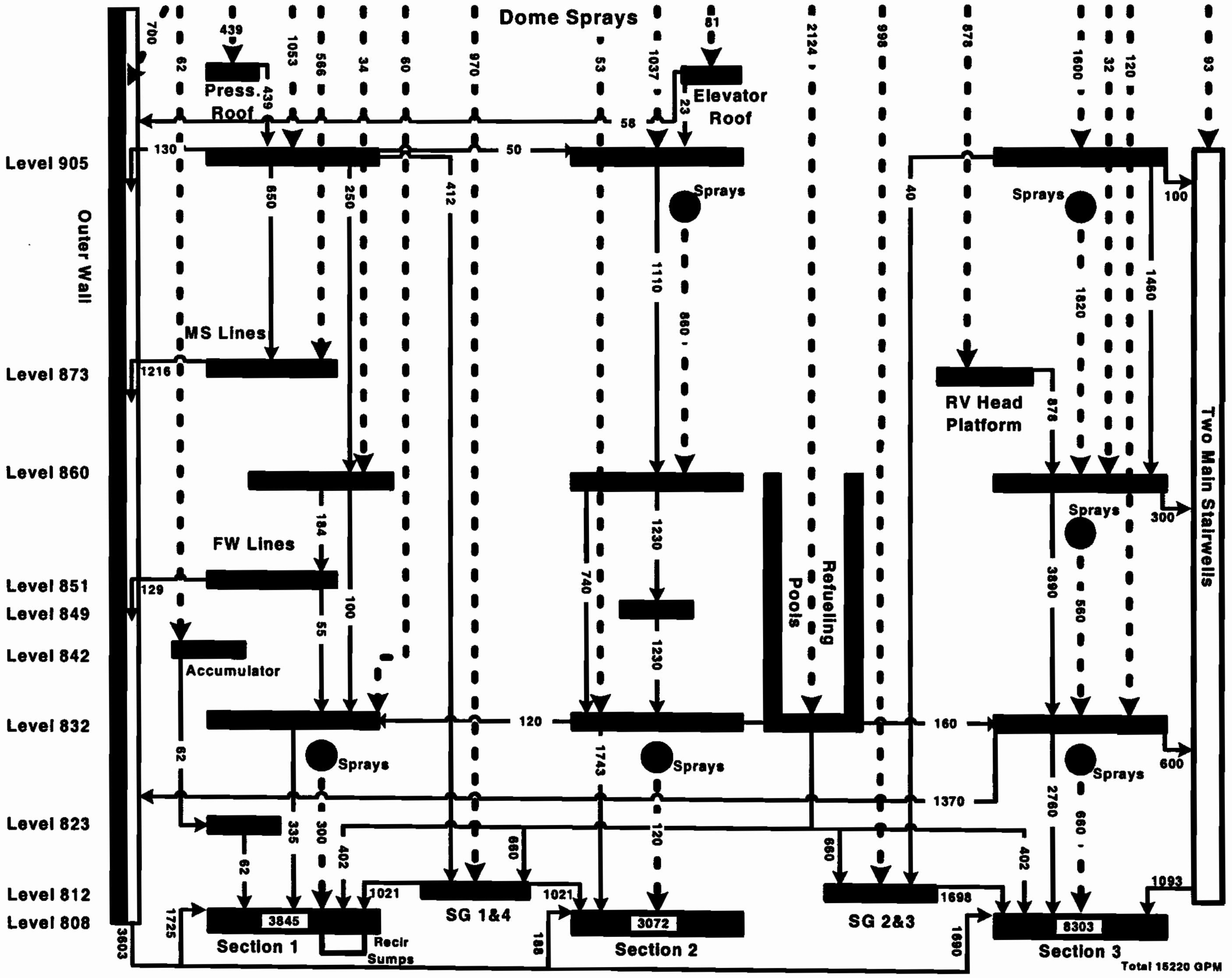

Figure 6. Spray-Water Drainage Schematic.

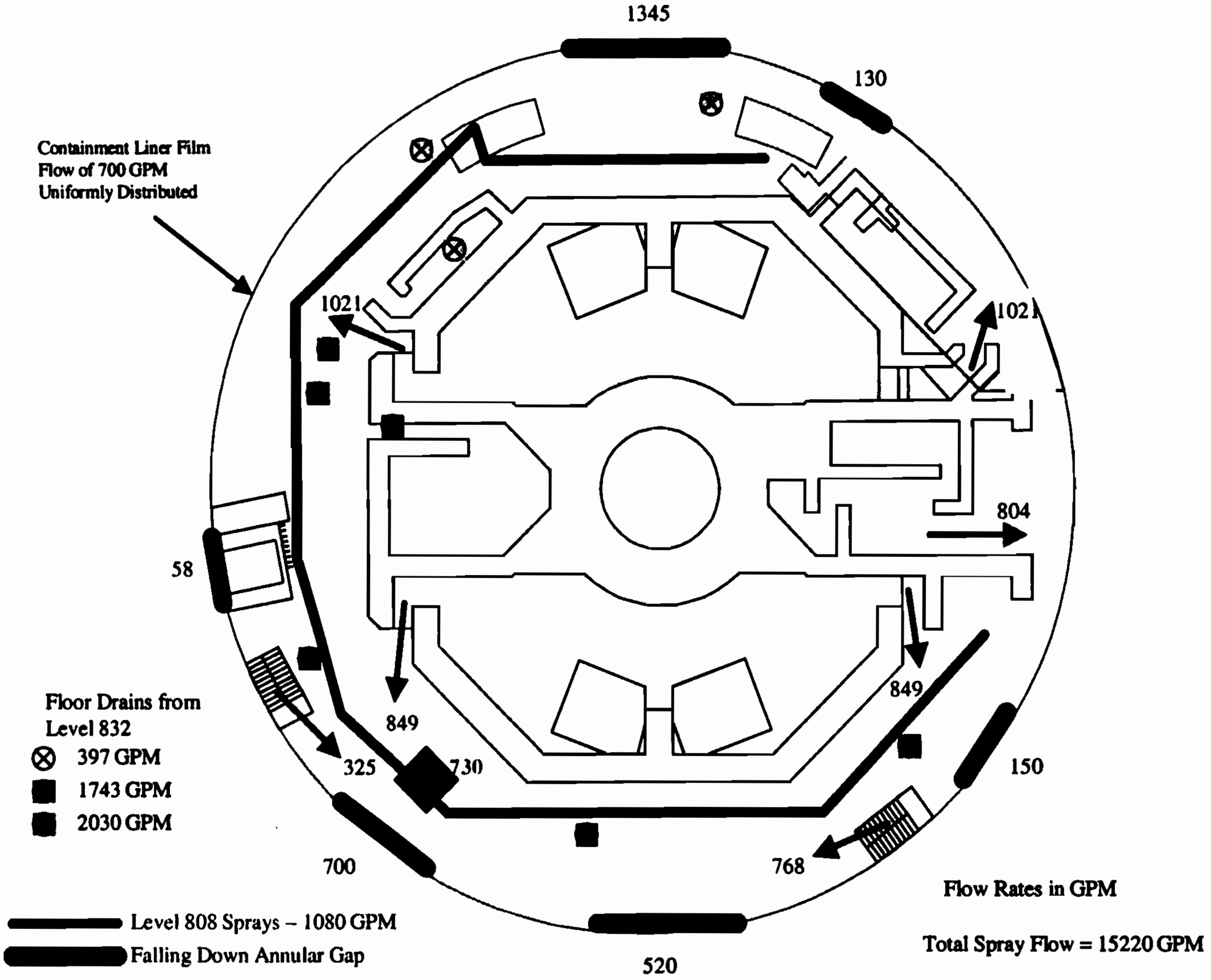

Figure 7. Spray-Water Drainage to Level 808 Sump Pool.

APPENDIX VII: Characterization of PWR Latent Debris

The United States (US) Nuclear Regulatory Commission (NRC) has recently initiated a study conducted through Los Alamos National Laboratory (LANL) and the University of New Mexico (UNM) to characterize latent debris samples collected at five individual volunteer plants. The focus of this work is to study physical attributes of dust and dirt, such as particulate-to-fiber mass ratio, size distributions of particulate, material and bulk densities, and hydraulic parameters, including the specific surface area. Because of variations in plant collection methods and sampling schemes, it is not possible to make estimates of total latent-debris inventories. This appendix documents preliminary results of that study that are relevant to the supplementary guidance provided by the staff in Section 3.5 of the safety evaluation report (SER).

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A total of five sets of samples were received at LANL for analysis, but only four were totally characterized. The fifth set was not characterized fully because it was dominated by paint chips generated from pressure washing and was therefore deemed to be unrepresentative of pressurized-water-reactor (PWR) containment debris. Material property data collected for the latent-debris samples establish the basis for preparation of a particulate-debris simulant that is suitable for large-scale head-loss testing at UNM. The objective of head-loss testing is to quantify the hydraulic properties of latent debris that are needed for the proper application of the NUREG/CR-6224 debris-bed head-loss correlation.

The experimental scope for sample characterization was as follows.

1. The debris was removed from its shipping container and transferred to plastic laboratory containers for gamma-spectrum counting.
2. The "fiber" and "particle" fractions were separated from the remaining (or "other") debris items by manual manipulation, sieving, and water rinsing.
3. Particulate size distributions were obtained by graduated sieving.
4. The weight of fine particles attached to swiping (masslin) **need to correct throughout SER** cloth or filter paper was determined by mass balance and comparisons of clean collection media to soiled collection media.
5. The fiber thickness/diameter was determined by scanning electron microscopy (SEM) and microphotographic statistics.
6. The material and bulk densities of fibers were estimated by mass measurement combined with volume estimates obtained from water displacement and direct measurement in graduated columns, respectively.
7. Particle surface area and density measurements were taken using state-of-the-art nitrogen adsorption techniques.
8. Scanning electron microscope/energy-dispersive spectroscopy (EDS) methods were used to characterize the chemical composition of representative particulate and fiber samples.

Figure VII-1 illustrates a typical variety of composition and proportion between particulate, fiber, and other larger pieces that are assumed to have minimal transport potential. All plants submitted multiple samples ranging from a few grams to several thousand grams that exhibited similar characteristics. For some plants, the samples had to be combined to obtain meaningful measurements; for others, each individual sample could be fully characterized.

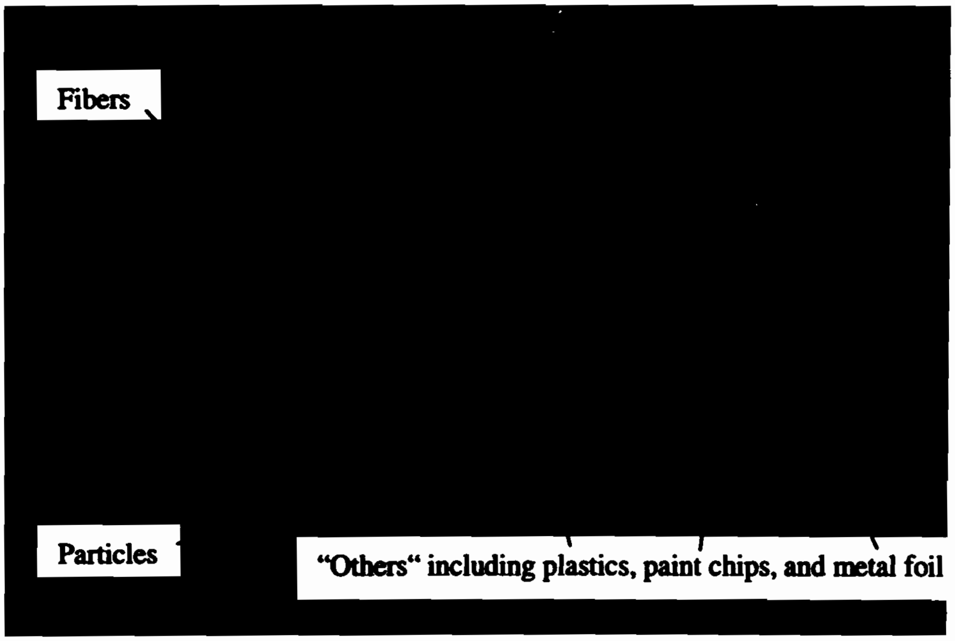

Figure VII-1. Representative Latent-Debris Components from a Single Volunteer Plant.

Objects larger than a 0.132-in.-mesh-size sieve were classified as a debris type "other" than particulate or fiber. This category of size, composition, and characteristics should be removed from any plant-specific samples that are collected before applying any mass fractions reported in this appendix. Larger latent debris types are not assumed to be transportable at recirculation pool velocities and so do not contribute to long-term increases in sump-screen head loss. However, any of this debris category that is present on the sump-pool floor may readily transport to the sump during pool fill-up. Table VII-1 presents the range of particulate and fiber mass fractions that were measured for samples that were characterized after the larger pieces were removed. From these data come the generic recommendation that 15% of the transportable latent debris be assumed to be fiber.

Each volunteer plant used a different collection method and sampling scheme. When separating particulates by wet sieving into fractions (>2 mm, 500 µm to 2 mm, 75 µm to 500 µm, and <75 µm), it became apparent by comparing plants that scraping and bristle-brush collection were not effective at capturing the smaller particulate fractions. This conclusion was further reinforced by SEM photos of filter papers and cloth swipes that showed significant loadings of particles <10 µm in diameter. High-efficiency particulate air (HEPA) filter vacuuming with the brush attachments or manual swiping with lint-free (masslin) cloth are recommended collection methods for characterizing plant-specific latent debris loadings.

Table VII-1. Particulate and Fiber Mass Fractions for Volunteer Plants A–D

Plant	Particle Weight	Fiber Weight	% Particle	% Fiber
A	5.42	1.04	84	16
B1	214	20	91	9
B2	369	64	85	15
B3	390	37	91	9
B4	592	47	93	7
B5	792	34	96	4
B6	122	50	71	29
B Total	2479	252	91	9
C	13.77	0.76	95	5
D1	2.51	0.47	84	16
D2	0.29	0	100	0
D3	12.45	0.28	97	3
D4	34.34	2.20	94	6
D6	5.56	0.1	98	2
D8	9.15	0.09	99	1
D10	11.98	0.74	94	6
D15	74.92	7.0	91	9
D Total	151.2	10.88	93	7

Sample Range	Total Particulate	71%–100%
	Total Fiber	0%–29%
Plant Range	Particulate	84%–95%
	Fiber	5%–16%

The material density of characterized fibers was found by water displacement measurements of 10 plant samples to range between 1.0 to 1.9 g/cm^3. The mean value of 1.5 g/cm^3 is recommended for use if needed in generic latent-debris assessments. However, a more relevant parameter of fiber is the dry-bed bulk density that can be used to estimate the volume of fiber needed to form a 1/8-in.-thick thin bed across the wetted-screen area of a given sump configuration. This property and the suggested application is comparable to the use of the as-manufactured bulk density for fiberglass insulation.

The dry-bed density of latent fiber depends greatly on the amount of compaction applied for the measurement. Several alternatives were tried, but ultimately the staff recommends using the fiberglass density of 2.4 lbm/ft^3 = 38.4 kg/m^3 as a surrogate for dry latent debris. Similarly, fiberglass hydraulic properties should also be used as a surrogate for latent fiber. These recommendations are supported by the following rationale. First, in cases where fiberglass debris is present on the screen, minor inaccuracies in the latent fiber properties will not affect head-loss calculations. Second, where latent fiber is the dominant fibrous debris source and there is sufficient quantity to form a thin-bed filter, maximum head loss will be dominated by the properties of particulates captured on the fiber bed. Again, the difference between the actual hydraulic behavior of latent fiber and the presumed properties of fiberglass will not affect head-loss calculations adversely.

Particulate densities for each size fraction and volunteer plant were measured very accurately using the Brunauer-Emmett-Teller (BET) nitrogen adsorption method. Densities of particulates in the debris range from 2 to 4 g/cm^3 with only a few exceptions, and densities for most of the samples range between 2.5 and 3.0 g/cm^3, regardless of their particle size. These data form the basis of the recommendation for a nominal latent particulate density of 2.7 g/cm^3.

A nominal size distribution of particulates found in the latent debris samples was used as a starting point to develop a formula for surrogate particulate debris that could be tested in a vertical-flow test loop at UNM. This apparatus permits measurement of pressure drop across a debris bed of known composition under a range of water velocities. Hydraulic parameters of the debris bed can then be inferred from differential pressure data by iteratively applying predictive correlations until the model results envelop a range of observed data. Material-specific parameter values, such as the specific surface area that are inferred in this manner, are only appropriate for use with the particular head-loss formula with which they were derived. In this case, the NUREG/CR-6224 head-loss correlation was applied. Microporous flow-resistance tests were performed on both the latent-debris samples and the surrogate formula to confirm that the surrogate could produce reasonably representative yet conservative hydraulic behavior.

Equivalent mass fractions of common sand and clay-based soil were used to recreate the size distribution of the latent particulate. Over a set of well-conditioned head-loss tests where the surrogate particulate was tested in combination with fiberglass insulation, the specific surface area of the surrogate was estimated to be 106,000 ft^{-1}. Analyses of these tests were complicated by penetration of the debris bed by extremely fine clay silt that continued to circulate in the test loop. Within the range of the tests where flow velocities at the screen are <0.2 ft/s (uncompressed fiber bed) and the estimated particulate-to-fiber mass ratios cannot exceed 3, the estimated particulate loading on a postulated debris bed can be reduced by 7.5% (one-quarter of the <75-μm mass fraction) to accommodate realistic debris-bed penetration of latent fine particulates.

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The surrogate debris formula was further refined by eliminating the latent-debris fraction with nominal dimensions >2 mm because the particles (sand grains) are not likely to transport at pool velocities <0.5 ft/s that may exist near the screen under recirculation conditions. This size fraction represents ~22% of the particulate mass on average that can be discounted from the particulate inventory that is available for long-term transport under recirculation. This size fraction may be subjected to high-velocity transport during fill-up, and so the fractional decrease was only recommended for latent-particulate inventories residing above the flood level.

APPENDIX VIII: FORMATION AND PREDICTION OF THIN BED HEAD LOSSES AND BEHAVIOR OF COMPACTED CALCIUM SILICATE

VIII.1 INTRODUCTION

Relatively high head losses have occurred across relatively thin layers of debris consisting of fibrous and particulate debris, whereas substantially thicker debris beds have cause lesser head losses. This behavior has been referred to as the "thin-bed effect" where the head loss per unit thickness of debris is relatively high. Such debris beds have caused head losses high enough to threaten BWR ECCS sump recirculation pumps with modest quantities of debris on the strainers and such debris beds can threaten PWR sump recirculation sump screens as well. These types of debris beds have occurred operationally at nuclear power plants, have been created during head loss testing, and have been analytically simulated with the NUREG/CR-6224 head loss correlation.

VIII.2 OPERATIONAL INCIDENCES

Two operational strainer clogging events occurred at the Perry Nuclear Power Plant (PNPP) and one event at the Limerick Generating Station, Unit 1, whereby in each event a high head loss occurred with a relatively thin layer of debris present on the strainers.

Perry Nuclear Power Plant On May 22, 1992, during a refueling outage inspection at the PNPP, debris was found on the suppression pool floor and on the RHR suction strainers. This is discussed in NRC IN-93-34, "Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment," May 6, 1993. In addition, the buildup of debris on the strainers caused an excessive differential pressure across the strainers and resulted in deformation of the strainers. PNPP replaced the strainers and cleaned the suppression pool. Then in March 1993, several safety relief valves lifted and the RHR was used to cool the suppression pool. The strainers were subsequently inspected and found covered with debris. A test of the strainers in the as-found condition was terminated when the pump suction pressure dropped to zero. The debris on the strainers consisted of glass fibers (from temporary drywell cooling filters inadvertently dropped into the suppression pool); corrosion products, and other materials filtered from the pool water by the glass fibers adhering to the strainer surfaces [IN-93-34, Supplement 1]. The suppression pool debris also consisted of general maintenance types of materials and a coating of fine dirt that covered most of the surface of the strainers and the pool floor. Fibrous material acted as a filter for suspended particles, a phenomenon not previously recognized by the NRC or the industry. This event suggested that filtering of small particles, such as suppression pool corrosion products (sludge), by the fibrous debris would result in significantly increased pressure drop across the strainers.

Limerick Generating Station, Unit 1 Another event occurred at the Limerick Generating Station, Unit 1 on September 11, 1995. This is discussed in NRC IN-95-47, "Unexpected Opening of a Safety/Relief Valve and Complications Involving Suppression Pool Cooling Strainer Blockage," November 30, 1995. A safety relief valve (SRV) opened on Unit 1 while at 100% power. Before the SRV opened, Limerick had been running Loop A of the RHR in suppression pool cooling mode. The operators initiated a manual scram in response to the SRV opening, and a second loop (Loop B) of suppression pool cooling. Approximately 30 minutes later, fluctuating motor current and flow were observed on Loop A. The cause was believed to be cavitation and Loop A was secured. Following the event, inspection by a diver revealed a thin mat of material covering the Loop A strainer. The mat consisted of fibrous material and sludge. The Loop B strainer had a similar covering, but to a lesser extent. Limerick subsequently removed about 635 kg of debris from the pool. Similar to the PNPP events, the mat of fibers on the strainer surface converted the strainer into a filter, collecting sludge and other material on the strainer surface.

These strainer clogging events caused substantial loss of pump flow and fluctuating conditions indicated cavitation due to debris beds consisting primarily of fibers and corrosion products. The debris bed descriptions "coating of fine dirt" and "thin mat of material" describe thin beds of debris. The conclusion is that relatively thin layers of debris caused relatively high head losses. Following these events, the thin-bed effect behavior has been experimentally replicated and analytically simulated, which has resulted in an understanding of how such thin layer caused such high head losses.

VIII.3 PHENOMENOLOGICAL DESCRIPTION

The head loss across a bed of debris is directly related to the porosity of that debris bed, i.e., the lower the porosity the higher the head loss. For fibers similar to low-density fiberglass insulation such as NUKON™, the porosity of a bed of these fibers typically ranges from 90% to 99% depending upon the mechanical compression of the bed by the frictional drag caused by the flow. The porosity decreases with the compression of the debris. The porosity of a bed of particulate (without any fibers), however, is substantially less than the porosity of a fibrous debris bed. The iron oxide corrosion products sludge that is typically formed in BWR suppression pools has a porosity of about 80%. In sludge, the particulate cannot be compressed significantly because the particles are hardened and the particles are already in contact with other particles. The porosity varies with types and size of the particulate. Another example, common sand in soils has porosity in the neighborhood of 40% to 46%.

When fibrous and particulate debris are mixed, the porosity of the mixture depends upon the relative quantities of fiber and particulate and the mechanical compression of the fibers. When quantities of particulate are relatively small compared to the quantities of fibrous debris, the individual particles are trapped in the fibers such that the particles do not generally interact. Such a debris bed has been referred to as a mixed debris bed for which an example is shown in the photo in Figure VIII.1. The particles contributed to the head loss but the particles still resist flow individually.

As the quantity of particulate relative to the quantity of fibers increases (typically referred to as the particulate to fiber mass ratio), the contribution of particulate increases and head loss increases. As head loss increases, the fibrous bed further compacts thereby reducing the spacing between the fibers, which also increases the ability of the bed to filter finer particulate from the flow. Eventually, further increases in the particulate to fiber mass ratio results in increasing particle interaction. When this interaction reaches its maximum limit, based on the particulate bulk density or sludge density, further compaction becomes difficult. As this maximum limit is approached, the bed porosity approaches that of the particulate sludge which is substantially less than the fibrous debris and the head loss increases correspondingly. Once the porosity of the debris bed approaches the porosity of the particulate sludge, high head loss can occur in a thin layer of debris, i.e., the thin bed effect. A definition of the thin-bed effect is then:

The thin-bed effect refers to the debris bed condition in a fibrous/particulate bed of debris whereby a relatively high head loss can occur due to a relatively thin layer of debris, by itself or embedded as a stratified layer within other debris, because the bed porosity is dominated by the particulate and the bed porosity approaches that of the corresponding particulate sludge.

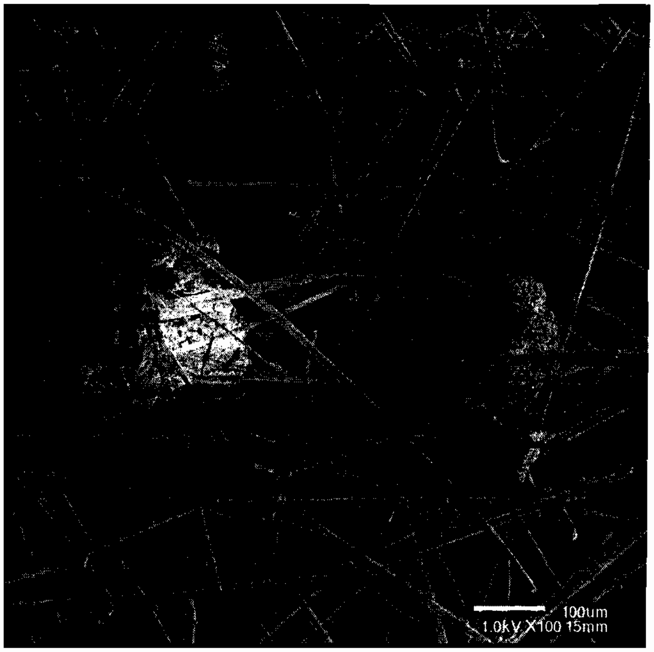

Figure VIII.1. Example of Particulate Embedded in Fibrous Debris[1]

During the Perry and Limerick events, relatively small quantities of fibrous debris and relatively large quantities of corrosion product particulate debris were discovered in each suppression pool. When the recirculation pumps were operated, both the fiber and particulate would have been drawn to the strainers but initially the particulate would pass through the strainers whereas the fibers preferentially filtered from the flow. Once sufficient fibers were accumulated on the screens, the fibers filtered the particulate. Subsequent accumulation would then have been dominated by the particulate such that the resultant accumulation would appear to be a layer of iron oxide sludge. As such, the pump had to draw water through this layer of sludge which had porosity near that of about 80%. Fibers would have been interspersed throughout the bed but likely were concentrated nearer the screen surface and those fibers would have been tightly compressed by associated pressures. The bed would have a high particulate to fiber mass ratio, which is characteristic of thin-beds involving typical hardened particles.

