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Vol. 24, No. 2

Abstracts for Publications in the NUREG-Series

Annual Compilation for 1999

U.S. Nuclear Regulatory Commission
Office of the Chief Information Officer
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ABSTRACT

This journal includes all publications NRC prepares in its NUREG series: reports, including those prepared for international agreements; brochures; conference proceedings; and books. The entries in this compilation are indexed for access by staff and contractor-prepared publications and NRC originating organizations.

In Vol. 23, No. 1, NUREG-0304, the title was changed from "Regulatory and Technical Reports (Abstract Index Journal)." The NRC is discontinuing publication of this journal with this issue, Vol. 24, No. 2.

PREFACE

The U.S. Nuclear Regulatory Commission (NRC) compiles bibliographic data and abstracts for publications in the NUREG-series available to the public. After the end of each calendar year, the NRC has been publishing an annual compilation of this information. However, the NRC is discontinuing publication of the compilation with this issue, which has a slightly different format than previous issues.

NRC is discontinuing this publication because, since November 1999, the public can electronically access the information previously published in this compilation. This information is contained in NRC's Agencywide Document Access and Management System (ADAMS), accessed through NRC's Web site <http://www.nrc.gov/NRC/ADAMS/index.html>. Additionally, you may access a table of publications issued in 2000 that are in ADAMS at Web address <http://www.nrc.gov/NRC/NUREGS/adamspubs.html>.

The "Bibliographic Data Sheets" for these publications are sequenced according to their NUREG-series number: publications including reports or brochures prepared by the staff designated (NUREG-XXXX) or (NUREG/BR-XXXX); conference proceedings designated (NUREG/CP-XXXX); reports prepared by an NRC contractor designated (NUREG/CR-XXXX); and publications resulting from international agreements designated (NUREG/IA-XXXX).

The four indexes for the Bibliographic Data Sheets included in this volume are titled:

- Staff-Prepared Publications
- NRC Originating Organization for Staff-Prepared Publications
- Contractor-Prepared Publications
- NRC Originating Organization for Contractor-Prepared Publications

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The four indexes are followed by a "Bibliographic Data Sheet" for each publication and the entries in the originating organization indexes will contain for each publication—

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NUREG/CR-6655, V02: SENSITIVITY AND UNCERTAINTY ANALYSES APPLIED TO CRITICALITY SAFETY VALIDATION, November 1999
NUREG/CR-6656: INFORMATION ON HYDROLOGIC CONCEPTUAL METHODS, PARAMETERS, UNCERTAINTY ANALYSIS, AND DATA SOURCES FOR DOSE ASSESSMENTS AT DECOMMISSIONING SITES, December 1999
NUREG/CR-6658: TRAC-M: PROGRAMMER'S GUIDE, December 1999

NRC FORM 335 (2-89) NRCM 1102, 3201, 3202 — U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET** *(See instructions on the reverse)*	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.) NUREG-0040 Vol. 22 No. 3
2. TITLE AND SUBTITLE Licensee Contractor and Vendor Inspection Status Report Quarterly Report July-September 1998	3. DATE REPORT PUBLISHED MONTH: March YEAR: 1999 4. FIN OR GRANT NUMBER
5. AUTHOR(S)	6. TYPE OF REPORT Quarterly 7. PERIOD COVERED *(Inclusive Dates)* July - September 1998

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Division of Inspection Program Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This periodical covers the results of inspection performed by the NRC's Quality Assurance, Vendor Inspection and Maintenance Branch, that have been distributed to the inspected organizations during the period from July through September 1998.

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1. REPORT NUMBER
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NUREG-0040
Vol. 22, No. 4

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October - December 1998

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Division of Inspection Program Management
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This periodical covers the results of inspections performed by the NRC's Quality Assurance, Vendor Inspection and Maintenance Branch, that have been distributed t the inspected organizations during the period October through December 1998.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

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BIBLIOGRAPHIC DATA SHEET *(See instructions on the reverse)*		NUREG-0040 Vol. 23 No. 1
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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

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NUREG-0040
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2. TITLE AND SUBTITLE

Licensee Contractor and Vendor Inspection
Status Report
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April - June 1999

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5. AUTHOR(S)

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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This periodical covers the results of inspections performed by the NRC's Quality Assurance, Vendor Inspection, Maintenance and Allegations Branch, that have been distributed to the inspected organizations during the period from April through June 1999.

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-0090, Vol. 21

2. TITLE AND SUBTITLE

Report to Congress on Abnormal Occurrences, Fiscal Year 1998

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May	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT

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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

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10. SUPPLEMENTARY NOTES

Harriet Karagiannis, NRC Project Manager

11. ABSTRACT *(200 words or less)*

Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence (AO) as an unscheduled incident or event that the Nuclear Regulatory Commission (NRC) determines to be significant from the standpoint of public health or safety. The Federal Reports Elimination and Sunset Act of 1995 requires that AOs be reported to Congress on an annual basis. This report includes those events that NRC has determined to be AOs during fiscal year 1998.

This report addresses six AOs. Five of these events involved NRC licensees and one involved an Agreement State licensee.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Nuclear Power Plant; Exposure; Misadministration; Gaseous Diffusion Plant.

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NUREG-0304
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2. TITLE AND SUBTITLE

Abstracts for Publications in the NUREG-Series

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10. SUPPLEMENTARY NOTES

L. L. Stevenson, Project Manager

11. ABSTRACT *(200 words or less)*

This journal includes all formal reports in the NUREG-series prepared by the NRC staff and contractors; proceedings of conferences and workshops; as well as international agreement reports. The entries in this compilation are indexed for access by title and abstract, secondary report number, personal author, subject, NRC organization for staff and international agreements, contractor, international organization, and licensed facility.

In Vol. 23, No. 1, of NUREG-0304, the title was changed from "Regulatory and Technical Reports (Abstract Index Journal)."

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

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NUREG-0304
Vol. 24, No. 1

2. TITLE AND SUBTITLE

Abstracts for Publications in the NUREG-Series
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10. SUPPLEMENTARY NOTES

L. L. Stevenson, Project Manager

11. ABSTRACT *(200 words or less)*

This journal includes all publications NRC prepares in its NUREG series: reports, including those prepared for international agreements; brochures; conference proceedings; and books. The entries in this compilation are indexed for access by staff and contractor-prepared publications and NRC originating organizations.

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12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

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2. TITLE AND SUBTITLE Directory of Certificates of Compliance for Radioactive Materials Packages Report of NRC Approved Packages	3. DATE REPORT PUBLISHED MONTH: October YEAR: 1999 4. FIN OR GRANT NUMBER
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Spent Fuel Project Office
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission and mailing address.)*

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

The purpose of this directory is to make available a convenient source of information on packagings approved by the U.S. Nuclear Regulatory Commission. To assist in identifying packaging, an index by Model Number and corresponding Certificate of Compliance Number is included at the front of Volumes 1 and 2. An alphabetical listing by user name is included in the back of Volume 3 of approved Quality Assurance programs. The reports include a listing of all users of each package design and approved Quality Assurance programs prior to the publication date.

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Directory of Certificates of Compliance for Radioactive Materials Packages

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Spent Fuel Project Office
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
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Directory of Certificates of Compliance for Radioactive Materials Packages

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Inventory Difference Data
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5. AUTHOR(S)

Thomas N. Pham

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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

NRC is committed to the periodic publication of licensed fuel cycle facility inventory difference data, following Agency review of the information and completion of any related investigations. Information in this report includes inventory difference data for active fuel fabrication facilities possessing more than one effective kilogram of special nuclear material.

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nuclear fuel facilities, inventory difference, physical inventories, special nuclear material accounting

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10. SUPPLEMENTARY NOTES

E.B. Morris, NRC Project Manager

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Juanita F. Beeson, Project Manager

11. ABSTRACT *(200 words or less)*

The guidance in this publication is for the staff and contractors who prepare manuscripts to be published in the NUREG series for the U.S. Nuclear Regulatory Commission (NRC). This revision 2 to NUREG-0650, "Publishing Documents in the NUREG Series," is retitled "Preparing NUREG-Series Publications. It gives more concise and up-to-date guidance, including certain Internet and World Wide Web addresses. It describes how to cite references to electronic information and, in addition, refers the NRC staff to online style guidance for Web site publishing.

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Division or Risk Analysis and Applications
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

*Science Applications International Corporation
301 Laboratory Road
Oak Ridge, TN 37830

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11. ABSTRACT *(200 words or less)*

This report summarizes the occupational exposure data that are maintained in the U.S. Nuclear Regulatory Commission's (NRC) Radiation Exposure Information and Reporting System (REIRS). The bulk of the information contained in the report was compiled from the 1998 annual reports submitted by six of the seven categories1 of NRC licensees subject to the reporting requirements of 10 CFR 20.2206. Since there are no geologic repositories for high level waste currently licensed, only six categories will be considered in this report. Annual reports for 1998 were received from a total of 288 NRC licensees, of which 105 were operators of nuclear power reactors in commercial operation. Compilations of the reports submitted by the 288 licensees indicated that 132,032 individuals were monitored, 65,070 of whom received a measurable dose (Table 3.1). The collective dose incurred by these individuals was 16,383 person-rem which represents a 17% decrease from the 1997 value. The number of workers receiving a measurable dose also decreased, resulting in the average measurable dose of 0.25 rem for 1998. The average measurable dose is defined to be the total collective dose (TEDE) divided by the number of workers receiving a measurable dose.2 These figures have been adjusted to account for transient reactor workers. In 1998, the annual collective dose per reactor for light water reactor licensees (LWRs) was 125 person-rem. This represents a 26% decrease from the value reported for 1997. The annual collective dose per reactor for boiling water reactors (BWRs) was 190 person-rem and, for pressurized water reactors (PWRs), it was 92 person-rem. Analyses of transient worker data indicate that 23,061 individuals completed work assignments at two or more licensees during the monitoring year. The dose distributions are adjusted each year to account for the duplicate reporting of transient workers by multiple licensees. In 1998, the average measurable dose calculated from reported data was 0.21 rem. The corrected dose distribution resulted in an average measurable dose of 0.25 rem.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

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U.S. Nuclear Regulatory Commission
Washington, DC 20555–0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

Digests and indexes for issuances of the Commission, the Atomic Safety and Licensing Board Panel, the Administrative Law Judges, the Directors' Decisions, and the Decisions on Petitions for Rulemaking are presented.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

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1. REPORT NUMBER (Assigned by NRC. Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-0800

2. TITLE AND SUBTITLE

NUREG-0800, STANDARD REVIEW PLAN, Office of Nuclear Reactor Regulation
Chapter 13, "Conduct of Operations," Section 13.1.1, "Management and Technical Support Organization," and Section 13.1.2-13.1.3, "Operating Organization"

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Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
WASHINGTON, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

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10. SUPPLEMENTARY NOTES

THIS IS A REVISION TO EXISTING GUIDANCE CONTAINED IN THE APRIL, 1996 DRAFT SRP (NUREG-0800)

11. ABSTRACT *(200 words or less)*

This is a revision to existing guidance as contained in NUREG-0800, Draft Revision 3, April, 1996. Chapter 13, "Conduct of Operations," Section 13.1.1, "Management and Technical Support Organization," and Section 13.1.2-13.1.3, "Operating Organization," were revised to include guidance for the staff to use in reviewing applications for license transfer in accordance with 10 CFR 50.80, " Transfer of Licenses."

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Human Factors Engineering
Management and Organization
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13. AVAILABILITY STATEMENT

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NUREG-0936
Vol. 17, No. 2

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Semiannual Report
July - December 1998

3. DATE REPORT PUBLISHED

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5. AUTHOR(S)

6. TYPE OF REPORT

Semiannual

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July - December 1998

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Division of Administrative Services
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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

The NRC Regulatory Agenda is a compilation of all rules on which the NRC has recently completed action, or has proposed action, or is considering action, and all petitions for rulemaking which have been received by the Commission and are pending disposition by the Commission. The Regulatory Agenda is updated and issued semiannually.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Compilation of rules
Petitions for rulemaking

13. AVAILABILITY STATEMENT

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NRC FORM 335 (2-89) NRCM 1102, 3201, 3202 — U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET** *(See instructions on the reverse)*	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.) NUREG-0936 Vol. 18, No. 1
2. TITLE AND SUBTITLE NRC Regulatory Agenda Semiannual Report January - June 1999	3. DATE REPORT PUBLISHED MONTH: August YEAR: 1999 4. FIN OR GRANT NUMBER
5. AUTHOR(S)	6. TYPE OF REPORT Semiannual 7. PERIOD COVERED *(Inclusive Dates)* January - June 1999

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

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1. REPORT NUMBER (Assigned by NRC. Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-0940, PART I
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Enforcement Actions: Significant Actions Resolved
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11. ABSTRACT *(200 words or less)*

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12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

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NUREG-0940, PART II
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Reactor Licensees
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12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

Technical Specification, Quality Assurance, Radiation Safety Program, Safety Evaluation

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14. SECURITY CLASSIFICATION

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NUREG-0940, PART III
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12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Diagnostic Radiopharmaceuticals, Teletherapy, Brachytherapy, Radiation Safety Program, Safety Evaluation, Quality Management Program, HDR

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Semiannual Progress Report
January - June 1999

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14. SECURITY CLASSIFICATION

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NUREG-0940, PART 3
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Enforcement Actions: Significant Actions Resolved
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January - June 1999

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2. TITLE AND SUBTITLE

Nuclear Regulatory Legislation
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4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT

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7. PERIOD COVERED *(Inclusive Dates)*

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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

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Washington, DC 20555-0001

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This document is a compilation of nuclear regulatory legislation and other relevant material through the 105th Congress, 2d Session. This compilation has been prepared for use as a resource document, which the NRC intends to update at the end of every Congress.

The contents of NUREG-0980 include The Atomic Energy Act of 1954, as amended; Energy Reorganization Act of 1974, as amended, Uranium Mill Tailings Radiation Control Act of 1978; Low-Level Radioactive Waste Policy Act; Nuclear Waste Policy Act of 1982; and NRC Authorization and Appropriations Acts. Other materials included are statutes and treaties on export licensing, nuclear non-proliferation, and environmental protection.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Nuclear Regulatory Legislation; Atomic Energy Act; Energy Reorganization Act; Nuclear Waste Policy Act, NRC Authorization and Appropriations Acts; Statutes, Treaties, Agreements on Export Licensing and Nuclear Non-Proliferation

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Office of The General Counsel
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Washington, DC 20555-0001

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

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2. TITLE AND SUBTITLE

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4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT

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Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

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10. SUPPLEMENTARY NOTES

Supersedes Interim Revision 8

11. ABSTRACT *(200 words or less)*

NUREG-1021 establishes the policies, procedures, and practices for examining licensees and applicants for reactor operator and senior reactor operator licenses at power reactor facilities pursuant to Title 10, Part 55, of the Code of Federal Regulations (10 CFR Part 55). It is intended to assist NRC examiners and facility licensees to better understand the processes associated with initial and requalification examinations. The standards also ensure the equitable and consistent administration of examinations for all applicants. These standards are for guidance purposes and are not a substitute for the operator licensing regulations (i.e., 10 CFR Part 55), and they are subject to revision or other changes in internal operator licensing policy.

This revision implements an amendment to 10 CFR Part 55 that allows facility licensees to prepare the entire operator licensing examination and to proctor and grade the written portion of the examination. The NRC will prepare the examinations at least four times per year to maintain the proficiency of its examiners, as necessary to ensure quality, and upon written request by facility licensees consistent with NRC staff availability.

This revision will become effective concurrent with the associated amendment to 10 CFR Part 55 or at an earlier date agreed upon by the facility licensee and its NRC Regional Office.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

operator licensing, examination standards

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2. TITLE AND SUBTITLE A Compilation of Reports of the Advisory Committee on Reactor Safeguards: 1998 Annual		3. DATE REPORT PUBLISHED MONTH: April YEAR: 1999 4. FIN OR GRANT NUMBER
5. AUTHOR(S)		6. TYPE OF REPORT Compilation 7. PERIOD COVERED *(Inclusive Dates)* Jan. thru Dec. 1998

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Advisory Committee on Reactor Safeguards
U. S. Nuclear Regulatory Commission
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9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above", if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This compilation contains 59 ACRS reports submitted to the U. S. Nuclear Regulatory Commission (NRC), or to the NRC Executive Director for Operations, during calendar year 1998. It also includes a report to the Congress on the NRC Safety Research Program. In addition, a report to the Commission on the NRC Safety Research Program, NUREG-1625, Volume 1, is included by reference only. All reports have been made available to the public through the NRC Public Document Room, the U. S. Library of Congress, and the Internet at http://www.nrc.gov/ACRSACNW. The reports are organized in chronological order.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Nuclear Reactors
Nuclear Reactor Safety
Reactor Operations
Safety Engineering
Safety Research

13. AVAILABILITY STATEMENT: Unlimited

14. SECURITY CLASSIFICATION
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U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER
(Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1187
Vol. 2

2. TITLE AND SUBTITLE

Performance Indicators for Operating
Commercial Nuclear Power Reactors

Data Through September 1998

3. DATE REPORT PUBLISHED

MONTH	YEAR
January	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT

Technical

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Office for Analysis and Evaluation of Operational Data
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as 8. above.

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This Nuclear Regulatory Commission (NRC) report provides performance indicator data, accounting for the different operational conditions, through September 1998 for 105 reactors. There are eight NRC Performance Indicators for Operating Commercial Nuclear Power Plants: (1) automatic scrams while critical, (2) safety system actuations, (3) significant events, (4) safety system failures, (5) forced outage rate, (6) equipment forced outages per 1000 commercial critical hours, (7) collective radiation exposure, and (8) cause codes. This report is based on data extracted from Licensee Event Reports (LERs) submitted in accordance with 10 CFR 50.73, immediate notifications to the NRC Operations Center in accordance with 10 CFR 50.72, monthly operating reports in accordance with plant technical specifications, and screening of operating experience by NRC staff. Radiation exposure data are obtained from the Institute of Nuclear Power Operations (INPO). Graphical presentation of each plant's data, including trends and deviations analyses are provided, as well as tabulated summaries of the data. The trends and deviations analyses and tabulated summaries have been presented and calculated accounting for the plant's operational conditions.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

10 CFR 50
performance indicators
data
radiation exposure
licensee event report

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

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unclassified

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unclassified

15. NUMBER OF PAGES

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3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, If any.)

NUREG-1350
Vol. 11

2. TITLE AND SUBTITLE

Nuclear Regulatory Commission
Information Digest
1999 Edition

3. DATE REPORT PUBLISHED

MONTH	YEAR
November	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

Timothy Pulliam

6. TYPE OF REPORT

Annual

7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION – NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Division of Planning, Budget, and Analysis
Office of the Chief Financial Officer
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION – NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

Same as 8, above.

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

The Nuclear Regulatory Commission Information Digest (digest) provides a summary of information about the U.S. Nuclear Regulatory Commission (NRC), NRC's regulatory responsibilities, NRC licensed activities, and general information on domestic and worldwide nuclear energy. The digest, published annually, is a compilation of nuclear- and NRC-related data and is designed to serve as a quick reference to major facts about the agency and the industry it regulates. In general, the data cover 1976 through 1998, with exceptions noted. Information on generating capacity and average capacity factor for operating U.S. commercial nuclear power reactors is obtained from monthly operating reports that are submitted directly to the NRC by the licensee. This information is reviewed by the NRC for consistency only and no independent validation and/or verification is performed.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

Information Digest
NRC Facts

13. AVAILABILITY STATEMENT

Unlimited

14. SECURITY CLASSIFICATION

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Unclassified

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Unclassified

15. NUMBER OF PAGES

16. PRICE

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1415
Vol. 12, No. 1

2. TITLE AND SUBTITLE

Office of the Inspector General
Semiannual Report to Congress
April 1, 1999 - September 30, 1999

3. DATE REPORT PUBLISHED

MONTH	YEAR
December	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT

Semiannual

7. PERIOD COVERED *(Inclusive Dates)*

4/1/99 - 9/30/99

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Office of the Inspector General
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as 8. above.

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

The Inspector General Act of 1978, as amended, requires that Inspectors General submit a "Semiannual Report to Congress" summarizing program activities. The Inspector General's report is submitted to the Chairman of the NRC not later than April 30 and October 31 for each reporting period. The Chairman comments on the report and prepares the NRC's Semiannual Report to Congress as required by the Act. the Chairman then submits the agency's report and the OIG's report to Congress no later than November 30 and May 31, respectively.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

THE OIG Semiannual Report to Congress

13. AVAILABILITY STATEMENT

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14. SECURITY CLASSIFICATION

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1423, Volume 9

2. TITLE AND SUBTITLE

A Compilation of Reports of the Advisory Committee on Nuclear Waste - July 1998 - June 1999

3. DATE REPORT PUBLISHED

MONTH	YEAR
August	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT

Compilation

7. PERIOD COVERED *(Inclusive Dates)*

July 1998 - June 1999

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Advisory Committee on Nuclear Waste
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as above

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This compilation contains 10 reports issued by the Advisory Committee on Nuclear Waste (ACNW) during the Eleventh year of its operation. The reports were submitted to the Chairman and Commissioners of the U. S. Nuclear Regulatory Commission. All reports prepared by the Committee have been made available to the public through the NRC Public Document Room, the U. S. Library of Congress, and the internet at http://www.nrc.gov/ACRSACNW.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Nuclear Waste Management
High-Level Radioactive Waste
Low-Level Radioactive Waste
Safety Engineering
Safety Research

13. AVAILABILITY STATEMENT

Unlimited

14. SECURITY CLASSIFICATION

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1. REPORT NUMBER
(Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1437, Vol. 1,
Addendum 1

2. TITLE AND SUBTITLE

Generic Environmental Impact Statement for License Renewal of Nuclear Plants
Main Report
Section 6.3--Transportation
Table 9.1 Summary of findings on NEPA issues for license renewal of nuclear power plants
Final Report

3. DATE REPORT PUBLISHED

MONTH	YEAR
August	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT

Regulatory

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as above

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This addendum to NUREG-1437, Generic Environmental Impact Statement for License Renewal of Nuclear Plants, documents the staff's analysis of the potential cumulative impacts of transporting spent nuclear fuel in the vicinity of a single high-level waste repository, and summarizes the staff's analyses undertaken to determine whether the environmental impacts of the transportation of higher enrichment and higher burnup spent nuclear fuel are consistent with the values of 10 CFR 51.52, Table S-4. The intent of the study is a generic analysis of the cumulative impacts associated with transportation of spent nuclear fuel as a result of nuclear power plant license renewal. The results of the analysis will be used to amend 10 CFR Part 51.53 and Appendix B to Subpart A of 10 CFR Part 51, and is not intended to support any other regulatory decision by the NRC. This addendum also includes an appendix that summarizes comments on the draft of the addendum, and documents the staff's responses to those comments.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Generic Environmental Impact Statement
License renewal
Nuclear Power Plant
Environmental Protection
Spent Nuclear Fuel

13. AVAILABILITY STATEMENT

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14. SECURITY CLASSIFICATION

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3201, 3202

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1. REPORT NUMBER
(Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1437, Supplement 1

2. TITLE AND SUBTITLE

Generic Environmental Impact Statement for License Renewal of Nuclear Plants

Supplement 1 Regarding the Calvert Cliffs Nuclear Power Plant

Final Report

3. DATE REPORT PUBLISHED

MONTH	YEAR
October	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT

Environmental Impact Statement

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as above

10. SUPPLEMENTARY NOTES

Docket Nos. 50-317 and 50-318

11. ABSTRACT *(200 words or less)*

This final supplemental environmental impact statement (SEIS) has been prepared in response to an application submitted to the Nuclear Regulatory Commission (NRC) by Baltimore Gas and Electric Company (BGE) to renew the operating licenses for the Calvert Cliffs Nuclear Power Plant (CCNPP) Units 1 and 2 for an additional 20 years under 10 CFR Part 54. The supplemental environmental impact statement includes the staff's analysis that considers and weighs the environmental effects of the proposed action, the environmental impacts of alternatives to the proposed action, and the alternatives available for reducing or avoiding adverse impacts. It also includes the staff's recommendations regarding the proposed action.