The formation of a thin-bed is somewhat variable. In laboratory testing, fibrous debris has been introduced prior to the particulate, in conjunction with the particulate, and after the particulate. During an actual event, the fibers and particulate debris would arrive in a mixed concentration that would likely vary with time, depending upon such factors as pool turbulence and relative densities. It is highly unlikely that the fiber could all arrive at the screen in advance of the particulate. If the particulate arrived at screen before significant fibers then it would pass through the screen, i.e., the fibers are required to filter the particulate. Calcium silicate is a possible exception to this rule because this material has its own fiber component, and that fiber component must be on the screen to filter the fine particles. The efficiency of the particle filtration depends upon the thickness of the fibrous debris and on its porosity. Further, the porosity of the fibers depends upon how tightly it is compacted by the flow, e.g., a fibrous bed will filter more efficiently at a flow velocity of 1 ft/s than it will at 0.25 ft/s given the same thickness of fibers.

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[1] Previously unpublished post-test SEM photo taken during the conduct of the NRC sponsored calcium silicate head loss tests [LA-UR-04-1227].

From a practical standpoint, a certain minimum thickness of fibers is needed to uniformly cover a strainer surface and to subsequently filter the particulate. For NUKON™ fibrous insulation debris, studied extensively during the BWR strainer clogging resolution, it was recommended in NUREG/CR-6224 that a 1/8-inch of fibrous debris bed thickness (based on the original bulk density generally referred to in head loss analyses as the as-manufactured density) is required. The 1/8-inch recommendation was based on experimental observations that typically at lesser thicknesses, it appeared the bed does not have the required structure to bridge the strainer holes and filter the sludge particles. During an NRC sponsored head loss test program [NUREG/CR-6367], five tests were conducted with 1/8-inch fibrous debris beds (formed with shreds of NUKON™ debris) and iron oxide particulate with mass ratios ranging from 10 to 60. The head losses associated with these tests were minor due to the inability of the fibrous debris to filter sufficient particulate. In addition, tests sponsored by Pennsylvania Power and Light Company [Brinkman] demonstrated low head losses for thin fibrous debris beds, i.e., beds nearly as thin as 1/8-inches.

When the BWROG conducted their head loss tests [URG], many of the tests were conducted by introducing the particulate into the test apparatus prior to introducing the fibrous debris, then allowing sufficient time for the particulate to become thoroughly dispersed. Once this was accomplished, the fibrous debris was introduced to allow a fibrous debris bed to slowly form, which subsequently filtered particulate from the flow once sufficient fibrous debris collected on the test screen. This type of bed formation created debris beds that were well intermixed, although it cannot be guaranteed that the bed was completely homogeneous. Like the Perry and Limerick events, it is likely that some fibrous debris is concentrated at the screen to hold onto the particulate. Another aspect of particulate filtration is the particle size. Within any particulate distribution, some particles may be so fine that these particles pass through the fibrous debris bed whereas the larger particles are readily trapped. The in-between sizes could have varied behavior such that some of these particles may be alternately trapped and freed, thereby contributing to homogeneity. The fineness of the particles that become firmly trapped depends upon the tightness of the fiber matrix. When a thin-bed is formed, the filtration process becomes more efficient and more of the fine particulate is filtered from the flow due to its associated reduced porosity. It should be noted that on a per mass basis, the finer particles have a substantially greater impact on the head loss, i.e., the resistance to flow is correlated with surface area and smaller particles have more surface area per unit volume than do larger particles (specific surface area).

The method of introducing the particulate debris prior to the fibrous debris is likely more realistic with respect to actual plant conditions, however, many tests have been conducted where the fibrous debris was introduced prior to the particulate. When conducting thin-bed debris tests, it is advantageous to establish as uniform a fibrous debris bed as reasonably possible before significant head loss is achieved and this can be achieved more easily when the particulate is not involved with the fibrous bed formation. When the fibrous and particulate debris are introduced at the same time, the debris bed tends towards homogeneity for thicker debris but can lead to lesser head losses for thin-bed formations compared to establishing the fibrous debris bed first at flow velocities sufficient to compact the fiber prior to the arrival of the particulates.

Establishing a fibrous debris bed first and then introducing the particulate can create a more stratified debris bed (sometimes referred as a sandwich configuration) especially if the fibrous bed is compacted by a higher rate of flow prior to introducing the particulate. Such stratified beds have been achieved[2]. Such a configuration is analogous to a typical coffee filter where the filter corresponds to the fibers and the particulate is the coffee grounds. Although, a truly stratified bed is not the anticipated plant accident condition debris bed, it is useful for determining specific debris head loss properties and generally leads to more severe head losses than the truly mixed debris beds.

This discussion has so far focused on particulate that can be characterized as hardened, i.e., the particles do not deform under the pressures encountered in a debris bed and are therefore considered solid. Head loss testing using calcium silicate insulation debris as the particulate has encountered behavior that is apparently different from the behavior of hardened particulate.

The calcium silicate insulation tested was manufactured primarily from diatomaceous earth (DE) and lime (calcium carbonate) in roughly equal portions (~90% of the total mixture). The remaining 10% consisted of small quantities of fiberglass fibers and a binder added for strength. The components were mixed, shaped, and baked, whereby the DE and lime reacted to form the CalSil in a porous <u>crystal lattice structure</u> that provides good insulation properties. The particulate debris created from the destruction of this insulation was examined under a scanning electron microscopy (SEM), which showed substantial very fine particulate and indicated voiding within the particles. An example SEM photo is shown in Figure VIII.2 where the magnification is indicated in the upper left corner by a white bar scaled to 20 microns.

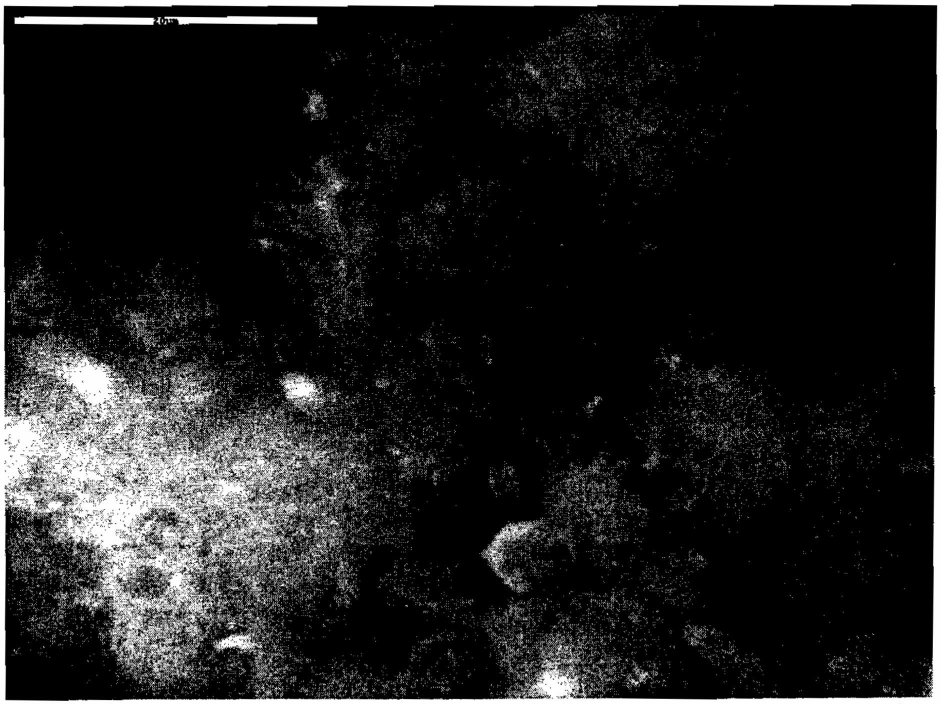

Figure VIII.2. Pre-test SEM Photo of Calcium Silicate Particulate Debris

Due to the porous crystal lattice structure of the particulate, it is likely that these particles could deform under pressure. A post-test photo shown in Figure VIII.3 indicated that the calcium

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[2] As an example, during the conduct of the surrogate latent particulate head loss tests documented in LA-UR-04-3970, the fibrous and particulate debris for an intended mixed debris bed test was inadvertently introduced separately instead of the being premixed as intended. Because the particulate was a course sand (75 to 500 μm), the particulate essentially remained in place above the fibrous layer.

silicate particulate appeared to have been pressed into a near continuous mat, which likely resulted in substantial reduction in porosity. If this particulate does deform under pressure, then the porosity through the continuous mat could decrease considerably. Also, note that its properties of specific surface area and densities would not necessarily remain constant during this process. The debris bed in the test associated with Figure VIII.3 created a relatively high head loss across a thin layer of debris.

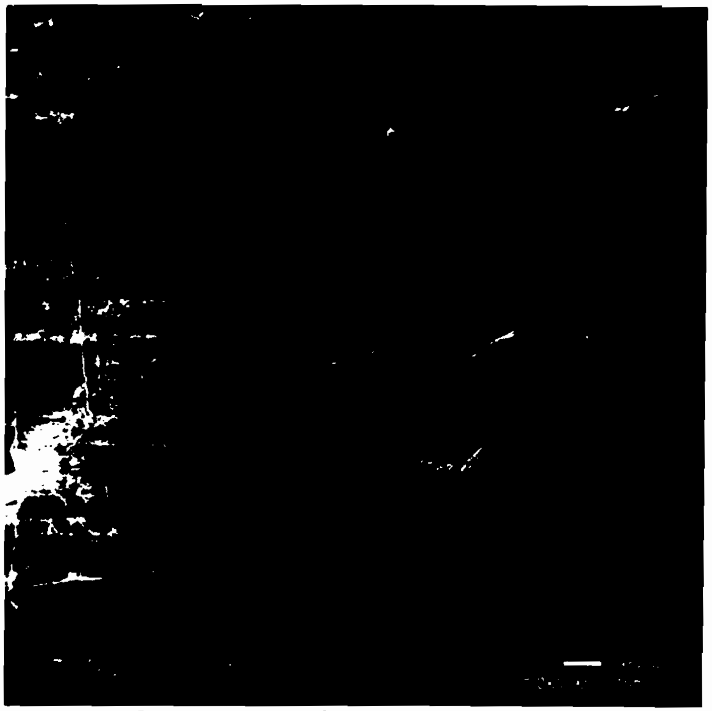

Figure VIII.3. Post-Test SEM Photo of Calcium Silicate Debris from a Thin-Bed Test

A calcium silicate debris bed can form in the same manner as a hardened particulate debris bed however calcium silicate is less dependent upon having a source of fibrous debris to filter it from the flow because calcium silicate has its own fibers (roughly 10% by mass). If the screen has a small enough mesh, it is likely that a bed of calcium silicate could form without any other additional fibers added to the bed. At first, the calcium silicate particulate might pass through the screen while the fibers from the calcium silicate accumulate. Then, if the fiber accumulation is sufficient, the fiber would filter the calcium silicate particulate. Existing test data is not sufficient to define the size of the screen mesh needed to form a calcium silicate only debris bed. In addition, larger pieces of calcium silicate debris would be filtered by the screen. To ensure conservative predictions, it is prudent to assume that calcium silicate only debris beds will form unless adequate data is obtained to conservatively demonstrate otherwise.

Filtration efficiency is also an important aspect of head loss behavior. As porosity decreases, finer sized particles may be filtered than before. When a thin-bed exists, the filtration efficiency will increase, so that the smaller particulate is filtered.

The parameters that affect the formation of a thin-bed debris bed and the resultant head loss include:

1. The existence of a sufficient quantity of fiber to filter the particulate from the flow.
2. The porosity of a bed of the fibers.
3. The quantities of particulate.

4. The size distribution and densities of the particulate that affect its porosity, specific surface area, and filtration efficiency.
5. Whether or not the particulate is hardened or can deform under pressure.
6. Sump screen mess size.
7. Flow approach velocity.

VIII.4 THIN-BED HEAD LOSS TESTING

The thin-bed effect has been demonstrated during head loss testing during various testing programs. The following examples are discussed to provide additional insights into thin-bed formations. The associated analyses were performed using the NUREG/CR-6224 head loss correlation.

BWROG Test 7 [Ref. URG Technical Support Document Vol. 1] In this test, a truncated cone strainer with a screen area 18 ft^2 was tested by first introducing 60 lbm of iron oxide corrosion products into the test tank followed by 1 lbm of NUKON™ insulation debris about an hour later at a pump flow of 5000 GPM and a water temperature of 63 °F. The corresponding screen approach velocity was 0.62 ft/s. The particulate was allowed to circulate and become distributed before the fibers were added. Following the addition of the fibrous debris, the head loss increased rapidly to about 32 ft-water. The uncompressed thickness of the fibrous debris without the particulate would have been 0.28-inches but the debris bed formed with the particulate would have been about 0.63-inches thick if complete filtration were assumed. Based on the accepted sludge density of the corrosion products of 65 lbm/ft^3 and the material density of 324 lbm/ft^3, the porosity would have been about 80%. Analysis indicated the head loss should have been about 200 ft-water, which is much higher than the head loss actually measured. It is likely that holes developed in the debris due to the high pressure differentials that relieved the pressure across the bed. It was noted in NUREG/CR-6224 that damage occurs to the fibrous bed whenever pressure drops exceed approximately 50 ft-water/in. (note that 32-ft-water/0.63-in = 50.8 ft-water/in). The debris bed in this test was essentially a layer of corrosion products held in place by the NUKON™ fibers. This debris bed consists primarily of a layer of particulate, its porosity would essentially be that of the sludge, and the resultant head loss is high, therefore this bed is a thin-bed debris bed. It demonstrates the higher head losses that can be created by a thin-bed even with the bed penetrations.

BWROG Test 8 [Ref. URG Technical Support Document Vol. 1] In this test, a truncated cone strainer with a screen area 18 ft^2 was tested by first introducing 3 lbm of NUKON™ insulation debris followed by approximately an hour later by 16 lbm of iron oxide corrosion products into the test tank at a pump flow of 5000 GPM and a water temperature of 61 °F. The corresponding screen approach velocity was 0.62 ft/s. The particulate to fiber mass ratio was 5.3. The uncompressed thickness of the fibrous debris without the particulate would have been 0.83-inches and the debris bed formed with the particulate alone would have been about 0.16-inches thick if complete filtration were assumed. Based on the accepted sludge density of the corrosion products of 65 lbm/ft^3 and the material density of 324 lbm/ft^3, the porosity would have been about 80%. The measured head loss at 5000 GPM was quoted greater than 41.7 ft-water. It is likely that the debris bed lost integrity at these high head losses which is indicated by the reported test measurement. In this test, the fibrous debris bed was formed at relatively high flow velocities prior to introducing the particulate, therefore it is apparent that the fiber was well compacted prior to the arrival of the particulate and that the bed likely remained substantially stratified.

Latent Particulate (Surrogate) Test 17 [LA-UR-04-3970] In this test, 15 gm of NUKON™ and 200 gm of particulate (less than 75 microns) were introduced into the test apparatus (fiber was introduced first, then the particulate). It should be noted that the particulate used included a clay component that appeared to break up in water into very fine particles. Post-test analyses of water clarity data indicated that approximately half of the particulate was not filtered from the flow, primarily due to the extreme fineness of the particulate, therefore the subsequent analyses assumed approximately 58 gm of particulate was in the debris bed, which resulted in a particulate to fiber mass ratio of 3.8 in the debris bed. A substantial uncertainty exists regarding the accuracy of the determination of the percentage of particulate not filtered from the flow. The 15-gm of NUKON™ formed a thin layer of fibrous debris 0.23-inches thick (at the as-manufactured density) but only about 0.07-inches thick at full test compression (analytical estimate). The particulate in the bed by itself would have formed a layer about 0.055-inches thick. At a flow approach velocity of 0.25 ft/s and a temperature of 94 °F, the measured head loss was 15.8 ft-water. Under these conditions, the analytically determined porosity was 77% which is only slightly higher than the porosity of the particulate by itself, i.e., one minus the sludge density of 39-lbm/ft^3 divided by the particle material density of 166.6-lbm/ft^3 (1–39/166.6=0.766). At faster approach velocities than the 0.25 ft/s that produced 15.9 ft-water head loss, the debris bed deteriorated, most likely due to the high pressure differential across the bed. Although substantial uncertainty is associated with the determination that approximately half of the particulate did not filter from the flow, a relatively high head loss was caused by a relatively thin layer of fibrous/particulate debris with bed porosity near that of the particulate alone.

Latent Particulate (Surrogate) Test 16 [LA-UR-04-3970] In this test, 15 gm of NUKON™ bed and 600 gm of sand particulate ranging from 75to 500 microns were introduced into the test apparatus (fiber was introduced first, then the particulate). Filtration of this relatively course sand was essentially complete. The particulate to fiber mass ratio for this test was 40. The resultant bed of debris, approximately 0.23-inches thick, shown in Figure VIII.4, consisted mostly of the coarse sand with the fibers compressed underneath the sand (stratified). At a flow approach velocity of 0.46 ft/s and a temperature of 96.5 °F, the measured head loss was 9.9 ft-water. Under these conditions, the analytically determined porosity was 41% which is only slightly higher than the porosity of the particulate by itself, i.e., one minus the sludge density of 99-lbm/ft^3 divided by the particle material density of 166.6-lbm/ft^3 (1 – 99/166.6=0.406). At a velocity of 0.25 ft/s, the head loss was 4.4 ft-water compared to 15.9 ft-water for latent particulate Test 17 even though the porosity of the coarse sand was much less than that of the fine particulate, due to the much smaller specific surface area of the coarse sand compared to the fine particulate. Although the porosity of the bed in Test 16 was much lower than the porosity in Test 17 (41% compared to 77%), the head loss for Test 16 was much lower than the head loss for Test 17. The reason for the result was the very much lower specific surface area of the coarse sand in Test 16 compared to the very fine particulate in Test 17.

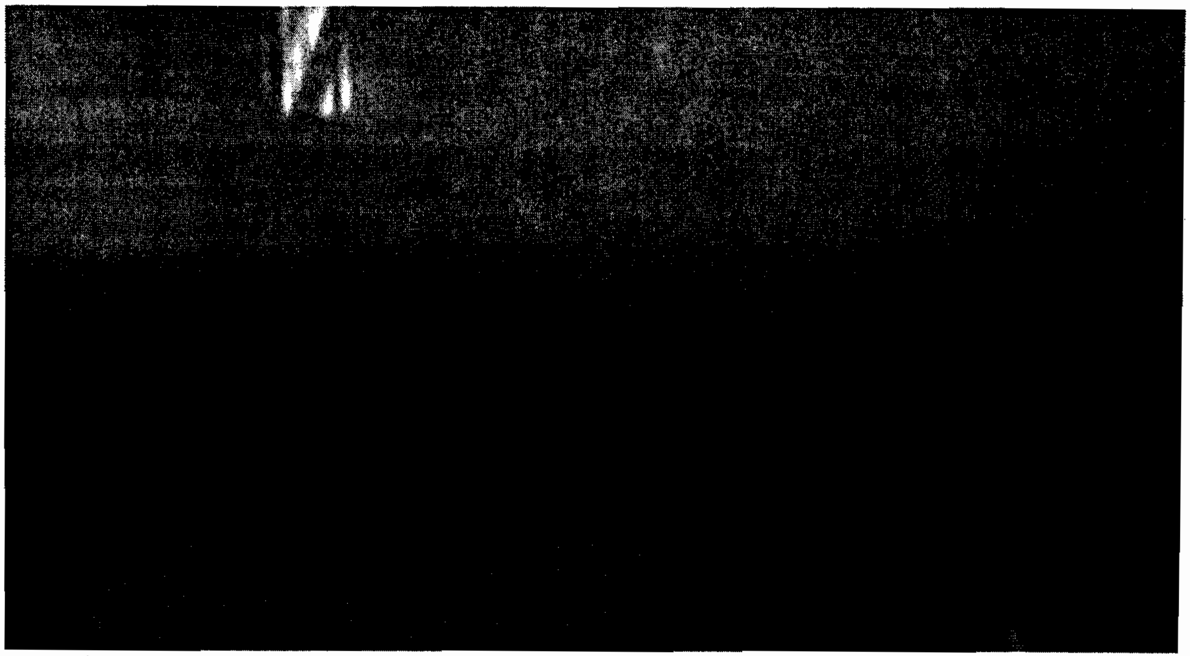

Figure VIII.4. Debris Bed for the Surrogate Latent Particulate Head Loss Test 16.

<u>Calcium Silicate Test 6H [LA-UR-04-1227]</u> In this test, 15 gm of NUKON™ bed and 7.5 gm of calcium silicate insulation particulate was introduced into the test apparatus (fiber was introduced first, then the particulate). The particulate to fiber mass ratio was 0.5. Post-test analyses of water clarity data indicated that all but the very finest particulate had filtered from the flow. The 15-gm of NUKON™ formed a thin layer of fibrous debris 0.23-inches thick (at the as-manufactured density) but test bed under full compression was substantially thinner. A photo of this debris bed is shown in Figure VIII.5. At a flow approach velocity of 0.4 ft/s and a temperature of 110 °F, the measured head loss was 12.7 ft-water. Analysis was used to deduce both the specific surface area and the sludge density for the calcium silicate. In the analysis, the sludge density was adjusted in the simulation until the particulate packing limit coincided with the rapid rise in head loss observed in the test data which occurred when the approach velocity was increased beyond 0.35 ft/s. The working theory for the analysis of the calcium silicate thin-bed tests was that the rapid increases in the head losses as velocities were increased was caused by the formation of a relatively continuous layer of matted calcium silicate as shown in the post test SEM photo shown in Figure VIII.3 that illustrates an apparent matted layer of calcium silicate. Under these conditions, the bed porosity apparently rapidly decreased with a corresponding increase in the bed's ability to filter finer particulate, which was demonstrated by the water clarity data. Under these conditions, the analytically determined porosity was 88% which is significantly higher than the porosity of the particulate by itself, i.e., one minus the sludge density of 22-lbm/ft^3 divided by the particle material density of 115-lbm/ft^3 (1 – 22/115=0.808). The most astounding feature of this thin bed test was the fact that such high head losses were achieved with a particulate to fiber mass ratio of only 0.5 even though the porosity apparently did not drop below approximately 0.8. To achieve such high head losses, the specific surface area had to be much higher than those determined for the hardened particulate. The analytically deduced specific surface area was 800,000-ft^2/ft^3. The higher specific surface areas were attributed to both the relative fineness of the particulate and internal voiding of the particles whereby some flow potentially moved through these voids at higher pressure differentials.

Figure VIII.5. Debris Bed for the Calcium Silicate Head Loss Test 6H

This set of relatively thin and relatively high head loss tests illustrate the formation of debris beds whereby the head loss is driven primarily by the porosity of the particulate compacted into a sludge for four distinctly different particulate materials and a variety of particulate to fiber mass ratios. The two corrosion product tests were tests where the head losses became so high the debris bed probably developed penetrations that relieved the head loss, however these thin-bed tests serve to illustrate how easily extreme head loss can occur. The latent thin-bed tests illustrate the differences between two distinctly different particulate size distributions. The calcium silicate test illustrated the potential effect of particulate deformation. Tests of this nature have been used to achieve an understanding of the thin-bed effect.

VIII.5 ANALYTICAL APPROACH TO PREDICTING THIN BEDS

For a head loss correlation to successfully predict the thin-bed behavior, as well as the porosity of a mixed debris bed, the correlation must have a debris bed porosity model that simulates not only mixed debris beds but also the porosity of the particulate by itself when enough of the particulate is in the debris bed to form a particulate layer. The NUREG/CR-6224 head loss correlation porosity model contains a debris packing limiting equation to limit bed compaction whenever the head loss and/or high quantities of particulate cause the bed compaction to reach the limit. The correlation porosity model includes a bed compaction term. When the particulate to fiber mass ratios become significantly large, the bed porosity from the porosity model approaches the porosity of the respective sludge.

The NUREG/CR-6224 correlation and its associated constitutive equations (porosity, compression function, and compression limiting) were developed assuming a uniform and a homogenous debris bed. Under thin-bed conditions, the fibrous debris could well be non-uniform because fiber would accumulate first before the particulate would filter from the flow; therefore a layer of fiber next to the screen is likely. However, in a thin-bed, the bed generally contains so much particulate that the fiber contribution to the head loss is small thereby making the non-uniformity of the fibrous debris far less important.

Table 1 compares head loss prediction using the NUREG/CR-6224 correlation with the thin-bed tests presented herein. For the two corrosion products thin-bed tests, the tests were apparently conducted with so much particulate that penetrations developed in the beds such that head loss were substantially less than if a uniform debris bed had been maintained. As such, the NUREG/CR-6224 correlation over predicted the head loss results by a substantial margin. For the two latent (surrogate) particulate debris tests (i.e., < 75 micron and 75 to 500 micron particulate), the NUREG/CR-6224 correlation was used to estimate specific surface area that agreed well with other tests including mixed bed tests in that test series. The latent sludge density and porosities determined experimentally agreed well with the correlation predictions. For the calcium silicate head loss tests, input parameters were recommended for the NUREG/CR-6224 correlation that would cause the correlation to bound the head losses even though the packing processes whereby the calcium silicate comes together to form a sort of matting layer are not well enough understood to formulate a model for those processes.

The NUREG/CR-6224 correlation was developed assuming the particulate properties would not be altered under head loss pressures, i.e., constant densities and specific surface areas. With a particulate capable of deforming under pressure, the densities and surface areas are not necessarily constant. Therefore, the correlation should not necessarily be expected to predict accurately the behavior of calcium silicate when compacted together in a thin-bed configuration. Therefore, the analytical approach is to estimate a bounding head loss. The bounding recommendation for calcium silicate were primarily based on the results of Test 6H, which produced the most severe head loss conditions, i.e., the bounding specific surface area. Although, a limited number of valid calcium silicate head loss tests were conducted to determine the most severe head loss conditions, the set of applicable tests support the use of Test 6H in making bounding head loss recommendations. Supporting tests accomplished the following:

1. One test essentially reproduced the results with Test 6H.
2. Two tests bracketed the thickness of the Test 6H fibrous debris bed, i.e., one test was slightly thinner and another slightly thicker. The associated head loss parameters were more severe for Test 6H. In the thinner bed test, the filtration efficiency dropped off substantially relative to the efficiency of Test 6H. In the thicker bed test, the fibrous debris bed was thick enough that the amount of compaction needed to form a thin-bed matting of calcium silicate apparently did not occur within the flow capacity of the test apparatus.
3. One test used the same quantity of fibrous debris as Test 6H but significantly more calcium silicate. In this test, the data indicated a lower specific surface area than the 800,000/ft deduced from Test 6H was needed to simulate the test results even though the head losses were higher for this test.

Based on these results, it was judged that the 800,000/ft specific surface area bounded the test results and that Test 6H represents the more limited debris bed configuration. The recommended 880,000/ft specific surface area (in conjunction with a sludge density of 22 lbm/ft^3) included a 10% enhancement as a safety factor due to experimental uncertainties and variances in calcium silicate manufacturing.

In summary, the thin-bed effect, originally recognized with respect to the response of ECCS long-term cooling systems at nuclear power plants after three BWR operational events where strainer clogging occurred, has been experimentally reproduced for a variety of particulate debris. The experimental data was subsequently used to study the physical processes whereby recommendations can be made for the application of the NUREG/CR-6224 correlation to this type of debris bed accumulation.

VIII.6 REFERENCES FOR APPENDIX VIII

K. W. Brinkman and P. W. Brady, "Results of Hydraulic Tests on ECCS Strainer Blockage and Material Transport in a BWR Suppression Pool," Pennsylvania Power and Light Company, EC-059-1006, Rev 0, May 1994.

Table VIII.1. Comparison of Thin-Bed Test Data with NUREG/CR-6224 Simulations

Test No. and Particulate	Experimental Parameters					NUREG/CR-6224 Head Loss Simulation Results				Comments
	Fiber[3] & Particulate[4] Bed Thicknesses (inches)	P/F Mass Ratio	Approach Velocity (ft/s)	Sludge Porosity & Density (lbm/ft_3)	Experimental Head Loss (ft-water)	NUREG/CR-6224 Head Loss Prediction (ft-water)	Compacted Fiber/Part. Bed Thickness (inches)	Bed Porosity	Particulate Specific Surface Area (ft^2/ft^3)	
BWROG Test 7 Corrosion Products	0.28 (Fiber) 0.62 (Part.)	60	0.62	0.8 65	32	203	0.63	0.8	183,000	Debris Bed Damage Probable
BWROG Test 8 Corrosion Products	0.83 (Fiber) 0.16 (Part.)	5.3	0.62	0.8 65	> 41.7	83.3	0.19	0.8	183,000	Debris Bed Damage Probable
Latent Particulate Test 17 (< 75 μm)	0.23 (Fiber) 0.06 (Part.)	3.8	0.25	0.77 39	15.8	15.9	0.07	0.77	277,000	Uncertain in Debris Filtration Fraction
Latent Particulate Test 16 (75-500 μm)	0.23 (Fiber) 0.23 (Part.)	40	0.46	0.41 99	9.9	10.9	0.23	0.41	10,800	Stratified Debris Bed
Calcium Silicate Test 6H	0.23 (Fiber) 0.01 (Part.)	0.5	0.40	0.81 22	12.7	12.7	0.04	0.86	800,000	Bound Upper Head Losses

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[3] Experimental fiber beds thickness are based on the as-manufactured density without any bed compression.

[4] Experimental particulate bed thickness estimate assumed an equivalent thickness of particulate without the fiber present.

Review of NEI Guidance Appendices

Review of Appendix A, "Defining Coating Destruction Pressures and Coating Debris Sizes for DBA-Qualified and Acceptable Coatings in Pressurized Water Reactor (PWR) Containments"

The Appendix A test program outlined the Industry's effort to determine the minimum coatings destruction pressure and provide information relative to coating debris sizes generated from within the ZOI. Testing utilized high pressure water to determine the jet effect on qualified coatings. A 3500 psig high pressure washer with a heated reservoir was used to simulate the LOCA jet. Test duration was 60 seconds. A 15 degree waterjet tip and angles of attack directing the waterjet normal to the surface and at 45 to the surface were used from multiples distances. Surface temperatures were measured during testing and were in the ranges of 150F and 80F. Coatings were applied to both steel and concrete substrates. The coating systems are characterized as untopcoated inorganic zinc (steel substrate only), inorganic zinc primer with epoxy topcoat (steel substrate only) and a self-priming epoxy all of which are representative of coating systems currently employed as qualified systems in power plants.

Testing concluded that erosion was the primary mode of coating degradation from interaction with the waterjet in all test cases. The untopcoated inorganic zinc coating failed at a distance up to 3 times greater than the epoxy. The industry concluded that a damage pressure of 333 psig for untopcoated IOZ and 1000 psig for epoxy systems should be used as the corresponding coating destruction pressures. Testing showed that an elevated surface temperature impacted the amount of coating degradation and increased fluid jet temperature resulted in coating degradation at lower jet pressures.

The test program was a good first attempt to define the destruction pressure and debris characteristics associated with LOCA jet interaction with qualified coatings. There appears to be little, if any, other test data available which attempts to define the impingement effects of a LOCA jet on coatings. This lack of data was identified in the GR. The test protocols used in the past and as currently specified in ASTM D3911 to DBA qualify coatings for nuclear service specifically prohibits fluid impingement onto the coated sample surface. The staff believes that the Appendix A test did provide valuable information by identifying erosion as a destruction mechanism for coatings and that the debris size would be characteristic of the basic material constituent under the conditions modeled during the test. The staff also believes that the test illustrated the effect that temperature plays in coating degradation. However, the staff positions is that the test did not provide sufficient justification supporting the destruction pressures and corresponding ZOI identified in the GR. No method was provided which could be used to correlate the waterjet test conditions with LOCA jet conditions. No test data was offered combining both the effects of mechanical insult and elevated temperatures (LOCA initial conditions). Nor was data provided on the effects of rapid thermal transients or pressure shock on the performance of qualified coatings. Therefore, the staff found the waterjet testing to be inconclusive.

The staff believes that a test program should be considered which will accurately estimate the coating ZOI based upon a representative LOCA jet (pressure and temperature) interacting with surfaces covered by qualified coatings. Such a test should combine the erosion effects of a water laden steam jet with the combined thermal and pressure transients associated with a LOCA. Coatings which can be correlated to qualified plant coatings should be used for the testing. This includes aging the coating to account for the effects of normal plant operation and the effects of radiation exposure. Provisions should also be established for characterization of coating debris and assessment of the failure mechanism. Such testing could lead to an understanding that debris may be generated in forms other than small particulate from erosion which may ultimately lead to a more realistic assessment of the coating debris contribution.

Review of Appendix B, “Example of a Latent Debris Survey”

This Appendix in the GR provides a simplified example of a method for determining the amount of latent debris on containment surfaces. This Appendix does not contain new or unique information, and is not totally consistent with Section 3.5 of the GR where the detailed guidance for evaluating Latent Debris is contained. In the evaluation of Section 3.5, the staff provides a more comprehensive and accurate method for evaluation of Latent Debris. As such, a separate evaluation of this Appendix is not required.

Review of Appendix C, "Comparison of Nodal Network and CFD Analysis"

The staff has reviewed the Appendix C comparison between the nodal network and CFD methods and finds that the conclusion of a "good comparison" is not supported by independent analysis and evaluations. The error values reported are computed by subtracting flow rates of the nodal network from the CFD and dividing by the total flow in the containment pool. The flows computed for the network sections are approximately 1000 gpm (order of magnitude). The total flow is 21,000 gpm, more than an order of magnitude larger than the individual flow rates, and almost 2 orders of magnitude larger than the flow difference between the two methods. The staff does not consider this approach to be a valid method for comparing nodal network results to those achieved with CFD analysis.

The staff finds that normalizing the flow error between the two methods by the total recirculation flow rate is incorrect, and minimizes the significance of the errors between the two methods. Particles/debris respond to local velocities, not normalized values. Comparison of the nodal values to the CFD values shows that there is quite a discrepancy in the associated local velocity values and discrepancies can also exist with respect to flow direction.

Also, in the information presented in the GR it is not clear how the flow channels were selected. In Figure 4-4 of the GR, the flow channels were determined by using the CFD analysis and essentially encapsulate the high velocity regions. Where the velocities are uniform across the channel, the comparison is fairly good in absolute terms, but not their "error" terms. When there is a gradient of velocity across the channel, the difference in the CFD versus nodal network velocity is quite large. Without the CFD analysis, the GR does not provide guidance for selecting the channel network. Even when the CFD results are known, the nodal network does not give a reasonable answer. The staff finds that relying on such a method for general use, where the flows are not known a priori is a difficult method to implement.

Appendix C does not provide a reference for the nodal analysis method used, nor is the method explicitly defined in the document. There is discussion about friction factors, having to choose a velocity for the Reynolds number assumed for the flow and needing to iterate to arrive at the correct velocity, but there are no equations or methodology outlined to follow. These conditions should be included and appropriate references cited for both the methodology and previously published applications to this type of flow problem.

Other issues the staff identified in Appendix C include:

A description of how the CFD flow rates were calculated for the nodal sections shown in Figure 4-4 is not provided.

Figure 4-4 is not a "composite" of the CFD results, it is exactly the case for a large loca break in the lower right quadrant, not a composite of all break locations and flows.

On Page C-4 – the threshold velocities quoted are to initiate motion of debris, not to sustain motion. The velocity required to sustain the debris motion may in fact be much lower, i.e., starting vs. rolling friction.

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Review of Appendix D, "Isobar Maps for Zone of Influence Determination"

The staff evaluation of GR Section 3.4.2.2 compared the ZOI isobars set forth in Appendix D of the GR with isobars independently calculated using the methodology of ANSI/ANS 58.2-1988. The comparison showed good agreement between the calculations for downrange behavior (Zone 3), but discrepancies exist in Zones 1 and 2. As indicated in Figure 3-1 of this SER, it appears that contour termination points on the centerline are not accurate and that the quadratic behavior of the Zone 2 isobar equations is not implemented correctly. These differences will have a negligible effect on volume integrals for jet pressures less than 20 psig, but may become more of a concern for higher pressures near the break. To quantify the magnitude of the difference, Table D-1 presents a comparison of ZOI radii computed from both methods. In particular, the GR approach may not have preserved the system stagnation pressure throughout the volume of the liquid core region as specified by the standard. However, in application of the calculated values as documented in Table 3-1 of the GR, the recommended value of 1.0 is provided for both the 1000 and 333 psig destruction pressures. The staff considers that using the recommended value of 1.0 is necessary for these pressures for a conservative treatment.

Table D-1. Comparison of Computed Spherical ZOI Radii from Independent Evaluations of the ANSI Jet Model

Impingement Pressure (psig)	ZOI Radius/Break Diameter	
	Guidance Report	SER Appendix I
1000	0.24	0.89[a]
333	0.55	0.90
190	1.11	1.05
150	1.51	1.46
40	3.73	4.00
24	5.45	5.40
17	7.72	7.49
10	12.07	11.92
6	16.97	16.95
4	21.53	21.60

[a] The core volume at stagnation pressure P0 gives a minimum possible ZOI radius of 0.88 diameters.

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Review of Appendix E, “Additional Information Regarding Debris Head Loss”

The GR Appendix E contains additional information regarding the estimation of head loss associated with debris beds. The supporting Appendix E repeats the text found in Section 4.2.5, and provides tables that summarize available domestic and international head loss testing and results. No head loss refinements are offered other than those given in Section 3.7.2.3.2.3. (See SER Section 3.7.2.3.2.3, “Thin Fibrous Beds,” for the staff evaluation of that section.)

Presentation to the Advisory Committee on Reactor Safeguards

UNITED STATES NUCLEAR REGULATORY COMMISSION

ACR-700 Pre-Application Review

October 7, 2004

Presented by:
Belkys Sosa, RNRP/NRR
James Kim, RNRP/NRR

UNITED STATES NUCLEAR REGULATORY COMMISSION

Purpose

- Brief the Committee on the status of the ACR-700 pre-application review
- To provide information to the Committee on the major issues identified in the pre-application safety assessment report (PASAR) for the ACR-700 design
- To request ACRS letter on the assessment of the ACR-700 design and the feasibility of completing the Design Certification Review

UNITED STATES NUCLEAR REGULATORY COMMISSION

Agenda

◆ Opening Remarks	5 min
◆ ACR Pre-Application Review Overview	5 min
◆ ACR-700 Review Issues	10 min
◆ Coolant Void Reactivity (CVR)	10 min
◆ Feedback / Questions	30 min
◆ AECL Presentation	20 min
◆ Closing Remarks	10 min

UNITED STATES NUCLEAR REGULATORY COMMISSION

ACR-700 Pre-Application Review Overview

- Approach was to identify concerns, not to try to resolve issues
  - ◆ Acquire familiarity with ACR-700 design - Phase 1 of pre-application review
  - ◆ Develop understanding of differences between ACR-700 and plants already operating or reviewed
  - ◆ Identify existing regulations that may not be met by the ACR-700
  - ◆ Identify new regulations needed to ensure adequate protection provided by the ACR-700 design
  - ◆ Conduct technical interactions with the Canadian Nuclear Safety Commission (CNSC) as added resource in the review process

Pre-Application Review Scope
Focus Topics (FT)

- **Class 1 pressure boundary design (FT1)**
- Design basis accidents and acceptance criteria (FT2)*
- **Computer codes and validation adequacy (FT3)**
- Severe accident definition and adequacy of supporting R&D (FT4)
- Design philosophy and safety-related systems (FT5)
- Canadian design codes and standards (FT6)
- Distributed control systems and safety critical software (FT7)
- **On-power fueling (FT8)**
- **Confirmation of negative void reactivity (FT9)**
- Preparation for Standard Design Certification Docketing (FT10)
- ACR PRA Methodology (FT11)
- ACR Technology Base (FT12)
- **Fuel design (FT13)**

Note: **Underline items are Key Focus Topics as defined by AECL**

*Designated as NRC priority

FT5, FT10, and FT12 do not have distinct sections in the PASAR

ACR-700 Pre-Application Safety Assessment Report (PASAR)

- **Review Scope**
  - Discuss what was reviewed and what guidance it was reviewed against, to the extent that the guidance exists.
- **Regulatory Issues**
  - Discuss regulatory issues, such as rules, rulemaking, or exemptions that will need to be resolved.
- **Policy Issues**
  - Discuss policy issues that will need upper management or Commission guidance for resolution.
- **Technical Issues**
  - Discuss technical issues identified that will require further data, tests, inspections, analyses, or codes.
- **Conclusion**
  - Discuss the feasibility of successfully completing design certification.

UNITED STATES NUCLEAR REGULATORY COMMISSION

ACR-700 Pre-Application Schedule

Phase 1 (Complete)	**June 2002 – July 2003**
Phase 2 (On-Going)	**August 2003 – October 2004**
ACRS Information Briefing (Complete)	**January 13, 2004**
Draft PASAR to ACRS (Complete)	**September 16, 2004**
ACRS Full Committee Meeting	**October 7, 2004**
ACR-700 PASAR Due	**October 30, 2004**
Phase 3 (Transition Phase)	**November 2004 – Design Certification Application**

ACR-700 Design Review Issues

- Class 1 Pressure Boundary Design (FT1)
  - Regulatory Issue – 10 CFR 50.55a requires the use of ASME for design and inservice inspection of safety related components
    - For areas where the ASME Code requirements are not applicable or need to be supplemented, the staff will evaluate the acceptability of Canadian CAN/CSA N285 series standards.
  - Regulatory Issue – ACR-700 does not have a ferritic reactor vessel - Per 10 CFR 52.48, the technical requirements specified in 10 CFR 50.61(pressurized thermal shock (PTS)), 10 CFR Part 50 Appendix G, Sections IV.A.1 and IV.A.2 (fracture toughness), and 10 CFR Part 50, Appendix H (materials surveillance) are not technically relevant.
    - The staff will develop review guidance and requirements related to maintaining the integrity of reactor assembly components.
  - PASAR discusses various issues on degradation mechanism that will require additional information and further review for resolution.

ACR-700 Design Review Issues

- Design-Basis Accidents and Acceptance Criteria (FT2)
    - AECL proposed a three-tier risk-informed reactor accident classification scheme
    - The staff recommends to adopt a probabilistic event selection for ACR-700.
    - Severe channel flow blockage (SCFB) in a fuel channel and stagnation feeder break (SFB) are limited core damage accidents (LCDA) that may be classified as DBAs.
    - As an alternative to meeting the requirements of 10CFR 50.34 the staff may propose a mechanistic fission product source term for Commission consideration.

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ACR-700 Design Review Issues

- Computer Codes and Validation Adequacy (FT3)
  - The current physics codes (WIMS-IST, DRAGON-IST, RFSP-IST) need modifications and revalidation for ACR-700 conditions
  - Experimental database on header/feeder inventory and flow distribution, horizontal fuel bundle thermal hydraulics and RD-14M integral tests is required for successful completion of design certification
  - Modifications of test facilities (RD-14M, CWIT, LASH) may be required to correctly scale to ACR-700 design

ACR-700 Design Review Issues

- Severe Accident Definition and Adequacy of Supporting Research and Development (FT4)
  - The NRC PIRT process identified a number of key technical issues that must be addressed for successful completion of design certification
  - The NRC PIRT process also identified potential deficiencies in the experimental database used to validate the analysis codes
  - MELCOR will be modified to model the unique ACR-700 configuration for independent validation
  - No severe accident experimental work by NRC is anticipated provided the results of AECL's planned experiments are available to support design certification review

ACR-700 Design Review Issues

- **Canadian Design Codes and Standards (FT6)**
  - SECY-03-0047 has direct applicability to the use of Canadian codes and standards for ACR-700.
  - Commission directed the staff to review international codes and standards only as part of an application or pre-application review of non-LWRs.
  - The review of Canadian codes and standards will have a significant impact on the time and technical resources of the staff during the design certification review

ACR-700 Design Review Issues

- **Distributed Control Systems and Safety Critical Software (FT7)**
  - The staff raised a question on how the design complies with the NRC position on defense-in-depth; since it appears that the trip setpoints for both shut down systems (SDS) are the same
  - The staff questioned whether both SDS1 and SDS2 are developed to meet the same systems, functional, and software requirements
  - AECL's presentation to the ACRS in January 2004 indicated the reliability of the safety critical software is demonstrated through particular quantitative reliability goals (assessed by trajectory-based random testing of the software). This may raise an issue since current NRC position does not provide for the use of digital reliability goals

ACR-700 Design Review Issues

- **On-Power Fueling (FT8)**
  - The staff compared the design of the ACR-700 on-power fueling systems to the design-related regulations in 10 CFR Part 50 and 52
  - The staff determined that existing regulations are adequate to support design certification of on-power fueling for the ACR-700
  - The on-power fueling process could be a relatively high-probability initiator of limited core damage accidents

ACR-700 Design Review Issues

- Confirmation of Negative Void Reactivity (FT9)
  - SECY-93-092 may have direct applicability to ACR-700.
    - If ACR-700 reference design does not eliminate the potential for substantially positive void reactivity during the initial checkerboard voiding of alternate fuel channels in large-break loss-of-coolant accidents (LBLOCAs).
  - What levels of confidence (e.g., 95/95) are needed for establishing compliance with GDC 11.

ACR-700 Design Review Issues

- PRA Methodology (FT11)
  - Treatment of limited core damage accidents (LCDAs).
  - Risk objectives should be expanded to address both LCDAs and severe core damage accidents (SCDAs).

LCDAs – accidents that involve a subset of the fuel (e.g., local power/cooling mismatches at full power such as single channel accidents, or global power/cooling mismatches at decay power such as LOCA followed by failure of the ECCS).

SCDAs – accidents that involve the entire core

UNITED STATES NUCLEAR REGULATORY COMMISSION

ACR-700 Design Review Issues

- ACR CANFLEX Fuel Design (FT13)
  - Design Certification process for ACR-700 fuel will deviate from past practice
    - AECL does not have a reference, CNSC-approved ACR-700 fuel design or fuel performance methodology.
    - ACR-700 fuel design criteria deviates from SRP 4.2.
    - ACR-700 design and operating conditions deviate from operational CANDUs.
  - AECL's limited in-reactor experience database for higher burnup SEU fuel bundle designs may necessitate reliance on on-going irradiation programs which will not be completed until 2009.

Conclusions

- Staff has prepared carefully for review of ACR-700 design certification application
- Based on the information provided by AECL during the pre-application review, the staff identify a number of issues that will require more detailed information for resolution but did not identify any issues that would preclude certification of the ACR-700 design
- Staff is preparing a SECY paper to inform the Commission of the issues identified in the pre-application review in preparation for the ACR-700 design certification application

ACR-700 Pre-Application Safety Assessment Report (PASAR)

Focus Topic 9
"Confirmation of Negative Void Reactivity"

Donald E. Carlson, Dr.-Ing.
RES/DSARE/ARREB

Presented to
NRC Advisory Committee on Reactor Safeguards
October 7, 2004

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ACR-700 PASAR FT9
"Confirmation of Negative Void Reactivity"

Review Highlights:

- NRC PIRT identified "checkerboard" voiding of alternate channels in ACR-700 large LOCAs.
- AECL design analysis reported a full-core CVR of -7 mk, but did not address checkerboard CVR.
- NRC calculation results on the reviewed design:
  - Consistent with AECL on negative full-core CVR
  - Showed checkerboard CVR to be positive
  - Positive checkerboard CVR results consistent among various independent calculations

ACR-700 PASAR FT9
"Confirmation of Negative Void Reactivity"

Key Points:

- The pre-application review was based on a preliminary ACR-700 nuclear design.
- In a large-break LOCA, negative fuel temperature feedback could tend to limit the power surge induced by positive checkerboard void reactivity effects.
- Positive checkerboard void reactivity <u>could</u> be reduced or eliminated by changing the nuclear design (e.g., more Dy, higher enrichment).