Based on the analysis and findings in the Generic Environmental Impact Statement, the Environmental Report submitted by BGE, consultation with other Federal and State agencies, its own independent review, and its consideration of public comments, the NRC staff recommends that the Commission determine that the adverse environmental impacts of license renewal for CCNPP Units 1 and 2 are not so great that preserving the option of license renewal for energy planning decisionmakers would be unreasonable.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Calvert Cliffs Nuclear Power Plant
Supplement to the Generic Environmental Impact Statement
License Renewal
National Environmental Policy Act
NEPA

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

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unclassified

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U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1437, Supplement 2

2. TITLE AND SUBTITLE

Generic Environmental Impact Statement for License Renewal of Nuclear Plants

Supplement 2 Regarding the Oconee Nuclear Station

Final Report

3. DATE REPORT PUBLISHED

MONTH	YEAR
December	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT

Technical

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Wasnington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as 8. above

10. SUPPLEMENTARY NOTES

Docket Numbers 50-269, 50-270, 50-287

11. ABSTRACT *(200 words or less)*

This final supplemental environmental impact statement (SEIS) has been prepared in response to an application submitted to the Nuclear Regulatory Commission (NRC) by Duke Energy Corporation (Duke) to renew the operating licenses for the Oconee Nuclear Station (ONS) Units 1, 2, and 3 for an additional 20 years under 10 CFR Part 54. The supplemental environmental impact statement includes the staff's analysis that considers and weighs the environmental effects of the proposed action, the environmental impacts of alternatives to the proposed action, and alternatives available for reducing or avoiding adverse impacts. It also includes the staff's recommendations regarding the proposed action.

Based on the analysis and findings in the Generic Environmental Statement, the environmental report submitted by Duke, consultation with other Federal and State agencies, its own independent review, and its consideration of public comments, the NRC staff recommends that the Commission determine that the adverse environmental impacts of license renewal for ONS Units 1, 2, and 3 are not so great that preserving the option of license renewal for energy planning decisionmakers would be unreasonable.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

License Renewal
National Environmental Policy Act
NEPA
Oconee Nuclear Station
Supplement to the Generic Environmental Statement

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

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(This Report) unclassified

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3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

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1. REPORT NUMBER
(Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1512, Supplement 1

2. TITLE AND SUBTITLE

Final Safety Evaluation Report Related to Certification of the AP600 Standard Design, Supplement 1

3. DATE REPORT PUBLISHED

MONTH	YEAR
December	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

Jerry N. Wilson, PE

6. TYPE OF REPORT

Final Safety Evaluation Report

7. PERIOD COVERED *(Inclusive Dates)*

September 1998 - December 1999

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Division of Regulatory Improvement Programs
Off ice of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as above

10. SUPPLEMENTARY NOTES

Docket No. 52-003 / This report supplements the FSER issued in September 199 8

11. ABSTRACT *(200 words or less)*

This supplement to the final safety evaluation report (FSER) for the AP600 Stan dard Plant Design documents the NRC staff's review of the changes to the AP600 design documentation since the issuance of t he FSER and the resolution of the confirmatory items identified in Section 1.9 of the FSER. This supplement also provides an evaluation of the design change described in Section 6.0 of this report. The discussions are supplementary to, and not in lieu of, the discussions in the FSER, unless otherwise noted.

On the basis of the evaluation described in the AP600 FSER (NUREG-1512) and thi s report, the NRC staff concludes that the confirmatory issues in NUREG-1512 are resolved, the AP600 design documentation (up to and including the 12/99 revision to the AP600 DCD) is acceptable, and Westinghouse's application for design certifi cation meets the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP600 Standa rd Plant Design.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Advanced Reactor Design
Advanced Light Water Reactor (ALWR)
Design Certification
Design Control Document (DCD)
Final Design Approval (FDA)
Final Safety Evaluation Report (FSER)
Passive Plant Design
Standardization

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

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unclassified

15. NUMBER OF PAGES

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16. PRICE

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NRCM 1102,
3201, 3202

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1. REPORT NUMBER
(Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1531
Vol. 1

2. TITLE AND SUBTITLE

Final Environmental Impact Statement Related to Reclamation of the Uranium Mill Tailings at the Atlas Site, Moab, Utah:
Source Material License No. SUA 917

3. DATE REPORT PUBLISHED

MONTH	YEAR
March	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

T.J. Blasing, S.M. Chin, Clay E. Easterly, Myron H. Fliegel, Gerald K. Eddlemon, Robert O. Johnson, Roger L. Koodsma, Michael C. Layton, Lance N. McCold, Allan T. Mullins, Carl H. Petrich, Harry D. Qualres, Robert M. Reed, William J. Reich, William P. Staub, James W. Van Dyke, Phillip J. Walsh

6. TYPE OF REPORT

Final Environmental Impact Statemen

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Division of Waste Management, Office of Nuclear Material Safety and Safeguards
United States Nuclear Regulatory Commission, Washington, DC 20555

Oak Rdge National Laboratory, Oak Ridge, Tennessee 37831

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Waste Management
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

Docket No. 40-3453

11. ABSTRACT *(200 words or less)*

This Final Environmental Impact Statement (FEIS) has been prepared by the Nuclear Regulatory Commission (NRC), Office of Nuclear Material Safety and Safeguards, to address potential environmental impacts associated with a request by Atlas Corporation to amend its existing NRC License No. SUA-917 to reclaim in place an existing uranium mill tailings pile near Moab, Utah. The proposed reclamation would allow Atlas to reclaim the tailings pile for permanent disposal and long-term custodial care by a government agency in its current location on the Moab site. The FEIS describes and evaluates (1) the purpose of and need for the proposed action, (2) alternatives considered, (3) potentially affected environmental resources, (4) environmental consequences of the proposed action, and (5) costs and benefits associated with reclamation alternatives.

The analysis of impacts presented in the FEIS indicates that the Atlas proposed on-site reclamation with recommended mitigation, will significantly reduce the impact of contaminants entering the Colorado River, but a rigorous determination of whether the proposed action will meet the FWS ammonia concentration requirements specified in the Final Biological Opinion, cannot be made without additional data and analyses by the applicant. All other environmental aspects of the proposed action are acceptable. The FEIS compares the proposed on-site reclamation to an alternative of moving the tailings to an alternative site on Klondike Flat. NRC staff's analysis finds that no aspect of the relocation alternative would have a potentially significant, adverse, long-term environmental or socioeconomic impact. Some of the short-term impacts, including radiation doses associated with moving the tailings, would be greater for the relocation alternative. Thus, the short-term impacts and the significantly higher economic cost of moving the tailings are the major disadvantages of the relocation alternative.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Atlas, uranium, mill tailings, Moab, reclamation, uranium mill, tailings

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

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unclassified

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unclassified

15. NUMBER OF PAGES

16. PRICE

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NRC FORM 335 (2-89) NRCM 1102, 3201, 3202

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1531
Vol. 2

2. TITLE AND SUBTITLE

Final Environmental Impact Statement Related to Reclamation of the Uranium Mill Tailings at the Atlas Site, Moab, Utah:
Source Material License No. SUA 917

3. DATE REPORT PUBLISHED

MONTH	YEAR
March	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

T.J. Blasing, S.M. Chin, Clay E. Easterly, Myron H. Fliegel, Gerald K. Eddlemon, Robert O. Johnson, Roger L. Koodsma, Michael C. Layton, Lance N. McCold, Allan T. Mullins, Carl H. Petrich, Harry D. Quarles, Robert M. Reed, William J. Reich, William P. Staub, James W. Van Dyke, Phillip J. Walsh

6. TYPE OF REPORT

Final Environmental Impact Statemen

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Division of Waste Management, Office of Nuclear Material Safety and Safeguards
United States Nuclear Regulatory Commission, Washington, DC 20555

Oak Rdge National Laboratory, Oak Ridge, Tennessee 37831

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above", if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Waste Management
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

Docket No. 40-3453

11. ABSTRACT *(200 words or less)*

This Final Environmental Impact Statement (FEIS) has been prepared by the Nuclear Regulatory Commission (NRC), Office of Nuclear Material Safety and Safeguards, to address potential environmental impacts associated with a request by Atlas Corporation to amend its existing NRC License No. SUA-917 to reclaim in place an existing uranium mill tailings pile near Moab, Utah. The proposed reclamation would allow Atlas to reclaim the tailings pile for permanent disposal and long-term custodial care by a government agency in its current location on the Moab site. The FEIS describes and evaluates (1) the purpose of and need for the proposed action, (2) alternatives considered, (3) potentially affected environmental resources, (4) environmental consequences of the proposed action, and (5) costs and benefits associated with reclamation alternatives.

The analysis of impacts presented in the FEIS indicates that the Atlas proposed on-site reclamation with recommended mitigation, will significantly reduce the impact of contaminants entering the Colorado River, but a rigorous determination of whether the proposed action will meet the FWS ammonia concentration requirements specified in the Final Biological Opinion, cannot be made without additional data and analyses by the applicant. All other environmental aspects of the proposed action are acceptable. The FEIS compares the proposed on-site reclamation to an alternative of moving the tailings to an alternative site on Klondike Flat. NRC staff's analysis finds that no aspect of the relocation alternative would have a potentially significant, adverse, long-term environmental or socioeconomic impact. Some of the short-term impacts, including radiation doses associated with moving the tailings, would be greater for the relocation alternative. Thus, the short-term impacts and the significantly higher economic cost of moving the tailings are the major disadvantages of the relocation alternative.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Atlas, uranium, mill tailings, Moab, reclamation, uranium mill, tailings

13. AVAILABILITY STATEMENT

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14. SECURITY CLASSIFICATION

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1532, Supplement 1

2. TITLE AND SUBTITLE

Final Technical Evaluation Report for the Proposed Reclamation Plan for the Atlas Corporation Moab Mill; Source Material License No. SUA 917

3. DATE REPORT PUBLISHED

MONTH	YEAR
April	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

Michael Layton, Daniel Rom

6. TYPE OF REPORT

Final Technical Evaluation Report

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Division of Waste Management
Office of Nuclear Material Safety and Safeguards
United States Nuclear Regulatory Commission
Washington, DC 20555

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as above

10. SUPPLEMENTARY NOTES

Docket No. 40-3453

11. ABSTRACT *(200 words or less)*

The supplement to the Final Technical Evaluation Report (FTER), pertaining to the proposed revised reclamation plan submitted by Atlas Corporation, summarizes the U.S. Nuclear Regulatory Commission staff's review of additional site data developed for the U.S. Fish and Wildlife Service (FWS) as a part of the section 7 Endangered Species Act (ESA) consultation process; and additional analyses performed by the Center for Nuclear Waste Regulatory Analyses (CNWRA) regarding performance of the proposed reclamation. The additional data were developed by the FWS in order for it to issue a Final Biological Opinion in accordance with the ESA. The results and conclusions of the Section 7 consultation are presented in the Final Environmental Impact Statement (FEIS). The FTER was issued on March 1997 and contained staff's analysis and conclusions that the proposed on-site reclamation of the existing uranium mill tailings pile complies with the requirements contained in Title 10 Code of Federal Regulations (CFR) Part 40. The additional data and analyses did not conflict with staff's analysis or conclusion of acceptability contained in the FTER; however, a supplement is needed to update the information contained in the FTER.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Atlas, uranium, mill tailings, Moab, reclamation, uranium mill, tailings

13. AVAILABILITY STATEMENT

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14. SECURITY CLASSIFICATION

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(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG–1552, Supplement 1

2. TITLE AND SUBTITLE

Fire Barrier Penetration Seals in Nuclear Power Plants

3. DATE REPORT PUBLISHED

MONTH	YEAR
January	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

C. S. Bajwa and K. S. West

6. TYPE OF REPORT

Technical

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8. PERFORMING ORGANIZATION – NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Division of Systems Safety Analysis
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555–0001

9. SPONSORING ORGANIZATION – NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

Same as above

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

Nuclear power plants use the "defense in depth" concept of echelons of fire protection to achieve a high degree of fire safety. The objective of this concept is to (1) prevent fires from starting; (2) rapidly detect, control, and extinguish those fires that do occur; and (3) protect structures, systems, and components important to safety so that a fire that is not promptly extinguished will not prevent the safe shutdown of the plant. Fire barrier penetration seals, which are but one part of one element of the fire protection defense-in-depth concept, are designed to maintain the fire rating of a barrier where penetrating items pass through the barrier. This is essential if the barrier is to confine a fire to the area in which it started or to protect plant systems and components within an area from a fire outside the area. On the basis of everything it found and considered, it is the staff's judgment that, overall, the issue of potential fire barrier penetration seal deficiencies does not affect safety. For the reasons given in this paper, typical penetration seal deficiencies do not necessarily equate to a lack of adequate protection or result in undue risk to public health and safety. It is the staff's opinion that continued licensee upkeep of existing penetration seal programs and continued NRC inspections are adequate (1) to ensure that penetration seal problems are discovered and resolved and (2) to maintain public health and safety.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

Fire Barrier Penetration Seals
Silicone Foam
Fire Barriers

13. AVAILABILITY STATEMENT

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14. SECURITY CLASSIFICATION

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1556
Vol. 6

2. TITLE AND SUBTITLE

Consolidated Guidance About Materials Licenses
Program-Specific Guidance About 10 CFR Part 36 Irradiator Licenses

Final Report

3. DATE REPORT PUBLISHED

MONTH	YEAR
January	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

J. D. Jones, W. T. Loo, E. H. Reber, M.E. Schwartz, P.C. Vacca.

6. TYPE OF REPORT

Final

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Office of Nuclear Material Safety and Safeguards
Division of Industrial and Medical Nuclear Safety
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as above

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

NRC is using Business Process Redesign (BPR) techniques to redesign its materia ls licensing process, as described in NUREG-1539, "Methodology and Findings of the NRC's Materials Licensing Process Redesign." A critical element of the new process is consolidating and updating numerous guidance documents into a NUREG- series of reports.

This NUREG report is the sixth program-specific guidance developed to support a n improved materials licensing process. It is intended for use by applicants, licensees, NRC license reviewers, and other NRC personnel. It combines and updates the guidance for applicants and licensees previously found in Draft Regulatory Guid e DG-0003, "Guide for the Preparation of Applications for Licenses for Non-Self-Contained Irradiators," dated January 19 94, and the guidance for licensing staff previously found in NMSS Policy and Guidance Directive, FC 84-23, "Standard Rev iew Plan for Licenses for the Use of Panoramic Dry Source-Storage Irradiators, Self-Contained Wet Source-Storage, an d Panoramic Wet Source-Storage Irradiators," dated December 27, 1984. In addition, this report also contains pertinent information found in Technical Assistance Requests and Information Notices.

On April 23, 1998, (63 FR 20224), NRC announced the availability of draft NUREG 1556, Vol. 6, and requested comments on it. The final document contains a compilation of the comments and the staff's responses. The comments were considered in preparing the final NUREG Report.

Applicants for NRC licenses should use this guidance when preparing application s for NRC licenses. NRC staff will use this final report in reviewing these applications.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

materials
licensees
irradiator
risk-informed
performance-based
program-specific

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG 1556
Volume 7

2. TITLE AND SUBTITLE

Consolidated Guidance about Materials Licenses:
Program-Specific Guidance About Academic, Research and Development, and Other Licenses of Limited Scope Including Gas Chromatographs and X-Ray Fluorescence Analyzers

Final Report

3. DATE REPORT PUBLISHED

MONTH	YEAR
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4. FIN OR GRANT NUMBER

5. AUTHOR(S)

M.L. Fuller, R.P. Hays, A.S. Lodhi, G.W. Purdy

6. TYPE OF REPORT

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8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Office of Nuclear Material Safety and Safeguards
Division of Industrial and Medical Nuclear Safety
US Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

Same

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

The United States Nuclear Regulatory Commission (NRC) is consolidating and updating numerous guidance documents into a single comprehensive repository as described in NUREG - 1539, "Methodology and Findings of the NRC's Materials Licensing Process Redesign," dated April 1996, and draft NUREG - 1541, "Process and Design for Consolidating and Updating Materials Licensing Guidance," dated April 1996. NUREG - 1556, Vol. 7, "Consolidated Guidance about Materials Licenses: Program - Specific Guidance about Academic, Research & Development, and Other Licenses of Limited Scope," dated December 1999, is the seventh program-specific guidance developed for the new process and is intended for use by applicants, licensees, and NRC staff and will also be available to Agreement States. This document combines and updates the guidance for applicants and licensees previously found in: (1) Regulatory Guide 10.2, Revision 1, "Guidance To Academic Institutions Applying For Specific Byproduct Material Licenses of Limited Scope," dated December 1976; (2) Regulatory Guide 10.7, "Guide For the Preparation of Applications For Licenses For Laboratory and Industrial Use of Small Quantities of Byproduct Material," dated August 1979; and (3) Draft Regulatory Guide FC 405-4, "Guide for the Preparation of Applications for Licenses for the Use of Sealed Sources in Gas Chromatography Devices and X-Ray Fluorescence Analyzers," dated February 1985. This report takes a more risk-informed, performance-based approach to the information needed to support an application for the use of byproduct material. This final report should be used in preparing academic, research & development, and other licenses of limited scope (ARDL) license applications. NRC staff will use this final report in reviewing these applications.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

ALARA
chromatography
laboratory
risk-informed
veterinary

13. AVAILABILITY STATEMENT

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14. SECURITY CLASSIFICATION

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1. REPORT NUMBER
(Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1556
Vol. 11

2. TITLE AND SUBTITLE

Consolidated Guidance About Materials Licenses
Program-Specific Guidance About Licenses of Broad Scope

Final Report

3. DATE REPORT PUBLISHED

MONTH	YEAR
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4. FIN OR GRANT NUMBER

5. AUTHOR(S)

James P. Dwyer, Orysia Masnyk Bailey, James R. Mullauer, Torre M. Taylor

6. TYPE OF REPORT

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7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Office of Nuclear Material Safety and Safeguards
Division of Industrial and Medical Nuclear Safety
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as above

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

NRC is using Business Process Redesign (BPR) techniques to redesign its materia ls licensing process, as described in NUREG-1539, "Methodology and Findings of the NRC's Materials Licensing Process Redesign." A critical element of the new process is consolidating and updating numerous guidance documents into a NUREG- series of reports.

This NUREG report is the eleventh program-specific guidance developed to suppor t an improved materials licensing process. It is intended for use by applicants, licensees, NRC license reviewers, and oth er NRC personnel. It updates the guidance for applicants and licensees previously found in Draft Regulatory Guide DG-0005, da ted October 1994. Included in this document is a new option for Type A licenses of Broad Scope to have increase flexibilit y to make changes in some program areas. This option is discussed in detail in Chapter 1.

On December 9, 1998, (63 FR 67946), NRC announced the availability of draft NUR EG 1556, Vol. 11, and requested comments on it. The final document contains a compilation of the comments and the staff's responses. The comments were considered in preparing the final NUREG Report.

Applicants for NRC licenses should use this guidance when preparing application s for NRC licenses. NRC staff will use this final report in reviewing these applications.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

materials
licensees
irradiator
risk-informed
performance-based
program-specific

13. AVAILABILITY STATEMENT

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NRC FORM 335 (2-89) NRCM 1102, 3201, 3202 — U.S. NUCLEAR REGULATORY COMMISSION — **BIBLIOGRAPHIC DATA SHEET** *(See instructions on the reverse)*	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.) NUREG 1556 Volume 12
2. TITLE AND SUBTITLE Consolidated Guidance About Materials Licenses - Program-Specific Guidance Abou t Possession Licenses for Manufacturing and Distribution Draft for comment	3. DATE REPORT PUBLISHED MONTH: July YEAR: 1999
	4. FIN OR GRANT NUMBER
5. AUTHOR(S) R. E. Zelac, D. J. Collins, C. F. Gill, L. Manning, III, M. E. Schwartz	6. TYPE OF REPORT Draft
	7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Office of Nuclear Material Safety and Safeguards
Division of Industrial and Medical Nuclear Safety
US Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

As part of its redesign of the materials licensing process, NRC is consolidatin g and updating numerous guidance documents into a single comprehensive repository as described in NUREG-1539, "Methodology and Findings of the NRC's Materials Licensing Process Redesign," dated April 1996, and draft NUREG-1541, "Process a nd Design for Consolidating and Updating Materials Licensing Guidance," dated April 1996. Draft NUREG-1556, Vol. 12, "C onsolidated Guidance about Materials Licenses: Program-Specific Guidance about Possession Licenses for Manufacturin g and Distribution," dated July 1999, is the twelth program-specific guidance document developed for the new process and is intended for use by applicants, licensees, and NRC staff, and will also be available to Agreement States. This draft rep ort takes a more risk-informed, performance-based approach to licensing for manufacturing and distribution, and reduces the information (amount and level of detail) needed to support an application. Note that this document is strictly for public comment and is not for use in preparing or reviewing applications until it is published in final form.