ACR-700 PASAR FT9
"Confirmation of Negative Void Reactivity"

Technical Insights on ACR-700 CVR (1 of 4):

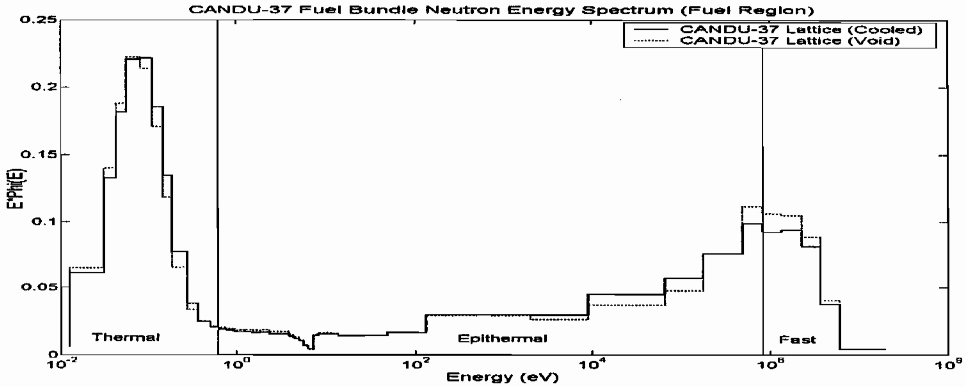
- ACR-700 coolant void reactivity is:
  - A combination of large positive and large negative effects
  - Nonlinear with partial versus full voiding (i.e., positive during checkerboard voiding)
  - Sensitive to void distribution within and between channels
  - Sensitive to core design, operating parameters, and fuel burnup
- Confirmatory measurements of coolant void reactivity in an operating core are inherently difficult
- Hence, the importance of code validation based on benchmarks against representative critical experiment data and nuclide assays from PIE of irradiated fuel
- Further development of code methods and validation data may be needed specifically for checkerboard CVR

ACR-700 PASAR FT9
"Confirmation of Negative Void Reactivity"

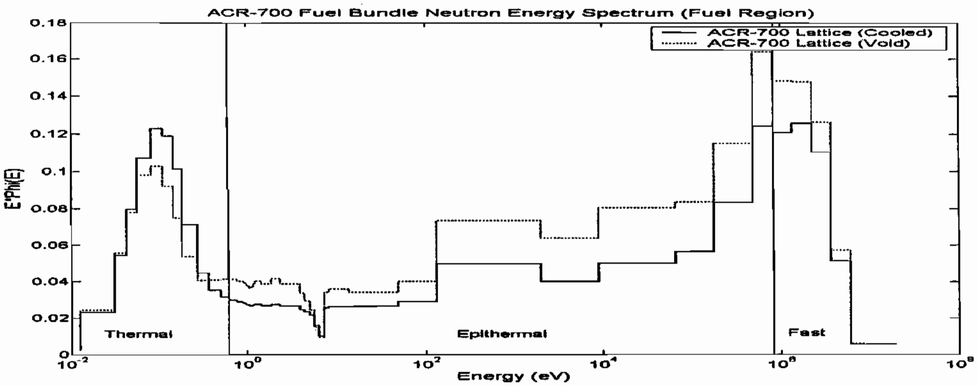
Technical Insights on ACR-700 CVR (2 of 4):

Neutron
Spectral Shift
on Voiding

<u>NU CANDU</u>



<u>ACR-700</u>



ACR-700 PASAR FT9
"Confirmation of Negative Void Reactivity"
Technical Insights on ACR-700 CVR (3 of 4):

Calculated Coolant Void Reactivity Components (mk) for Fresh Fuel in Core Lattices of Conventional CANDU and Reference ACR-700

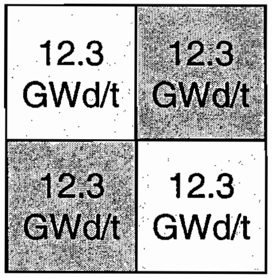
4-Factor Spectral Component	WIMS-AECL*	HELIOS-1.8			
	Conventional CANDU		Reference ACR-700		
	Full Voiding		Full Voiding	50% Uniform Voiding	Checkerboard Voiding
fast fission factor	+4.4	+4.0	+35.7	+15.4	+14.6
resonance escape probability	+6.8	+7.8	**-72.4**	**-34.8**	**-28.7**
thermal utilization factor	+3.3	+4.3	+39.9	+18.7	+20.7
reproduction factor	+1.8	+1.4	-5.2	-1.8	-3.1
Lattice Coolant Void Reactivity	+16.3	+17.5	**-2.1**	**-2.5**	**+3.5**

*From Whitlock, 1995

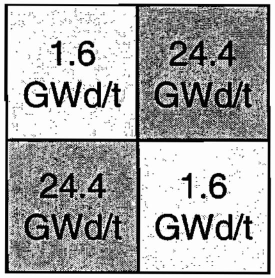
ACR-700 PASAR FT9
"Confirmation of Negative Void Reactivity"

Technical Insights on ACR-700 CVR (4 of 4):

Checkerboard CVR in Uniform versus Mixed Burnup Lattices



12.3 GWd/t	12.3 GWd/t
12.3 GWd/t	12.3 GWd/t



1.6 GWd/t	24.4 GWd/t
24.4 GWd/t	1.6 GWd/t

2x2 Burnup Case	Coolant Void Reactivity (mk)		
	CB-1 Voiding	CB-2 Voiding	Full Voiding
Uniform Burnup 12.3 GWd/t	**+4.7**		**-3.4**
Mixed Burnup 1.6 / 24.4 GWd/t	**+2.0**	**+6.5**	**-3.4**

ACR-700 PASAR FT9
"Confirmation of Negative Void Reactivity"

PASAR Main Conclusions:

- The reviewed preliminary nuclear design of ACR-700 does not have negative void reactivity in large LOCAs.
- Nuclear design changes could reduce LOCA void reactivity.
- CVR bias and uncertainties are potentially large in relation to nominal values, and AECL's ongoing experimental work is essential to their quantification.

ACR-700 PASAR FT9
"Confirmation of Negative Void Reactivity"

Path Forward:

In preparation for the ACR-700 design certification review, the staff will continue:

- Development of PARCS code models, coupled with TRACE (and MELCOR), for simulating ACR-700 operations and accidents, including the combined effects of void and Doppler reactivity
- Interactions with AECL to assess the applicability and adequacy of the existing and planned sets of experiments for validating code predictions of CVR and other safety-relevant nuclear effects, providing timely identification of gaps and technical issues

10

AECL Technologies Presentation to ACRS on ACR-700

October 7, 2004

Glenn Archinoff
Manager ACR Licensing
AECL Technologies
Rockville, MD

Outline

- Pre-Application Phase summary and path forward
- Reactor physics codes
- Coolant void reactivity

Pre-Application Review

- OBJECTIVE: To determine if the ACR-700 design can be certified in the US in a timely manner, emphasizing:
  - Aspects of the ACR-700 design that are not directly addressed by NRC regulations
  - Aspects of the underlying technology base that are new to NRC staff
- ACTIVITIES
  - Phase 1 – Familiarization, NRC review of documents, meetings
  - Phase 2 – Responses to RAIs, detailed technical meetings, address focus topics

Results

- The main objective of the Pre-Application phase has been met
- AECL's view is that the ACR-700 design will meet applicable US regulations
  - For CANDU-specific aspects where US regulations do not exist, Canadian requirements meet the intent of US regulations and will be applied
- NRC staff now familiar with ACR-700 technology
  - Will facilitate timely review of Design Certification application
- There are still issues to address

Path Forward – Transition Phase

- Overall objective is to achieve high confidence in the acceptability of the Design Certification application
- Smaller set of focus topics for Transition Phase
  - Physics codes and coolant void reactivity
  - Evaluation models
  - Fuel
  - Safety Analysis
  - Thermal Hydraulics
  - Class 1 pressure boundary
  - Plus others to be determined based on discussion with NRC staff

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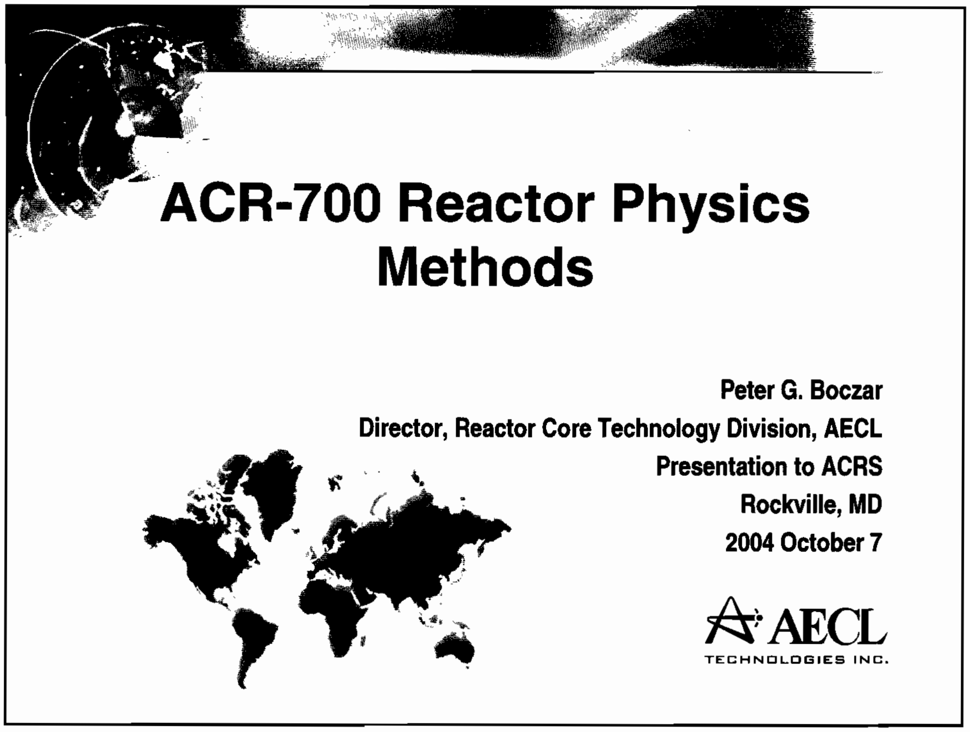

ACR-700 Reactor Physics Methods

Peter G. Boczar
Director, Reactor Core Technology Division, AECL
Presentation to ACRS
Rockville, MD
2004 October 7

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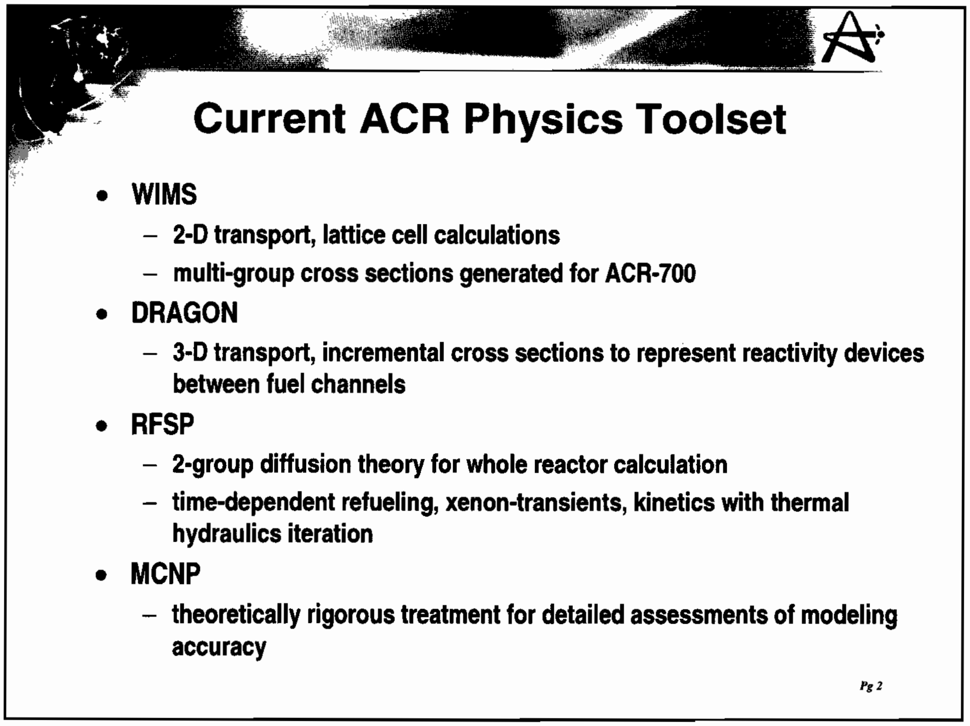


Current ACR Physics Toolset

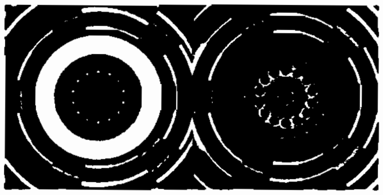
- WIMS
  - 2-D transport, lattice cell calculations
  - multi-group cross sections generated for ACR-700
- DRAGON
  - 3-D transport, incremental cross sections to represent reactivity devices between fuel channels
- RFSP
  - 2-group diffusion theory for whole reactor calculation
  - time-dependent refueling, xenon-transients, kinetics with thermal hydraulics iteration
- MCNP
  - theoretically rigorous treatment for detailed assessments of modeling accuracy

Assessment of Toolset

- Key ACR physics phenomena
  - tighter coupling between adjacent lattice cells
  - heterogeneity between adjacent cells
  - leakage
- Our assessment to date
  - toolset is adequate for most applications
  - enhancement desired for certain heterogeneous configurations

Enhancements to Physics Codes

- WIMS 3.0
  - improved resonance treatment
  - more detailed geometrical representation
  - multi-cell capability

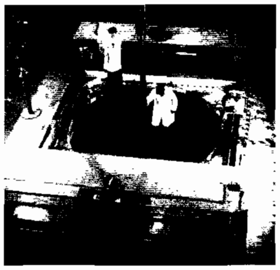


- RFSP
  - micro-depletion model for isotopic evolution calculations (burnup reflecting local parameters and history)
  - specific enhancements being assessed and under development to address heterogeneity between adjacent cells

ACR Physics Analysis Approach

- Will use WIMS 3.0
- Enhancements to RFSP
  - as they become available
- Modeling uncertainties assessed through specific detailed MCNP analysis
  - bundle powers/channel powers in steady state
  - reactivity, powers during LOCA

Qualification of Physics Toolset



- ACR-700 specific experiments in ZED-2
- Past experiments in other critical facilities
- NRU irradiations
- MCNP for “filling in the gaps”
- Independent assessments to confirm the adequacy of both modeling, and the toolset qualification

Conclusions

- Current toolset, including MCNP, is adequate for core physics design
  - MCNP analysis for situations having significant spatial heterogeneity (such as checkerboard voiding)
- Physics toolset is being enhanced to capture heterogeneity between adjacent cells
- Physics toolset qualification based on
  - extensive measurements in ZED-2
  - past measurements in other critical facilities
  - NRU irradiations
  - benchmarks against MCNP

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ACR-700 Coolant Void Reactivity

Ben Rouben
Manager, Reactor Core Physics Branch
Manager, ACR Physics
Presented to ACRS
Rockville, MD
2004 October 7

AECL
TECHNOLOGIES INC.

Design for Void Reactivity

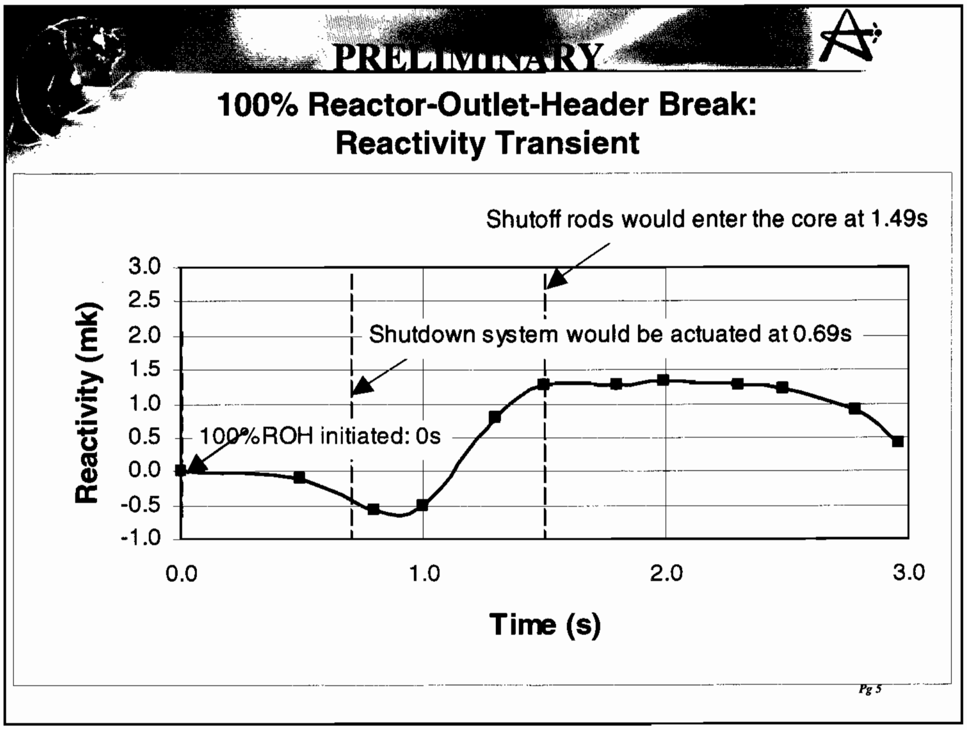
- The safety objective for the choice of the void reactivity in the ACR-700 is to provide a good balance of nuclear protection between loss-of-coolant accidents (LOCAs) and fast-cooldown accidents.
- The requirement stemming from this objective is to keep the power transient before reactor trip mild for all design basis accidents, including LOCA or steamline breaks.

Checkerboard Void Reactivity

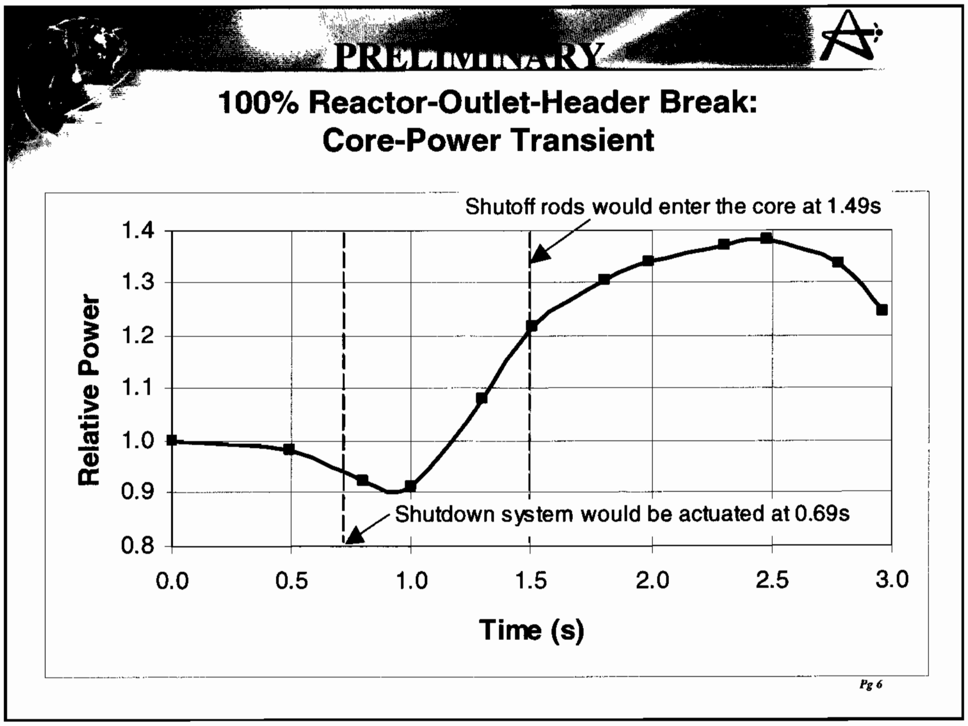
- In the ACR-700, the design of the reactor coolant system consists of two passes in a figure-of-eight, with coolant flowing in checkerboard fashion in opposite direction in neighboring channels.
- In a loss-of-coolant accident (LOCA), one pass will generally void faster than the other.
- Different coolant density in neighboring channels leads to spectrum heterogeneity, which can result in a "checkerboard" void reactivity which can be different from the reactivity generated by the same average voiding but distributed uniformly in the core.
- Note that the "extreme" case of 100% coolant density in one pass, 0% density in the other does not physically occur.

LOCA Analysis

- The next two slides show our current results for:
  - the system reactivity in the first 3 seconds of a 100% Reactor-Outlet-Header-Break Large LOCA (void reactivity was calculated with MCNP)
  - the resulting core power transient <u>without shutdown-system action</u>.


PRELIMINARY

100% Reactor-Outlet-Header Break: Reactivity Transient


PRELIMINARY

100% Reactor-Outlet-Header Break: Core-Power Transient

Conclusions

- MCNP gives us a good handle on the physics of checkerboard voiding.
- We are working to further develop other codes in our toolset to enhance their capability to model heterogeneity.
- The effect of checkerboard voiding in a LOCA is a mild power transient.
- The power transient is self-limiting and turns over within a few seconds.

AECL
TECHNOLOGIES INC.

(11)

Resolution of GSI 185

Advisory Committee on Reactor Safeguards
October 7, 2004

Jack Rosenthal, Chief
SMSAB
Office of Research

Issue

- Given a small break LOCA, can a volume of diluted water form in the primary system
  - To get boron dilution requires narrow break-size range and other independent equipment failures
- If so, can the diluted water be transported to the core
  - Restart of RCP or natural circulation
  - Mixing of diluted slug
- If transported to the core, can the core become recritical, and if so, at what power level
  - Event must occur early in fuel cycle
- If so, can fuel damage result

Our assessment had five components:
probabilistic risk assessment;
thermal hydraulic system analysis;
mixing and transport analysis;
core criticality analysis;
and fuel behavior.

- A number of RES infrastructure programs were essential to resolving this generic issue. Experiments on fluid-to-fluid mixing conducted some years ago at the University of Maryland. Participation in the international SETH-PKL program. Use of the RELAP5/PARCS computer code. Validation of that code via comparison with calculations from the Kurchatov Institute (Russia). Participation in Cabri (France) and NSRR (Japan) fuel testing programs.

Probability Considerations for Boron Dilution

- ➤ Initiating event.
  - ➤ Small LOCA (1.4 to 2.0-inch) frequency ~2 E-4/yr.
  - ➤ Stuck open pressurizer PORVs and SRVs ~2 E-3/yr.
- ➤ Small LOCAs alone do not cause substantial dilution of loop seals. Insufficient time spent in dilution mode. Must be additional failures and/or operator errors.
- ➤ Subset *boron dilution small break LOCAs* is lower probability. For example, involves failure of HPI: one train 1E-2, two trains 1E-3

To Form a Diluted Loop Seal

- Open small break. Don't inject HPI, or at least degrade it, until inventory drops into the hot leg (~60% of initial inventory)
- Then close the break, forcing the steam generators to act as the heat sink. During this time, HPI remains off to prevent refill.
- Reflux condensation must proceed for ~ one hour.
  - In PKL experiments, one hour was required to dilute the loop seals from their initial value of 1000 ppm to below 50 ppm.
  - University of Maryland loop experiments also required 70 to 90 minutes to dilute loop seals.
- Best prospect is a stuck-open pressurizer PORV or SRV that later recloses, with coincident failure of HPI.

Restart Reactor Coolant Pump

- Framatome B&W EOPs instruct operators not to restart a RCP unless:
  - Stable subcooled natural circulation has gone on for at least 60 minutes.
  - Core exit subcooling > 30F and P > 200 psia.
- Objective is to prevent RCP restart until well after possible diluted loop seals have been swept by natural circulation.

Probability Considerations for Boron Dilution

	Occurrence	Current Study
p_1	Small break LOCA	~2 E-3 (includes SBLOCA and stuck-open pressurizer valves)
p_2	Early in fuel cycle	2 E-1
p_3	Slug formation	1 E-2 [GSI report]
p_4	Restart RCP	1 E-2 [HF evaluation]
	$p_1 \times p_2 \times p_3 \times p_4$	< 1 E-7

Consequences: B&W 40 m^3 Slug

RELAP5/PARCS Calculated Result	Restart NC	Restart RCP
Fuel enthalpy increase in the first maximum power pulse	25 cal/g	30 cal/g
Fuel enthalpy after multiple power pulses	90 cal/g	185 cal/g
Peak power	500%	2700%
Maximum fuel centerline temperature	2000C	> 2800C (melting)
Minimum DNBR	1.3	< 1

Conclusions

- No recriticality for Combustion Engineering and Westinghouse based on relatively small loop seal volume
- B&W loop seal volume factor of 10 greater
- For B&W, low probability, low consequence
- Issue resolved without need for regulatory actions

(12)

ANALYSIS OF BORON DILUTION TRANSIENTS IN PWRS

Presentation to
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission

October 7, 2004

David Diamond
Brookhaven National Laboratory
Energy Sciences and Technology Department

Brookhaven Science Associates
U.S. Department of Energy

BROOKHAVEN NATIONAL LABORATORY

OUTLINE OF PRESENTATION

- Objectives of study
- Reactor analysis methodology
  - RELAP5/PARCS
- Results
- Conclusions

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OBJECTIVES OF BNL PROJECT

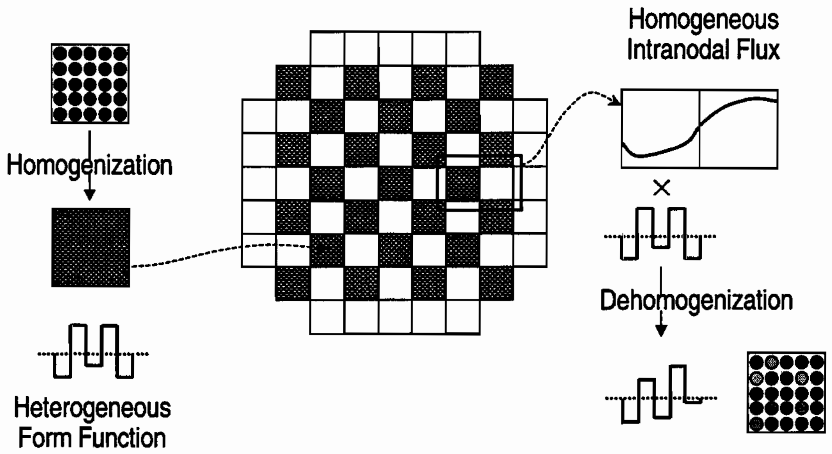
- To understand the consequences of a boron dilution event as defined in GSI-185
  - Deterministic calculations of fuel enthalpy (pellet radial average at any location in the core) as a function of time
    - Peak fuel enthalpy used as failure criterion
  - Parametric studies to determine the effect of assumptions e.g., flow rate, boron concentration, reactor type

---

REACTOR ANALYSIS METHODOLOGY

- RELAP5 for system thermal-hydraulics
- PARCS (Purdue Advanced Reactor Core Simulator) calculates the neutron kinetics and hence the power distribution as a function of time
  - Neutron balance for two neutron energy groups
  - Six groups of delayed neutron precursors
  - Each assembly represented as a uniform composition
  - Cross sections a function of variables that change during a transient
    - Fuel temperature (Doppler effect)
    - Moderator density
    - Boron concentration
    - Presence (or absence) of control rods
    - Presence of special nuclides: Xe, Sm