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NUREG 1556
Volume 13

2. TITLE AND SUBTITLE

Consolidated Guidance about Materials Licenses:
Program-Specific Guidance about Commercial Radiopharmacy Licenses

Final Report

3. DATE REPORT PUBLISHED

MONTH	YEAR
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4. FIN OR GRANT NUMBER

5. AUTHOR(S)

J. Cameron, D.B. Howe, J. Montgomery, P.Henderson

6. TYPE OF REPORT

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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Office of Nuclear Material Safety and Safeguards
Division of Industrial and Medical Nuclear Safety
US Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

NRC is consolidating and updating numerous guidance documents into a single com prehensive repository, as described in NUREG-1539, "Methodology and Findings of the NRC's Materials Licensing Process Redesign," dated April 1996, and draft NUREG-1541, "Process and Design for Consolidating and Updating Materials Licens ing Guidance," dated April 1996. NUREG-1556, Vol. 13, "Consolidated Guidance about Materials Licenses: Program- Specific Guidance about Commercial Radiopharmacy Licenses," dated September 1999, is the thirteenth program-specif ic guidance developed for the new process and is intended for use by applicants, licensees, and NRC staff, and will also be available to Agreement States. This document combines and updates the guidance found in "Draft Regulatory Guide DG- 0006" (previously issued as FC 410-4), "Guide for the Preparation of Applications for Commercial Nuclear Pharmacy Lice nses" (March 1997), and Standard Review Plan 85-14, "Standard Review Plan for Applications for Nuclear Pharmacy License s." This report takes a more risk-informed, performance-based approach to licensing commercial radiopharmacies and reduces the information (amount and level of detail) needed to support such an application.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Radiopharmacy
Dosage
Unit dose
risk-informed

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2. TITLE AND SUBTITLE Consolidated Guidance About Materials Licenses - Program-Specific Guidance About Licenses Authorizing Distribution to General Licensees Draft for comment	3. DATE REPORT PUBLISHED MONTH: September — YEAR: 1999 4. FIN OR GRANT NUMBER
5. AUTHOR(S) D. Wiedman, B. Parker, S. Minnick, J. McCausland	6. TYPE OF REPORT Draft 7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Office of Nuclear Material Safety and Safeguards
Division of Industrial and Medical Nuclear Safety
US Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This draft guide has been developed in parallel with the proposed rulemaking on 10 CFR Parts 30, 31, 32, 170, and 171, "Requirements for Certain Generally Licensed Industrial Devices Containing Byproduct Material." Comments received in response to publication of this draft will be considered in developing the final guide. Finalization of the guidance will continue to parallel the rulemaking; resulting in a guidance document that is consistent with the final rule.

As part of its redesign of the materials licensing process, NRC is consolidating and updating numerous guidance documents into a single comprehensive repository as described in NUREG-1539, "Methodology and Findings of the NRC's Materials Licensing Process Redesign, " dated April 1996, and draft NUREG-1541, "Process and Design for Consolidating and Updating Materials Licensing Guidance, dated April 1996. Draft NUREG-1556, Vol. 16, "Consolidated Guidance About Materials Licenses: Program-Specific Guidance About Licenses Authorizing Distribution to General Licensees, " dated September 1999, is the sixteenth program-specific guidance document developed for the new process and is intended for use by applicants, licensees, and NRC staff, and will also be available to Agreement States. This draft report takes a more risk-informed, performance-based approach to licensing for distribution to general licensees, and reduces the information (amount and level of detail) needed to support an application. Note that this document is strictly for public comment and is not for use in preparing or reviewing applications until it is published in final form.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*	13. AVAILABILITY STATEMENT: unlimited
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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1574, Rev. 1

2. TITLE AND SUBTITLE

Standard Review Plan on Antitrust Reviews

Draft Report for Comment

3. DATE REPORT PUBLISHED

MONTH	YEAR
October	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

M.J. Davis

6. TYPE OF REPORT

Technical

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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as 8. above.

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

The Nuclear Regulatory Commission is issuing this draft revision to the standard review plan on antitrust reviews to describe the procedure used to implement the antitrust review and enforcement process prescribed in Sections 105 and 186 of the Atomic Energy Act of 1954, as amended. This draft SRP revision reflects the recent policy change to discontinue antitrust reviews of license transfer applications.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Standard Review Plan
Antitrust

13. AVAILABILITY STATEMENT

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14. SECURITY CLASSIFICATION

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1577

2. TITLE AND SUBTITLE

Standard Review Plan on Power Reactor Licensee Financial Qualifications and Decommissioning Funding Assurance

3. DATE REPORT PUBLISHED

MONTH	YEAR
February	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

Robert S. Wood

6. TYPE OF REPORT

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Division of Reactor Program Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Comission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above" if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same As Above

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

The NRC is issuing this Standard Review Plan to describe the process it uses to review the financial qualifications and the methods of providing decommissioning funding assurance required of power reactor license applicants and existing licensees.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Standard Review Plan
Financial Qualifications
Decommissiong Funding Assurance

13. AVAILABILITY STATEMENT

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14. SECURITY CLASSIFICATION

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1600

2. TITLE AND SUBTITLE

General Statement of Policy and Procedure for NRC Enforcement Actions

Enforcement Policy - November 9, 1999

3. DATE REPORT PUBLISHED

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November	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

Office of Enforcement

6. TYPE OF REPORT

Regulatory

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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Office of Enforcement
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as 8 above.

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This document includes the U.S. Nuclear Regulatory Commission's (NRC's or Commission's) revised General Statement of Policy and Procedure for Enforcement Actions (Enforcement Policy) as it was published in the Federal Register on November 9, 1999 (64 FR 61142). Because the policy is often revised without being reissued as a NUREG series publication, this current document is identified by date instead of revision number. The policy that preceded this November 1999 publication was NUREG-1600, Rev. 1. The Enforcement Policy is a general statement of policy explaining the NRC's policies and procedures in initiating enforcement actions, and the responsibilities of the presiding officers and the Commission in reviewing these actions. This policy statement is applicable to enforcement matters involving the radiological health and safety of the public, including employees' health and safety, the common defense and security, and the environment. The NRC publishes the policy statement in NRC's NUREG series to foster its widespread dissemination. However, this is a policy statement and not a regulation. The Commission may deviate from this statement of policy and procedure as appropriate under the circumstances of a particular case. As a living policy statement, revisions are noticed in the Federal Register. The NRC's Office of Enforcement maintains the current policy statement on its homepage on the Internet at www.nrc.gov/OE/. This revision (1) revises the approach for assessing the significance of violations, (2) changes guidance to conform to recent revisions to the requirements of 10 CFR 50.59, "Changes, tests, and experiments" (64 FR 53582; October 4, 1999), (3) updates the policy to reflect the Deputy Executive Director for Reactor Programs and the Deputy Executive Director for Materials, Research and State Programs as the principal enforcement officers of the NRC, (4) corrects the schedule for exercising enforcement discretion for findings involving the completeness and accuracy of licensee Final Safety Analysis Reports (FSAR), (5) consolidates the guidance on dispositioning Severity Level IV violations as either Notices of Violation or Non-Cited Violations, (6) reorganizes existing guidance on the relationship between safety and compliance to improve clarity, (7) consolidates changes to the Enforcement Policy since May 1998, and (8) edits and restructures existing guidance to assure consistency with recent policy changes and to facilitate maintenance of the living policy statement.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

NRC enforcement guidance
NRC Enforcement Policy
NRC enforcement program
NRC enforcement responsibilities

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NRC FORM 335 (2-89) NRCM 1102, 3201, 3202 U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET** *(See instructions on the reverse)*	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.) NUREG-1605
2. TITLE AND SUBTITLE Risk Profile Methodology of Plant Configurations and Pilot Applications: Lessons Learned	3. DATE REPORT PUBLISHED MONTH: January YEAR: 1999 4. FIN OR GRANT NUMBER
5. AUTHOR(S) J.W. Chung, USNRC S-M Wong, BNL J.E. Riley, SAIC	6. TYPE OF REPORT Technical 7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Division of Systems Safety Analysis
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Brookhaven National Laboratory
Upton, NY 11973

Science Applications International Corporation
4920 El Camino Road
Los Altos, CA 94022

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as 8. above.

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This report summarizes a risk-based methodology for developing time-dependent risk profiles of plant operational configurations, and lessons learned from pilot plant visits. The methodology focuses on the determination of plant configuration risk profiles using plant operating records, or actual real-time equipment unavailabilities due to plant operational evolutions. This methodology employs state-of-the-art PRA methods and risk models, and uses available computer codes and various quantification techniques. The risk profile emphasizes the relative risk changes with respect to a reference risk level that is defined on the basis of plant baseline configuration. The limitations of risk models and sensitivity to risk contributors and of computation methods are discussed within the scope of developing risk profiles, and lessons learned from the pilot studies are documented. Results and lessons learned from the trial applications at five pilot plant sites are also presented. The pilot plants for these studies are the South Texas Project, Comanche Peak, Crystal River 3, Brunswick 2, D.C. Cook, and San Onofre 2 nuclear stations. The trial application studies demonstrated that insights from configuration risk profiles can provide a risk perspective of plant performance and risk trending. These insights can be used for risk-informed regulatory initiatives and other applications.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*	13. AVAILABILITY STATEMENT unlimited
PRA methodology risk profile risk management plant configuration	14. SECURITY CLASSIFICATION *(This Page)* unclassified *(This Report)* unclassified
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BIBLIOGRAPHIC DATA SHEET

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1. REPORT NUMBER
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NUREG-1609

2. TITLE AND SUBTITLE

Standard Review Plan for Transportation Packages for Radioactive Material

3. DATE REPORT PUBLISHED

MONTH	YEAR
May	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

Spent Fuel Project Office, Office of Nuclear Material Safety and Safeguards

6. TYPE OF REPORT

Final

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Spent Fuel Project Office, Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as above.

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

The Standard Review Plan for Transportation Packages for Radioactive Material provides guidance for the review and approval of applications for packages used to transport radioactive material (other that irradiated nuclear fuel) under 10 CFR Part 71.

This document is intended for use by the U.S. Nuclear Regulatory Commission (NRC) staff. Its objectives are to (1) summarize 10 CFR Part 71 requirements for package approval, (2) describe the procedures by which the NRC staff determines that these requirements have been satisfied, and (3) document the practices developed by the staff in previous reviews of package applications.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Transportation, Standard Review Plan, 10 CFR Part 71, Radioactive Material Packaging

13. AVAILABILITY STATEMENT

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1. REPORT NUMBER
(Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG -1615

2. TITLE AND SUBTITLE

Physical Protection Requirements for Categories I, II, and III Material at Fuel Cycle Facilities

3. DATE REPORT PUBLISHED

MONTH	YEAR
March	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

C. Brown

6. TYPE OF REPORT

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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Division of Fuel Cycle Safety and Safeguards
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Fuel Cycle Safety and Safeguards
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This NUREG presents the primary physical protection requirements, issued by the U.S. Nuclear Regulatory Commission (NRC), under Title 10 of the U.S. Code of Federal Regulations, applicable to unirradiated Categories I, II, and III material at fixed sites. Category I refers to formula quantities of strategic special nuclear material. Category II refers to quantities and types of special nuclear material of moderate strategic significance. Category III refers to quantities and types of special nuclear material of low strategic significance. The requirements are presented in a modular format in relation to the various physical protection functional areas.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Physical Protection requirements for Categories I, II, III material in a modular format

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1635, Volume 2

2. TITLE AND SUBTITLE

Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program

A Report to the U. S. Nuclear Regulatory Commission

3. DATE REPORT PUBLISHED

MONTH	YEAR
July	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT

Annual

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Fiscal Year 1999

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Advisory Committee on Reactor Safeguards
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as above

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

In 1998, the Advisory Committee on Reactor Safeguards (ACRS) submitted to the Nuclear Regulatory Commission (NRC) a comprehensive report of the NRC's Safety Research Program, NUREG-1635, Vol. 1, "Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program," which documented the ACRS conclusions and recommendations. The ACRS continues to support the conclusions and recommendations of that report. The present report is more modest in scope and is intended to provide additional information concerning the research needed to support Commission programs, especially risk-informed regulation. Not all research programs are included in this report and our observations are often limited to certain aspects of a given program.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Nuclear Reactors
Nuclear Reactor Safety
Reactor Operations
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Safety Research

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG–1636

2. TITLE AND SUBTITLE

Regulatory Perspectives on Model Validation in High-Level Radioactive Waste Management Programs: A Joint NRC/SKI White Paper

3. DATE REPORT PUBLISHED

MONTH	YEAR
March	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

N.A. Eisenberg[a], M.P. Lee[a], M.V. Federline[a], S. Wingefors[b], J. Andersson[b], S. Norrby[b], B. Sagar[c], and G.W. Wittmeyer[c]

6. TYPE OF REPORT

Technical

7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION – NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

[a]Division of Waste Management, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555–0001
[b]Office of Nuclear Waste Safety, Swedish Nuclear Power Inspectorate (Statens Kärnkraftinspektion), Stockholm, Sweden
[c]Center for Nuclear Waste Regulatory Analyses, San Antonio, Texas, USA

9. SPONSORING ORGANIZATION – NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

Division of Waste Management, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555–0001
Office of Nuclear Waste Safety, Swedish Nuclear Power Inspectorate

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

Validation should be an important aspect of the regulatory uses of mathematical models in the safety assessments of geologic repositories for the disposal of spent nuclear fuel and other high-level radioactive wastes. Because models for a geologic repository performance assessment cannot be tested over the spatial scales of interest and long time periods for which the models will make estimates of performance, the usual avenue for model validation—e.g., comparison of model estimates with actual data at the space-time scales of interest—is precluded. Further complicating the model validation process in HLW programs are the uncertainties inherent in describing the geologic complexities of potential disposal sites, and their interactions with the engineered system, with a limited set of generally imprecise data, making it difficult to discriminate between model discrepancy and inadequacy of input data. A successful strategy for model validation should attempt to recognize these difficulties, address their resolution, and document the resolution in a careful manner. The end result of validation efforts should be a documented enhancement of confidence in the model to an extent that the model's results can aid in regulatory decision-making. The level of validation needed should be determined by the intended uses of these models.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

conceptual models
confidence building
decision-making
geologic repository
high-level radioactive waste
mathematical models
model validation
performance assessment
probabilistic risk assessment
safety assessment

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(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1648

2. TITLE AND SUBTITLE

Lessons Learned From Maintenance Rule Baseline Inspections

3. DATE REPORT PUBLISHED

MONTH	YEAR
October	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

S. M. Wong, F. X. Talbot, R. M. Latta, R. P. Correia, T. R. Quay

6. TYPE OF REPORT

Final

7. PERIOD COVERED *(Inclusive Dates)*

7/96 through 7/98

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

DIvision of Inspection Program Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as Above

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This report summarizes the lessons learned from 68 maintenance rule (MR) baseli ne inspections (MRBIs)conducted at plants with operating licenses in accordance with Title 10, Part 50, of the Code of Fe deral Regulations (CFR) and 3 MRBIs conducted at plants with decommissioning certifications in accordance 10 CFR 50.82(a)(2) . The MRBIs were conducted between July 15, 1996, and July 10, 1998. In general, these MRBIs revealed that licensees i mplemented the requirements of the maintenance rule , 10 CFR 50.65, by following the guidance in NUMARC 93-01, "I ndustry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants", as endorsed by NRC Regul atory Guide 1.160, "Montoring the Effectiveness of Maintenance at Nuclear Power Plants." Additionally, licensee s effectively determined which structures, systems, and components (SSCs) at each site were within the scope of the MR. T he use of the expert panels was effective in determining which SSCs were risk significant. The results of the MRBIs also in dicated that the several expert panels performed other MR activities that exceeded the guidance in NUMARC 93-01. Whe n setting goals or performance measures (criteria) in accordance with 10 CFR 50.65(a)(1) or (a)(2), respectively, most licensees considered the risk insights from the probabilisitic risk assessments (PRAs). However, early MRBIs revealed that som e licensees did not have adequate technical justification for deviating from SSC reliability and availability assumptions i n the PRAs when establishing goals or performance criteria and did not adequately assess planned and emergent maintenance activit ies. Most licensees' individual self-assessments were a MR program implementation strength. Most licensees als o established reasonable plans and methods to periodically evaluate the effectiveness of equipment performance and preventive maintenance, including the balance between reliability and availability.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Maintenance Rule
Maintenance Rule Baseline Inspections

13. AVAILABILITY STATEMENT

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14. SECURITY CLASSIFICATION

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1. REPORT NUMBER
(Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1649

2. TITLE AND SUBTITLE

New NRC Reactor Inspection and Oversight Program

3. DATE REPORT PUBLISHED

MONTH	YEAR
February	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT

Technical

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Office of Public Affairs
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as 8. above

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

The Nuclear Regulatory Commission is revamping its inspection and oversight program for commercial nuclear power plants. The new program takes into account improvements in the performance of the nuclear industry over the past twenty years; the desire of the NRC to apply more objective, timely, safety-significant criteria in assessing performance; and the agency's need to effectively regulate the industry with a smaller staff and budget. The new program will be used at eight nuclear power plants on a pilot basis, beginning in June 1999. The experience of this pilot program will be used to evaluate and, if needed, modify the new processes before they are extended to all plants in January 2000.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

reactor inspection
nuclear power plant performance

13. AVAILABILITY STATEMENT

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14. SECURITY CLASSIFICATION

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1649, Rev. 1

2. TITLE AND SUBTITLE

New NRC Reactor Inspection and Oversight Program

3. DATE REPORT PUBLISHED

MONTH	YEAR
May	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT

Technical

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Office of Public Affairs
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as organization in Block 8.

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

The Nuclear Regulatory Commission is revamping its inspection and oversight program for commercial nuclear power plants. The new program takes into account improvements in the performance of the nuclear industry over the past 20 years; the desire of the NRC to apply more objective, timely, safety-significant criteria in assessing performance; and the agency's need to effectively regulate the industry with a smaller staff and budget. The new program will be used at eight nuclear power plants on a pilot basis, beginning in June 1999. The experience of this pilot program will be used to evaluate and, if needed, modify the new processes before they are extended to all plants in January 2000.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

reactor inspection
nuclear power plant performance

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

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NRC FORM 335 (2-89) NRCM 1102, 3201, 3202	U.S. NUCLEAR REGULATORY COMMISSION	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)
BIBLIOGRAPHIC DATA SHEET (See instructions on the reverse)		NUREG-1667
2. TITLE AND SUBTITLE		3. DATE REPORT PUBLISHED
The Impact of the Heat Transfer Coefficient on Pressurized Thermal Shock		MONTH: February YEAR: 1999
		4. FIN OR GRANT NUMBER
5. AUTHOR(S)		6. TYPE OF REPORT
C.F. Boyd, T. Dickson*		Technical
		7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Division of Systems Technology
Office of Nuclear Regulatory Research
U.S.Nuclear Regulatory Commission
Washington, DC 20555-0001

*Oak Ridge National Laboratory
Oak Ridge, TN 37831-8056

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

Same as above

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

A study is completed which quantitatively defines values for the downcomer heat transfer coefficients which induce conduction-limited heat transfer during postulated pressurized thermal shock transients. The results are useful for assessing whether fracture mechanics predictions of pressurized thermal shock will be sensitive to uncertainty or to variations in the heat transfer coefficient. Idealized thermal transients are used along with several fixed levels of the heat transfer coefficients to access sensitivity of the results. Thermal-hydraulic results are relatively insensitive to the value of the heat transfer coefficient. Variations in the fracture mechanics predictions can be significant, hopwever. Material fracture toughness and the stress intensity factor for a fixed flaw are determined using the FAVOR code for various levels of the heat transfer coefficient and for several prescribed cooldown rates. On the basis of the fracture mechanics results, values of the heat transfer coefficients which result in conduction-limited behavior determined as a function of the cooldown rate. Bounds are defined based upon the fracture mechanics solutions reaching 85, 90, and 95 percent of the asymptotic conduction-limited solution. Conclusions are directly applicable only under tthe assumptions and limitations of this study.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)	13. AVAILABILITY STATEMENT
Fracture Mechanics	unlimited
Pressurized Thermal Shock	14. SECURITY CLASSIFICATION
Thermal Hydraulics	(This Page) unclassified
Heat Transfer Coefficient	(This Report) unclassified
Conduction Limited	15. NUMBER OF PAGES
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1. REPORT NUMBER
(Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, If any.)

NUREG-1668
Vol. 2

2. TITLE AND SUBTITLE

NRC Sensitivity and Uncertainty Analyses for a Proposed HLW Repository at Yucca Mountain, Nevada, Using TPA 3.1

Results and Conclusions

3. DATE REPORT PUBLISHED

MONTH	YEAR
March	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

M.S. Jarzemba*, R.B. Codell**, L.M. Deere*, J.R. Firth**, C. Lui**, S. Mohanty*, K. Poor*, J.Weldy*, V. Colten-Bradley**

6. TYPE OF REPORT

Technical

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

* Center for Nuclear Waste Regulatory Analyses, 6220 Culebra Road, San Antonio, TX 78228-0510
** Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Waste Management
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

The total-system performance assessment computer code was developed to assist the U.S. Nuclear Regulatory Commission staff in evaluating the safety case for the proposed geologic repository at Yucca Mountain, Nevada. This report: (i) presents results of process and system-level sensitivity and uncertainty analyses to determine the parameters that have the most influence on repository performance; (ii) estimates the relative importance of the key elements of subsystem abstraction; and (iii) focuses staff reviews of the U.S. Department of Energy total-system performance assessments. Results of system-level analyses are based on peak dose in the time period of interest (TPI) to a receptor group 20 kilometers from the proposed repository. For a TPI of 10,000 years, and an inner waste package of Alloy 625, the most influential parameters were the rates of corrosion and water ingress into the package. With Alloy C-22, the same parameters were found to be influential, along with those affecting damage to waste packages from seismicity. For a TPI of 50,000 years, parameters dictating the transport of neptunium and americium from the waste form and through the geosphere were found to be most influential, with parameters dictating the rate of water ingress into the package somewhat less important. The influential parameters resulting from these analyses were compared to the current key elements of subsystem abstraction used by the NRC staff to focus work on items important to overall repository performance.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Computer code
Dose
Geologic repository
High-level radioactive waste
Performance assessment
Receptor group
Sensitivity and uncertainty analysis
Transport
Waste package
Disruptive Consequences
EPA Standard
Ground water
Parameter
Probabilstic risk assessment
Scenario
Source term
Yucca Mountain

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1. REPORT NUMBER *(Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)*

NUREG-1705

2. TITLE AND SUBTITLE

Safety Evaluation Report
Related to the License Renewal of
Calvert Cliffs Nuclear Power Plant, Units 1 and 2

3. DATE REPORT PUBLISHED

MONTH	YEAR
December	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT

Technical

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as item 8 above.

10. SUPPLEMENTARY NOTES

Docket Numbers 50-317 and 50-318 D.L. Solorio, NRC Project Manager

11. ABSTRACT *(200 words or less)*

This safety evaluation report (SER) documents the technical review of the Calvert Cliffs Nuclear Power Plant (CCNPP), Units 1 and 2 license renewal application (LRA) by the U.S. Nuclear Regulatory Commission staff. The Baltimore Gas and Electric Company (BGE) requested renewal of the Class 104b operating licenses for the Calvert Cliffs units for a period of 20 years beyond the current expiration dates. By letter dated April 8, 1998, BGE submitted a LRA for Calvert Cliffs required by Part 54 of Title 10 of the Code of Federal Regulations.

On the basis of its evaluation of the LRA the staff concludes that: (1) actions have been identified and have been or will be taken with respect to managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require an aging management review under 10 CFR 54.21(a)(1), and (2) actions have been identified and have been or will be taken with respect to time-limited aging analyses that have been identified to require review under 10 CFR 54.21(c). Accordingly, the staff finds that there is reasonable assurance that the activities authorized by a renewed license will continue to be conducted in accordance with the current licensing basis for the CCNPP, Units 1 and 2 during the period of extended operation.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

License Renewal
Part 54
Calvert Cliffs Nuclear Power Plant
Safety Evaluation Report
50-317
50-318
Scoping
aging management

102

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

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U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER
(Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1706

2. TITLE AND SUBTITLE

Year 2000 Readiness in U.S. Nuclear Power Plants

3. DATE REPORT PUBLISHED

MONTH	YEAR
September	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

Michael E. Waterman, Deirdre W. Spaulding

6. TYPE OF REPORT

Technical

7. PERIOD COVERED *(Inclusive Dates)*

Jan. 1996 - March 2000

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Division of Engineering
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission and mailing address.)*

Same as above

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This report describes the results of NRC staff reviews of licensee nuclear power plant (NPP) year 2000 (Y2K) readiness activities conducted at each nuclear power plant. The results of the staff reviews are integrated with the July 1, 1999, licensee responses to Generic Letter (GL) 98-01, Supplement 1, "Year 2000 Readiness of Computer Systems at Nuclear Power Plants," and licensee follow-up reports of Y2K readiness. All licensees of NPPs reported in response to GL 98-01, Supplement 1, that there are no Y2K-related problems that directly affect the performance of safety systems. The Nuclear Regulatory Commission has confirmed by onsite reviews that at all 103 U.S. nuclear power plants there are no Y2K-related problems which affect the performance of safety systems needed to safely shut down the plants. As of September 1, 1999, the staff concludes that licensees of 75 of the 103 plants have completed all activities to ensure computer systems and digital embedded components that support plant operations are "Y2K ready. Licensees for 28 plants have additional work to complete on a few non-safety systems or components that support plant operations and administrative functions. These licensees provided scheduled completion dates for their plants. Typically, the licensee is completing the remainingY2K work after July 1, 1999, because the work requires a plant outage scheduled for the fall of 1999 or because the licensee is waiting for delivery of a replacement component. All licensees are expected to be Y2K ready by December 16, 1999.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Year 2000
Y2K
Y2K readiness
Y2K Ready
Y2K Compliant
Nuclear power plant
Contingency Plan
Contingency Planning

103

13. AVAILABILITY STATEMENT

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U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER
(Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1709

2. TITLE AND SUBTITLE

Selection of Sample Rate and Computer Wordlength in Digital Instrumentation and Control Systems

Draft for Comment

3. DATE REPORT PUBLISHED

MONTH	YEAR
August	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

T. W. Jackson

6. TYPE OF REPORT

Technical

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Division of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as above

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

Digital sampling of analog signals adds two types of errors, aliasing and finit e wordlength error, to the sampled version of the signal. Aliasing is characterized by high frequency components misrepresented a s low frequency components in the sampled signal. It is greatly influenced by the sample rate, and may lead to degraded performance in monitoring, alarm, control, and protection systems. Since computer wordlengths are finite in length, digital s ystems are limited in their capability to represent real number values. Finite wordlength errors related to round-off, truncation, and data conversion have the potential to adversely impact the performance of digital instrumentation and control (I&C) s ystems.

The Office of Nuclear Regulatory Research is investigating good engineering pra ctices and review guidance regarding aliasing and finite wordlength errors in nuclear facilities. Hazards associated with th ese errors are minimized through proper design and selection of sample rates and computer wordlengths. This document provides the regulatory background, theoretical information, practical issues, best engineering practices, review guidance, and examples associated with sample rate and computer wordlength selection. This information is used by NRC staff to identi fy improper treatment of aliasing and finite wordlength error in digital I&C systems.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Analog-to-Digital Converters, Accuracy, Calibration, Calibration Standards, Com puters, Data Acquisition, Data Processing, Digital Filters, Digital Frequency Analysis, Digi tal Systems, Digital-to-Analog Converters, Electronic Equipment, Errors, Frequency Analysis, Reactor Instrumentation, Reactor Protection Systems, Real Time Systems, Resolution, Sig nal-to-Noise Ratio, Time Delay, Time Resolution, Time Series Analysis, Tolerance

13. AVAILABILITY STATEMENT

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14. SECURITY CLASSIFICATION

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(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1711
Draft

2. TITLE AND SUBTITLE

Nuclear Byproduct Material Risk Review: Regulatory and Other Bases for Barriers to Dose

Draft for Comment

3. DATE REPORT PUBLISHED

MONTH	YEAR
July	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

E. Ullrich

6. TYPE OF REPORT

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7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Division of Industrial and Medical Nuclear Safety
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as above

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This project responded to NRC's Direction Setting Issue 12, Risk-Informed, Performance-Based Regulation. Its scope was limited to nuclear byproduct materials as defined in Section 11.e(1) of the Atomic Energy Act of 1954 and Title 10 of the Code of Federal Regulations (CFR), Section 30.4. 10 CFR Parts 30 through 36 and 39 address regulation of those materials. The goal was to document current (i.e., mid-1999) regulatory and other bases (e.g., license conditions, good practices) that support physical and procedural barriers intended to limit dose from materials regulated under 10 CFR Parts 30 through 36 and 39. The process involved (1) use of a list of nuclear byproduct material systems based on how the nuclear byproduct material was used, (2) for each system, use of a list of barriers to dose, both physical and procedural for workers and for the public, (3) review of NRC regulations, licenses, etc. to identify regulatory or other support for the barriers present in each system, and (4) documentation of the results of that review in tabular form.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Radiation Protection
Radioactive Materials
Risk Assessment

13. AVAILABILITY STATEMENT

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14. SECURITY CLASSIFICATION

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1712
Draft

2. TITLE AND SUBTITLE

Nuclear Byproduct Material Risk Review: Results of Survey of NRC and Agreement State Materials Licensing and Inspection Personnel

Draft Report for Comment

3. DATE REPORT PUBLISHED

MONTH	YEAR
July	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

D. Serig, J. Lubinski, E. Ullrich, J. Randall, N. Daugherty

6. TYPE OF REPORT

Draft

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Division of Industrial and Medical Nuclear Safety
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as above.

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This project responded to NRC's Direction Setting Issue 12, Risk Informed, Performance-Based Regulation. Its scope was limited to nuclear byproduct materials as defined in Section 11.e(1) of the Atomic Energy Act of 1954 and Title 10 of the Code of Federal Regulations (CFR), Section 30.4. 10 CFR Parts 30 through 36 and 39 address regulation of those materials. The goal was to confirm and augment information on nuclear byproduct material systems obtained from other sources. The process involved (1) use of a list of nuclear byproduct material systems based on how the nuclear byproduct material was used, (2) a survey of Agreement State materials licensing and inspection personnel concerning typical annual doses to workers for the various systems, safety of each system under various conditions, the types and frequencies of incidents occurring at each system, definitions of safety, and opinions about the appropriate bases for regulatory decision making, and (3) summarization of the respondent's answers to those questions.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Radiation Protection
Radioactive Materials
Risk Assessment

13. AVAILABILITY STATEMENT

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14. SECURITY CLASSIFICATION

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1717
Draft

2. TITLE AND SUBTITLE

Systematic Radiological Assessment of Exemptions for Source and Byproduct Materials

Draft Report for Comment

3. DATE REPORT PUBLISHED

MONTH	YEAR
December	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

S. Schneider, NRC
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G.D. Kerr*
P.A. Scofield**
F.R. O'Donnell**
C.R. Mattsen, NRC
S.J. Cotter**
J.S. Bogard*
J.S. Bland***
C. Wiblin***

6. TYPE OF REPORT

Draft Report for Comment

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Division of Risk Analysis and Applications
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

*Oak Ridge National Laboratory
Life Sciences Division
1060 Commerce Park
Oak Ridge, TN 37830

**Office of Environmental Compliance, ORNL

***J. Stewart Bland Associates, Inc.
788 Sonne Drive
Annapolis, MD 21401

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as 8. above.

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This report is an assessment of potential radiation doses associated with the current exemptions for byproduct and source material in Title 10, of the Code of Federal Regulations (CFR). Doses were estimated for the normal life cycle of a particular product or material, covering distribution and transport, intended or expected routine use, and disposal using dose assessment methods consistent with the current requirements in 10 CFR Part 20. In addition, assessments of potential doses due to accidents and misuse were estimated. Also presented is an assessment of potential radiological impacts associated with selected products containing byproduct material which currently may only be used under a general license and may be potential candidates for exemption from licensing requirements.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

exemption
dose assessment
Part 30
Part 40

13. AVAILABILITY STATEMENT

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/BR-0006
Rev. 4

2. TITLE AND SUBTITLE

Instructions for Completing Nuclear Material Transfer Reports

3. DATE REPORT PUBLISHED

MONTH	YEAR
October	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT

Brochure

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as 8. above.

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

NRC regulations require each licensee who ships, receives or adjusts their physical inventory or source or special nuclear material to document and report such activities. The documentation is submitted using the DOE/NRC Form-741 or 741A. Licensees may need to provide additional information on some imports or exports of source or special nuclear material. The additional information is reported using DOE/NRC Form 740M. This NUREG contains the reporting instructions for licensees to follow in preparing these forms.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Nuclear Materials Management and Safeguards System
Nuclear Material Transaction Report
Concise Note

13. AVAILABILITY STATEMENT

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NRC FORM 335 (2-89) NRCM 1102, 3201, 3202	U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET** *(See instructions on the reverse)*	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.) NUREG/BR-0007 Rev. 3
2. TITLE AND SUBTITLE	Instructions for the Preparation and Distribution of Material Status Reports	3. DATE REPORT PUBLISHED — MONTH: October; YEAR: 1999
		4. FIN OR GRANT NUMBER
5. AUTHOR(S)		6. TYPE OF REPORT: Brochure
		7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as 8. above.

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

NRC regulations require each licensee who is authorized to possess at any one time and location special nuclear material (SNM) in a quantity totaling more than 350 grams of contained uranium-235, uranium-233, or plutonium, or any combination thereof, to prepare and submit in computer readable format reports concerning SNM received, produced, possessed, transferred, consumed, disposed of, or lost. This NUREG contains the reporting instructions for licensees to follow in making these reports.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*	
Nuclear Materials Management and Safeguards System	13. AVAILABILITY STATEMENT: unlimited
Material Status Report	14. SECURITY CLASSIFICATION *(This Page)*: unclassified
Material Balance Report	*(This Report)*: unclassified
Physical Inventory Listing	15. NUMBER OF PAGES
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2. TITLE AND SUBTITLE Proceedings of the Twenty-Sixth Water Reactor Safety Information Meeting Plenary Sessions, Pressure Vessel Research, Severe Accident Research, Fission Product Behavior, Nuclear Materials Issues and Health Effects Research, Materials Integrity Issues	3. DATE REPORT PUBLISHED MONTH: June YEAR: 1999 4. FIN OR GRANT NUMBER A3988
5. AUTHOR(S) Compiled by Susan Monteleone, Brookhaven National Laboratory	6. TYPE OF REPORT Conference Proceedings 7. PERIOD COVERED *(Inclusive Dates)* October 26-28, 1998

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as 8. above.

10. SUPPLEMENTARY NOTES

S. Nesmith, NRC Project Manager Proceedings prepared by Brookhaven National Laboratory

11. ABSTRACT *(200 words or less)*

This three-volume report contains papers presented at the Twenty-Sixth Water Reactor Safety Information Meeting held at the Bethesda Marriott Hotel, Bethesda, Maryland, October 26-28, 1998. The papers are printed in the order of their presentation in each session and describe progress and results of programs in nuclear safety research conducted in this country and abroad. Foreign participation in the meeting included papers presented by researchers from France, Germany, Italy, Japan, Norway, Russia, Sweden and Switzerland. The titles of the papers and the names of the authors have been updated and may differ from those that appeared in the final program of the meeting.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*	13. AVAILABILITY STATEMENT: unlimited
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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/CP-0166
Vol. 2

2. TITLE AND SUBTITLE

Proceedings of the Twenty-Sixth Water Reactor Safety Information Meeting

Digital Instrumentation and Control, The Halden Program, Structural Performance, and PRA Methods and Applications

3. DATE REPORT PUBLISHED

MONTH	YEAR
June	1999

4. FIN OR GRANT NUMBER

A3988

5. AUTHOR(S)

Compiled by Susan Monteleone, Brookhaven National Laboratory

6. TYPE OF REPORT

Conference Proceedings

7. PERIOD COVERED *(Inclusive Dates)*

October 26-28, 1998

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as 8. above.

10. SUPPLEMENTARY NOTES

S. Nesmith, NRC Project Manager Proceedings prepared by Brookhaven National Laboratory

11. ABSTRACT *(200 words or less)*

This three-volume report contains papers presented at the Twenty-Sixth Water Reactor Safety Information Meeting held at the Bethesda Marriott Hotel, Bethesda, Maryland, October 26-28, 1998. The papers are printed in the order of their presentation in each session and describe progress and results of programs in nuclear safety research conducted in this country and abroad. Foreign participation in the meeting included papers presented by researchers from France, Germany, Italy, Japan, Norway, Russia, Sweden and Switzerland. The titles of the papers and the names of the authors have been updated and may differ from those that appeared in the final program of the meeting.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

BWR Type Reactors - Reactor Safety, Nuclear Power Plants - Reactor Safety, PWR Type Reactors - Reactor Safety, Reactor Safety - Meetings

13. AVAILABILITY STATEMENT

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NRC FORM 335 (2-89) NRCM 1102, 3201, 3202 — U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET** *(See instructions on the reverse)*	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.) NUREG/CP-0166 Vol. 3
2. TITLE AND SUBTITLE Proceedings of the Twenty-Sixth Water Reactor Safety Information Meeting Thermal Hydraulic Research, Plant Aging I - Plant Life Management, High Burn-up Fuel, Plant Aging II - Cable Aging	3. DATE REPORT PUBLISHED MONTH: June YEAR: 1999 4. FIN OR GRANT NUMBER A3988
5. AUTHOR(S) Compiled by Susan Monteleone, Brookhaven National Laboratory	6. TYPE OF REPORT Conference Proceedings 7. PERIOD COVERED *(Inclusive Dates)* October 26-28, 1998

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as 8. above.

10. SUPPLEMENTARY NOTES

S. Nesmith, NRC Project Manager   Proceedings prepared by Brookhaven National Laboratory

11. ABSTRACT *(200 words or less)*

This three-volume report contains papers presented at the Twenty-Sixth Water Reactor Safety Information Meeting held at the Bethesda Marriott Hotel, Bethesda, Maryland, October 26-28, 1998. The papers are printed in the order of their presentation in each session and describe progress and results of programs in nuclear safety research conducted in this country and abroad. Foreign participation in the meeting included papers presented by researchers from France, Germany, Italy, Japan, Norway, Russia, Sweden and Switzerland. The titles of the papers and the names of the authors have been updated and may differ from those that appeared in the final program of the meeting.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)* BWR Type Reactors - Reactor Safety, Nuclear Power Plants - Reactor Safety, PWR Type Reactors - Reactor Safety, Reactor Safety - Meetings	13. AVAILABILITY STATEMENT unlimited 14. SECURITY CLASSIFICATION *(This Page)* unclassified *(This Report)* unclassified 15. NUMBER OF PAGES 16. PRICE

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/CP-0167
CONF-980803

2. TITLE AND SUBTITLE

Proceedings of the 25th DOE/NRC Nuclear Air Cleaning and Treatment Conference

Held in Minneapolis, Minnesota
August 3-6, 1998

3. DATE REPORT PUBLISHED

MONTH	YEAR
April	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

M.W. First, Editor

6. TYPE OF REPORT

Conference Proceeding

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Harvard School of Public Health
665 Huntington Avenue
Boston, MA 02115

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

U.S. Department of Energy
Washington, DC 20585
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

International Society of Nuclear Air Treatment Technologies, Inc.
Batavia, OH 45103
Harvard School of Public Health
Boston, MA 02115

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This report contains papers presented at the 25th DOE/NRC Nuclear Air and Treatment Conference without associated discussions. Major topics are: (1) nuclear air cleaning issues, (2) waste management, (3) nuclear codes and standards, (4) control room safeguards.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

nuclear air cleaning
waste management
nuclear codes and standards
control room safeguards

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5. AUTHOR(S) Conference Papers by various authors; Compiled by Susan Monteleone, BNL	6. TYPE OF REPORT Transactions of conference on safety research 7. PERIOD COVERED *(Inclusive Dates)* October 25-27, 1999

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Same as Item 8 above.

10. SUPPLEMENTARY NOTES

S. Nesmith, NRC Project Manager; Transactions prepared by Brookhaven National Laboratory

11. ABSTRACT *(200 words or less)*

This contains summaries of papers to be presented at the Twenty-Seventh Water Reactor Safety Information Meeting held at the Bethesda Marriott Hotel, Bethesda, Maryland, October 25-27, 1999. The summaries briefly describe the programs and results of nuclear safety research sponsored by the Office of Nuclear Regulatory Research, U.S. NRC. Summaries of invited papers concerning nuclear safety issues from U.S. government laboratories, the electric utilities, the nuclear industry, and from foreign governments and industry are also included. the summaries have been compiled in one report to provide a basis for meaningful discussion and information exchange during the course of the meeting, and are given in the order of their presentation in each session.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*	13. AVAILABILITY STATEMENT
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1. REPORT NUMBER
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NUREG/IA-0151

2. TITLE AND SUBTITLE

Verification of RELAP/MOD3 With Theoretical and Numerical Stability Results on single-Phase, Natural Circulation in a Simple Loop

3. DATE REPORT PUBLISHED

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5. AUTHOR(S)

J.C. Ferreri, ARN
W. Ambrosini, USP

6. TYPE OF REPORT

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9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

The theoretical results given by Pierre Welander are used to test the capability of the RELAP5 series of codes to predict instabilities in single-phase flow. These results are related to the natural circulation in a loop formed by two parallel adiabatic tubes with a point heat sink at the top and a point heat source at the bottom. A stability curve may be defined for laminar flow and was extended to consider turbulent flow. By a suitable selection of the ratio of the total buoyancy force in the loop to the friction resistance, the flow may show instabilities. The solution was useful to test two basic numerical properties of the RELAP5 code, namely: a) convergence to steady state flow-rate using a "lumped parameter" approximation to both the heat source and sink and, b) the effect of nodalization to numerically damp the instabilities. It was shown that, using a simple volume to lump the heat source and sink, it was not possible to reach convergence to steady state flow rate when the heated (cooled) length was diminished and the heat transfer coëfficient increased to keep constant the total heat transferred to (and removed from) the fluid. An algebraic justification of these results is presented, showing that it is a limitation inherent to the numerical scheme adopted. Concerning the effect of nodalization on the damping of instabilities, it was shown that a "reasonably fine" discretization led, as expected, to the damping of the solution. However, the search for convergence of numerical and theoretical results was successful, showing the expected nearly chaotic behavior. This search lead to very refined nodalizations. The results obtained have also been verified by the use of simple, ad hoc codes. A procedure to assess the effects of nodalizations on the prediction of instabilities threshold is outlined in this report. It is based on the experience gained with the aforementioned simpler codes.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

RELAP5/MOD3

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/IA-0152

2. TITLE AND SUBTITLE

RELAP5/MOD3.2 Post Test Analysis and Accuracy Quantification of
Lobi Test BL-34

3. DATE REPORT PUBLISHED

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4. FIN OR GRANT NUMBER

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5. AUTHOR(S)

F.D'Auria, M. Frogheri*, W. Giannotti

6. TYPE OF REPORT

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7. PERIOD COVERED *(Inclusive Dates)*

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

The present document deals with the RELAP5/MOD3.2 analysis of the small break LOCA experiment BL-34 performed in LOBI/MOD2 facility. LOBI/MOD2 was a PWR simulator (Integral Test Facility) installed at JRC Joint Research Center) in Ispra Establishment (I). Volume scaling and core power scaling factors are 1/712, with respect to the KWU Siemens 1300 MWe (3900 MWt) standard reactor. The experiment is originated by a small break in the cold leg (2" equivalent break area in the plant) without the actuation of the high pressure injection system. Low pressure injection system actuation occurs after core dry-out and accumulators intervention is foreseen when primary pressure falls below 4 MPa. The RELAP5 code has been extensively used at University of Pisa; the nodaliztion of LOBI facility has been qualified through the application of the version RELAP5/MOD2 to the same experiment and another test performed in the same facility. Sensitivity analyses have been addressed to the influence of several parameters (like discharge break coefficient, time of accumulators start, etc.) upon the predicted transient evolution. Qualitative and quantitative code calculation accuracy evaluation has been performed.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

RELAP5/MOD3.2
LOCA
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2. TITLE AND SUBTITLE RELAP5/MOD3.2 Post Test Analysis and Accuracy Quantification of Lobi Test BL-44	3. DATE REPORT PUBLISHED MONTH: February YEAR: 1999 4. FIN OR GRANT NUMBER D6227
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U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

The present document deals with the RELAP5/MOD3.2 analysis of the small break LOCA experiment BL-44 performed in LOBI/MOD2 facility. LOBI/MOD2 was a PWR simulator (Integral Test Facility) installed at JRC Joint Research Center) in Ispra Establishment (I). Volume scaling and core power scaling factors are 1/712, with respect to the KWU Siemens 1300 MWe (3900 MWt) standard reactor. The experiment is originated by a small break in the cold leg (2" equivalent break area in the plant) without the actuation of the high pressure injection system. Low pressure injection system actuation occurs after core dry-out and accumulators intervention is foreseen when primary pressure falls below 4 MPa. The RELAP5 code has been extensively used at University of Pisa; the nodaliztion of LOBI facility has been qualified through the application of the version RELAP5/MOD2 to the same experiment and another test performed in the same facility. Sensitivity analyses have been addressed to the influence of several parameters (like discharge break coefficient, time of accumulators start, etc.) upon the predicted transient evolution. Qualitative and quantitative code calculation accuracy evaluation has been performed.