HOMOGENIZATION AND DEHOMOGENIZATION



Homogeneous Intranodal Flux

Homogenization

×

Dehomogenization

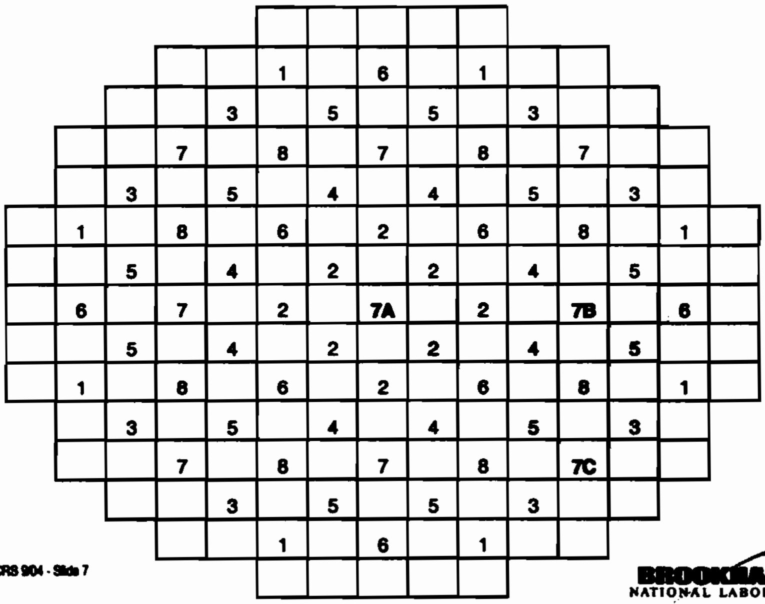
Heterogeneous Form Function

PWR MODEL FOR BORON DILUTION EVENT

- **TMI-1 Core Model at Beginning-of-Cycle**
  - **Babcock & Wilcox design, 177 15x15 FAs, 2772 MW$_{th}$**
  - **2x2 mesh/assembly**
  - **28 axial meshes**

- **Starting point for boron dilution**
  - **All banks inserted (control and shutdown)**
  - **Fuel, moderator at 500 K, 2500 ppm boron**
  - **~15$ shutdown**
  - **Transient boundary conditions**
    - **Boron concentration at lower plenum from mixing model**
    - **Flow rate based on nat'l circulation or one pump restart**

TMI-1 CORE LAYOUT WITH CONTROL BANKS



					1		6		1					
				3		5		5		3				
			7		8		7		8		7			
		3		5		4		4		5		3		
	1		8		6		2		6		8		1	
		5		4		2		2		4		5		
	6		7		2		7A		2		7B		6	
		5		4		2		2		4		5		
	1		8		6		2		6		8		1	
		3		5		4		4		5		3		
			7		8		7		8		7C			
				3		5		5		3				
					1		6		1					

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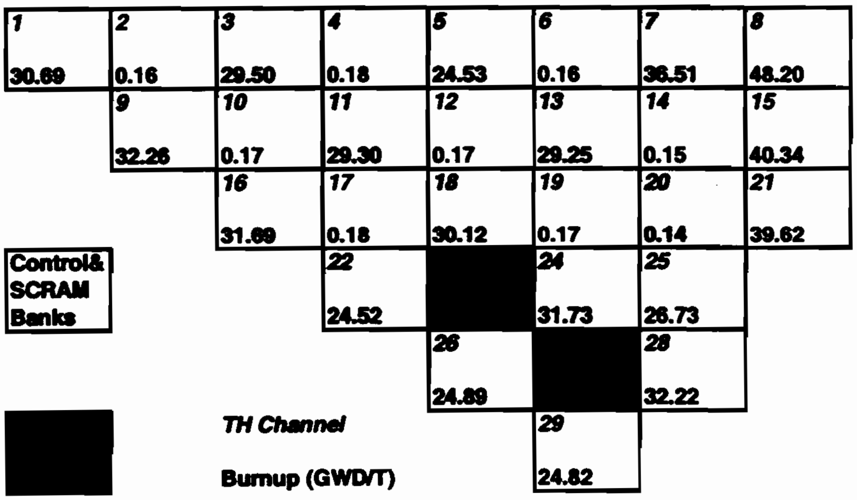
PWR CORE MODEL:
TMI-1 at BOC, HZP, ARI, 500 K, 2500 ppm, 3% Flow,



1 30.69	*2* 0.16	*3* 29.50	*4* 0.18	*5* 24.53	*6* 0.16	*7* 36.51	*8* 48.20
	9 32.26	*10* 0.17	*11* 29.30	*12* 0.17	*13* 29.25	*14* 0.15	*15* 40.34
		16 31.69	*17* 0.18	*18* 30.12	*19* 0.17	*20* 0.14	*21* 39.62
			22 24.52	■	*24* 31.73	*25* 26.73	
				26 24.89	■	*28* 32.22	
					29 24.82		

Control& SCRAM Banks

■

TH Channel

Burnup (GWD/T)

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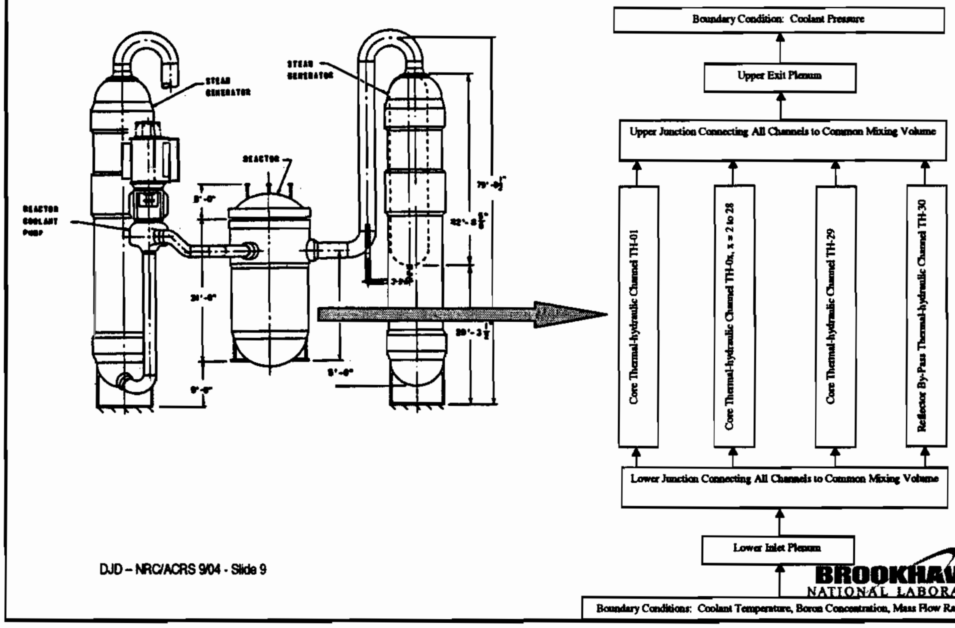
MODELING THE GSI-185 EVENTS



Boundary Condition: Coolant Pressure

Upper Exit Plenum

Upper Junction Connecting All Channels to Common Mixing Volume

Core Thermal-hydraulic Channel TH-01

Core Thermal-hydraulic Channel TH-0x, x = 2 to 28

Core Thermal-hydraulic Channel TH-29

Reflector By-Pass Thermal-hydraulic Channel TH-30

Lower Junction Connecting All Channels to Common Mixing Volume

Lower Inlet Plenum

Boundary Conditions: Coolant Temperature, Boron Concentration, Mass Flow Rate

DJD – NRC/ACRS 9/04 - Slide 9

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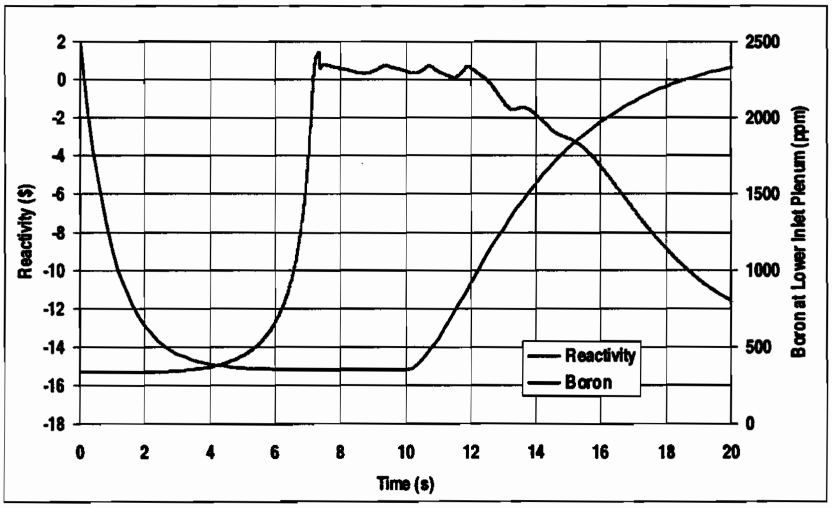
BORON CONCENTRATION AND REACTIVITY 25% FLOW



DJD – NRC/ACRS 9/04 - Slide 10

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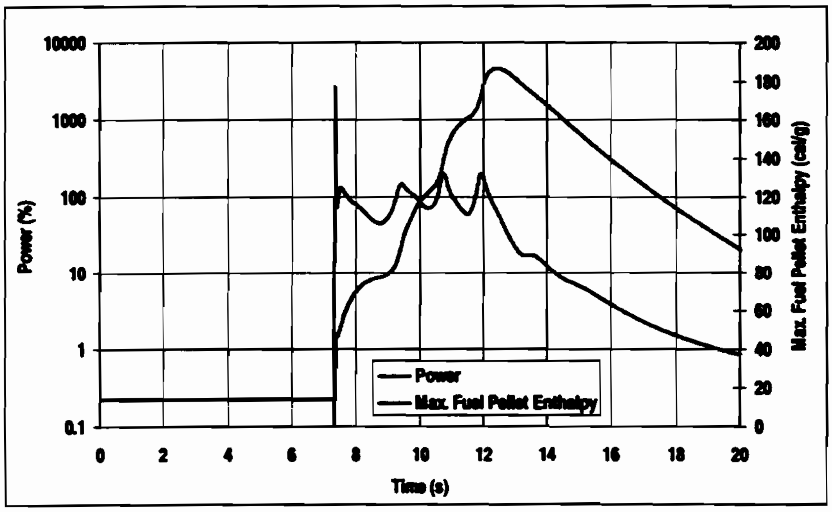
POWER AND MAX. FUEL PELLET ENTHALPY 25% FLOW



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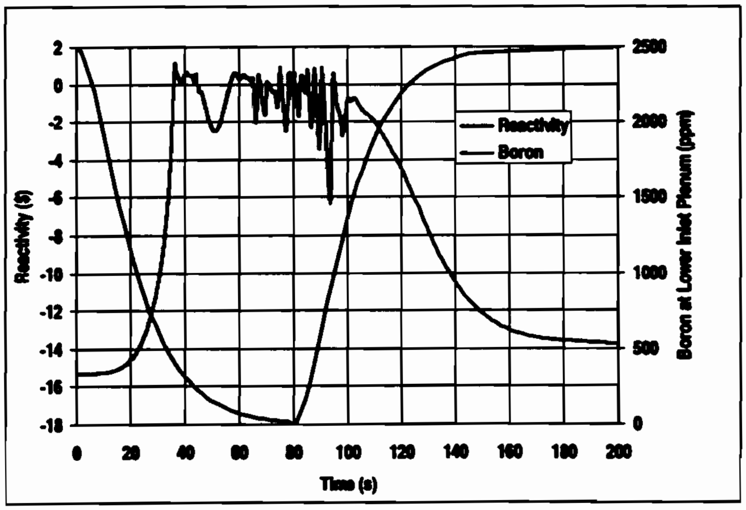
BORON CONCENTRATION AND REACTIVITY 3% FLOW



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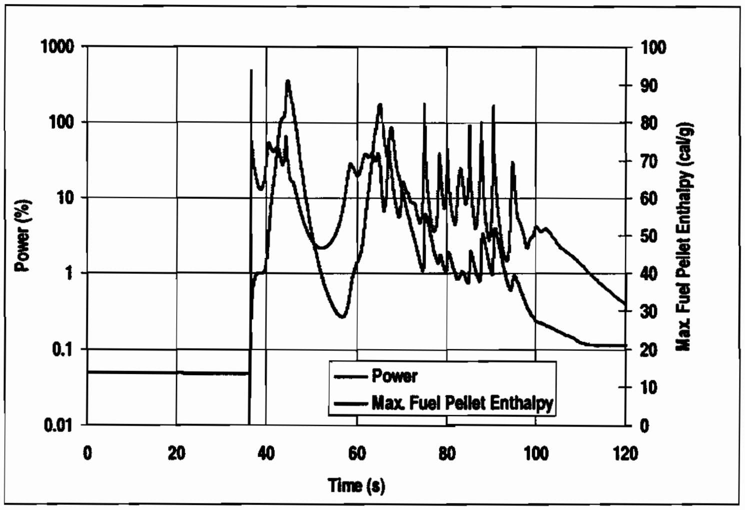
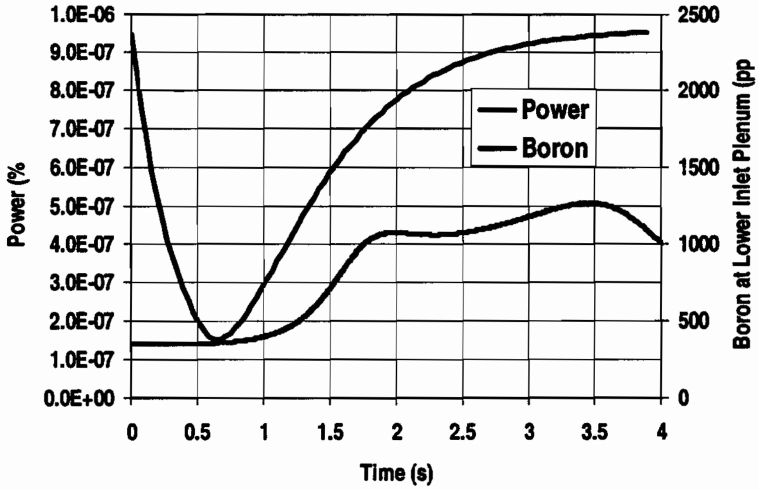
PEAK POWER AND FUEL ENTHALPY 3% FLOW



SIMULATION FOR W COLD LEG DESIGN



SUMMARY OF RESULTS

3% Flow

- Peak reactivity ~ $1.14
- Peak power ~480% at ~ 36.4 s
- Initial $\Delta h_{f\text{-max}}$ ~ 26 cal/g
- Peak $\Delta h_{f\text{-max}}$ ~ 77 cal/g at ~ 45 s
- Peak at bottom in fresh fuel

3% TO 25% FLOW IN 10 s

- Peak reactivity ~ $1.44
- Peak power ~ 2700% at ~ 7.3 s
- Initial $\Delta h_{f\text{-max}}$ ~ 33 cal/g
- Peak $\Delta h_{f\text{-max}}$ ~ 173 cal/g at ~ 12.3 s
- Peak at bottom in fresh fuel

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CONCLUSIONS

- RELAP5/PARCS is a viable method for this analysis
- Fuel enthalpy increase only significant if
  - Volume of diluted water is large enough
  - Rate of injection is large enough
- Effect only possible in first ~20% of cycle

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13

Tools: Deborate Mixing

ACRS Full Committee

Marino di Marzo
RES-DSARE-SMSAB

October 7, 2004

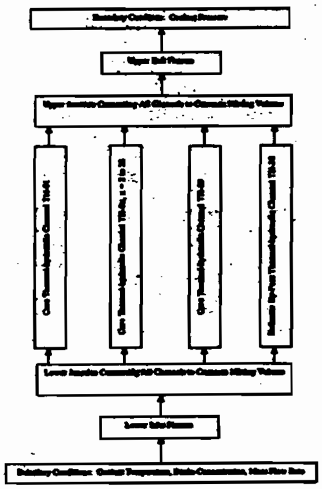
Mixing models (MM)

- ☐ Levenspiel (1962) identifies two bounding conditions for mixing
    - ■ Plug flow
    - ■ Backmix flow
- ☐ Plug flow is simply a time shift of the original input
- ☐ Backmix flow is given as:

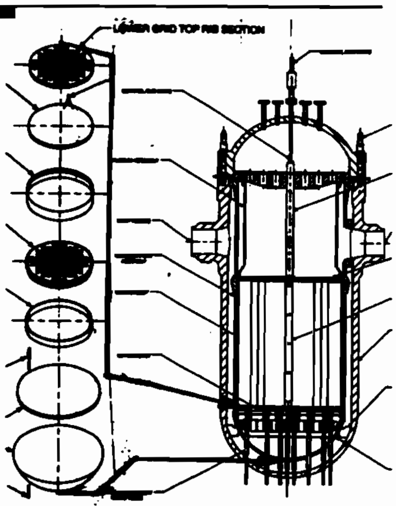
$$C(\theta)=\frac{V_S}{V_C}\int_0^{\theta}\left[C(\lambda)-C_0\right]\exp\left[\frac{V_S}{V_C}(\lambda-\theta)\right]d\lambda+C_0$$

$$\tau=\frac{V_S}{\dot{V}} \Rightarrow \theta=\frac{t}{\tau} \quad and \quad \lambda=\frac{\gamma}{\tau}$$

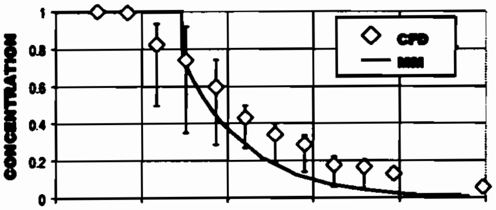
RELAP5/PARCS model



RELAP5 represents in-vessel mixing as: plug flow in the core backmix flow in the lower head and plug flow in the downcomer

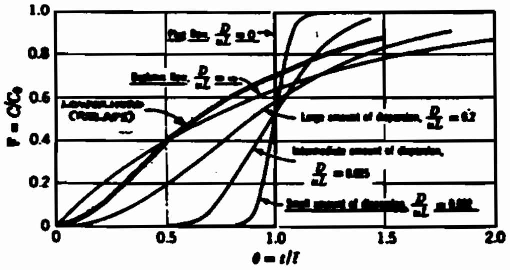


In-vessel mixing

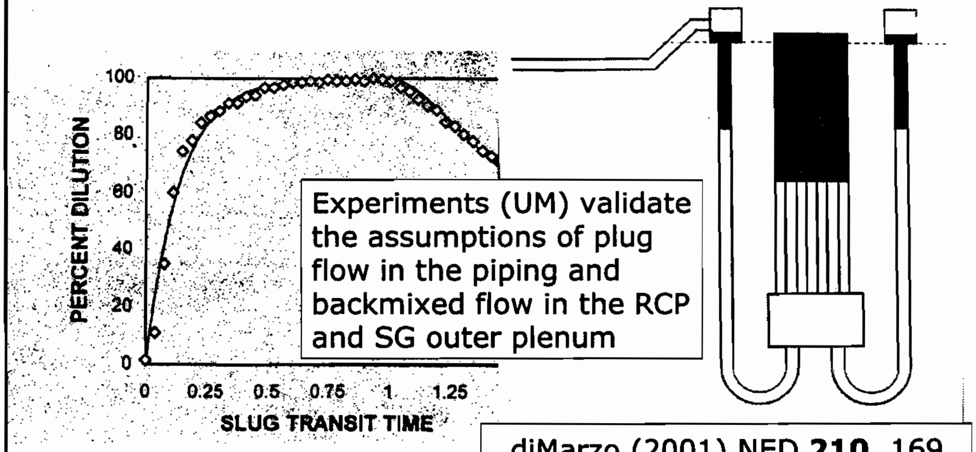


Experiments (UM) and CFD computations (RES) at reduced scale are compared with the MM

The MM provide a simple, scalable representation of the in-vessel mixing



Ex-vessel mixing



Experiments (UM) validate the assumptions of plug flow in the piping and backmixed flow in the RCP and SG outer plenum

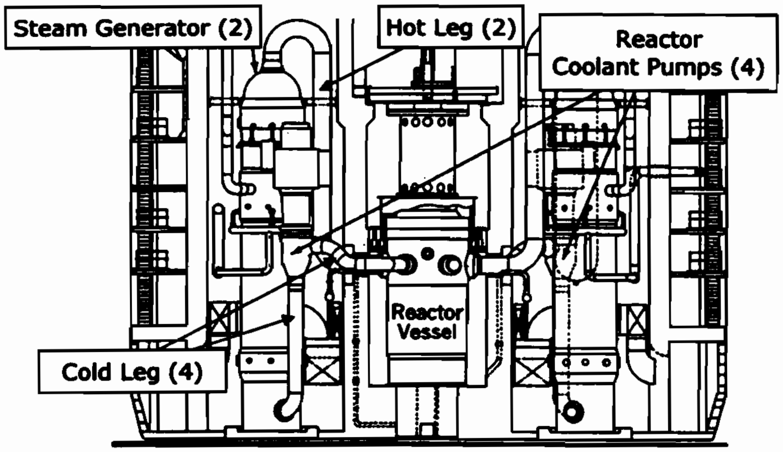
diMarzo (2001) NED **210**, 169

Mixing conclusions

- ☐ The RELAP5/PARCS model of the in-vessel mixing is a reasonable (albeit conservative) representation of the mixing in the reactor vessel
- ☐ The MM are used to generate the boundary conditions for the RELAP5/PARCS calculations in accordance with the appropriate scenarios of concern
- ☐ The ex-vessel results provide the concentration and flow time-dependent input into the lower head volume of the RELAP5 model

Backup slides

B&W Plant (Oconee)



MITIGATING SYSTEMS PERFORMANCE INDEX

PRESENTATION TO ACRS

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U.S. NUCLEAR REGULATORY COMMISSION

October 7, 2004

⑦

Purpose and Objective of Meeting

- Provide status of MSPI implementation
- Describe resolution of key technical issues and response to public comments on technical report
- Request ACRS Letter on MSPI methodology

Contents of Presentation

- MSPI Overview
  - Purpose of Meeting
  - Overall Conclusions on MSPI
  - Background
- Technical Discussion
  - Description of MSPI
  - Resolution of Key Technical Issues
  - Task Force on PRA Adequacy
  - Public Comments on Technical Report
- Summary
  - Conclusions
  - Request for Letter
- MSPI Status
  - Implementation Status
  - Future Challenges with Implementation

Conclusions on MSPI

- The MSPI is a robust performance indicator that can differentiate risk significant changes in system performance.

- The MSPI has been tested and evaluated in a pilot plant program. Its performance characteristics, strengths, and limitations are documented and accounted for.

- The MSPI provides a better overall measure of system performance than the existing Safety System Unavailability (SSU) Performance Indicator. The MSPI addresses problems associated with the SSU.

Background

- MSPI evolved from feasibility study of Risk-Based Performance Indicators (NUREG-1753).
- NRR issued User Need Request to RES to support development of risk-informed indicator that includes unreliability and safety system unavailability.
- MSPI formulated to address known issues with current indicator
    - Use of fault exposure time in the SSU Performance Indicator
    - Omission of unreliability elements from indicator
    - Definition of unavailability inconsistent with Maintenance Rule and INPO/WANO indicators
    - Cascading of cooling water support systems failures
    - Thresholds that do not recognize plant-specific features.
- Twelve-month Pilot Program initiated September 2002.
- ACRS subcommittees briefed on July 8, 2003 and April 14, 2004 regarding status of pilot and RES-recommended improvements to method.
    - No open items.

Overview of MSPI Features

- Eliminates known problems with existing SSU Indicator.
- Accounts for both unavailability and unreliability of a system, weighted relative to their Risk-Importance.
- Uses plant PRA model to derive Risk-Importance weightings. Hence, captures plant-specific configuration and performance.
- Identifies changes in equipment performance while taking into account expected performance variations.
- MSPI data are consistent with PRA methods and Maintenance Rule data. Data to be integrated with Consolidated Data Entry (CDE) Program under INPO.

Definition of MSPI

The MSPI is ***a measure of the deviation of actual plant system unavailability and component unreliabilities from historical baseline values, where each element is weighted by plant-specific risk importance measures.***

MSPI Technical Approach

- MSPI monitors risk impact (i.e., approximate change in CDF) of changes in performance of selected mitigating systems, which accounts for plant-specific design and performance data.

- MSPI consists of two elements, system unavailability and system unreliability. MSPI is the sum of changes in a simplified CDF evaluation resulting from changes in system unavailability and system unreliability relative to baseline values.

- MSPI = UAI + URI where

  UAI: system unavailability index due to changes in train unavailability

  URI: system unreliability index due to changes in component unreliability

- The risk impact of changes in mitigating system performance on plant-specific CDF is estimated using plant-specific performance data and Fussell-Vesely importance measures.

List of MSPI Monitored Systems

BWRs	PWRs
HPCI/HPCS (high pressure coolant injection/core spray)	HPSI (high pressure safety injection)
RCIC (reactor core isolation cooling) or Isolation Condenser	AFW (auxiliary feedwater or equivalent)
RHR (residual heat removal)	RHR
EAC (emergency AC power)	EAC
Support System Cooling (ESW + CCW)	Support System Cooling

Resolution of Key Technical Issues

- Frontstop
  - Expected performance variation should *not* result in crossing a performance threshold.
  - The *frontstop* is a mechanism that minimizes likelihood that one failure beyond baseline in 3-year period results in White. However, index could still become White with one or even zero failures if there is significant system unavailability.
  - Decision to move forward with use of *frontstop* as recommended in draft NUREG report.

Resolution of Key Technical Issues (cont.)

- Backstop
  - Some systems and/or components within systems may be of sufficiently low risk significance that extraordinarily high number of failures would be necessary to cross MSPI performance threshold.
  - The *backstop* is a mechanism that results in White indication if component type exhibits statistically significant departure from expected number of failures in 3-year period, regardless of risk-significance.
  - Sufficient number of failures in short-time would still trip threshold before 3-year period is over.
  - Decision to move forward with use of *backstop* as recommended in draft NUREG report.

Resolution of Key Technical Issues (cont.)