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RELAP5/MOD3.2 LOCA PWR	unlimited
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NUREG/IA-0154

2. TITLE AND SUBTITLE

RELAP5/MOD3.2 Post Test Analysis and Accuracy Quantification of SPES Test SP-SB-03

3. DATE REPORT PUBLISHED

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4. FIN OR GRANT NUMBER

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5. AUTHOR(S)

F.D'Auria, M. Frogheri*, W. Giannotti

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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

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Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

The present document deals with the RELAP5/MOD3.2 analysis of the small break LOCA experiment SP-SB-03 performed in SPES facility. SPES is a PWR simulator (Integral Test Facility) installed at SIET center in Piacenza (IT). Volume scaling and core power scaling factors are 1/427, with respect to the Westinghouse 900 MWe standard reactor. The experiment is originated by a small break in the cold leg (2" equivalent break area in the plant) without the actuation of the high pressure injection system. Low pressure injection system actuation occurs after core dry-out. The RELAP5 code has been extensively used at University of Pisa; the nodaliztion of SPES facility has been qualified through the application of the version RELAP5/MOD2 to the same experiment and another test performed in the same facility. Sensitivity analyses have been addressed to the influence of several parameters (like discharge break coefficient, time of accumulators start, etc.) upon the predicted transient evolution. Qualitative and quantitative code calculation accuracy evaluation has been performed.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

RELAP5/MOD3.2
LOCA
PWR

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2. TITLE AND SUBTITLE

RELAP5/MOD3.2 Post Test Analysis and Accuracy Quantification of SPES Test SP-SB-04

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Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

The present document deals with the RELAP5/MOD3.2 analysis of the small break LOCA experiment SP-SB-04 performed in SPES facility. SPES is a PWR simulator (Integral Test Facility) installed at SIET center in Piacenza (IT). Volume scaling and core power scaling factors are 1/427, with respect to the Westinghouse 900 MWe standard reactor. The experiment is originated by a small break in the cold leg (2" equivalent break area in the plant) without the actuation of the high pressure injection system. The test starts from full power and is the counterpart of the test SP-SB-03, that started at an initial power roughly equal to 10% of nominal power. Low pressure injection system actuation occurs after core dry-out. The RELAP5 code has been extensively used at University of Pisa; the nodaliztion of SPES facility has been qualified through the application of the version RELAP5/MOD2 to the same experiment and another test performed in the same facility. Sensitivity analyses have been addressed to the influence of several parameters (like discharge break coefficient, time of accumulators start, etc.) upon the predicted transient evolution. Qualitative and quantitative code calculation accuracy evaluation has been performed.

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RELAP5/MOD3.2
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PWR

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NUREG/IA-0156
Volume 1

2. TITLE AND SUBTITLE

Data Base on the Behavior of High Burnup Fuel Rods with Zr-1%Nb Cladding and UO2 Fuel (VVER Type) under Reactivity Accident Conditions

Volume 1 Review of Research Program and Analysis of Results

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5. AUTHOR(S)

L. Yegorova

6. TYPE OF REPORT

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9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

The present report contains a data base used for analyzing the behavior of three types of VVER fuel rods (fresh fuel rods; fuel rods with fresh fuel and irradiated cladding; high burnup fuel rods) which have been tested in the IGR under reactivity accident conditions. The basic test parameters are as follows: capsule tests with stagnant water or air coolant under ambient conditions; pressurized fuel rods; fuel burnup: 0 and 48 MWd/kg U; pulse width - about 700 ms.

The presented data base includes the results of reactor tests of 25 fuel rods as well as results of pre- and post-test examinations of fuel rods, computer simulations of fuel rod behavior under test conditions; in addition, the report presents the results of special out-of-pile tests carried out to measure mechanical properties of Zr-1%Nb cladding.

The report consists of three volumes, each volume contains the following information:

Volume 1:Brief description of the test program, testing and analytical techniques and summary of results;

Volume 2:Description and validation of procedures used to obtain the data base. Summarization of test results as supported by mechanical properties of Zr-1%Nb cladding;

Volume 3:Data base consisting of: parameters of VVER fuel rods before and after irradiation at the NovoVoronezh Nuclear Power Plant; parameters of fresh and refabricated fuel rods before and after IGR tests; results of out-of-pile mechanical tests of non-irradiated and irradiated Zr-1%Nb cladding.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

reactivity initiated accident
impulse graphite reactor (Kazakhstan Atomic Energy Agency)
Zirconium-1% Niobium alloy
FRAP-T6 & SCANAIR computer codes
material properties
cladding strain and ballooning
refabrication of VVER fuel rods
post-test examinations

120

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/IA-0156
Volume 2

2. TITLE AND SUBTITLE

Data Base on the Behavior of High Burnup Fuel Rods with Zr-1%Nb Cladding and UO2 Fuel (VVER Type) under Reactivity Accident Conditions

Volume 2 Description of Test Procedures and Analytical Methods

3. DATE REPORT PUBLISHED

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4. FIN OR GRANT NUMBER

W6500

5. AUTHOR(S)

L. Yegorova, V. Asmolov, G. Abyshov, et. al.
A. Bortash, L. Maiorov, et. al.
V. Smirnov, A. Goryachev, V. Prokhorov
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State Research Centre "Research Institute of Atomic Reactors" Dimitrovgrad Russia
Institute of Atomic Energy of NNC Semipalatinsk Kazakhstan

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This volume of the report contains the following information on test conditions and basic procedures used to develop the data base:

- objects and conditions of the IGR reactor tests;
- test parameters of fuel rods before, during and after the RIA tests;
- parameters of fuel rods during RIA tests calculated by FRAP-T6 and SCANAIR computer codes;
- measured mechanical properties of Zr-1%Nb cladding obtained due to special tests;
- parameters of Zr-1%Nb cladding failure of the ballooning type measured under the burst test conditions;
- input data with original material properties of Zr-1%Nb cladding for the MATPRO package and SCANAIR code.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

reactivity initiated accident
impulse graphite reactor (Kazakhstan Atomic Energy Agency)
Zirconium-1% Niobium alloy
FRAP-T6 & SCANAIR computer codes
material properties
cladding strain and ballooning
energy deposition
post-test examinations

13. AVAILABILITY STATEMENT

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2. TITLE AND SUBTITLE Data Base on the Behavior of High Burnup Fuel Rods with Zr-1%Nb Cladding and UO2 Fuel (VVER Type) under Reactivity Accident Conditions Volume 3 Test and Calculation Results	3. DATE REPORT PUBLISHED MONTH: July YEAR: 1999 4. FIN OR GRANT NUMBER W6500
5. AUTHOR(S) L. Yegorova, G. Abyshov, V. Malofeev, et. al. A. Bortash, M. Kalugin, et. al. V. Smirnoiv, A. Goryachev, V. Prokhorov, et. al. V. Pakhnitz, A. Vurim	6. TYPE OF REPORT IPSN 99/08-2 NSI RRC KI-2179 7. PERIOD COVERED *(Inclusive Dates)*

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Russian Research Center "Kurchatov Institute" Moscow 123182 Russia
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Institute of Atomic Energy of NNC Semipalatinsk Kazakhstan

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U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

The present report contains a data base used for analyzing the behavior of three types of VVER fuel rods (fresh fuel rods; fuel rods with fresh fuel and irradiated cladding; high burnup fuel rods) which have been tested in the IGR under reactivity accident conditions. The basic test parameters are as follows: capsule tests with stagnant water or air coolant under ambient conditions; pressurized fuel rods; fuel burnup: 0 and 48 MWd/kg U; pulse width - about 700 ms.
The presented data base includes the results of reactor tests of 25 fuel rods as well as results of pre- and post-test examinations of fuel rods, computer simulations of fuel rod behavior under test conditions; in addition, the report presents the results of special out-of-pile tests carried out to measure mechanical properties of Zr-1%Nb cladding.
The report consists of three volumes, each volume contains the following information:
Volume 1:Brief description of the test program, testing and analytical techniques and summary of results;
Volume 2:Description and validation of procedures used to obtain the data base. Summarization of test results as supported by mechanical properties of Zr-1%Nb cladding;
Volume 3:Data base consisting of: parameters of VVER fuel rods before and after irradiation at the NovoVoronezh Nuclear Power Plant; parameters of fresh and refabricated fuel rods before and after IGR tests; results of out-of-pile mechanical tests of non-irradiated and irradiated Zr-1%Nb cladding.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*	13. AVAILABILITY STATEMENT
reactivity initiated accident impulse graphite reactor (Kazakhstan Atomic Energy Agency) Zirconium-1% Niobium alloy FRAP-T6 & SCANAIR computer codes material properties cladding strain and ballooning refabrication of VVER fuel rods post-test examinations	unlimited 14. SECURITY CLASSIFICATION *(This Page)* unclassified *(This Report)* unclassified 15. NUMBER OF PAGES 122 16. PRICE

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Contrast of RELAP5/MOD3.2 Results From Different Computing Platforms

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5. AUTHOR(S)

A. López, CNA
J. M. Sierra, CDI

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Control Data Ibérica, SA
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Spain

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Office of Nuclear Regulatory Research
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Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

T. Lee, NRC Project Manager

11. ABSTRACT *(200 words or less)*

RELAP5/MOD3.2 was installed in all major UNIX workstations that include DEC, HP, IBM, SGI and SUN, besides that of CONVEX, and a common sample case was run on each of them. When these results were analyzed, substantial differences were observed in the magnitudes of selected parameters such as; pressure, mass flow, temperature, etc. The level of optimization adopted for compilation, the chosen maximum time step size and the inherent precision of the platforms under consideration are identified as possible causes for the discrepancies. Those discrepancies, however, lied within acceptable limits when results of different machines were produced by non-optimized version of RELAP5 and a maximum time step of about 0.125 seconds was used. To the users, it is recommended that a test be run whenever a new version of the code, operation system and/or compiler level is installed. The code developer needs to deliver a code which is compliant with ANSI standard for FORTRAN. The developer is also urged to evaluate the maximum time step size and provide a (or a set of) test to be used as an acceptance/rejection criterion for the new version of RELAP5.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

RELAP5
Different Platforms

13. AVAILABILITY STATEMENT

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14. SECURITY CLASSIFICATION

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NUREG/IA-0162

2. TITLE AND SUBTITLE

Test LOBI-BL06: Post-Test Analysis and RELAP5/MOD3.2.1 Code Performance Assessment

3. DATE REPORT PUBLISHED

MONTH	YEAR
April	1999

4. FIN OR GRANT NUMBER

D6227

5. AUTHOR(S)

T. Fiore, P. Marsili

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Agenzia Nazionale per la Protezione dell' Ambiete (ANPA)
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Office of Nuclear Regulatory Research
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Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

G. Rhee, NRC Project Manager

11. ABSTRACT *(200 words or less)*

This report deals with the results of the "post-test analysis" of the test BL-06 performed in the LOBI/MOD2 test facility. LOBI/MOD2 is an integral test facility that represents, at approximately 1:712 scale, a four loop (KWU design, 1300 MWe) PWR. The test BL-06 simulates a 1% cold leg break LOCA, with the main coolant pumps switched off very late in the transient. The calculations have been realized with the code RELAP5/MOD3.2.1. The uncertainty evaluation of the calculation result has been performed using a specific method developed by Pisa University.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

LOBI Test BL-06
Cold Leg SBLOCA
RELAP5/MOD3.2.1 Assessment
LOBI/MOD2 Facility
KWU 4 Loop Design

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

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NUREG/IA-0163

2. TITLE AND SUBTITLE

A Study of the Dispersed Flow Interfacial Heat Transfer Model of RELAP5/MOD2.5 and RELAP5/MOD3

3. DATE REPORT PUBLISHED

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5. AUTHOR(S)

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G. Th. Analytis
S. N. Aksan

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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

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*ETHZ, Nuclear Engineering Laboratory
Swiss Federal Institute of Technology
8092 Zurich Switzerland

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

G. Rhee, NRC Project Manager

11. ABSTRACT *(200 words or less)*

For conditions of low quality at the quench front and low mass flux, the interfacial heat transfer calculated assuming a uniform distribution of droplets over the cross-sectional area of the channel is necessarily overpredicted, and the vapour superheat is strongly underpredicted. The purpose of the present paper is to show that limitations of the 1-D approach, obtained from the steady-state analyses of slow reflooding experiments, has some impact on the performance of the 1-D transient computer codes like RELAP5/MOD2.5 and RELAP5/MOD3. As an example, the transient analysis of a low flooding rate experiment in a tube was performed. An early completion of the quench process and a fast desuperheating of the vapour at the tube exit was obtained by both codes. The too high quench front velocity (four times higher than in the experiment) could not, however, be put univocally in relation to the underprediction of the vapour temperature, and the consequent increase of the precursory cooling, as many coupled thermal and hydraulic transient effects prevailed. Quasi steady-state analyses of two runs, where the boundary conditions for the post-dryout region could be better controlled for a predetermined position of the quench front, were thus performed. These analyses show that the vapour superheat at the tube exit is strongly underpredicted, confirming the limitations of the 1-D model for interfacial heat transfer in the dispersed flow region.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Assessment of RELAP5, Reflood Heat Transfer
Dispersed Flow, Post-CHF Heat Transfer
Interfacial Heat Transfer Model
RELAP5/MOD2.5
RELAP5/MOD3

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NUREG/IA-0164
RRC KI 2180
IPSN 99/10

2. TITLE AND SUBTITLE

Modification of USNRC's FRAP-T6 Fuel Rod Transient Code for High Burnup VVER Fuel

3. DATE REPORT PUBLISHED

MONTH	YEAR
May	1999

4. FIN OR GRANT NUMBER

W6500

5. AUTHOR(S)

A. Shestopalov, K. Lioutov, L. Yegorova, G. Abyshov, K. Miktiouk

6. TYPE OF REPORT

Technical

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1995-1998

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Russian Research Center - Kurchatov Institute
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Moscow 12382
Russia

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

The USNRC's transient fuel rod code, FRAP-T6, has been modified to analyze pulse tests with high burnup VVER fuel rods in the Impulse Graphite Reactor (IGR). New and modified models of separate phenomena have been developed, including models for heat transfer from the cladding to the stagnant coolant, the effect of fission gas swelling on the fuel cladding gap, and the conditions of base irradiation. Thermal and mechanical properties for the VVER's Zr-1%Nb cladding were added to MATPRO-V11, which is used by the FRAP-T6 code. Changes in the input data file are described and a sample calculation is presented with the modified code. A FORTRAN listing for the new and modified models is given in an Appendix A.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

reactivity initiated accident
impulse graphite reactor
cladding-to-coolant heat transfer model
FRAP-T6 computer code
material properties
cladding strain and ballooning

13. AVAILABILITY STATEMENT

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14. SECURITY CLASSIFICATION

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NUREG/IA-0165
RRC KI 2181
IPSN 99/09

2. TITLE AND SUBTITLE

Modification of IPSN's SCANAIR Fuel Rod Transient Code for High Burnup VVER Fuel

3. DATE REPORT PUBLISHED

MONTH	YEAR
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4. FIN OR GRANT NUMBER

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5. AUTHOR(S)

K. Mikitiouk, A. Shestopalov, K. Lioutov, L. Yegorova, G. Abyshov

6. TYPE OF REPORT

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Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

The IPSN's transient fuel rod code SCANAIR, has been modified to analyze pulse tests with high-burnup VVER fuel rods in the Impulse Graphite Reactor (IGR). New and modified models of separate phenomena have been developed, including models for gas plenum temperature, heat transfer from the cladding to the stagnant coolant, the effect of the cladding strain rate on the yield stress, and interruption of the cladding mechanical calculation. Thermal and mechanical properties for the VVER's fuel and Zr-1%Nb cladding were added to the SCANAIR data base on material properties. Changes in the input data file are described, and a sample calculation is presented with the modified code. A FORTRAN listing for the new and modified models is given in an Appendix A.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

reactivity initiated accident
impulse graphite reactor
SCANAIR computer code
material properties
cladding-to-cooolant heat transfer model
cladding strain and ballooning

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NUREG/IA-0166

2. TITLE AND SUBTITLE

RELAP5/MOD3.2 Assessment Using GERDA Small Break Test, 1605AA

3. DATE REPORT PUBLISHED

MONTH	YEAR
July	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

W. Tietsch

6. TYPE OF REPORT

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Division of Systems Technology
Office of Nuclear Reactor Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This report presents the results of RELAP5/MOD3.2 assessment on the integral (composite) GERDA Test 1605 AA which features a complete sequence of a 10 cm2 Reactor Vessel Leak transient conducted on April 12, 1983. The purpose of the assessment analysis was to determine whether the tested RELAP version can predict the major phenomena of this complex transient and to provide some useful information both to code developers and analysts for the application on SBLOCA transients in 'Once-Through-Steam-Generator' Plants. The Small Break Loss of Coolant Accident (SBLOCA) Test Facility GERDA (Geradrohr Dampferzeuger Anlage) was designed to evaluate the post SBLOCA thermal-hydraulic events expected to occur in the German Mulheim-Karlich plant. The scaled in volume 1:1 in height Test Facility was built and tested by B&W at the Alliance Research Centre (ARC) under contract of BBR (Brown Boveri Reaktor GmbH) now ABB Reaktor GmbH. The objective of the whole test program was to obtain detailed experimental data for the evaluation of single and composite SBLOCA phenomena and for the verification and refinement of the analytical tools and models used to predict plant performance during SBLOCA transients. Additionally, this report presents some base calculations identified as conditioning calculations of OTSG stready state and transient behaviour which is one of the essentials for the prediction of phenomena observed in the SBLOCA scenarios.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

RELAP5/MOD3.2
GERDA
Small Break Loss of Coolant Accident

13. AVAILABILITY STATEMENT

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14. SECURITY CLASSIFICATION

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NUREG/IA-0167

2. TITLE AND SUBTITLE

Assessment Study of RELAP5/MOD3.2 Based on the Kalinin NPP Unit - 1 Stop of Feedwater Supply to the Steam Generator No. 4

3. DATE REPORT PUBLISHED

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October	1999

4. FIN OR GRANT NUMBER

W6500

5. AUTHOR(S)

S. S. Pylev

6. TYPE OF REPORT

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Russian Research Center - Kurchatov Institute
Kurchatov Square 1
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Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This report has been prepared as a part of the Agreement on Research Participation and Technical Exchange under the Code Assessment and Maintenance Program. The data collected by Kalinin NPP Unit-1 Data Acquisition System and In-Vessel Monitoring System during stop of feedwater supply on the SG-4 transient have been analyzed by using the RELAP5/MOD3.2 and RELAP5/MOD3.2.2 Beta codes. Kalinin NPP-1 is a Russian designed four loop pressurized water reactor (VVER-1000, project V-338) rated at 1000MWe. RELAP5 code calculation results were compared with plant data. Sensitivity studies were carried out to investigate the effects of modeling on major thermal-hydraulic parameters and to examine a new heat transfer model for horizontal tube bundles.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

RELAP5
Stean Generator
Assessment
Kalinin

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NUREG/IA-0168

2. TITLE AND SUBTITLE

Assessment of RELAP5/MOD3.2 for Thermohydraulic Processes in Heated Rod Bundles With Tight Lattice at CKTI Test Facility

3. DATE REPORT PUBLISHED

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October	1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

A.S. Devkin

6. TYPE OF REPORT

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Nuclear Safety Institute
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123182, Moscow
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9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of System Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This report presents the results of RELAP5/MOD3.2 assessment in the prediction of two-phase hydrodynamics and heat transfer in rod bundle model with tight lattice. The experiments have been carried out at the CKTI (St. Petersburg) test facility. The peculiarities of these researches were the non-standard geometrical characteristics of the 55-rod bundle - close packed assembly and small hydraulic diameters, and also the parameters - small flow rates (down to zero) and low pressure (from 0.23 up to 2.0 MPa). The assessment of RELAP5/MOD3.2 code was done for two cases: for steady state conditions at small or zero flowrates at the inlet of the rod bundle and for boil-off and reflooding processes. The comparison of calculated results with experimental ones shows that there was a good coordination between computed and experimental values of void fractions and bundle wall temperatures for almost all the tests.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

RELAP5
Rod Bundles
CKTI Test Facility
Assessment

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14. SECURITY CLASSIFICATION

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2. TITLE AND SUBTITLE Analysis of the KS-1 Experimental Data on the Behavior of the Heated Rod Temperatures in the Partially Uncovered VVER Core Model Using RELAP5/MOD3.2	3. DATE REPORT PUBLISHED MONTH: November — YEAR: 1999 4. FIN OR GRANT NUMBER
5. AUTHOR(S) V.A. Vinogradov, A.Y. Balykin	6. TYPE OF REPORT Technical 7. PERIOD COVERED *(Inclusive Dates)*

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Nuclear Safety Institute
Russian Research Center "Kurchatov Institute"
123182, Moscow
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9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of System Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This report has been prepared as a part of the Agreement on Research Participation and Technical Exchange under the International code application and Maintenance Program. KS-1 Test 35-1 data on the behavior of the heated rod temperatures in the partially uncovered VVER Core model were simulated with RELAP5/MOD3.2 to assess the code, especially its non-equilibrium (unequal phase temperatures) heat t transfer models for modeling phenomena in partially uncovered core under Small Break LOCA conditions. The test has been carried out at experimental section KS-1 of the test facility KS (RRC KI) in 1991. KS-1 experimental section (VVER Loop model) includes models of all main elements of VVER type reactor, loop hot leg model and cold leg simulator, and also horizontal SG tube bundle simulator with passive heat removal. Core model consists of 19 electrically heated rod simulators with diameter 9 mm and height 2.5m. Test 35-1 models thermal and hydraulic processes during reflux condenser mode in primary circuit with low mixture level in partially uncovered VVER core under conditions of small residual heat power, middle pressure and counter current flow in the core. First a study of the effect of the hydraulic nodalization to the code calculations was performed using different number of hydraulic volumes for Core model. after the choice of proper nodalization and maximum user-specified time step, base case calculations were done for the test. The differences between code predictions for behavior of rod simulator temperatures along the height of Core model and test data are described and analyzed. Sensitivity studies were carried out to investigate the effects of modeling on the behavior of the rod simulator temperatures along the height of Core model.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*	13. AVAILABILITY STATEMENT unlimited
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NUREG/IA-0170

2. TITLE AND SUBTITLE

RELAP5/MOD3.2 Post Test Calculation of the PKL-Experiment
PKLIII-B4.3

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4. FIN OR GRANT NUMBER

5. AUTHOR(S)

L. Karner

6. TYPE OF REPORT

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9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

Division of System Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

The PKL III test facility (Primar-Kreis-Lauf) simulates a typical 1300 MWe Pressurized Water Reactor of Siemens/KWU. In test B4.3, the influence of non-condensables on heat transfer in the steam generators during reflux condenser conditions was investigated. This report presents the results of a post-test analysis of PKL III-B4.3 using RELAP5/MOD 3.2. A description of the input model is given, and the correspondence of measured and calculated results is discussed.. The results of the calculation show differences in the distribution of nitrogen in the primary system compared to the experiment. When nitrogen was injected into the hot leg of the primary system, the heat transfer in the affected steam generator decreased. In contrast to the experiment RELAP calculated that the volumes in the adjacent steam generator did not get the full amount of nitrogen and that nitrogen was transported from the steam generators to other locations during the course of the transient. Thus, the heat transfer in these steam generators later increased in contradiction to the measured values. In the steam generator tubes, RELAP calculated that the nitrogen accumulated in the descending part as predicted in the experiment. For the ascending U-tubes RELAP predicted that the nitrogen was transported to the loop seal, which was not seen in the experiment. Fluctuations occurred during the course of the RELAP5/MOD 3.2 analysis of the PKLIII B4.3 experiment. This phenomenon may be the main reason, that RELAP calculates the transport of nitrogen from the steam generators into the system and predicts finally a homogeneous distribution of nitrogen in the primary system. The analysis performs an in kind contribution to the CAMP contract.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

PKL III Test Facility
PWR
non-condensables
steam-generator
RELAP5/MOD3.2

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

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NUREG/CR-4667, Vol. 26
ANL-98/30

2. TITLE AND SUBTITLE
Environmentally Assisted Cracking in Light Water Reactors
Semiannual Report January 1998—June 1998

3. DATE REPORT PUBLISHED

MONTH	YEAR
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5. AUTHOR(S)
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W. K. Soppet, R. V. Strain, and C. L. Trybus/ANL

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9700 South Cass Avenue
Argonne, IL 60439

Subcontractor:
OECD Halden Reactor Project
Halden N-1751, Norway

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Division of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES
M. B. McNeil, NRC Project Manager

11. ABSTRACT (200 words or less)
This report summarizes work performed by Argonne National Laboratory on fatigue and environmentally assisted cracking (EAC) in light water reactors from January 1998 to June 1998. Topics that have been investigated include (a) fatigue of austenitic stainless steels (SSs), (b) irradiation-assisted stress corrosion cracking of austenitic SSs, and (c) EAC of Alloys 600 and 690. Fatigue tests were conducted on cast austenitic SSs to establish the effects of light water reactor (LWR) environments on the fatigue life of these steels. Slow-strain-rate-tensile tests were conducted in simulated boiling water reactor (BWR) water at 288°C on SS specimens irradiated to a low and medium fluence in the Halden reactor, and the results were compared with similar data from a control-blade sheath and neutron-absorber tubes irradiated in BWRs to the same fluence levels. Crack-growth-rate (CGR) tests are being conducted on compact-tension specimens from several heats of Alloys 600 and 690 to evaluate the resistance of these alloys to environmentally assisted cracking in LWR environments. CGR correlations were developed as a function of loading and environmental parameters.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating this report.)*
Corrosion Fatigue
Crack Growth
Irradiation-Assisted Stress Corrosion Cracking
Radiation-Induced Segregation
Stress Corrosion Cracking
Types 304, 304L, 316, and 316NG Stainless Steel
Alloys 600 and 690

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1. REPORT NUMBER
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ANL-99/11

2. TITLE AND SUBTITLE
Environmentally Assisted Cracking in Light Water Reactors.
Semiannual Report July 1998—December 1998

3. DATE REPORT PUBLISHED

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4. FIN OR GRANT NUMBER
W6610

5. AUTHOR(S)
O. K. Chopra, H. M. Chung, E. E. Gruber,
T. F. Kassner, W. E. Ruther, W. J. Shack,
J. L. Smith, W. K. Soppet, and R. V. Strain

6. TYPE OF REPORT
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7. PERIOD COVERED *(Inclusive Dates)*
July 1998-December 1998

8. PERFORMING ORGANIZATION – NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Argonne National Laboratory
9700 South Cass Avenue
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9. SPONSORING ORGANIZATION – NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES
M. B. McNeil, NRC Project Manager

11. ABSTRACT (200 words or less)

This report summarizes work performed by Argonne National Laboratory on fatigue and environmentally assisted cracking (EAC) in light water reactors from July 1998 to December 1998. Topics that have been investigated include (a) environmental effects on fatigue S–N behavior of primary pressure boundary materials, (b) irradiation-assisted stress corrosion cracking of austenitic stainless steels (SSs), and (c) EAC of Alloys 600 and 690. Fatigue tests have been conducted to determine the crack initiation and crack growth characteristics of austenitic SSs in LWR environments. Procedures are presented for incorporating the effects of reactor coolant environments on the fatigue life of pressure vessel and piping steels. Slow-strain-rate tensile tests and posttest fractographic analyses were conducted on several model SS alloys that were irradiated to ≈0.3 and 0.9 x 10^{21} n·cm^{-2} (E > 1 MeV) in helium at 289°C in the Halden reactor. The results have been used to determine the influence of alloying and impurity elements on the susceptibility of these steels to irradiation assisted stress corrosion cracking. Fracture toughness J–R curve tests were also conducted on two heats of Type 304 SS that were irradiated to ≈0.3 x 10^{21} n·cm^{-2} in the Halden reactor. Crack-growth-rate tests have been conducted on compact-tension specimens of Alloys 600 and 690 under constant load to evaluate the resistance of these alloys to stress corrosion cracking in LWR environments.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating this report.)*
Corrosion Fatigue
Crack Growth
Irradiation-Assisted Stress Corrosion Cracking
Radiation-Induced Segregation
Stress Corrosion Cracking
Types 304, 304L, 316, and 316NG Stainless Steel
Alloys 600 and 690

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1. REPORT NUMBER *(Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)*

NUREG/CR-5362, Vol. 1

2. TITLE AND SUBTITLE

Discrete Event Simulation as a Risk Analysis Tool for
Remote Afterloading Brachytherapy

Main Report

3. DATE REPORT PUBLISHED

MONTH	YEAR
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4. FIN OR GRANT NUMBER

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5. AUTHOR(S)

R. Archer, J. Keller, S. Archer, S. Scott-Nash

6. TYPE OF REPORT

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

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9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Systems Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

P. M. Lewis, NRC Project Manager

11. ABSTRACT *(200 words or less)*

Remote Afterloading Brachytherapy (RAB) is a cancer treatment process that uses ionizing radiation from radioactive materials to retard or destroy tumors. RAB uses a computer-controlled device to insert radioactive sources in or adjacent to the tissue to be exposed. Human errors and equipment failures in the RAB process can lead to the administration of incorrect radiation doses, the insertion of radioactive sources into the incorrect location in the body, or a treatment delivered to the wrong patient.

Discrete Event Simulation (DES) is a technique for simulating and studying processes that can be described by a sequence of events that each have a distinct beginning and end. It is particularly well suited to modeling series of human tasks. This study used DES modeling to evaluate the potential for human errors in the process and responses made by humans to hardware and software failures. Using a task analysis of RAB combined with estimates of the probability of the errors that can occur in the RAB process, the events (previous tasks and errors) that lead to the errors, and information about potential equipment failures, a DES model of the process was developed. This project demonstrated that the DES model can be used to examine process safety through the identification of the human errors that are most likely to occur and the points in the process most vulnerable to severe consequences.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

remote afterloading brachytherapy
radiation medicine
human error
human factors
Task Network Modeling
Discrete Event Simulation
safety
ionizing radiation

13. AVAILABILITY STATEMENT

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14. SECURITY CLASSIFICATION

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Division of Systems Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

P. M. Lewis, NRC Project Manager

11. ABSTRACT *(200 words or less)*

Remote Afterloading Brachytherapy (RAB) is a cancer treatment process that uses ionizing radiation from radioactive materials to retard or destroy tumors. RAB uses a computer-controlled device to insert radioactive sources in or adjacent to the tissue to be exposed. Human errors and equipment failures in the RAB process can lead to the administration of incorrect radiation doses, the insertion of radioactive sources into the incorrect location in the body, or a treatment delivered to the wrong patient.

Discrete Event Simulation (DES) is a technique for simulating and studying processes that can be described by a sequence of events that each have a distinct beginning and end. It is particularly well suited to modeling series of human tasks. This study used DES modeling to evaluate the potential for human errors in the process and responses made by humans to hardware and software failures. Using a task analysis of RAB combined with estimates of the probability of the errors that can occur in the RAB process, the events (previous tasks and errors) that lead to the errors, and information about potential equipment failures, a DES model of the process was developed. This project demonstrated that the DES model can be used to examine process safety through the identification of the human errors that are most likely to occur and the points in the process most vulnerable to severe consequences.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*	13. AVAILABILITY STATEMENT
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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/CR-5392
SCIE-NRC-373-98

2. TITLE AND SUBTITLE

Elements of an Approach to Performance-Based Regulatory Oversight

3. DATE REPORT PUBLISHED

MONTH	YEAR
January	1999

4. FIN OR GRANT NUMBER

J6040

5. AUTHOR(S)

R.W. Youngblood, R.N.M. Hunt, E.R. Schmidt,
J. Bolin, F. Dombek, D. Prochnow

6. TYPE OF REPORT

Technical

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

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11140 Rockville Pike
Rockville, MD 20852

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Regulatory Applications
Office of Nuclear Regulatory Research
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Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

N.P. Kadambi, NRC Project Manager. available in paper and CD

11. ABSTRACT *(200 words or less)*

This report discusses an approach to performance-based regulatory oversight. One key issue in developing a performance-based approach is choosing a collection of performance measures that is highly results-oriented, and will support the capability to detect and act upon emerging performance problems before they lead to adverse consequences. A related issue is the role of institutional factors, and how to reflect institutional factors in a results-oriented, performance-based approach. These issues are explored through discussion of examples. Based on these discussions, an approach is recommended. The approach entails (1) careful formulation of a safety case, which shows what the challenges are to plant safety and what the plant capability is for responding to those challenges, (2) allocation of performance goals over elements of the safety case, (3) formulation of a "diamond tree," which is an integrated, hierarchical presentation of hardware, human, and institutional performance areas that indicates how institutional performance supports the safety case, and (4) application of the diamond tree to select a set of performance measures that is as results-oriented as possible, given the levels and kinds of performance needed in order to support the safety case, and the need to respond to emergent problems before adverse consequences develop.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

performance-based regulation
Probabilistic Risk Analysis
Probabilistic Safety Analysis
diamond tree
risk-informed regulation

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13. AVAILABILITY STATEMENT

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/CR-5452
SCIE-NRC-350

2. TITLE AND SUBTITLE

Demonstration of Pressurized Thermal Shock Thermal-Hydraulic Analysis With Uncertainty

RELAP5/MOD3 and TRAC-PF1/MOD2 Calculations in Support of Regulatory Guide 1.154 Revisions

3. DATE REPORT PUBLISHED

MONTH	YEAR
March	1999

4. FIN OR GRANT NUMBER

W7630

5. AUTHOR(S)

D. Palmrose

6. TYPE OF REPORT

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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

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11140 Rockville Pike, Suite 500
Rockville, MD 20852

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Systems Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

C. Boyd, NRC Project Manager

11. ABSTRACT *(200 words or less)*

The development of an improved pressurized thermal shock thermal-hydraulic methodology is presented. The improved methodology uses the latest capabilities of thermal-hydraulic codes to model plant-specific features, two and three-dimensional component models, guidance on treating cold leg stratification, and a modified code scaling, assessment, and uncertainty (CSAU) methodology to determine the bounds in uncertainty of key thermal-hydraulic parameters. This study uses the same thermal-hydraulic code packages (RELAP5 and TRAC-PF1), cold leg thermal stratification code (REMIX) and case plant (H. B. Robinson Unit 2) that was used in integrated PTS analyzes of the 1980's. Several key items were found to be important to the analytical results. These items are: 1) accurate modeling of plant-specific features which influence the effect of PTS; 2) the accurate modeling of choked or critical flow and multi-dimensional flow patterns in the downcomer; 3) establishment of a specific criteria for cold leg stratification in the integral system code to determine when to use a cold leg thermal stratification code (i.e., REMIX): 4) integration of the system code results (RELAP5) with the downcomer plume results from REMIX; and 5) bounding the uncertainty by concentrating on a small set of key PTS parameters.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Pressurized Thermal Shock
Thermal-Hydraulic Codes
Thermal-Hydraulic Analysis
Cold Leg Stratification
Uncertainty
Multi-Dimensional Flow Patterns

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INEEL/EXT-97-00740

2. TITLE AND SUBTITLE
Reliability Study: Westinghouse Reactor Protection System, 1984 – 1995

3. DATE REPORT PUBLISHED

MONTH	YEAR
April	1999

4. FIN OR GRANT NUMBER
E8246

5. AUTHOR(S)
S. A. Eide, S. T. Beck, M. B. Calley, W. J. Galyean, C. D. Gentillon, S. T. Khericha, S. D. Novack, T. E. Wierman

6. TYPE OF REPORT
Technical

7. PERIOD COVERED (Inclusive Dates)
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8. PERFORMING ORGANIZAITON - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)
Idaho National Engineering and Environmental Laboratory
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9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above"; If contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)
Safety Programs Division
Office for Analysis and Evaluation of Operational Data
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)
This report documents an analysis of the safety-related performance of the reactor protection system (RPS) at U.S. Westinghouse commercial reactors during the period 1984 through 1995. Westinghouse RPS designs analyzed in this report include those with solid state protection system trains and Analog Series 7300 or Eagle-21 channels. The analysis is based on a four-loop plant design. RPS operational data were collected for all U.S. Westinghouse commercial reactors from the Nuclear Plant Reliability Data System and Licensee Event Reports. A risk-based analysis was performed on the data to estimate the observed unavailability of the RPS, based on a fault tree model of the system. An engineering analysis of trends and patterns was also performed on the data to provide additional insights into RPS performance. RPS unavailability results obtained from the data were compared with existing unavailability estimates from Individual Plant Examinations and other reports.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)
Reactor protection system, RPS, PWR, RPS operational events, probabilistic risk assessments, plant evaluations, system unreliability

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2. TITLE AND SUBTITLE

Reliability Study: General Electric Reactor Protection System, 1984-1995

3. DATE REPORT PUBLISHED

MONTH	YEAR
May	1999

4. FIN OR GRANT NUMBER

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5. AUTHOR(S)

S.A. Eide, S.T. Beck, M.B. Calley, W.J. Galyean, C.D. Gentillon, S.T. Khericha, T.E. Wierman

6. TYPE OF REPORT

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7. PERIOD COVERED *(Inclusive Dates)*

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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Idaho National Engineering and Environmental Laboratory
Lockheed Martin Idaho Technologies Company
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9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Safety Programs Division
Office for Analysis and Evaluation of Operational Data
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This report documents an analysis of the safety-related performance of the reactor protection system (RPS) at U.S. General Electric commercial reactors during the period 1984 through 1995. General Electric RPS designs analyzed in this report include those with relay-based trip systems. The analysis is based on a BWR/4 plant design. RPS operational data were collected for all U.S. General Electric commercial reactors from the Nuclear Plant Reliability Data System and Licensee Event Reports. A risk-based analysis was performed on the data to estimate the observed unavailability of the RPS, based on a fault tree model of the system. An engineering analysis of trends and patterns was also performed on the data to provide additional insights into RPS performance. RPS unavailability results obtained from the data were compared with existing unavailability estimates from Individual Plant Examinations and other reports.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Reactor protection system, RPS, BWR, RPS operational events, probabilistic risk assessments, plant evaluations, system unreliability

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2. TITLE AND SUBTITLE

Reliability Study:
High-Pressure Coolant Injection System, 1987-1993

3. DATE REPORT PUBLISHED

MONTH	YEAR
September	1999

4. FIN OR GRANT NUMBER
E8246

5. AUTHOR(S)

G.M. Grant, W.S. Roesener, D.G. Hall, C.L. Atwood, C.D. Gentillon, T.R. Wolf

6. TYPE OF REPORT
Technical

7. PERIOD COVERED *(Inclusive Dates)*
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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

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9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Risk Analysis & Applications
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U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES
T. R. Wolf, NRC Project Manager

11. ABSTRACT *(200 words or less)*

This report documents an analysis of the safety-related performance of the high-pressure coolant injection (HPCI) system at U.S. commercial boiling water reactor plants during the period 1987-1993. Both a risk-based analysis and an engineering analysis of trends and patterns were performed on data from HPCI system operational events to provide insights into the performance of the HPCI system throughout the industry and at a plant-specific level. Comparison was made to Probabilistic Risk Assessment/Individual Plant Evaluations for 23 plants to indicate where operational data either support or fail to support the assumptions, models, and data used to develop HPCI system unreliability.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

High-Pressure Coolant Injection, HPCI, Boiling Water Reactors, BWR, High-Pressure Coolant Injection Operational Events, Probabilistic Risk Assessment, PRA, Plant Evaluation, High-Pressure Coolant Injection System Unreliability

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NUREG/CR-5500, Volume 5

2. TITLE AND SUBTITLE

Reliability Study:
Emergency Diesel Generator Power System, 1987-1993

3. DATE REPORT PUBLISHED

MONTH	YEAR
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4. FIN OR GRANT NUMBER

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5. AUTHOR(S)

G.M. Grant, J.P. Poloski, A.J. Luptak C.D. Gentillon, W.J. Galyean

6. TYPE OF REPORT

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7. PERIOD COVERED *(Inclusive Dates)*

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Division of Risk Analysis & Applications
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

T. R. Wolf, NRC Project Manager

11. ABSTRACT *(200 words or less)*

This report documents an analysis of the reliability of emergency diesel generator (EDG) power systems at U.S. commercial nuclear plants during the period 1987-1993. To evaluate EDG power system performance, estimates are given of individual EDG train reliability to supply emergency as power to the safety-related bus. The estimates are based on EDG train performance data that would be typical of an actual response to a low-voltage condition on a safety-related bus for averting a station blackout event. A risk-based analysis and an engineering analysis of trends and patterns are performed on data from EDG operational events to provide insights into the reliability performance of EDGs throughout the industry and at a plant-specific level. Comparisons are made to EDG train statistics from Probabilistic Risk Assessments, Individual Plant Examinations, and NUREG reports, representing 40% of the U.S. commercial nuclear power plants. In addition, EDG train reliability estimates and associated uncertainty intervals are compared to station blackout target reliability goals.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Emergency Diesel Generator Power System, EDG, Boiling Water Reactor, BWR, Pressurized Water Reactor, PWR, Emergency Diesel Generator Power System Operational Events, Probabilistic Risk Assessment, PRA, Plant Evaluation, Emergency Diesel Generator Power System Unreliability

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NUREG/CR-5500, Volume 6

2. TITLE AND SUBTITLE

Reliability Study:
Isolation Condenser System, 1987-1993

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5. AUTHOR(S)

G.M. Grant, J.P. Poloski, C.D. Gentillon, W.J. Galyean

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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Idaho National Engineering and Environmental Laboratory
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9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Risk Analysis & Applications
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

T. R. Wolf, NRC Project Manager

11. ABSTRACT *(200 words or less)*

This report documents an analysis of the safety-related performance of the isolation condenser systems at U.S. commercial boiling water reactor plants during the period 1987-1993. Both a risk-based analysis and an engineering analysis of trends and patterns were performed on data from isolation condenser system operational events to provide insights into the performance of the system throughout the industry and at a plant-specific level. Comparisons were made to Probabilistic Risk Assessments and Individual Plant Evaluations for all the plants that have an isolation condenser system to indicate where operational data either support or fail to support the assumptions, models, and data used to develop system unreliability.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Isolation Condenser System, IC, Boiling Water Reactor, BWR, Isolation Condenser System Operational Events, Probabilistic Risk Assessment, PRA, Plant Evaluation, Isolation Condenser System Unreliability

13. AVAILABILITY STATEMENT

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14. SECURITY CLASSIFICATION

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NRC FORM 335 (2-89) NRCM 1102, 3201, 3202	U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET** *(See instructions on the reverse)*	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.) NUREG/CR-5500, Volume 7
2. TITLE AND SUBTITLE	Reliability Study: Reactor Core Isolation Cooling System, 1987-1993	3. DATE REPORT PUBLISHED — MONTH: September; YEAR: 1999
		4. FIN OR GRANT NUMBER: E8246
5. AUTHOR(S)	J.P. Poloski, G.M. Grant, C.D. Gentillon, W.J. Galyean, W.S. Roesener	6. TYPE OF REPORT: Technical
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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

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9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Risk Analysis & Applications
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

T. R. Wolf, NRC Project Manager

11. ABSTRACT *(200 words or less)*

This report documents an analysis of the safety-related performance of the reactor core isolation cooling (RCIC) system at U.S. commercial boiling water reactor plants from 1987-1993. Both a risk-based analysis and an engineering analysis of trends and patterns were performed on RCIC operating data to provide insights into the performance of the RCIC system throughout the industry and at a plant-specific level. Comparisons were made to Probabilistic Risk Assessments and Individual Plant Evaluations (PRA/IPEs) for the 29 plants having a RCIC system to indicate where operating data either support or fail to support the assumptions, models, and data used to develop the RCIC system unreliability estimates provided by the PRA/IPEs.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*	
Reactor Core Isolation Cooling System, RCIC, Boiling Water Reactor, BWR, Reactor Core Isolation Cooling System Operational Events, Probabilistic Risk Assessment, PRA, Plant Evaluation, Reactor Core Isolation Cooling System Unreliability	13. AVAILABILITY STATEMENT: unlimited
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1. REPORT NUMBER
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NUREG/CR-5500, Volume 8

2. TITLE AND SUBTITLE

Reliability Study:
High-Pressure Core Spray System, 1987-1993

3. DATE REPORT PUBLISHED

MONTH	YEAR
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4. FIN OR GRANT NUMBER
E8246

5. AUTHOR(S)

J.P. Poloski, G.M. Grant, C.D. Gentillon, W.J. Galyean

6. TYPE OF REPORT
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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Idaho National Engineering and Environmental Laboratory
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9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Risk Analysis & Applications
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES
T. R. Wolf, NRC Project Manager

11. ABSTRACT *(200 words or less)*

This report documents an analysis of the performance of the high-pressure core spray (HPCS) system at U.S. commercial boiling water reactor plants during the period 1987-1993. Both a reliability analysis and an engineering analysis of trends and patterns were performed on data from HPCS system operational events to obtain insights into the performance of the HPCS system throughout the industry and at a plant-specific level. Comparisons were made to probabilistic risk assessments and individual plant examinations for the eight plants to indicate where operational data either support or fail to support the assumptions, models, and data used to develop the HPCS system unreliability estimates.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

High-Pressure Core Spray System, RCIC, Boiling Water Reactor, BWR, High-Pressure Core Spray System Operational Events, Probabilistic Risk Assessment, PRA, Plant Evaluation, High-Pressure Core Spray System Unreliability