- Short-term Backstop
  - Some concern expressed that a situation such as the four Salem-1 EDG failures in 3rd Quarter 2002 did not quite reach White threshold in MSPI, although a White finding in the Significance Determination Process (SDP).
  - An additional *short-term backstop* based on departure from expected number of failures over one or two quarters evaluated.
  - Conclusion that the short-term backstop would further complicate the index, was not in keeping with monitoring trend over three-year period, and decision to keep the SDP obviated the need.
  - Decision to move forward at this time without implementation of short-term backstop.

Resolution of Key Technical Issues (cont.)

- Constrained non-informative prior (CNIP)
  - Some concern that Bayesian formulation could mask plant-specific component performance.
  - CNIP demonstrated to provide best false positive/false negative characteristics of priors considered in NUREG-1753.
  - With no prior, NUREG-1753 found to be too volatile leading to high false positive probability.
  - RES assessed other possibilities such as the *mixture prior* which have promise, but require much more data analysis, and more development and assessment is necessary.
  - Decision to move forward with use of CNIP as recommended in draft NUREG report.

MSPI PRA Quality Task Group

- To determine the PRA quality needed for the MSPI application.
- To identify the appropriate role of the ASME PRA Standard.
- Identifying process for documenting that the appropriate quality has been achieved.
- Identify the main modeling issues that give rise to variability among licensee models, and between licensee models and SPAR models. To identify which of these issues are most important to the MSPI.
- Consists of three staff from NRC (NRR, RES, Region I) and two from industry.

Public Comments on the MSPI Technical Report

- **Comments received from:**
  - **F. G. Burford, *Entergy***
  - **Mark Burzynski, *TVA***
  - **Fred Madden, *TXU Power***
  - **L. William Pearce, *FENOC***
  - **Anthony Pietrangelo, *NEI***
  - **Bill Vesely, *NASA***

- **Supportive of MSPI technical concepts.**
- **Nuclear industry representatives endorse all six recommendations in draft NUREG report.**
- **Comments on "cohort effects" from Dr. William Vesely addressed in Appendix M of report.**

Response Regarding Cohort Effects

- It is recognized that the MSPI is a *linearized approximation* to the change in CDF for a given change in system unavailability or unreliability.
- Plant-specific importance measures are derived and used as "weights" in the MSPI formulation that monitors deviation of system unavailability and component unreliabilities from historical baselines.
- The linear approximation is recognized to be valid for small deviations from the norm. An assessment found the formulation to generally be acceptable based on pilot plant performance data, though some close observation may be warranted once implemented.
- The formulation would clearly be inappropriate for other risk-informed applications such as on-line risk monitoring or technical specification changes where removal from service of high risk components could cause large factor increases in instantaneous CDF.

In Conclusion

- MSPI has been tested and evaluated in a pilot plant program, and discussed in numerous public meetings.
  - It addresses problems with currently used PIs.
  - Its capabilities, strengths, and limitations are documented and accounted for.
- MSPI is a robust performance indicator.
- MSPI has desirable qualities with respect to:
  - Plant-specific risk implications.
  - Proper treatment of reliability and availability.
  - Ability to capture system performance degradation.
  - Computation is structured and programmable.
  - MSPI is consistent with Maintenance Rule, Technical Specifications, and ROP SECY 99-007.
- PRA adequacy issues are being addressed by task force.

- **Request ACRS Letter on MSPI methodology**

Recent Staff MSPI Activities

- One year pilot of the MSPI completed in early 2004.
- Commission provided staff guidance in SRM's dated April 8 and May 27, 2004.
- Staff and stakeholders conduct monthly meetings on MSPI.
- NRR staff issued letter on September 15, 2004 to NEI documenting agreement to move forward with MSPI implementation.

Status of Remaining Technical Issues

- Staff and industry agree to retain frontstop and define the minimal set of PRA requirements and issues important to MSPI.

- Staff-industry task force created to identify important PRA issues that impact MSPI. Resolution of PRA-related issues by the task force will reduce the TI inspection burden on initial implementation.

- Working with industry to reach agreement on implementation details contained in the guidance documents.

- MSPI will be implemented at all sites at the same time. No partial or delayed implementation.

Future Challenges with MSPI Implementation

- Implement MSPI in a manner that minimizes interpretation issues and minimizes staff resource demands to oversee MSPI.
- Issue final MSPI guidance documents (99-02, Section 2.2 & Appendices F and G).
- Issue Staff Communication Plan and Regulatory Issues Summary.
- Assess re-alignment of Maintenance Rule guidance with MSPI (i.e., evaluation of the need to monitor UA during shutdown conditions).
- Conduct/participate in three public workshops.
- Conduct internal workshops/training.
- Develop MSPI TI and resolution processes to handle MSPI technical issues and disagreements.

REGULATORY STRUCTURE FOR NEW PLANT LICENSING, PART 1: TECHNOLOGY-NEUTRAL FRAMEWORK

"POLICY ISSUES"

PRESENTED TO
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

Presented by
Mary Drouin, Amarjit Singh, Tom King, Stuart Rubin
US Nuclear Regulatory Commission
John Lehner, Trevor Pratt, Vinod Mubayi
Brookhaven National Laboratory
Dennis Bley
Buttonwood Consulting, Inc.

October 8, 2004

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Purpose

- Information briefing on policy issues
- Secy Paper due to Commission in December, 2004
- Paper will discuss
  - Status of framework
  - Implementation of previously approved issues
  - Recommendations on integrated risk and containment
  - New policy issues
- Brief the full Committee , December 2, 2004 on paper, Requesting a letter

Background/History

- SECY-03-0047
- SRM on SECY-03-0047
- SECY-04-0103
- SECY-04-0157

Policy Issues

- Commission approved staff approach
    - Definition of defense-in-depth
    - Probabilistic approach for licensing basis
    - Licensing source term
    - Offsite emergency preparedness
- Commission requesting additional information
    - Integrated risk
    - Containment performance
- Potential New Issues
    - Level of Safety
    - Security
    - Selective Implementation

Defense-in-Depth

- Commission approved development of a description of Defense-in-Depth (DID) and to be incorporated in PRA policy statement
- Staff approach
    - Develop principles
    - Develop model
    - Implementation

Probabilistic Approach for Licensing Basis

- Commission approved use of probabilistic criteria for identification of elements to be considered in design and safety classification
- Staff Approach
  - Identify/define event sequence categories
  - Two category SSC classification scheme
  - Replace SFC with event sequences from design-specific PRA
- Consistent with Safety Goal Policy
- Need "living" PRA

Licensing Source Term

- Commission approved use of scenario-specific source terms
- Staff proposes a flexible, performance-based approach
- Burden on applicant to develop the technical basis

Offsite Emergency Preparedness

- Commission approved that no change to emergency preparedness requirements are needed in the near term
- Staff approach requires a baseline emergency preparedness capability

Integrated Risk

- Commission asked the staff to provide further details on the options for, and associated impacts of, requiring modular reactor designs account for the integrated risk posed by multiple reactors
- Staff approach, metrics associated with both accident prevention and mitigation need to be considered
- ACRS
  - Recommend QHOs apply to site as a whole
  - Differing views presented on how to treat CDF
  - ACRS view expands scope

Integrated Risk (Cont'd)

- Staff evaluation limited to modular reactors
- Non modular reactor not an issue in near term requiring Commission direction
- Staff recommendation
  - Not address risk for non-modular reactors
  - For modular reactors, integrated risk should be considered:
    - Accident prevention independent of reactor barrier level and
    - Accident mitigation that allows for consideration of the affected power level

Background (Non-LWR Containment Functional Performance Requirements and Criteria)

The Commission SRM direction on SECY-03-0047:

- Develop options for non-LWR containment functional performance *requirements* and *criteria*,
- Account for such features as core, fuel and cooling system designs
- Interact with industry experts and other stakeholders to develop the options
- Submit options and recommendations for Commission decision

Policy Development Activities

- NRC Workshop on Regulatory Structure for New Plant Licensing Framework for a Risk-Informed Regulatory Structure for Advanced Reactors, November 2003
- NRC Workshop on Non-LWR Containment Functional Performance, January 2004
- SECY-04-0103, Status of Response to the June 26, 2003, Staff Requirements Memorandum on Policy Issues Related to Licensing Non-LWR Designs, June 23, 2004
- NRC Workshop on Framework and Non-LWR Containment Functional Performance, July 2004

“Containment” Functional Roles

- Protect SSCs important to safety from internal/external events
- Ensure physical support of SSCs important to safety
- Protect onsite workers from radiation
- Provide physical protection for SSCs important to safety
- Remove heat to prevent SSCs important to safety from exceeding design or safety limits
- Reduce radionuclide releases to the environs (including limiting core damage)

Technology-Neutral "Containment"

- Third Level Radionuclide Barrier (Main Focus)
- Prevention Functions
- Mitigation Functions
- Building (Structural) Elements
- System Elements

Adopt technology-neutral term: TLPMBS

- Third Level Prevention Mitigation Building System

TLPMBS Performance *Requirement*

- *Must reduce radionuclide releases adequately to ensure that the dose criteria are met for the selected events in the event categories.*

Development of TLPMBS Performance *Criterion*

Conform to framework by using:

- Frequency-consequence curve to limit risk
- Probabilistic approach to identify design-basis events; deterministic engineering judgment to bound uncertainties
- Scenario-specific mechanistic source term; deterministic engineering judgment to bound uncertainties
- Best-estimate deterministic analysis for design-basis events and uncertainty analysis, with 95% confidence level for meeting criteria
- Defense-in-depth to address "random" (stochastic) uncertainties and "state-of-knowledge" (e.g., completeness) uncertainties

Development of TLPMBS Performance *Criterion* (cont.)

Use Defense-in-Depth Principles:

- Balance accident prevention with accident mitigation
- Key safety functions (e.g., control fission product release, chemical attack) <u>not</u> dependant on a single element of design, construction or operation
- Account for uncertainties in SSC (e.g., fuel and heat removal systems, pressure boundary) and human performance
- Use two elements of defense-in-depth:
  - Rationalist – for model and parameter uncertainties
  - Structuralist – for completeness uncertainties

Options demonstrate/provide progressive mitigation capability to reduce radionuclide release to the environment (Defense-in-Depth)

Alternative Functional Performance *Criterion* (Options)

1. Must be adequate to meet the dose criteria.
2. Must be adequate to meet the dose criteria, include in the design-basis category bounding events with potential high consequence source terms.
3. Must be adequate to meet the dose criteria and have the capability for low leakage and controlled release of delayed accident source term.
4. Must be adequate to meet the dose criteria by being essentially leak-tight for both prompt and delayed accident source terms.

Level of Safety

- Commission approved the staff's recommendation on implementation of the Commission's expectation for enhanced safety in future reactors
- What is the criteria for achieving enhanced safety?
- Staff approach: develop requirements to achieve level of safety defined as Safety Goal QHOs
- Consistent with Advanced Reactor Policy Statement
- Plan to solicit stakeholders input

Security

- Raised as a potential new policy issue
- Issue being evaluated with recommendation to be proposed for Commission consideration
- Staff intends to implement Commission direction on this issue

Selective Implementation

- Raised as a potential new policy issue
- Not meant to preclude exemption process
- No longer considered a policy issue

Plan and Schedule

- Preliminary framework drafted
  - Work to date indicates the feasibility of developing a technology-neutral approach
- Issue working draft to public to engage stakeholder's input early in the process
  - Consistent with Commission direction in Advanced Reactors Policy Statement
    - "the Commission encourages the earliest possible interaction of applicants, vendors, and other government agencies,…"

Plan and Schedule (Cont'd)

- SECY paper on framework and policy issues due to Commission in December 2004
- Brief the ACRS, December 2, 2004, staff will request a letter

(16)

G:PPHandout

ACRS MEETING HANDOUT

Meeting No. 516	Agenda Item 11	**Handout No.:** 11.1

Title PLANNING & PROCEDURES/ FUTURE ACRS ACTIVITIES

Authors
JOHN T. LARKINS

List of Documents Attached

PLANNING & PROCEDURES MINUTES

11

Instructions to Preparer
1. Paginate Attachments
2. Punch holes
3. Place Copy in file box

From Staff Person
JOHN T. LARKINS

October 7, 2004
G:PlanPro(ACRS):ppmins.516

INTERNAL USE ONLY

SUMMARY MINUTES OF THE
ACRS PLANNING AND PROCEDURES SUBCOMMITTEE MEETING
October 8, 2004

The ACRS Subcommittee on Planning and Procedures held a meeting on October 8, 2004, in Room T-2B3, Two White Flint North Building, Rockville, Maryland. The purpose of the meeting was to discuss matters related to the conduct of ACRS business. The meeting was convened at 1:30 p.m. and adjourned at 3:40 p.m.

ATTENDEES
M. Bonaca
G. Wallis
S. Rosen

ACRS Staff
J. T. Larkins
S. Duraiswamy
J. Gallo
M. Snodderly
H. Nourbakhsh
M. Sykes
M. El-Zeftawy
C. Santos
J. Flack
S. Meador
M. Afshar-Tous
M. Weston
R. Caruso

NRC Staff
D. Diec, NRR

1) Review of the Member Assignments and Priorities for ACRS Reports and Letters for the October ACRS meeting

Member assignments and priorities for ACRS reports and letters for the October ACRS meeting are attached (pp. 7-12). Reports and letters that would benefit from additional consideration at a future ACRS meeting were discussed.

RECOMMENDATION

The Subcommittee recommends that the assignments and priorities for the October ACRS meeting be as shown in the attachment (pp. 7-12).

2) Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through December 2004 is attached (pp. 7-12). The objectives are to:

- Review the reasons for the scheduling of each activity and the expected work product and to make changes, as appropriate
- Manage the members' workload for these meetings
- Plan and schedule items for ACRS discussion of topical and emerging issues

During this session, the Subcommittee also discussed and developed recommendations on items included in Section IV of the Future Activities list (pp. 13-16).

RECOMMENDATION

The Subcommittee recommends that the members provide comments on the anticipated workload. Changes will be made, as appropriate.

3) Proposed ACRS Meeting Dates for CY 2005

Proposed ACRS meeting dates for CY 2005 are included in the attached calendar (pp. 17-28) and summarized below.

Meeting No.	Dates
---	January 2005 (No meeting)
519	February 10-12, 2005
520	March 3-5, 2005
521	April 7-9, 2005
522	May 5-7, 2005
523	June 1-3, 2005
524	July 6-8, 2005
---	August 2005 (No meeting)
525	September 7-10, 2005
526	October 6-8, 2005
527	November 3-5, 2005
528	December 8-10, 2005 (Changed)

These proposed meeting dates were provided to the members during the September ACRS meeting for feedback.

As suggested by Dr. Apostolakis and agreed to by several members, the December meeting dates have been changed from December 1-3 to 8-10.

During the September meeting, Dr. Bonaca and Mr. Rosen suggested changing the March meeting dates from March 3-5 to 10-12, 2004. In view of the fact the Regulatory Information Conference is scheduled for March 8-10, coupled with the fact the Commission meeting with the ACRS is scheduled for March 3, 2004, the March meeting dates have not been changed.

The Committee needs to approve the meeting dates for CY 2005 during the October 2004 ACRS meeting.

RECOMMENDATION

The Subcommittee recommends that the members provide feedback on the proposed dates and that the Committee approve dates for CY 2005 ACRS meetings during the October 2004 ACRS meeting.

4) ACRS Retreat in 2005

During its September 2004 meeting, the Committee discussed whether to have a retreat in 2005, and deferred its decision until after evaluating the topics proposed by the members for discussion at the retreat. Topics proposed by several members are attached (pp. 29-35) and also listed below under two categories (Process Issues and Technical Issues). The members who proposed the issues are identified in the parenthesis.

Process Issues

- Incoming Chairman's agenda (DAP)
- Alternative approach to review license renewal applications in view of Mr. Leitch's retirement (DAP)
- Should the ACRS take on fewer issues and spend more time on each? (DAP)
- Assessment of the Subcommittee structure and member assignments (DAP)
- Should the ACRS continue to defer voicing and/or documenting its concerns regarding the quality of science and engineering that goes into regulations, more importantly into the regulatory process? (DAP)
- Does the ACRS have a proactive role in nuclear safety issues? If so, how does it originate? (VHR)
- Can the ACRS better influence the NRC safety research (i.e., topics and quality) (VHR)
- Could the report writing be conducted so that a draft is available before the review starts? (This is done sometimes now). Perhaps, the review could begin by going around the table and hearing the main point of view of each member. (VHR)
- What will it take for the ACRS to recommend disapproval of a license renewal application? (TSK)
- Interactions with the NRC staff outside the formal Subcommittee and full Committee meetings (GBW)
- What are the limits to "working with the staff" (suggested by the Commission on certain issues) (GBW):

  - Acting like members of the staff, or its consultants, in developing solutions to the problems
  - Stepping into management gaps and trying to organize the staff and set objectives
  - Planning activities or strategies
  - Doing joint brainstorming or analysis of alternative approach to issues
  - Being peer reviewers
- Should we hold discussions with the staff only at formal meetings or through formal communications, so that any criticism or disagreement is entirely out in the open (GBW)
- Should we workout anything that appears at all potentially disagreeable in private and have more apparent consensus in public? (GBW)
- Agenda for the Quadripartite meeting (should be a focused meeting - perhaps on risk-informed regulation or on divergence) (TSK)

Technical Issues

- Extended Power Uprate Issues (MVB)
- Power uprates - Is there any limit other than component capacities? Is there not a risk limit? (TSK)
- Is a design-basis accident a useful concept for future reactors? Should the concept be abandoned in favor of an examination of risk? Would not abandoning the design-basis concept lead to technology neutral regulations? (DAP)
- What should be the role (if any) of design-basis accidents in future regulations? (TSK)
- Why do we think redefining the LBLOCA size is O.K.? (TSK)
- What is the ACRS position on a technology neutral regulatory framework? (TSK):
  - risk acceptance criteria
  - appropriate F-C curves
  - defense in depth - in particular the question of whether a containment is always required
- Any preliminary concerns about design certification of ACR-700 (TSK)
- What should ACRS recommend on sump blockage? (TSK)
- What more should the ACRS do about security issues - particularly with respect to spent fuel? (TSK):
  - pool
  - dry cask storage
  - transportation
  - interim storage facility
- Shouldn't the ACRS do some more complaining about air oxidation? (TSK)
- Organizational factors (Safety Culture) (TSK/SLR)

Proposed Locations

- Baltimore, MD
- Salt Lake City, UT
- Missoula, MT
- Santa Barbara, CA
- San Diego, CA
- Datmouth, NH

- Cambridge, MA (MIT)
- St. Louis
- New Orleans
- Las Vegas, NV (Yucca Mountain)
- Phoenix
- Florida

Since we will be operating under continuing resolution, holding the retreat out of town will be subject to the availability of resources.

RECOMMENDATION

The Subcommittee recommends that an expanded meeting of the Planning and Procedures Subcommittee be held during the week of January 24, 2005, to discuss some process issues and certain regulatory issues. The Planning and Procedures Subcommittee will select a list of process and regulatory issues and send it to the members for feedback. The Committee should decide on the dates and location for the meeting.

5) State of Vermont Request to the ACRS

On September 17, 2004, Mr. David O'Brien, NRC State Liaison Officer for Vermont, sent a letter to Dr. Bonaca, ACRS Chairman, (pp. 36-45) requesting on behalf of the State of Vermont that:

- The ACRS specifically review, as part of Entergy's request for extended power uprate for the Vermont Yankee Nuclear Plant, Entergy's request to change Vermont Yankee's design basis to take credit for containment overpressure to demonstrate the adequacy of its emergency core cooling and containment spray pumps, and the NRC staff's policy of granting such requests. (The reasons for the above request are included in the attachment.)

- State of Vermont be allowed to present its case before the ACRS Subcommittee and full Committee, along with the Applicant and staff, in the Committees' deliberations of Vermont Yankee extended power uprate.

- One or both of the Subcommittee and full Committee meetings regarding the Vermont Yankee extended power uprate be held in the vicinity of the nuclear plant.

Also, attached for your information is an e-mail, dated September 21, 2004, sent to Dr. Larkins by Mr. Blanch regarding Vermont Yankee power uprate (pp. 46-50).

RECOMMENDATIONS

The Subcommittee recommends the following:

- The Thermal-Hydraulic Phenomena Subcommittee should hold a meeting in the vicinity of the plant to review the Vermont Yankee extended power uprate application, and the issues raised by the State of Vermont.

- Representatives of the State of Vermont should be provided an opportunity to express their views during the meeting.
- A response should be sent to Mr. David O'Brien.

10) TRACE Computer Code - Anonymous Letters

Drs. Wallis and Ransom have each received a copy of an anonymous letter, alleging inconsistencies in the development of the TRACE code and the review of thermal-hydraulic codes in general by the ACRS (pp. 51-55). This letter is somewhat similar to the e-mail sent to Drs. Wallis and Ransom previously.

RECOMMENDATION

The Subcommittee recommends that the ACRS Executive Director refer this matter to the NRC Executive Director for Operations for disposition.

7) Member Issue

Marriott hotel, across from the White Flint Subway Station, will be opening soon. During the September meeting, there was a discussion about members staying at this hotel when they come to the ACRS meetings. However, no decision was made. There are several positive aspects staying at this hotel, such as:

- ACRS Office is walking distance from the hotel.
- Members don't have to depend on the subway train or other modes of transportation to come to the meeting especially in the event of inclement weather.
- If the meeting ran late because of not completing certain agenda items within the allocated time, and if there is a need to start the next day meeting little early, staying in a hotel close to the ACRS Office would be helpful.
- We can start the meetings at 8:00 a.m., as needed, and recess the meeting early to enable the members to travel to Bethesda for dinner, if they desire.

RECOMMENDATION

The Subcommittee recommends that the members decide whether they would like to stay at the Marriott hotel in Rockville.

ANTICIPATED WORKLOAD
OCTOBER 7-9, 2004

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Kress		El-Zeftawy	Safety Assessment Report Associated with the Pre-application review of the ACR-700 Design	B	To provide Committee's views	**Draft 1**
		El-Zeftawy	Technology Neutral Framework for Future Plant Licensing **[INFORMATION BRIEFING]**	—	—	—
Powers	Rosen/Kress	Nourbakhsh/ Duraiswamy	Assessment of the Quality of the NRC Research Projects on Sump Blockage and on MACCS Code	Report to be complied in November	—	—
		Nourbakhsh/ Duraiswamy	Divergence in Regulatory Approaches Between U.S. and Other Countries	A	To respond to the Commission SRM	**Draft 1**
Ransom	Kress	Caruso	Proposed Resolution of GSI-185, "Control of Recriticality Following Small-Break LOCAs in PWRs"	A	To support the staff schedule	**Draft 1**
Sieber	—	Weston	Mitigating System Performance Index (MSPI) Program	B	To provide Committee's reviews	**Draft 1**

ANTICIPATED WORKLOAD
OCTOBER 7-9, 2004 (Cont'd)

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Wallis	—	Caruso	Safety Evaluation Report for the Evaluation Guidelines Regarding Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at PWRs	A	To support the staff schedule	**Draft 1**
	Ford	Santos	Response to the August 25, 2004 EDO Response to the May 21, 2004 ACRS Letter on Resolution of Certain Items Identified by the ACRS in NUREG-1740, "Voltage-Based Alternative Repair Criteria	A	To respond to the EDO	**Draft 1**

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ANTICIPATED WORKLOAD
NOVEMBER 4-6, 2004

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	Rosen/Kress	Nourbakhsh/ Duraiswamy	Assessment of the Quality o the NRC Research Projects on Sump Blockage and on MACCS Code	A	To support the staff schedule	**Draft**
Bonaca	—	Savio/Major	Safeguards and Security Matters	—	—	—
		Santos	**Subcommittees Report** — Interim Review of the Farley License Renewal Application — Subcommittee Mtg. November 3, 2004	—	—	—
Kress	—	El-Zeftawy	AP1000 Lessons Learned Report	A	To identify issues stemming from the ACRS review of AP1000	**Draft**
		El-Zeftawy	Status of Early Site Permit Reviews - **INFORMATION BRIEFING**	—	—	—
Rosen	—	Sykes	Proposed Rule on Post-Fire Operator Manual Actions	A	To support the staff schedule	—
Shack	—	Snodderly	Proposed Rule for Risk-Informing 10 CFR 50.46	A	To support the staff schedule	—

ANTICIPATED WORKLOAD NOVEMBER 4-6, 2004 (Cont'd)

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Sieber		Weston	Proactive Materials Degradation Assessment Program - **INFORMATION BRIEFING**	—	—	—
		Weston	Significant Operating Events and Grid Reliability	B	To provide Committee's views	—

ANTICIPATED WORKLOAD
DECEMBER 2-4, 2004

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	Shack	Snodderly	Draft Final NUREG Documenting the Expert Elicitation on LBLOCA Frequencies **(TENTATIVE)**	A	To support staff schedule	—
	—	Snodderly	**Subcommittee Report** - Status of the Development of Draft NUREG on Treatment of Uncertainties - Subc. Mtg. 11/16/04	— —	— —	— —
	—	Snodderly	**Subcommittee Report** - Draft ANS Standard on External Events, Subc. Mtg. 11/15/04	—	—	—
Bonaca	—	Savio/Major	Safeguards and Security Matters	A	To provide Committee's views	—
		Santos	**Subcommittee Report** - Interim Review of the Arkansas Unit 2 License Renewal Application - Subc. Mtg 12/1/04	—	—	—
Kress	—	El-Zeftawy	Proposed Rule for AP1000 Design Certification	Possible Larkinsgram	—	—
	—	El-Zeftawy	Commission Paper on "Regulatory Structure for New Plant Licensing, Part 1: Technology Neutral Framework"	A	To support staff schedule	—
Shack	Apostolakis/ Wallis	Nourbakhsh/Santos	PTS Technical Basis Reevaluation Project	A	To support staff schedule	—

ANTICIPATED WORKLOAD
DECEMBER 2-4, 2004 (Cont'd)

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Sieber	–	Sykes	Generic Correspondence Regarding the use of Ultrasonic Flow Measurement Systems (**TENTATIVE**)	A	To support staff schedule	–
	–	Weston	Proposed Revisions to Management Directive 6.4. Generic Issue Process	Possible Larkinsgram	–	–
Wallis	–	Caruso	Power Uprate for the Waterford Nuclear Plant (**TENTATIVE**)	A	To support staff schedule	–

Items Requiring Committee Action

1 **Review Draft NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities"** (Open)

Member: Stephen Rosen **Engineer:** Marvin Sykes

Estimated Time:

Purpose: Determine a Course of Action

Priority: High

Requested by: RES J.S. Hyslop

In a letter dated September 17, 2004 from Charles Ader to John Larkins, RES submitted the draft NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities" for ACRS consideration prior to its release for public review and comment. The report presents the results of the collaborative program with EPRI and RES to develop state-of-art methods for conducting fire risk analysis at nuclear power plants. The studies included in the report address the full breadth of fire risk analysis technical issues for power operations, and include consideration of large early release frequency. The report excludes low power/shutdown operations, spent fuel pool accidents, sabotage, and PRA level 3 estimates of consequence.