13. AVAILABILITY STATEMENT
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1. REPORT NUMBER
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NUREG/CR-5503

2. TITLE AND SUBTITLE

Techniques for Identifying Faults and Determining Their Origins

3. DATE REPORT PUBLISHED

MONTH	YEAR
July	1999

4. FIN OR GRANT NUMBER

L2267

5. AUTHOR(S)

K.L. Hanson, K.I. Kelson, M.A. Angell[a], W.R. Lettis[b]

6. TYPE OF REPORT

Technical

7. PERIOD COVERED (Inclusive Dates)

September 1995 – March 1998

8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

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Walnut Creek, CA 94596

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

Division of Engineering Technology
Office of Nuclear Regulatory Research
US Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

E. Zurflueh, NRC Project Manager

11. ABSTRACT (200 words or less)

Satisfying regulatory criteria for siting nuclear power plants requires the ability to distinguish between tectonic and nontectonic faulting. Nontectonic faults can produce ground deformation but are not capable of producing significant earthquakes. These faults often have characteristics similar to those of tectonic faults, but they differ in terms of their causative forces and potential hazard. Tectonic faults, which may or may not be seismogenic, include primary structures capable of producing earthquakes and secondary structures that are produced by earthquakes but are not themselves capable of generating an earthquake. An understanding of the geologic, geomorphic, and tectonic processes that result in surface deformation is essential for developing criteria to identify and evaluate the seismogenic potential of faults. In this report, we (1) summarize the characteristics of faults resulting from tectonic and nontectonic mechanisms and (2) develop criteria to identify and differentiate tectonic and nontectonic faults. We find very few diagnostic criteria to differentiate tectonic from nontectonic faults. Determining the geologic context of a fault provides the best method for differentiating the origin of a fault. Observations and measurements of scale, geometry, and timing of fault movement are the most important attributes to understand in order to confidently assess the origin and, thus, potential hazard of a fault.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

tectonic faults
nontectonic faults
capable faults
seismogenic faults

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NRC FORM 335 (2-89) NRCM 1102, 3201, 3202 U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET** *(See instructions on the reverse)*	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.) NUREG/CR-5512, Volume 3
2. TITLE AND SUBTITLE Residual Radioactive Contamination From Decommissioning Parameter Analysis Draft for Comment	3. DATE REPORT PUBLISHED MONTH: October YEAR: 1999 4. FIN OR GRANT NUMBER W6227
5. AUTHOR(S) W.E. Beyeler, W.A. Hareland, F.A. Duran, T.J. Brown, E. Kalinina, D.P. Gallegos, P.A. Davis	6. TYPE OF REPORT 7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Sandia National Laboratories, P.O. Box 5800, MS 1345, Albuquerque, NM 87185

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Risk Analysis and Applications
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555 -0001

10. SUPPLEMENTARY NOTES

C. Daily, NRC Project Manager

11. ABSTRACT *(200 words or less)*

NUREG/CR-5512 is a multi-volume report describing a generic model which estimates potential radiation dose from exposure to residual radioactive contamination after the decommissioning of facilities licensed by the U.S. Nuclear Regulatory Commission. Individual volumes describe the generic scenarios, models, and parameter values for screening calculations, and the software that implements these calculations. This third volume describes the analysis used to define default parameter values for the Building Occupancy and Residential scenarios and the results of that analysis. Screening calculations are designed to support dose-based decisions without requiring information about specific site conditions. The range of conditions that might exist at licensed sites was used to develop distributions describing the variability in site-specific parameter values. These distributions were then used to derive distributions of potential dose values for unit concentrations of individual source radionuclides. Parameter values were then identified which produce dose values in the upper quantiles of the distributions for all source radionuclides. The resulting parameter values define a generic screening calculation that has a limited risk of underestimating a site-specific dose calculation based on the generic scenarios, models, and screening group. The distributions that underlie these parameter values provide a basis for developing site-specific parameter values.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*	13. AVAILABILITY STATEMENT
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1. REPORT NUMBER
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NUREG/CR-5512, Volume 4

2. TITLE AND SUBTITLE

Comparison of the Models and Assumptions used in the DandD 1.0, RESRAD 5.61, and RESRAD-Build 1.50 Computer Codes with Respect to the Residential Farmer and Industrial Occupant Scenarios Provided in NUREG/CR-5512

Draft for Comment

3. DATE REPORT PUBLISHED

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4. FIN OR GRANT NUMBER

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5. AUTHOR(S)

Rick Haaker (1), Theresa Brown (2), David Updegraff (3)

6. TYPE OF REPORT

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7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

(1) AQ Safety, Inc., 11024 Montgomery NE PMB# 294, Albuquerque, NM 87111

(2) Sandia National Laboratories, P.O. Box 5800, MS 1345, Albuquerque, NM 87185

(3)Gram Inc., 8500 Menaul NE, Albuquerque, NM 87112

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Office of Nuclear Regulatory Research
Division of Risk Analysis and Applications
U.S. Nuclear Regulatory Commission
Washington, DC 20555 -0001

10. SUPPLEMENTARY NOTES

C. Daily, NRC Project Manager

11. ABSTRACT *(200 words or less)*

This report provides a detailed comparison of the models, simplifying assumptions, and default parameter values implemented by the DandD 1.0, RESRAD 5.61, and RESRAD Build 1.50 computer codes. Each of these codes is a potentially useful tool for demonstrating compliance with the License Termination Rule, 10 CFR 20, Subpart E. The comparison was limited to the industrial occupant and residential farmer scenarios defined in NUREG/CR-5512, Volume 1. This report is intended to describe where and how the models and default parameter values in each of the codes differ for the specified scenarios. Strengths, weaknesses and limitations of the models are identified. The practical impacts of the identified differences to dose assessment results are discussed.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Decommission
Decontamination
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software
DandD
screening
dose assessment

148

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1. REPORT NUMBER
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NUREG/CR-5541
BNL-NUREG-52550

2. TITLE AND SUBTITLE

System Scaling for the Westinghouse AP600 Pressurized Water Reactor and Related Test Facilities

Analysis and Results

3. DATE REPORT PUBLISHED

MONTH	YEAR
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4. FIN OR GRANT NUMBER

W6670

5. AUTHOR(S)

W. Wulff, U.S. Rohatgi

6. TYPE OF REPORT

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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Brookhaven National Laboratory
Upton, NY 11973-5000

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Systems Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

D. Bessette, NRC Project Manager

11. ABSTRACT *(200 words or less)*

The global system scaling analysis for the Advanced Pressurized Water Reactor AP600 of Westinghouse has been performed in five main time phases for the 1-inch Cold -Leg break, to determine whether three related and already existing test facilities, namely the Advanced Plant Experiment (APEX) facility located at Oregon State University (OSU) at Corvallis, the Rig of Safety Assessment (ROSA) Large Scale Test Facility located in Tokai-mura, Japan, and the Simulatore per Esperienze di Sicurezza (simulator for Safety Experimental Analysis, SPES-2) located in Piacenza, Italy represent the AP600 Reactor. The scaling analysis is the top-down, global system analysis. It is intended to establish thermodynamic similarity between AP600, APEX, ROSA, and SPES at the level of overall system response and dynamic interaction between the system components. It is intended also to rank global transport processes according to their importance and to identify possible deviations from thermohydraulic similarity, or scale distortion.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Loss-of-Coolant-Computerized Simulation, PWR Type Reactors - Thermal Analysis, Westinghouse Standard Reactor - PWR Type Reactors, Comparative Evaluations, Cooling Systems, Heat Transfer, Reactor Components, Similarity, Scaling Criteria, Test Facilities, Global System Scaling, Process Priority, Scale Distortion, Advanced Passive Safety Systems, Multiphase Flow Systems, AP600

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1. REPORT NUMBER
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NUREG/CR-5563

2. TITLE AND SUBTITLE

A TECHNICAL BASIS FOR
REVISION TO ANCHORAGE CRITERIA

3. DATE REPORT PUBLISHED

MONTH	YEAR
March	1999

4. FIN OR GRANT NUMBER
W6454

5. AUTHOR(S)

Richard E. Klingner, Hakki Muratli and Mansour Shirvani

6. TYPE OF REPORT
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7. PERIOD COVERED *(Inclusive Dates)*
October 1, 1996-November 1, 1998

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Department of Civil Engineering ECJ 4.2
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Austin, Texas 78712

9. SPONSORING ORGANIZATION-NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.*

Division of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES
H. Graves, III, NRC Project Manager

11. ABSTRACT *(200 words or less)*

The objective of this project is to provide the US Nuclear Regulatory Commission (NRC) with a comprehensive technical basis that can be used to establish regulatory positions regarding fastening to concrete. The project consists of the following three tasks:
1) Prepare a report summarizing the guidance in documents such as:
 a) ACI 318 (fastening proposal);
 b) ACI 349, Appendix B;
 c) USI A-46, SQUG Report; and
 d) ACI 355.
2) Review and evaluate available sources of test data to establish trends in test results (for example, group and edge effects).
3) Prepare a comprehensive report that covers aspects of fastening design.
In this report, the guidance of existing documents is reviewed. Test results are evaluated for anchors under tensile and shear loads. Based on that evaluation, procedures are recommended for the evaluation and design of fasteners to concrete, including the effects of cracked concrete and dynamic loading.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

anchors, concrete, codes, design, earthquake, evaluation, fasteners, seismic

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14. SECURITY CLASSIFICATION
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NUREG/CR-5578

2. TITLE AND SUBTITLE
Instrumentation for the PUMA Integral Test Facility

3. DATE REPORT PUBLISHED

MONTH	YEAR
March	1999

4. FIN OR GRANT NUMBER
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5. AUTHOR(S)
S. T. Revankar, M. Ishii, Y. Mi, M. L, Bertodano, Y. Xu, S. Kelly, V. H. Ransom, Purdue University and J. T. Han, NRC

6. TYPE OF REPORT
Technical Report

7. PERIOD COVERED (Inclusive Dates)
06/01/97 - 05/31/98

8. PERFORMING ORGANIZATION – NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)
School of Nuclear Engineering
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West Lafayette, IN 47907-1290

9. SPONSORING ORGANIZATION – NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)
Division of Systems Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington DC 20555-0001

10. SUPPLEMENTARY NOTES
J. T. Han, NRC Project Manager

11. ABSTRACT (200 words or less)
The flow conditions in the PUMA (Purdue University Multi-Dimensional Integral Test Assembly) tests require special instrumentation. In this report, special instruments used in PUMA, impedance meter, modified magnetic flow meter, oxygen analyzer, conductivity probe, and vortex flow meter are presented. The impedance void-meter was developed to measure the average void fraction in pipes. The commercially available magnetic flow meter electrodes were modified to measure the liquid flow rate in two-phase flow. A multi-port gas sampling system using oxygen analyzers was developed for on line measurements of air concentration in steam at various components of PUMA. A commercial vortex flow meter was used to measure low steam flow rate in pipes. A conductivity probe was developed to measure the local void fraction in the reactor pressure vessel for water-steam at 1 MPa (150 psig) and 180°C (356°F) or below. These meters were tested and calibrated in the laboratory.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)
Two-phase instruments, PUMA, simplified boiling water reactor (SBWR) impedance meter, modified magnetic flow meter, oxygen analyzer, conductivity probe, vortex flow meter, two-phase measurement, void fraction, non-condensable measurement

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/CR-5582
SAND98-2047

2. TITLE AND SUBTITLE

Lower Head Failure Experiments and Analyses

3. DATE REPORT PUBLISHED

MONTH	YEAR
February	1999

4. FIN OR GRANT NUMBER

F6356

5. AUTHOR(S)

T.Y. Chu, M.M. Pilch, J.H. Bentz, J.S. Ludwigsen, W-Y Lu, L.L. Humphries*

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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

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Albuquerque, NM 87110-4204

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Systems Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

Program objectives are to characterize the mode, timing, and size of lower head failure (LHF). A scaling analysis was performed to guide the design and conduct of the experiment program. The experiment program consisted of eight 1/5 scale, tests simulating a PWR lower head. Each test was pressurized and heated from the inside until failure occurred. The test matrix addressed issues of heating patterns, lower head penetrations, lower head weldments, and system pressure. One replicate test was performed. It was found that experiment results were repeatable and that vessel rupture was attracted to regions of non-uniform temperature or non-uniform thickness. The material property database for SA533B1 steel was critically reviewed and curvefits were recommended for tensile properties and creep properties. A constitutive law for vessel deformation was developed and assessed against four of the LHR tests. It was concluded that a more complete material property testing program is required for a more quantitative validation of models against the LHF tests. The LHF database was used to assess the "engineering" methodologies commonly employed in Severe Accident Codes. The current database is notably lacking in tests with a prototypic temperature differential across the lower head.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

severe accidents, reactor pressure vessel failure, creep rupture, severe accident code, SA533B1 properties

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14. SECURITY CLASSIFICATION

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/CR-5593
SAND99-0676

2. TITLE AND SUBTITLE

Risk Comparisons of Scheduling Preventive Maintenence for Boiling Water Reactors During Shutdown and Power Operations

3. DATE REPORT PUBLISHED

MONTH	YEAR
July	1999

4. FIN OR GRANT NUMBER

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5. AUTHOR(S)

B.D. Staple, T.D. Brown, J.J. Gregory, J.L. LaChance

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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

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Albuquerque, NM 87185-0747

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

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Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

A. Buslik, NRC Project Manager

11. ABSTRACT *(200 words or less)*

With the nuclear industry focused on reducing the length of its downtime for refueling and maintenance, increased attention has been given to performing preventive maintenance during full-power operation rather than during periods when the plant is shut down, such as during refueling. This shift to on-line maintenance led the Nuclear Regulatory Commission to sponsor studies aimed at understanding the risks of carrying out preventive maintenance during power operation compared with the risk when preventive maintenance is performed during shutdown. The types of accidents, the availability of accident mitigating systems, the accident progression, the status of the containment, and the time available for accident recovery all differ in the various plant operational modes. Thus the impact of preventive maintenance on risk can vary from one mode to another and will in general be component specific.

This report presents a process developed to compare the risk impacts of performing preventive maintenance during full power operation and during shutdown modes of operation. The process includes the use of public risk measures in addition to core damage frequency as metrics for evaluating the effects on risk from performing preventive maintenance during different plant operational states. The effects of uncertainties in parameters is also included in order to provide a comprehensive evaluation. The report describes the results of the application of the process to one nuclear power plant and, where possible, extrapolates the results to other similar plants.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

on-line maintenance, probabilistic risk assessment, preventive maintenance, maintenance scheduling, nuclear power plant maintenance, Grand Gulf Nuclear Station, core damage frequency, risk measures, public risk, risk increase, risk contribution, full-power operation, refueling, cold shutdown

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1. REPORT NUMBER *(Assigned by NRC. Add Vol., Supp., Rev., and Addendum Numbers, if any.)*
NUREG/CR-5608
ANL-98/25

2. TITLE AND SUBTITLE
Irradiation-Assisted Stress Corrosion Cracking of Model Austenitic Stainless Steels Irradiated in the Halden Reactor

3. DATE REPORT PUBLISHED

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4. FIN OR GRANT NUMBER
W6610

5. AUTHOR(S)
H. M. Chung, W. E. Ruther, and R. V. Strain

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8. PERFORMING ORGANIZATION – NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*
Argonne National Laboratory
9700 South Cass Avenue
Argonne, IL 60439

9. SPONSORING ORGANIZATION – NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*
Division of Engineering Technology
Office of Nuclear Regulatory Research
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES
M. NcNeil, NRC Project Manager

11. ABSTRACT (200 words or less)

This report summarizes work performed by Argonne National Laboratory on irradiation-assisted stress corrosion cracking (IASCC) of model austenitic stainless steels (SSs) that were irradiated in the Halden reactor in simulation of boiling water reactor core internal components. Slow-strain-rate tensile tests were conducted on 16 austenitic stainless steel alloys that were irradiated to a fluence of $\approx 3 \times 10^{20}$ n cm^{-2} (E > 1 MeV) and 9 alloys irradiated to a fluence of $\approx 9 \times 10^{20}$ n cm^{-2}. Fractographic analysis by scanning electron microscopy was also conducted to determine susceptibility to irradiation-assisted stress corrosion cracking manifested by transgranular (TG) and intergranular (IG) fracture surface morphology. A high-purity heat of Type 316L SS exhibited the highest susceptibility to IASCC at the low fluence. Total elongation and susceptibility to IASCC at the low fluence could be correlated well with the nitrogen and silicon concentrations of the alloys. All alloys that contained low levels of nitrogen (<100 wppm) and silicon (<1.0 wt.%) exhibited low ductility and high susceptibility to IASCC. All alloys that contain either high silicon (>1.0 wt.%) or nitrogen >100 wppm exhibited high ductility, low percent TGSCC, and negligible percent IGSCC. Because practically all commercially fabricated steels contain nitrogen >100 wppm, this observation indicates that alloys that contain >1.0 wt.% silicon are effective in delaying the onset of and suppressing susceptibility to IASCC. Initial results from specimens irradiated to a fluence of $\approx 9 \times 10^{20}$ n cm^{-2} indicate that low concentrations of chromium and silicone and high concentration of oxygen in steels are conducive to higher susceptibility to IASCC.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating this report.)*
Boiling water reactor
In-core water chemistry
Core internal components
Irradiation-assisted stress corrosion cracking
Type 304 and 316 stainless steels
Transgranular stress corrosion cracking
Intergranular stress corrosion cracking
Work-hardening capability
Tensile ductility
Silicon content in austenitic stainless steel

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1. REPORT NUMBER *(Assigned by NRC. Add Vol., Supp., Rev., and Addendum Numbers, if any.)*
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ANL-98/29

2. TITLE AND SUBTITLE
Residual Stresses and Associated Stress Intensity Factors in Core Shroud Weldments

3. DATE REPORT PUBLISHED
MONTH: March YEAR: 1999

4. FIN OR GRANT NUMBER

5. AUTHOR(S)
J. Zhang, P. Dong, F. W. Brust, and W. J. Shack

6. TYPE OF REPORT
Technical

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION – NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*
Argonne National Laboratory
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9. SPONSORING ORGANIZATION – NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission.*
Division of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES
M. McNeil, NRC Project Manager

11. ABSTRACT (200 words or less)
Numerical models were developed to characterize weld residual stresses and the associated stress intensity factors at BWR core shroud welds. Detailed descriptions of the weld residual stresses have been obtained for the H4 cylinder-to-cylinder weld and the H8 weld, which joins the core shroud to a conical support shell.

An axisymmetric solid element model was used to characterize the detailed evolution of residual stresses in the H4 weld. In the analysis, a series of advanced weld modeling techniques were used to address some specific welding-related issues, such as material melting/re-melting and history annihilation. In addition, a 3-D shell element analysis was performed to quantify specimen removal effects on residual stress measurements based on a sub-structural specimen from a core shroud. Stress intensity factors were calculated for both axisymmetric circumferential (360°) and circumferential surface cracks.

In BWR-2 models, the core shroud has a conical support structure. The geometry of the H8 weld that joins the support and the shroud is more complex than the cylinder-to-cylinder geometry that is characteristic of most core shroud welds. In addition the H8 weld is made prior to the postweld heat treatment (PWHT) of the reactor vessel. The residual stress distribution at the H8 weld is then affected by the subsequent H7 weld. A detailed finite-element analysis was carried out to characterize the residual stresses in the H7 and H8 welds including the effect of the PWHT. A stress-intensity-factor-based fracture assessment, based on the residual stress solutions, for a crack growing from the bottom surface of the H8 weld was also performed.

The printed report contains gray scale versions of the stress contour plots. A Portable Document File (pdf) version of the report that contains full color versions of the contour plots is available at http://www.et.anl.gov/Reports/NUREGCR5675.PDF

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating this report.)*
Residual Stresses
Stress Intensity Factors
Core Shroud
Weldment
Stress Corrosion Cracking

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NRC FORM 335 (2-89) NRCM 1102, 3201, 3202 — U.S. NUCLEAR REGULATORY COMMISSION — **BIBLIOGRAPHIC DATA SHEET** (See instructions on the reverse)	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.) NUREG/CR-5698
2. TITLE AND SUBTITLE Comparing Monitoring Strategies at the Maricopa Environmental Monitoring Site, Arizona	3. DATE REPORT PUBLISHED MONTH: July YEAR: 1999 4. FIN OR GRANT NUMBER JCN W6151
5. AUTHOR(S) M.H. Young, A.W. Warrick, P.J. Wierenga, L.L. Hofmann and S.A. Musil	6. TYPE OF REPORT Technical 7. PERIOD COVERED (Inclusive Dates) 04/95 - 02/99

8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Department of Soil, Water and Environmental Science
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Tucson, AZ 85721

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

Division of Risk Analysis and Applications
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

T.J. Nicholson, NRC Project Manager

11. ABSTRACT (200 words or less)

The purpose of this document is to discuss the alternative monitoring strategie s used during field experiments at the Maricopa Environmental Monitoring site, Maricopa, AZ. The strategies selected could pot entially be incorporated into monitoring programs at low-level radioactive waste disposal facilities. The four monitori ng strategies include: Monitoring Trenches, Monitoring Islands, Borehole Monitoring, and Geophysical Monitoring. Strengths and weaknesses of each strategy were described with respect to installation, maintenance and replacement of monitori ng systems and instruments. Evaluation of the strategies was mostly qualitative in nature, but were supported by data collect ed during two, field-scale infiltration experiments in the vadose zone. Each of the strategies possess benefits and drawbacks, req uiring site specific analyses of site and environmental conditions during monitoring program design. The document also p resents the concept of primary performance measures (e.g., water content, water tension and solute concentration), each of which directly influences water movement and contaminant migration from disposal sites, and discusses the need to accurately convert field observations to these primary measures. Using multiple instruments whose data convert to the same primary pe rformance measure, could improve the confidence that changes in soil water conditions are real and not affected by t he monitoring systems themselves.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)	13. AVAILABILITY STATEMENT
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2. TITLE AND SUBTITLE Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels	3. DATE REPORT PUBLISHED MONTH: April YEAR: 1999
	4. FIN OR GRANT NUMBER W6610
5. AUTHOR(S) O. K. Chopra	6. TYPE OF REPORT Technical
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8. PERFORMING ORGANIZATION – NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

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9. SPONSORING ORGANIZATION – NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Engineering Technology
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Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

M. McNeil, NRC Project Manager

11. ABSTRACT (200 words or less)

The ASME Boiler and Pressure Vessel Code provides rules for the construction of nuclear power plant components. Figures I-9.1 through I-9.6 of Appendix I to Section III of the Code specify fatigue design curves for structural materials. Although effects of reactor coolant environments are not explicitly addressed by the design curves, test data indicate that the Code fatigue curves may not always be adequate for coolant environments. This report summarizes work performed by Argonne National Laboratory on the fatigue of austenitic stainless steels in light water reactor (LWR) environments. Existing fatigue S–N data have been evaluated to establish the effects of various material and loading variables, such as steel type, dissolved oxygen level, strain range, strain rate, and temperature, on the fatigue lives of these steels. Statistical models are presented for estimating the fatigue S–N curves as a function of material, loading, and environmental variables. Design fatigue curves have been developed for austenitic stainless steel components in LWR environments. The extent of conservatism in the design fatigue curves and alternative methods for incorporating the effects of LWR coolant environments into the ASME Code fatigue evaluations are discussed.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating this report.)*	13. AVAILABILITY STATEMENT Unlimited
Fatigue Strain–Life Curves Fatigue Design Curves LWR Environments Austenitic Stainless Steels Cast Austenitic Stainless Steels Fatigue Crack Initiation	14. SECURITY CLASSIFICATION *(This Page)* Unclassified *(This Report)* Unclassified
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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/CR-5719
ORNL/TM-13738

2. TITLE AND SUBTITLE

SEN1: A One-Dimensional Cross-Section Sensitivity and Uncertainty Module for Criticality Safety Analysis

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4. FIN OR GRANT NUMBER

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5. AUTHOR(S)

R.L. Childs

6. TYPE OF REPORT

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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Oak Ridge National Laboratory
Oak Ridge, TN 37831-6370

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

C.W. Nilsen, NRC Project Manager

11. ABSTRACT *(200 words or less)*

This report documents sensitivity analysis computer codes that have been developed for use with one-dimensional (1-D) and two-dimensional (2-D) calculational models. These codes provide a useful tool to aid criticality safety analysts in understanding the applicability of selected critical experiments to the validation of real systems.