Mr. Rosen recommends that the Committee review this document after reconciliation of public comments. The Planning and Procedures Subcommittee agrees with Mr. Rosen's recommendation.

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2 **Review of Draft Proposed Rule on Post-Fire Operator Manual Actions** (Open)

Member:	Stephen Rosen	**Engineer:**	Marvin Sykes
Estimated Time:	1 hour		
Purpose:	Determine a Course of Action		
Priority:	High		
Requested by:	NRR	David Diec	

On September 20, 2004, the staff submitted the draft proposed rule on post-fire operator manual action to the ACRS for consideration prior to its release for public review and comment. The proposed rule would provide an alternative method for satisfying 10 CFR 50 Appendix R requirements. Specifically, the proposed rule would allow for the use of approved manual operator actions as an acceptable alternative to achieve safe shutdown in lieu of adequate equipment and/or permanently installed barriers. In its letter, the staff requested that the ACRS waive the review of the draft proposed rule until after reconciliation of public comments.

The draft proposed rule language has been heavily criticized by external stakeholders. Industry representatives have suggested that promulgation of this proposed rule will have minimal impact and will not alleviate the need for plant-specific fire protection exemptions which have been identified as a primary contributor to the inconsistent application of fire protection standards and requirements. Public stakeholders have stated that the proposed rule to allow for the use of manual operator actions manual operator actions in lieu of passive barriers and equipment separation will result in a notable reduction in plant fire safety and may also have an adverse affect on the plants ability to reach hot shutdown during certain security events.

During its April 24, 2004 meeting, the Fire Protection Subcommittee discussed the proposed rule language and the staff's rulemaking plan. External stakeholders criticized the proposed rule. According to the staff, no significant changes have been made to the rule language to address the stakeholder concerns. According to the staff's schedule, the draft proposed rule should be submitted to the Commission in December 2004. If approved by the Commission, the draft proposed rule will be released for public review and comment.

The Planning and Procedures Subcommittee suggests that Mr. Rosen recommend a course of action on this matter during the October 2004 Planning and Procedures Subcommittee meeting. Mr. Rosen recommends holding a Fire Protection Subcommittee meeting on October 27, 2004. Subsequent to the meeting, he will prepare an ACRS letter for discussion at the November 2004 ACRS meeting.

3 Draft Final Regulatory Guide DG-1085, "Standard Format and Content Guide of Decommissioning Cost Estimates for Nuclear Power Plants," and NUREG-1713, "Standard Review Plan for Decommissioning Cost Estimates for Nuclear Power Reactors" (Open)

Member: Thomas Kress **Engineer:** Michael Snodderly

Estimated Time:

Purpose: Determine a Course of Action

Priority: High

Requested by: NRR Larry Pittiglio, NRR/DRIP

The Office of Nuclear Reactor Regulation is providing the ACRS and ACNW the opportunity to review draft final Regulatory Guide DG-1085, "Standard Format and Content Guide of Decommissioning Cost Estimates for Nuclear Power Plants," and NUREG-1713, "Standard Review Plan for Decommissioning Cost Estimates for Nuclear Power Reactors." DG-1085 and the associated Standard Review Plan provide guidance on performing decommissioning cost estimates as set forth in 10 CFR Parts 2, 50, and 51. As stated in an October 15, 2001 Larkinsgram, the Committee considered these two documents and decided not to review them. The Committee had no objection to issuing them for public comment. The Project Manager stated in an October 5, 2004 e-mail that the public comments were minor/editorial in nature. He said that the regulation requires cost estimates to be submitted at different time periods and the revisions clarified the required costs estimates.

Dr. Kress recommends that the Committee not review this matter and that this matter be referred to the ACNW for consideration.

4 Vermont Yankee - Request from State of Vermont (Open)

Member: Graham Wallis **Engineer:** Ralph Caruso

Estimated Time:

Purpose: Determine a Course of Action

Priority: High

Requested by: NRR Rick Ennis

The State of Vermont has sent a letter to the ACRS asking it
(1) to specifically consider the technical issue of Containment overpressure, and determine whether the staff decision to allow overpressure credit is acceptable,
(2) to allow the State to present its views on the use of containment overpressure at the Committee meetings to discuss the uprate request,
(3) to hold one or both of the Sub- and Full Committee meetings regarding the uprate in the vicinity of the plant.

The Committee needs to discuss this request, and determine how it will respond.

5 **TRACE Computer code - Anonymous Letters** (Open)

Member: Victor Ransom **Engineer:** Ralph Caruso

Estimated Time:

Purpose: Determine a Course of Action

Priority: Medium

Requested by: RES J. Rosenthal

Drs. Wallis and Rnasome have each recevied a copy of an anonymous letter, alleging inconsistencies in the development of the TRACE T/H code and the review of T/H codes in general by the ACRS. ACRS needs to consider what to do with this letter. A similar anonymous email that was recevied by Dr. Wallis in 2003 was forwarded to the EDO for his information/action.

6 **Seabrook Power Uprate** (Open)

Member: Graham Wallis **Engineer:** Ralph Caruso

Estimated Time:

Purpose: Review & Comment

Priority: High

Requested by: NRR L. McMurtrey

Seabrook has submitted a request for a 5.2% power uprate. NRR staff has requested a waiver of ACRS review of this application, IAW the NRR-ACRS MOU on power uprates. This request is similar to one that was received, and approved by ACRS, in 2003, regarding the Kewaunee power uprate. Plant modifications needed to support the Seabrook uprate are limited to steam turbine changes. ACRS needs to determine whether to accept the waiver request.

January 2005

ACRS/ACNW CALENDAR YEAR

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
						1
2	3 NO ACRS/ACNW MEETINGS	4	5	6	7	8
9 Payperiod 3	10	11	12	13	14	15
16	17 Martin Luther King Holiday	18	19	20 Inaugaration Day	21	22
23 Payperiod 4	24	25	26	27 Retreat	28 Retreat	29
30	31					

February 2005

ACRS/ACNW CALENDAR YEAR

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
		1	2	3	4	5
6 Payperiod 5	7	8	9 NWTRB (Las Vegas)	10 519th ACRS Meeting	11	12
13	14 Health Physics Society (New Orleans)	15	16	17	18	19
20 Payperiod 6	21 Presidents Holiday	22	23 156th ACNW Meeting	24	25	26
27	28					

March 2005

ACRS/ACNW CALENDAR YEAR

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
		1	2	3 520th ACRS Meeting	4	5
6 Payperiod 7	7	8 Regulatory Information Conference	9	10	11	12
13	14	15 157th ACNW Meeting	16	17	18	19
20 Payperiod 8	21	22	23	24	25 Good Friday	26
27 Easter Sunday	28	29	30	31		

April 2005

ACRS/ACNW CALENDAR YEAR

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
					1	2
3 Payperiod 9	4	5	6	7 521st ACRS Meeting	8	9
10	11	12	13	14	15	16
17 Payperiod 10	18	19 158th ACNW Meeting	20	21	22	23
24	25	26	27	28	29	30

20

0/7/2004

May 2005

ACRS/ACNW CALENDAR YEAR

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
1 Passover (ends) Payperiod 11 ACNW trip to Japan (tentative)	2	3	4	5 522nd ACRS Meeting	6	7
8	9	10 NWTRB (DC)	11	12	13	14
15 Payperiod 12	16	17 159th ACNW Meeting	18	19	20	21
22	23 AGU	24 Passover (begins)	25	26	27	28
29 Payperiod 13	30 Memorial Day Holiday	31				

21

0/7/2004

June 2005

ACRS/ACNW CALENDAR YEAR

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
			1 523rd ACRS Meeting	**2**	**3**	**4**
5 ANS Summer Meeting (San Diego, CA)	**6**	**7**	**8**	**9**	**10**	**11**
12 Payperiod 14	**13**	**14** 160th ACNW Meeting	**15**	**16**	**17**	**18**
19	**20**	**21**	**22**	**23**	**24**	**25**
26 Payperiod 15	**27**	**28**	**29**	**30**		

July 2005

ACRS/ACNW CALENDAR YEAR

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
					1	2
3	4 Independence Day Holiday	5	6 524th ACRS Meeting	7	8	9
10 Payperiod 16 Health Physics Society (Spokane, WA)	11	12	13	14	15	16
17	18	19 161st ACNW Meeting	20	21	22	23
24 Payperiod 17	25	26	27	28	29	30
31						

23

August 2005

ACRS/ACNW CALENDAR YEAR

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
	1 NO ACRS/ACNW MEETINGS	2	3	4	5	6
7 Payperiod 18	8	9	10	11	12	13
14	15	16	17	18	19	20
21 Payperiod 19	22	23	24	25	26	27
28	29	30	31			

September 2005

ACRS/ACNW CALENDAR YEAR

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
				1	2	3
4 Payperiod 20	5 Labor Day Holiday	6	7	8 525th ACRS Meeting	9	10
11	12	13	14	15	16	17
18 Sukkot (begins) Payperiod 21	19	20 162nd ACNW Meeting NWTRB (Nevada)	21	22	23	24
25 Intl HLW Mgmt (Las Vegas, NV)	26	27	28	29	30	

October 2005

ACRS/ACNW CALENDAR YEAR

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
						1
2 Payperiod 22	3	4 Rosh Hashanah	5	6 526th ACRS Meeting	7	8
9	10 Columbus Day Holiday	11	12	13 Yom Kippur	14	15
16 Payperiod 23	17	18 Sukkot (begins) 163rd ACNW Meeting	19	20	21	22
23	24 Sukkot (ends)	25	26	27	28	29
30 Payperiod 24	31					

November 2005

ACRS/ACNW CALENDAR YEAR

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
		1	2	3 527th ACRS Meeting	4	5
6	7	8	9	10	11 Veterans Day Holiday	12
13 Payperiod 25 ANS Winter Meeting (DC)	14	15	16	17	18	19
20	21	22	23	24 Thanksgiving Day Holiday	25	26
27 Payperiod 26	28	29	30			

27

10/7/2004

December 2005

ACRS/ACNW CALENDAR YEAR

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
				1	2	3
4	5 AGU	6	7	8 528th ACRS Meeting	9	10
11 Payperiod 1	12	13 164th ACNW Meeting	14	15	16	17
18	19	20	21	22	23	24
25 Chanukah (begins) Payperiod 2	26 Christmas Day Holiday [observed]	27	28	29	30	31 chanukah (ends)

From: John Larkins
To: Dana A Powers
Date: 9/16/04 9:47AM
Subject: Re: Topics for Consideration at the ACRS Retreat

Good list of potential topics, hopefully it will stimulate some discussion.

>>> "Powers, Dana A" <dapower@sandia.gov> 09/15/04 07:13PM >>>
John,

The first priority for any ACRS retreat is, of course, for the incoming Chairman to set his agenda and the way he wants to progress for the year. Any other topic has to be deferred to set this new agenda. If there is time to address issues other than the incoming chairman's agenda, I would suggest the following:

1. We see frustration among several members (notably Wallis, Ford, and Apostolakis) on the quality of science and engineering that goes into regulations to be sure, but more importantly into the regulatory process. We continuously defer documenting objections that are not killers in favor of meeting the staff's milestones for completing an action. Or, we avoid raising the issue for fear of contaminating an approval of some licensee's application for fear our concerns over the level of science will somehow stain an ACRS approval. This leaves the concern unvoiced and presumably unknown to those outside the meeting. Should ACRS continue this way? Recall the Fermi I crisis that eventually led to making the ACRS a statutory body was precipitated, in part, by ACRS concern over the quality of science available to assess the safety of an LMFBR.

2. We have lost a lead member for the review of license renewal applications. We need to rethink our strategy. We could, of course, identify an alternative to take Leach's position and proceed as we have done for the last 4 years. An alternative approach is to assign each member a license renewal application. Other approaches may be imagined.

3. Staff is being pressed to revise 10CFR50.46 and the whole idea of design basis accidents. Is a design basis accident a useful concept for future reactors? Should the concept be abandoned in favor of an examination of risk? Risk as now practiced in the regulatory process does not address essential elements of defense in depth - notably the integrity of barriers and the capabilities of engineered safety systems. How would a risk-centric system be modified to address these issues and how must risk analyses be expanded? Would not abandoning the design basis concept lead to technology neutral regulations?

4. We take on a lot of activities based on staff requests and deadlines even though a majority of members are quite disinterested in the subject and contribute little to the debate. Should we take on fewer issues and spend more time on each?

5. We have not reassessed the issues addressed by subcommittees in some time. Should we not do this to ascertain if the subcommittee should survive and if the membership is appropriate?

Dana

From: <TSKress@aol.com>
To: <SAM@nrc.gov>
Date: 9/17/04 11:00AM
Subject: Suggested Retreat Items

Below are some Items I would suggest for the Retreat.

I Would like to keep a significant technical component to the retreat:

1. Why do we think re-defining the LBLOCA break size is OK?

2. What should be the role (if any) of design basis accidents in future regulations?

3. What will it take for ACRS to disapprove a license extension?

4. We need to discuss our position on a technology neutral regulatory framework:
   - risk acceptance criteria
   - appropriate FC curves
   - defense-in-depth -- in particular question of whether a containment is always required.

5. Any preliminary concerns about design certification of ACR700?

6. It may be too late -- but what should ACRS recommend on sump blockage?

7. We need to start thinking about an agenda for Quadripartite meeting -- I suggest we make it a focussed meeting -- perhaps on risk-informed regulation -- or divergence a la Dana Powers/Nourbakhsh.

8. What more should we do about security issues -- particularly with respect to spent fuel?
   - pool
   - dry cask storage
   - transportation
   - interim storage facility

9. Shouldn't we do some more complaining (whining?) about air oxidation behavior?

10. Power uprates -- is there any limit other than component capacities? Is there not a risk limit?

11. Organizational factors (Safety culture).

Cheers,
Tom Kress

GBW

Sherry,

We already have a lot of suggestions for topics to discuss at a possible retreat.

I suggest we revisit the question of interactions with the staff outside the formal subcommittee and ACRS meetings.

It is often useful to touch base with the staff one-on-one or sometimes two-on-two, for short discussions, clarifications, ideas for resolving problems or uncertainties, explanations of technical matters, knowing what issues are on the way etc. Some of this occurs via e-mail. We have worked hard at creating an atmosphere in which the staff trusts us and feels free to open up to us and not be defensive. At the same time we have tried to avoid anything that looked like an unofficial subcommittee meeting; indeed some of us maintain that such a thing would be inappropriate, if not illegal.

Ralph arranged an informal meeting with the LANL folks last week. While the discussion was mostly between two of us and was at times more frank than one would expect in a public meeting, there was an unusual, almost entirely silent, audience of a dozen or so people from NRR and RES. Now we are planning another informal meeting next week which I understand many of the T/H subcommittee will attend as well as quite a few staff. This is looking much like an unofficial Subcommittee meeting. I suppose that it is partly justified by the Commission's instruction that we "work with the staff" to resolve the sump blockage problem.

What are the limits to "working with the staff"? I think that in the past we have avoided:

* acting like members of the staff, or its consultants, in developing solutions to technical problems,

* stepping into management gaps and trying to organize the staff or set objectives,

* planning activities or strategies,

* doing joint brainstorming or analysis of alternative approaches to issues,

* being peer reviewers.

We often develop definite technical positions, but they are ours, not some joint product with the staff. One approach we could take is to only discuss these at formal meetings or through formal communications, so that any criticism or disagreement is entirely out in the open, perhaps even coming as a surprise to the staff. The other extreme is to work out anything that appears at all potentially disagreeable in private and have more apparent consensus in public.

If we let the Commission push us into "working with the staff" we risk becoming more like members of that staff, not an independent advisory committee. While it is appropriate for the Commission to ask us to help solve key problems, I doubt if we should be working in the trenches to do so, perhaps becoming subject to some of the constraints and influences that the staff experiences.

G.

From: <JDSIEBER@aol.com>
To: <Graham.B.Wallis@Dartmouth.EDU>, <SAM@nrc.gov>
Date: 9/18/04 4:20PM
Subject: Re: Suggested Retreat Items

I agree with Tom's and Graham's list of topics for the Retreat. If we put a little more thought into it, we could come up with something worthwhile.

As for places, I advise against going to the Northeast or Northwest in the winter. I think a warm place would be better. Perhaps, now is the time to visit Yucca Mountain, or Phoenix or Florida, whatever is left of it. San Diego is also OK.
Jack

CC: <JTL@nrc.gov>, <apostola@mit.edu>, <mvbonaca@snet.net>, <TSKress@aol.com>, <GMLeitch@aol.com>, <dapower@sandia.gov>, <wjshack@anl.gov>, <FPCTFord@aol.com>, <HistoryArt2004@aol.com>, <ransom@ecn.purdue.edu>, <rxc@nrc.gov>

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From: <HistoryArt2004@aol.com>
To: <jtl@nrc.gov>
Date: 9/17/04 5:47PM
Subject: RETREAT ITEMS

Topics
Safety Culture
Second (and Third?...) License Extensions???

Location
Anyplace except Baltimore
St. Louis
New Orleans

CC: <sxd1@nrc.gov>, <SAM@nrc.gov>

From: Victor Ransom <ransom@ecn.purdue.edu>
To: "Sherry Meador" <SAM@nrc.gov>
Date: 9/15/04 10:40PM
Subject: Re: Fwd: Retreat Items

Sherry/John,
A couple of rather general items:

Does the ACRS have a proactive role in nuclear safety issues? If so how does it originate?

Can the ACRS better influence NRC safety research? i.e. topics and quality.

Could the report writing be conducted so that a draft is available before the review starts? (this is done sometimes now)
Perhaps the review could begin by going around the table and hearing the main point of view of each member.
Then let the shredding begin!

Places to hold the retreat:
* Salt Lake City, UT
* Missoula, MT
* Santa Barbara, CA
* San Diego, CA
* Dartmouth, NH
* Cambridge, MA (MIT)

On the meeting format, I would like to see a morning meeting, afternoon off until 5:00 pm, then meet for social hour and dinner meeting until 9:00 pm. An activfty could be planned for the afternoon if desired, i.e. the French format.

At 10:36 AM 9/15/2004, you wrote:
>I misspelled your email address on the original e-mail, please see the
>attached message.

>Sherry
>
>
>Date: Wed, 15 Sep 2004 10:57:21 -0400
>From: "Sherry Meador" <SAM@nrc.gov>
>To: <Wjshack@anl.gov>,<Fpctford@aol.com>, <historyart2004@aol.com>,
>   <JDSIEBER@aol.com>, <tskress@aol.com>,
>   <Graham.b.wallis@dartmouth.edu>, <ransom@ecn-purdue.edu>,
>   <apostola@mit.edu>, <dapower@sandia.gov>, <Mvbonaca@snet.net>
>Cc: "Jenny Gallo" <JMG@nrc.gov>, "John Larkins" <JTL@nrc.gov>
>Subject: Retreat Items
>Mime-Version: 1.0
>Content-Type: text/plain; charset=US-ASCII
>Content-Disposition: inline
>
>Greetings.
>
>This is a reminder to e-mail Dr. Larkins with suggested retreat items for
>discussion and places to hold the retreat.

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September 17, 2004

Dr. Mario V. Bonaca, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

RE: State of Vermont Request to Consider the
Containment Overpressure Credit Policy

Dear Chairman Bonaca:

On behalf of the State of Vermont, by this letter I am requesting that the ACRS ("the Committee") specifically review, as part of Entergy's request for extended power uprate for the Vermont Yankee nuclear plant, Entergy's request to change Vermont Yankee's design basis to take credit for containment overpressure to demonstrate the adequacy of its emergency core cooling and containment spray pumps, and the NRC staff's ("the staff") policy of granting such requests. The reasons for the State of Vermont request are presented below.

We have reviewed Entergy's request for extended power uprate for Vermont Yankee. We question the prudence of removing the safety margin for the adequate functioning of emergency core cooling and containment cooling pumps provided by containment overpressure. That safety margin was established with good basis in the first Regulatory Guide, then Safety Guide 1.

From December 2003, we have conducted correspondence with the staff regarding the proposed removal of this important safety margin, but we have not yet received satisfactory answers to our questions, nor have we seen a convincing reason why such overpressure credit should be granted. Therefore, on August 30, 2004, we petitioned NRC for a hearing on the issue as it relates to Vermont Yankee. The technical contentions from our petition are included as Attachment A. Our letter to the staff of December 8, 2003, and the staff's answer on June 29, 2004 are provided as Attachment B and C, respectively.

At an early Vermont Yankee power uprate meeting at NRC headquarters, NRC staff asserted that the basis for considering containment overpressure credit was Regulatory Guide 1.82, Rev.

3 (then Draft Regulatory Guide 1107), which, in a very small portion, addresses the containment overpressure credit. I will now recount the history leading up to the overpressure credit statements in Regulatory Guide 1.82, Rev. 3. For most of this history, the NRC, industry and the Committee were adamant about retaining the defense-in-depth safety margin provided by not linking emergency core and containment cooling functions with containment performance. In the late-1990's, overpressure credit began to be granted selectively for a few cases based on need related to strainer debris loading. And then, with the advent of extended power uprates, it appears the NRC began granting overpressure credit whenever an Applicant asked for such credit.

- From the early 1970's, and before, until the mid-1990's, the principle expressed in Regulatory Guide 1.1 of not allowing containment overpressure credit for net positive suction head (NPSH) calculations seems to have been upheld.

- Vermont Yankee's design basis does not allow containment overpressure credit for demonstration of adequate NPSH.

- In the mid-1990's, as a result of ECCS suction strainer fouling at a number of boiling water reactors (BWRs), regulatory actions were taken to cause BWRs to modify suction strainers and recalculate the adequacy of NPSH for emergency core cooling and containment spray pumps. The Committee was highly involved in this review and Regulatory Guide 1.82, Rev. 2 (May 1996), was one of the regulatory products of this review[1].

- In 1996, all parties (industry, NRC staff, ACRS) agreed that overpressure credit should not be granted for NPSH calculations. Regulatory Guide 1.82, Rev. 2, is clear regarding no containment overpressure credit. Section 2.3.3.4 states:

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[1] In its recommendation of Regulatory Guide 1.82, Rev. 2, the Committee faulted the industry and the staff: "Finally, we continue to believe that the response of the staff and the BWR licensees to this important nuclear safety issue has been unacceptably slow." ACRS Letter of February 26, 1996, *Proposed Final NRC Bulletin 96-XX, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors" and an associated Draft Revision 2 of Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling following a Loss-of-Coolant Accident."*

> The NPSH available to the ECC pumps should be determined using conditions specified in the plant's licensing basis (e.g., Regulatory Guide 1.1).

The BWR strainer regulatory actions in the mid- and late-1990's are summarized by Los Alamos National Laboratory (LANL) in the NRC sponsored research paper, "*BWR ECCS Strainer Blockage Issue: Summary of Research and Resolution Actions,*" LA-UR-01-1595, March 21, 2001. The following are quotations from the LANL report regarding containment overpressure credit and uncertainty:

> Note that the BWROG does not recommend crediting containment overpressure in calculating NPSH margins. (page 4-2).
>
> The staff concurs that additional containment overpressure (other than an amount already approved by the staff for the existing licensing basis) should not be used as part of the resolution of this issue. (page 4-3).
>
> Some licensees discovered that they must take new credit for containment overpressure to meet the NPSH requirements of the ECCS and containment heat removal pumps and the overpressure being credited by licensees may be inconsistent with the plant's respective licensing basis. The staff further evaluated its position on use of containment overpressure in calculating NPSH margin and recommended that licensing basis changes not be used as a resolution option due to the substantial uncertainty associated with determining NPSH margin. (page x).
>
> The staff also noted that a good practice would be to maintain defense-in-depth because of the uncertainties associated with any resolution of this issue. (page 4-4).
>
> The Advisory Committee on Reactor Safeguards (ACRS) agreed with the BWROG. In a letter from the ACRS to the NRC Executive Director for Operations (EDO) explicitly stated, "We believe that allowing some level of containment

> overpressure is not an acceptable corrective action because adequate overpressure may not be present when needed." (page 1-14).
>
> However, due to incomplete guidance and inadequate supporting documentation or analysis in several areas, the staff was unable to determine if all of the methodologies, or combination of methodologies, were conservative. Similarly, much of the general guidance on "resolution options" also lacked sufficient detail for the staff to review. Since the staff lacked sufficient detail and supporting justification on many of the "resolution options," these were generally considered unacceptable without further supporting justification from a licensee or the BWROG. (page 4-2).

- Despite the 1996 determinations of industry, staff and the Committee, numbers of BWRs (but not Vermont Yankee) needed to rely on containment overpressure credit to demonstrate the adequacy of existing designs with revised ECCS strainer loadings. Accordingly, the NRC staff began granting this credit. The Committee concluded in its December 12, 1997 letter, *Credit for Containment Overpressure to Provide Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps*:

  > As a result of further review of this issue, we now concur with the NRC staff position that *selectively* granting credit for small amounts of overpressure for *a few cases* may be justified. (Emphasis added).

  This letter went on to express concern over the completeness of the staff's consideration for granting this credit and its assessment of risk probabilities.

- During the intervening eight years (between Rev. 2 and Rev. 3 of Regulatory Guide 1.82), the staff has granted overpressure credit to a number of plants and the expectation exists that PWRs will require containment overpressure credit as part of resolution of the pending PWR sump/strainer issues. Regulatory Guide 1.82, Rev. 3, as approved by the Committee, contains the following:

> **2.1.1.1** ECC and containment heat removal systems should be designed so that adequate available NPSH is provided to the system pumps, assuming the maximum expected temperature of the pumped fluid and no increase in containment pressure from that present prior to the postulated LOCAs. (See Regulatory Position 2.1.1.2.)
>
> **2.1.1.2** For certain operating BWRs for which the design *cannot be practicably altered*, conformance with Regulatory Position 2.1.1.1 may not be possible. In these cases, no additional containment pressure should be included in the determination of available NPSH than is necessary to preclude pump cavitation. [Emphasis added.]

In addition, the introductory portion of Regulatory Guide 1.82, Rev. 3, contains the following statement, at 8:

> Predicted performance of the emergency core cooling and the containment heat removal pumps *should be* independent of the calculated increases in containment pressure caused by postulated LOCAs in order to ensure reliable operation under a variety of possible accident conditions. . . However, for some operating reactors, credit for containment accident pressure *may be necessary*. This should be minimized to the extent possible. [Emphasis added.]

- Also during the intervening eight years, the staff has considered and granted BWR extended power uprate amendments which also included grants of containment overpressure credit[2]. For NPSH adequacy, it appears the staff's willingness to grant selective overpressure credit for sump/strainer loading resolution, has been changed into a general grant of overpressure credit for power uprate. It appears it has now become standard policy to grant containment overpressure credit to extended power uprate applicants whenever asked.

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[2] Extended power uprate creates more energy transfer during a postulated LOCA that results in higher sump (or torus) temperatures, in turn resulting in reduced available NPSH for ECCS and containment heat removal pumps.

Considering the history stated above, the State of Vermont believes the following:

1. We believe the staff has made a major policy change in reducing defense-in-depth by granting containment overpressure credit whenever an Applicant asked for such credit. Although we may not have seen all the related Committee documentation, we do not believe that ACRS has thoroughly reviewed and recommended this major policy change.

2. Considerable uncertainty continues to exists with regard to determination of BWR pump NPSH adequacy. These uncertainties include non-conservative assumptions in NPSH calculations, reliance on testing which may not conservatively reflect actual conditions, uncertainty and lack of margin in NPSH-required determinations, and non-consideration of chemical effects. Because of the magnitude, importance and significance of these uncertainties we believe that overpressure should be retained as a safety margin rather than used as a credit for NPSH adequacy[3].

3. We agree with the appropriate principle of extended power uprate - using certain safety margins, established many years ago, that are now understood to be excessive through experience or more exact calculations, to permit higher power levels. However, Vermont Yankee's overpressure credit request does not conform to this principle. Vermont Yankee's extended power uprate proposal uses all the available safety margin in NPSH, and then inappropriately seeks to change its design basis to use a properly reserved safety margin in order to achieve the increased power level.

4. For power uprate applications, the staff is not following its own guidance in Regulatory Guide 1.82, Rev. 3[4]. The guidance allows containment overpressure credit when necessary and when the design cannot be practicably altered. However, it appears the

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[3] We are aware of the ACRS interest in NRC staff independent verification of licensee assessments. Because of the importance of this NPSH issue, we asked the NRC staff by letter of June 9, 2004 to independently assess Vermont Yankee's overpressure credit request for LOCA, SBO, ATWS and Appendix R fire events. While the staff has not responded to our letter, it appears from RAI's that the staff is independently verifying the LOCA calculations.

[4] Vice Chairman Wallis queried the staff on this point during the Thermal Hydraulics Subcommittee consideration of Duane Arnold's extended power uprate, September 2001 - See Attachment D to this letter. We are not aware of additional ACRS attention to this matter.

> staff now grants containment overpressure credit whenever asked. For Vermont Yankee, containment overpressure credit is not necessary, because the power uprate is not necessary. Also, the Vermont Yankee's design can be practicably altered such that NPSH requirements are met without the need for overpressure credit[5].

It appears the NRC staff is indiscriminately granting containment overpressure credit for extended power uprate, contrary to the policy guidance regarding *need* and *practicable alteration* that are in the Regulatory Guide 1.82, Rev. 3 recommended by the Committee. This appears to be a significant deviation from the Committee's 1997 concurrence that containment overpressure credit be granted *selectively, in a few cases.* Vermont therefore considers this major policy change to be ripe for review and requests the following:

- If the Committee has provided recommendations on the use of containment overpressure credit different from the recommendation in its December 12, 1997 letter quoted above, and different from the guidance in Regulatory Guide 1.82, Rev. 2, we would appreciate it if the documentation of those recommendations would be identified for us.

- We request that the major policy change of granting overpressure credit, described above, be considered by the Committee in its deliberations regarding Vermont Yankee's extended power uprate. We believe the following questions should be considered in reviewing the issue:

  - Should the defense-in-depth provided by unlinked fission product barriers (the containment function and core cooling function) be surrendered (by linking the core cooling function to containment integrity) when there is no need to do so[6] and when the design can be practicably altered[7] to avoid this linkage?

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[5] The backfit rule, 10 C.F.R. §50.109, does not apply because the request for extended power uprate is a voluntary change.

[6] There is no need to change the overpressure credit design basis because it is not necessary for Vermont Yankee to increase its power level.

[7] We believe it is practicable for Vermont Yankee to modify its emergency core cooling system and containment cooling system such that overpressure credit is not required. A possible modification would be to install pumps with different NPSH requirements.

- Is there sufficient continued uncertainty[8] in the sum of:

  1) event calculations that develop the heat loading for torus water,
  2) calculations determining debris loading and strainer head loss,
  3) test results that may not conservatively reflect actual conditions,
  4) uncertainty in NPSH-required values, and
  5) uncertainty resulting from the adverse trend in as-found leakage in Vermont Yankee containment isolation valve leakage rate tests;

  such that the margin provided by containment overpressure should not be surrendered when there is no need to do so and when the design can be practicably altered to avoid this crediting?

- Is risk evaluation methodology sufficiently developed to account for uncertainties in granting overpressure credit, including those identified above, and are the results sufficiently certain, accurate, and reliable to justify using the margin provided by containment overpressure when there is no need to do so and when the design can be practicably altered to avoid this usage?

- Does the post-event, human factors confusion for operators, who would now have to both retain and reduce containment pressure after thirty-two years of training to reduce pressure, merit the containment pressure reliance when there is no need to do so and when the design can be practicably altered to avoid this confusion?

- If this major policy change regarding containment overpressure credit is recommended by the Committee, and if Vermont Yankee's change in design basis is considered, should the precedence created in Section 5.1.4 of Regulatory Guide 1.183[9], regarding the application of current licensing

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[8] The calculation uncertainty referred to here includes uncertainties in the methodology accepted by the staff and Committee for such calculations, in the application of this methodology by the Applicant, and in the assumptions and initial conditions chosen by the Applicant.

[9] Regulatory Guide 1.183, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*, July 2000:

5.1.4 Applicability of Prior Licensing Basis - The NRC staff considers the

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standards, be extended to the grant of extended power uprate, design basis changes for overpressure credit? This method establishes a precedence for evaluating voluntary, major changes of design bases according to current licensing standards.

- If the major policy change is recommended by the Committee that containment overpressure credit should be granted regardless of need or ability to alter design, should there be limits on the percentage or amount of overpressure credited? Is risk methodology sufficiently developed to allow a risk-informed decision on the percentage or amount of overpressure credit to allow? Does the Vermont Yankee request for design basis change fit within these limits?

- We request that Vermont be allowed to present on these issues before the Sub- and Full-Committees, along with the Applicant and staff, in the Committees' deliberations for Vermont Yankee extended power uprate.
- We request that one or both of the Sub- and Full-Committee meetings regarding Vermont Yankee extended power uprate be held in the vicinity of the nuclear plant.

We are aware the staff has granted containment overpressure credit for extended power uprate to other Applicants, and therefore will be reluctant to retreat from this policy. However, Vermont does not believe the implications of this major policy change, as represented by the questions above, have been fully considered. We would greatly appreciate if ACRS

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implementation of an AST to be a significant change to the design basis of the facility that is voluntarily initiated by the licensee. In order to issue a license amendment authorizing the use of an AST and the TEDE dose criteria, the NRC staff must make a current finding of compliance with regulations applicable to the amendment. The characteristics of the ASTs and the revised dose calculational methodology may be incompatible with many of the analysis assumptions and methods currently reflected in the facility's design basis analyses. The NRC staff may find that new or unreviewed issues are created by a particular site-specific implementation of the AST, warranting review of staff positions approved subsequent to the initial issuance of the license. This is not considered a backfit as defined by 10 CFR 50.109, "Backfitting." However, prior design bases that are unrelated to the use of the AST, or are unaffected by the AST, may continue as the facility's design basis.

considerations for the Vermont Yankee extended power uprate could document the Committee's recommendations regarding these questions.

We appreciate your consideration of our request. Please call me if you have questions.

Sincerely,

David O'Brien, Commissioner
NRC State Liaison Officer for Vermont

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From: "Paul Blanch" <pmblanch@comcast.net>
To: <jtl@nrc.gov>
Date: 9/21/04 8:43AM
Subject: FW: VY uprate controversy continues to simmer

Dr. Larkins:

I believe the ACRS members may be interested in the following discussion about a letter from the Commission to Vermont's Congressional delegation.

Paul M. Blanch

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West Hartford, CT 06117

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Office 860-236-0326

-----Original Message-----
From: Arnie Gundersen [mailto:arniegundersen@sailchamplain.net]
Sent: Tuesday, September 21, 2004 5:56 AM
To: Shadis Raymond; Peter Alexander; Paul Blanch; Lochbaum David; Gundersen Margaret
Subject: VY uprate controversy continues to simmer

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View Brattleboro Reformer

VY uprate controversy continues to simmer

By CAROLYN LORI

Reformer Staff

Tuesday, September 21, 2004 - BRATTLEBORO -- Controversy continues to swirl around Entergy Nuclear Vermont Yankee's uprate application.

Officials at the 32-year-old plant applied to the Nuclear Regulatory Commission last year to increase power generation by 20 percent. This is known as an extended power uprate and is the most allowed in the industry. Vermont Yankee is the oldest plant to have requested a 20 percent power boost.

On June 28, nuclear industry experts Paul Blanch and Arnold Gundersen wrote a six-page letter to the Vermont congressional delegation and the chairman of the Vermont Public Service Board, outlining their concerns with the application, as well as with the NRC's review process.

"Put in its simplest terms, we are convinced that the proposed uprate will make Vermont Yankee significantly less safe than it is today, and we are also convinced that the NRC has turned a deaf ear on the irrefutable facts that support this powerful statement," wrote Blanch and Gundersen.

In their letter, Gundersen and Blanch contend that safety margins at Vermont Yankee will be greatly reduced if the uprate occurs. The two also charged that the NRC has refused to show how the plant conforms to or deviates from current safety and design basis regulations.

The two asked that the delegation and PSB Chairman Michael Dworkin use their authority to demand answers from the NRC.

Earlier this month, Sens. Patrick Leahy, D-Vt., and James Jeffords, I-Vt., and Congressman Bernard Sanders, I-Vt., responded to the letter, saying that they had "asked for an official response to [the] letter from the agency that addresses each of the safety concerns that [were] raised."

According to Neil Sheehan, NRC spokesman for Region I, the official response came on Sept. 14.

Luis Reyes, NRC executive director of operations, wrote to the delegation stating that the issues raised by Blanch and Gundersen had already been raised by petitions to intervene filed by the state and the nuclear power watchdog group, the New England Coalition. The NRC has not responded formally to either party.

Sheehan said the letter explained that answers would be provided when the petitions are addressed.

"Until then it would be premature to comment on the issues raised," closed the letter.

Upon hearing about the NRC's response, Blanch accused the agency of "leading us around in circles."

One of the central safety issues in question has to do with containment overpressure. Under uprated conditions, the water in the containment tank will be warmer which will allow bubbles to form. If a loss of coolant accident were to occur, the bubbles in the water would interfere with the emergency pumps' functioning, possibly destroying them over time. Without the pumps, the water necessary to keep the core cool could not circulate, resulting in a meltdown.

According to Vermont Yankee engineers, during the type of loss of coolant accident postulated in the above scenario, there will be sufficient pressure in the tank to prevent the bubbles from forming. This is known as taking credit for overpressure.

Although NRC regulatory guides state that credit should be taken only when necessary and minimized to the extent possible, many plants have been allowed to do this in order to increase power generation.

Both the coalition and the state have included this issue in their petitions.

The state has gone one step further, however, calling on the Advisory Committee on Reactor Safeguards to "specifically review" this point.

The ACRS is an 11-member panel that is appointed by the NRC, but operates independently. All uprate applications go before the panel, which makes a recommendation to the NRC. So far all applications for increasing power generation have been approved by the committee.

In a letter addressed to Mario Bonaca, chairman of the ACRS, Vermont Department of Public Service commissioner David O'Brien questioned the NRC's decision to grant exceptions to its own regulatory guides.

"...The NRC, industry and the Committee [ACRS] were adamant about retaining the defense-in-depth safety margin provided by not linking emergency core and containment cooling functions with containment performance," O'Brien wrote. "And then, with the advent of extended power uprates, it appears the NRC began granting overpressure credit whenever an applicant asked for such credit."

The 10-page letter outlines the evolution of granting credit for overpressure and questions whether the ACRS "thoroughly reviewed and recommended this major policy change."

State nuclear engineer Bill Sherman first raised questions about the containment overpressure in December, prompting him to write a letter to the NRC about the issue. The federal regulator responded in May but not to the satisfaction of the department.

"The department has always vowed to have our safety concerns heard in the correct venue -- the NRC," said O'Brien in a press release. "Requesting the ACRS to review the containment overpressure issue in addition to requesting a hearing on the issue is a second avenue to having our questions resolved. We want to make sure that Vermont Yankee is safe if an uprate of power output is allowed."

Another issue raised in the department letter was the fact that the plant could be modified in such a way that taking credit for overpressure would be unnecessary.

When asked about the possibility of modifications being made to avoid this problem, Rob Williams, spokesman for Vermont Yankee, refused to answer, saying that he was not in a "position to speculate about that."

Williams did state, however, that the uprate application was "grounded in NRC regulations" and that ACRS review is a routine part of the regulatory review process.

"The ACRS was set up to give an independent view on safety matters that came before the NRC and we expect they will review this issue as well," said Williams.

Blanch, who has worked in the nuclear industry as an electrical engineer for 35 years, said that the department's appeal to the ACRS marked a first, as far as a state's involvement in nuclear regulatory affairs.

"It is extremely significant. I've never seen anything like this," he said.

While he expressed frustration with the NRC, Blanch was optimistic about how the ACRS might respond.

Raymond Shadis, technical advisor to the New England Coalition, was less hopeful.

"This is good. It can't be bad," he said. "But we should not raise our hope that this is a comprehensive answer."

Carolyn Lorié can be reached at clorie@reformer.com.

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ACRS/ACNW
US NRC

SEP 20 2004

Subject: The TRACE Code

Professor Wallis:

There are several issues that need to be addressed relative to the TRACE code. Generally, these issues are overlooked when NRC codes are under review. They are not however overlooked when the NRC reviews codes submitted by other organizations for review and approval. The issues are discussed following the next paragraph. The next paragraph is an aside that however needs to be addressed by the NRC and the ACRS T/H Subcommittee.

It is a very unfortunate situation in that the extreme adversarial environment that has been present at the NRC for the past thirty years or so makes this form of communication necessary. Free and open discussion of the technical issues that are truly important has not been possible for all these years. The ACRS T/H Subcommittee and its Consultants have been filled with persons who have individual agendas and who do not listen to the very people who know the most about the subject matter that is presented. Those organizations that submit codes for review have hundreds of thousands of very specialized and focused man-hours invested in their products. These people know exactly what is important for each and every application of their codes and experimental data. The ACRS and it consultants, on the other hand, generally do not have the time, or more importantly the inclination, to digest the material presented to the depth necessary to understand the important items that really matter in any application. The personal agendas of the T/H Subcommittee and its Consultants, as reflected in the ACRS transcripts, almost never are important to the practical issues of an application. Quite frankly, the T/H Subcommittee and its Consultants and the material on which they focus and the manner on which they discuss the material are the subject of many jokes and not-so-kind comments all over the industry.

The issues that are being overlooked relative to the TRACE code at this stage in its development include:

1. Independent verification of the coding.

2. A fundamental issue associated with the numerical methods used in the code.

Independent Verification

It is accepted procedure that software in computer codes must be verified before the models and methods are validated. Verification is the process of ensuring that the equations used in the code have been correctly coded. Validation is the process of ensuring that the correct equations have been chosen and coded. Verification must always precede validation, and that is the methodology applied to codes submitted to the NRC by all commercial organizations.

Generally, the computer codes developed under NRC funding have never undergone verification. Additionally, the validation procedure applied to these codes has not measured up to the standards required by the NRC for commercial organizations. Almost all the so-called "validation" or "assessment" calculations done with the NRC codes have not been done under an approved and qualified procedure with "frozen" versions of the software.

I have not seen that verification of the coding in the TRACE code is to be performed. To proceed to validation without verification invalidates the validation process. Additionally, it is not clear that the NRC has a qualified and approved Q/A plan in place for TRACE. Such plans are required of commercial organizations by the NRC.

Issues with the Numerical Methods

If the documentation for the numerical solution methods used in TRACE, both the code manuals and papers in the literature, are studied in detail the results will show that the basic SETS solution method is based on less-than-exact methodologies. Many solution orders for the equations were simply experimented with until one that "works" was discovered. While this approach is less than satisfactory from a theoretical view, it might be called an "engineering solution".

The following discussion is based on the documentation given in the (1) TRAC-P manual of NUREG/CR-5673, LA-12031-M, 1993, and (2) the draft of the manual for the first F-90 version of TRAC/M given in the LA-UR-00-910, 2000. The latter document is basically the former only re-arranged. The latter document is also, I think, a rough first cut at the TRAC part of the TRACE manual. The latter document, (2), will be cited in the following discussion, although the exact same material can be found in the former.

Generally, no multi-step method actually satisfies the original FDEs, and the many approximations used in the TRAC SETS method are somewhat out the the ordinary relative to almost all other numerical solution methods for transient compressible fluid flow. The lack of satisfying the original FDEs and the many approximations can only be appreciated after digging through the manuals in a very, very detailed study. But these are not the main issues here, however.

There is a basic problem that has never been addressed and it is kind of hard to dig out of the documentation. This basic problem is as follows. The numerical method does not solve for the void fraction in a way that can be theoretically justified. A reference to section 2.1.8.2.4 at the bottom of page 2-31 of (2) is the only clue in the manual for how the void fraction at the new-time level is obtained. Section 2.1.8.2.4 is on page 2-61 of (2), 30 pages from where it is needed. The material on page 2-61 states that a system of equations is set up to obtain the new-time level values for the void fraction and the other EOS variables. The system is based on the solution of products of void and density and void-density-energy from the mass and energy stabilizer step of the SETS method plus the EOS with pressure and temperature as independent variables. The discussion given in the manuals is correct and the system of non-linear equations can easily be discovered and the iterative Newton-Raphson method applied to the solution of the system to get the pressure, phase temperatures, and void fraction.

But, here is the basic issue. As discussed on page 2-61, the system of non-linear equations is treated as a system of un-coupled linear equations and a one-shot step, without iteration is all that is done. Note that coupled linear equations require iteration to obtain a solution. Most importantly, all the quantities determined by this one-shot rough estimate are discarded except for the void fraction. This means that none of the results from the one-shot evaluation will satisfy the equations from which they were obtained and only the void fraction from this "solution" is retained. Thus, just as in the case of the RELAP5 code, the non-linear EOS is not satisfied. Additionally one must wonder exactly what the "void fraction" "calculated" in the TRAC SETS manner actually represents.

I continue to investigate the properties of the numerical solution method used in the TRAC-P code, which is the same as in the current version of the TRACE code. The references are (1) TRAC-P manual of NUREG/CR-5673, LA-12031-M, 1993, and (2) the draft of the manual for the first F-90 version of TRAC/M given in the LA-UR-00-910, 2000. The latter document is basically the former only re-arranged. I have so far found out the following.

(1) At the top of page 2-20 of (2) the comments seem to say that the nonlinear equations for the fluid T/H are not solved for the single-phase liquid and single-phase vapor and non-condensable gas cases. That is when a single phase is in the flow system a linearized version of the basic equations is solved. Solution of the non-linear algebraic FDEs by an iterative method is not used; a one-shot through solution method is used instead. I recall that Wolfgang Wulff has written a long article about the use of linearized equations in T/H codes for safety analysis. My recollection is that he thought that this process was not fundamentally sound and that instead the fully non-linear T/H equations should be solved.

(2) I have been thinking some more about the fact that neither RELAP5 or TRAC/TRACE attempt to satisfy the non-linear EOS for the two-fluid model. The solution methods in both these codes are basically slight variations on the MAC-ICE-ALE methods developed at LANL almost three decades ago. While the two-phase equations, especially the non-conservative form of the "momentum" equations, can be made to look a lot like the momentum equation for single-phase flows for which the LANL methods were developed, there is a very significant departure from the single-phase equations. The two-fluid model

momentum equations contain the void fraction, for which there is no analogy in the LANL methods. What both RELAP5 and TRAC/TRACE apparently try to do in order to be able to use "two"- step solution methods, in contrast to fully-implicit methods, is not address completely the pressure-void coupling that is the essence of the two-fluid problem.

I further suspect that both codes have tried to solve the non-linear EOS in developmental versions of the codes but have found that that process leads to unsatisfactory performance of the numerical methods. I suspect this because both codes have apparently come to the same conclusion regarding the non-linear EOS. Other investigations of application of the MAC-ICE-ALE "two"-step methods, in which the non-linear EOS is solved, to the two-fluid model equations have found that the resulting methodology is not unconditionally stable. I do not have the software that could be used to investigate the properties and characteristics of such complex numerical methods as needed for the two-fluid model so that this hypothesis can be tested. A direct question to the developers of the codes might be the most efficient way to get a handle on how the solution methods have come to relay on non-solution of the EOS.

(3) The numerical methods used for both RELAP5 and TRAC/TRACE have been developed with the focus on the CPU time needed to complete a calculation. In REALP5 this has taken the form of solving systems of non-linear algebraic equations without iteration, in addition to not solving the non-linear EOS. In TRAC/TRACE this has taken the form of trying to avoid solution of coupled systems of equations by use of various time-levels during the several steps used in the solution. The number of different time-level values used in the SETS method is sometimes very bewildering. All of the following appear in the equations (2-70) through (2-91) on pages 2-22 through 2-31: old-time values, new-time values, tilde-level values and caret-level values. Plus, the spatial weighting factor for the flux terms, beta, takes on different values as a function of what is calculated to be happening in a given region of the flow field. Beta is a function of tilde- and old-time velocity and velocity-gradient values and phase-change rates and the formulation for beta changes during the various steps in the SETS method. Thus (1) the basic equations cannot be actually satisfied unless all the various time-level quantities attain the same numerical values, including beta; (2) beta can have different values in proximate regions of the flow; and (3) quantities at different time-levels, caret, tilde and new, are taken as the "solution" from the various steps in SETS. It seems to me that there is a high probability that convergence of the numerical equations to the continuous equations cannot be demonstrated.

I continue to investigate some properties of the TRAC-P code, which is the same as in the current version of the TRACE code. The references are (1) TRAC-P manual of NUREG/CR-5673, LA-12031-M, 1993, and (2) the draft of the manual for the first F-90 version of TRAC/M given in the LA-UR-00-910, 2000. The latter document is basically the former only re-arranged.

The authors of the TRAC codes have chosen to use the nomenclature "motion equation" for what is generally considered to be a momentum equation model. My experience is that all these systems-analysis codes generally return solutions consistent with the mechanical energy, or Bernoulli, equation.

There is an option in the TRAC codes to apply a COSINE factor to the momentum-flux term in the motion equation. The factor can be applied to the flux term at either, or both, contributions which appear in each cell edge (or link, flowpath, junction) motion equation. The factor is designed to be applied at locations at which the flowpath might split, or merge, such as at TEEs, and Ys.

This seems to be a problem as follows. If the motion equation returns the correct solution, ie returns the Bernoulli results, for one value of the COSINE factor, like the case for flows in straight pipes, it cannot return the correct solution for another value. Unless, of course, the motion equation is modified by some term. I do not see that the equation is modified in the manuals.

I suspect that if the COSINE factor is set to make the flux contribution null, pressure peaks will be present in the solution. This is the usual situation of stopping a flow so that the flux gives rise to a pressure increase.