SEN1 is a prototypic SCALE control module that faciiitates the application of sensitivity theory to criticality safety analysis. The XSDRNPM module uses the method of discrete ordinates to calculate k_{eff} for applications that are appropriate for 1-D modeling. Perturbation theory is used to determine the sensitivity of the calculated value of k_{eff} to the nuclear data used in the calculation as a function of nuclide, reaction type, and energy. The uncertainty in the calculated value of k_{eff}, resulting from uncertainties in the basic nuclear data used in the calculation, is estimated using energy-dependent relative covariance matrices processed from ENDF/B-V. Systems containing arrays of fuel pins may be analyzed using cell-weighted cross sections. The methods used in this work are based on the FORSS system developed at ORNL in the 1970s. The present work uses the XSDRNPM module and the problem-dependent cross-section processing capabilities of the SCALE system and is much more automated than the earlier FORSS system. Two-dimensional sensitivity analysis using the DORT code has also been developed and is described in the appendix.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

sensitivity, uncertainty, SCALE, criticality safety, FORSS

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)
NUREG/CR-5724
ORNL/SUB/98-SZ272V

2. TITLE AND SUBTITLE
Feasibility of Magnetostrictive Sensor Inspection of Containments

3. DATE REPORT PUBLISHED
MONTH: March
YEAR: 1999

4. FIN OR GRANT NUMBER
J6043

5. AUTHOR(S)
H. Kwun
(Southwest Research Institute)

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8. PERFORMING ORGANIZATION — NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

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9. SPONSORING ORGANIZATION — NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)
Division of Engineering Technology
Office of Nuclear Regulatory Research
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES
H. L. Graves, III, NRC Project Manager

11. ABSTRACT (200 words or less)

This report describes a study on the feasibility of using guided waves for long-range global inspection of containment metallic pressure boundaries in nuclear power plants. Of particular concern was the potential of the guided-wave approach for remotely inspecting regions that are inaccessible; e.g., regions where the metallic pressure boundary is backed by concrete on one or both sides.

The study included a literature review on long-range guided-wave inspection techniques, a modeling study of the behavior of guided waves in plates with different boundary conditions (e.g., freestanding and backed by concrete on one or both sides), and an experimental investigation of the feasibility of magnetostrictive sensor (MsS) technology for (1) generating and detecting guided waves in plates and (2) detecting a defect over a long range.

Study results showed (1) it is feasible to achieve long-range global inspection of plates, including regions that are inaccessible, using low-frequency guided waves and (2) that the MsS technique is well suited for this application. Recommendations are made to further test and develop the MsS technique for practical implementation for containment inspection in nuclear power plants.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

Compressional wave
containment
degradation
electromagnetic acoustic transducer
global inspection
guided waves
inspection
liner
magnetostrictive sensor
nondestructive examination
numerical modeling
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shear wave
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2. TITLE AND SUBTITLE

Paleoseismology Study Northwest of the New Madrid Seismic Zone

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5. AUTHOR(S)

M. Tuttle, J. Chester, R. Lafferty, K. Dyer-Williams, R.Cande

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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

University of Maryland
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College Park, MD 20745

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission and mailing address.)*

U.S. Nuclear Regulatory Commission
Division of Engineering Technology
Office of Nuclear Regulatory Research
Washington, D.C.20555-0001

10. SUPPLEMENTARY NOTES

Ernst G. Zurflueh, NRC Project Manager

11. ABSTRACT *(200 words or less)*

We surveyed more than 400 km of river cutbank and documented liquefaction features, including sand dikes and sandblows, at over 50 sites. In addition, we studied several structures, including the Bodenschatz-Lick, Eureka-House Springs, and St. Genevieve fault systems and the Farmington and Valmeyer anticlines, that are associated with hisotrical and instrumentally recorded seismicity. Although no recent faulting was found, possible earthquake sources for liquefaction include the St. Louis, Centralia, and New Madrid faults, the Valmeyer and Waterloo-Dupo anticlines, and the Du Quoin monocline.

We found liquefaction features along the Big Muddy, Cache, Marys, Meramec, and Kaskaskia Rivers and Mud, Shoal, and Silver Creeks, The distribution of liquefaction features suggests that a large to very large earthquake occurred east of St. Louis about 6,500 years ago and that another significant event occurred in the same area in the past 4,000 years. We propose three possibe earthquake scenarios to account for the observed pattern of liquefaction. Additional information is needed to better estimate the timing and magnitude of large prehistoric earthquake(s). A large earthquake in southwestern Illinois today would have a major impact on nearby urban areas including St. Louis and possibly Chicago. The seismic hazard remains poorly understood due to large uncertainties in timing, location, and magnitudes of large prehistoric earthquakes in this region.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

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archaeology
scenarios
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Illinois

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NUREG/CR-5733

2. TITLE AND SUBTITLE

Re-evaluation of Regulatory Guidance Provided in Regulatory Guides 1.142 and 1.143

3. DATE REPORT PUBLISHED

MONTH	YEAR
August	1999

4. FIN OR GRANT NUMBER

W6540

5. AUTHOR(S)

T.M. Adams, J.D. Stevenson, G.G. Thomas, G.A. Harstead

6. TYPE OF REPORT

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1/3/97 - 4/2/99

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Stevenson and Associates
9217 Midwest Avenue
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9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

H. Graves III, NRC Project Manager

11. ABSTRACT *(200 words or less)*

This report recommends changes in the Nuclear Regulatory Commission's (NRC) criteria now used in the design of Nuclear Concrete Structures other than Containment and the design of radioactive Waste Management Systems, Structures and Components. The authors judged that the best method by which the conclusions of this report can be implemented are through the issuance of revisions to existing Regulatory Guidelines. There have been several developments since the issuance of the last revisions of Regulatory Guidelines 1.142 and 1.143 (1981 and 1979 respectively) which provides guidance for the design of these items. ACI 349, which presents the code requirements for nuclear safety related concrete structures, has had two major revisions since the last issue of the Regulatory Guidelines. New advanced reactor design concepts have been put forward that include unique attributes that need to be considered. The Department of Energy (DOE) has been active in writing standards for the design of radioactive waste management systems. There have also been numerous changes to the Code of Federal Regulations since the issuance of Regulatory Guidelines 1.142 and 1.143. This report discussed these advancements and changes with respect to the scope of the Regulatory Guidelines and includes detailed recommended changes to Revision 1 of Regulatory Guidelines 1.142 and 1.143.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Regulatory Guide 1.142
Regulatory Guide 1.143
Concrete Design
ACI-349
Concrete Structures
Nuclear Waste Management
ANS 55.1
ANS 55.4
ANS 55.6
Nuclear Waste Processing
Nuclear Waste Management Systems
Nuclear Waste Management Structures
Nuclear Waste Management Components

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NUREG/CR-5738

2. TITLE AND SUBTITLE

Field Investigations for Foundations of Nuclear Power Facilities

3. DATE REPORT PUBLISHED

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November	1999

4. FIN OR GRANT NUMBER

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5. AUTHOR(S)

N. Torres, J.P. Koester, J.L. Llopis

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7. PERIOD COVERED *(Inclusive Dates)*

June 1995 - July 1999

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

U.S. Army Corps of Engineers
Waterways Experiment Station
3909 Halls Ferry Road
Vicksburg, MS 39810-6199

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

E.G. Zurflueh, NRC Project Manager

11. ABSTRACT *(200 words or less)*

This document provides a technical basis for revision of the U.S. Nuclear Regulatory Commission Regulatory Guide 1.132, Site Investigations for Foundations of Nuclear Power Facilities, reflecting current and state-of-the-art techniques related to field site investigations. The report summarizes the processes of acquiring geological, geophysical, geotechnical, and other kinds of relevant information that may affect the construction or performance of a building or other engineered structure at selected sites. Guidance is presented for in situ studies during the various stages of site characterization. Topics range from initial information gathering, literature review, and site reconnaissance investigations, to on-site testing and the collection and management of samples for laboratory testing. Specific laboratory tests and techniques for the engineering analysis of soils and specific requirements for liquefaction analysis are not addressed in this document but are covered in companion technical basis documents.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

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Sampling methods

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BIBLIOGRAPHIC DATA SHEET

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any)
NUREG/CR-5746
SAND99-1634

2. TITLE AND SUBTITLE

Direct Containment Heating Experiments at Low Reactor Coolant System Pressure in the Surtsey Test Facility

3. DATE REPORT PUBLISHED

MONTH	YEAR
July	1999

4. FIN OR GRANT NUMBER

W6162

5. AUTHOR(S)

T. K. Blanchat and M. M. Pilch (SNL), R. Y. Lee (USNRC), L. Meyer (FZK), and M. Petit (IPSN)

6. TYPE OF REPORT

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7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION – NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Sandia National Laboratories
Albuquerque, NM 87185-1139

Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Forschungszentrum Karlsruhe GmbH (FZK)
Karlsruhe, Germany

Institut de Protection et de Surete Nucleaire (IPSN)
Fontenay-aux-Roses
France

9. SPONSORING ORGANIZATION – NAME AND ADDRESS *(If NRC, type "Same as above", if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES
Richard Y. Lee, NRC Project Manager

11. ABSTRACT *(200 words or less)*

The Surtsey Test Facility at Sandia National Laboratories is used to perform scaled experiments for the Nuclear Regulatory Commission (NRC) that simulate high pressure melt ejection (HPME) accidents in a nuclear power plant (NPP). These experiments are designed to investigate melt dispersal from a reactor cavity and the resulting containment loads if the reactor pressure vessel (RPV) lower head fails while the reactor coolant system (RCS) is still at elevated pressures. Collectively, these phenomena are referred to as HPME from the RPV and direct containment heating (DCH). The DCH Supplementary tests described in this report provided additional information to establish baseline DCH and the potential contribution of hydrogen combustion to DCH loads. These tests were performed in a more prototypic manner where melt was driven from the RPV mockup with steam. These tests were also conducted at lower RCS pressure (~1 MPa) using different hole sizes (10 cm and 4 cm). The tests were conducted under a cooperative research program among the Forschungszentrum Karlsruhe, Institut de Protection et de Surete Nucleaire and the NRC. The proposed European Pressurized Water Reactor (EPR) cavity has a melt spreading room attached to the side and below the cavity that would be rendered useless if the melt is dispersed from the cavity to the upper dome area by the blowdown.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

low reactor coolant system (RCS) pressure
Combustion Engineering (CE)
Direct Containment Heating (DCH)
high pressure melt ejection (HPME)
Surtsey Test Facility
severe accident
pressurized water reactor
Calvert Cliffs
European Pressurized Water Reactor (EPR)

13. AVAILABILITY STATEMENT
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14. SECURITY CLASSIFICATION
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2. TITLE AND SUBTITLE Rates of Initiating Events at U.S. Nuclear Power Plants: 1987–1995	3. DATE REPORT PUBLISHED MONTH: February YEAR: 1999 4. FIN OR GRANT NUMBER E8246
5. AUTHOR(S) J.P. Poloski, D.G. Marksberry, C.L. Atwood, and W.J. Galyean	6. TYPE OF REPORT Technical 7. PERIOD COVERED (Inclusive Dates) 01/01/87—12/31/95

8. PERFORMING ORGANIZAITON - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

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P.O. Box 1625
Idaho Falls, ID 83415-3129

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above"; If contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

Safety Programs Division
Office for Analysis and Evaluation of Operational Data
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

This report was produced at the Idaho National Engineering and Environmental Laboratory for the U.S. Nuclear Regulatory Commission, Office for Analysis and Evaluation of Operational Data. Data for all unexpected reactor trips during power operations at commercial nuclear power plants from 1987 through 1995 were reviewed. Each event was reviewed and categorized according to the initial event and, additionally, was marked if certain other risk-significant events occurred, regardless of their position in the event sequence. The collected data were analyzed for time dependence, reactor-type dependence, and between-plant variance. Dependencies and trends are reported, along with the raw counts and the best estimate for 1995 initiating event frequencies. For some initiators whose frequencies are low enough that no events would be expected in the 1987–1995 period, additional operating experience and information from other sources were used to estimate frequencies. These included operating experience from U.S. and foreign reactors, as well as evaluation of engineering aspects of certain rare events, such as loss-of-coolant accidents (LOCAs). Results of engineering analyses of the operating experience are compared with probabilistic risk assessment/individual plant examinations (PRA/IPEs) and other regulatory issues.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)	13. AVAILABILITY STATEMENT
initiating event frequency, functional impact, initial plant fault, BWR, PWR, LOCA, operational experience, probabilistic risk assessments, individual plant examinations	Unlimited
	14. SECURITY CLASSIFICATION (This page) Unclassified (This report) Unclassified
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2. TITLE AND SUBTITLE Hanford Tank Waste Remediation System High-Level Waste Chemistry Manual	3. DATE REPORT PUBLISHED MONTH: May — YEAR: 1999 4. FIN OR GRANT NUMBER J5164
5. AUTHOR(S) R.T. Pabalan, M.S. Jarzemba, D.A. Pickett, N. Sridhar, J. Weldy[a] C.S. Brazel, J.T. Persyn, D.S. Moulton, J.P. Hsu, J. Erwin[b] T.A. Abrajano, Jr.[c] B. Li[d]	6. TYPE OF REPORT Technical 7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address, if contractor, provide name and mailing address)

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Troy, NY 12180-3590

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Morris Plains, NJ 07950

9. SPONSORING ORGANIZATION - NAME AND MAILING ADDRESS (If NRC, type "Same as above", if contractor, provide NRC Division, Office or Region, US. Nuclear Regulatory Commission, and mailing address)

Division of Fuel Cycle Safety and Safeguards
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

The U.S. Department of Energy (DOE) plans to privatize the waste treatment and immobilization operations of the Hanford Tank Waste Remediation System (TWRS) program. A Memorandum of Understanding has been established between the DOE and the Nuclear Regulatory Commission (NRC) for the first phase of the TWRS program. To assist the NRC in developing technical and regulatory tools for the TWRS privatization effort, the Center for Nuclear Waste Regulatory Analyses is providing the NRC with information and tools needed to assess the chemical, radiological, and criticality hazards of Hanford tank wastes and operations addressed under the privatization initiative. Of primary concern are those reactions that could occur during waste retrieval and processing, but potential reactions during continued interim storage are also important.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)		13. Availability Statement
Hanford	Ferrocyanide Safety Issue	Unlimited
Tank Waste Remediation System	Criticality Safety Issue	14. Security Classification (This Page)
Tank Waste	High Heat Safety Issue	Unclassified
Chemistry	Flowsheet Simulation	(This Report)
Flammable Gas Safety Issue	Thermodynamic Model	Unclassified
High Organic Safety Issue		15. NUMBER OF PAGES
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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/CR-6369, Vol. 1
SEA 97-3105-A:14

2. TITLE AND SUBTITLE

Drywell Debris Transport Study

3. DATE REPORT PUBLISHED

MONTH: September
YEAR: 1999

4. FIN OR GRANT NUMBER

W6325

5. AUTHOR(S)

D.V. Rao, C. Shaffer, E. Haskin

6. TYPE OF REPORT

Final

7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Science and Engineering Associates, Inc.
6100 Uptown Blvd. NE
Albuquerque, NM 87110

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

Division of Engineering Technology
Officeof Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

M. Marshall, NRC Project Manager

11. ABSTRACT (200 words or less)

This report describes results of the drywell debris transport study. The objective of the study is to develop a methodology for estimating fraction of LOCA generated fibrous insulation debris that would be transported from the location of their generation in the drywell to the suppression pool. The study decomposed the problem into several components that were amenable to resolution by the knowledge base that can be developed from separate effects experiments, analytical modeling and engineering calculations. Experiments and analytical studies were undertaken to compile the necessary knowledge base on debris transport during blowdown, washdown of debris by ECCS water flow and debris sedimentation on the drywell floor. Logic charts were used to link both experimental and analytical results. The results of the study were used to delineate plant features and transport phenomena that dominate debris transport in the BWR drywell. A separate logic chart was developed for each postulated accident scenario and generic plant type analyzed. The logic charts can be modified to take into account effects of the plant specific features. The overall method is comprehensible to engineers who are not experts in the subject of debris transport. Also, it is sufficiently flexible that new evidence and assumptions, related to debris size and distribution, can be easily accommodated.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

BWR Suction Strainers
Drywell Debris Transport
Transport Experiment
Computational Fluid Dynamics
Transport Fractions

13. AVAILABILITY STATEMENT

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14. SECURITY CLASSIFICATION

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/CR-6369, Vol. 2
SEA 97-3105-A:15

2. TITLE AND SUBTITLE

Drywell Debris Transport Study: Experimental Work

3. DATE REPORT PUBLISHED

MONTH	YEAR
September	1999

4. FIN OR GRANT NUMBER

W6325

5. AUTHOR(S)

D.V. Rao, C. Shaffer, B. Carpenter, D. Cremer, and J. Brideau

6. TYPE OF REPORT

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7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Science and Engineering Associates, Inc.
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Albuquerque, NM 87110

Subcontractor:
Alden Research Laboratory
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Holden, MA 01520

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

Division of Engineering Technology
Officeof Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

M. Marshall, NRC Project Manager

11. ABSTRACT (200 words or less)

This report describes three test programs undertaken as part of the DDTS to pro vide basic understanding regarding transport of insulation fragments in the drywell following a postulated LOCA. The first two tests focused on transport of debris by blowdown flow. They obtained data related to (a) inertial capture of insulatio n debris on typical BWR drywell structures while they are transported across the structures by the steam flow: and (b) degradati on of large insulation pieces captured on floor gratings when exposed to high velocity steam flow with suspended droplets. The se tests clearly established that wet floor gratings would capture significantly more debris than any other BWR drywell str uctures (e.g., pipes, I-beams and vents). The capture efficiency of all structures was found to be a strong function of debri s size and structural wetness, but a weak function of flow velocity and local flow patterns. Floor gratings possess 100% capture efficiency for insulation pieces larger than 6"x4". These large pieces do not degrade or are not forced through the grating clearan ces (1.5"x4") when subjected to high velocity droplet flow, even though the differential pressure across them is as high as 1 psid.

The third test program addressed the issue of washdown of debris previously cap tured on floor gratings by break over flow or containment spray flow during ECCS recirculation phase. These tests concluded that majority of the small debris pieces captured on various structures would be washed down by break flow or spray flow. On the other hand, erosion is the only available mechanism by which large pieces deposited on the floor gratings would be transported. In three hours, as much as 25% of the larger pieces can be eroded and transported to the suppression pool.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

BWR Suction Strainers
Drywell Debris Transport
Transport Experiment

13. AVAILABILITY STATEMENT

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14. SECURITY CLASSIFICATION

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NUREG/CR-6369, Vol. 3
SEA 97-3105-A:17

2. TITLE AND SUBTITLE

Drywell Debris Transport Study: Computational Work

3. DATE REPORT PUBLISHED

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4. FIN OR GRANT NUMBER

W6325

5. AUTHOR(S)

C. Shaffer, D.V. Rao, and J. Brideau

6. TYPE OF REPORT

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7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Science and Engineering Associates, Inc.
6100 Uptown Blvd. NE
Albuquerque, NM 87110

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

Division of Engineering Technology
Officeof Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

M. Marshall, NRC Project Manager

11. ABSTRACT (200 words or less)

This report describes various analyses conducted as part of the drywell debris transport study. The primary objective of these analyses was to identify controlling phenomena and critical data requirements. A secondary objective was to explore various options available to model debris transport in the drywell, and make judgements regarding the degree of accuracy to which each phenomenon should be modeled. These analyses decomposed the problem into several components that were amenable to resolution by well-proven analytical models. The analyses specific ally addressed the following phenomena that significantly impact debris transport: pressure vessel blowdown, containment th ermal-hydraulics (e.g., structural wetness, flow velocities in the drywell), debris removal by various capture mechanisms and de bris transport in the water pools formed on the drywell floor. The analytical tools used in the study included RELAP, MELCOR a nd CFD-2000. The results of some of the analyses were used to design the experiments conducted as part of the study and during the debris transport quantification process described in NUREG CR-6369, Vol 2

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

BWR Suction Strainers
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Computational Fluid Dynamics

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/CR-6380
ORNL/TM-13091

2. TITLE AND SUBTITLE

Assessment of the Fracture Behavior of Weld Material in Full-Thickness Clad Beams

3. DATE REPORT PUBLISHED

MONTH	YEAR
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4. FIN OR GRANT NUMBER

B0119

5. AUTHOR(S)

J. A. Keeney, B. R. Bass, W. J. McAfee, and P. T. Williams

6. TYPE OF REPORT

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8. PERFORMING ORGANIZATION – NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Oak Ridge National Laboratory
Oak Ridge, Tennessee 37831-6285

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Division of Engineering Technology
Office of Nuclear Regulatory Research
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

S. N. Malik, NRC Project Manager

11. ABSTRACT (200 words or less)

A Heavy-Section Steel Technology testing program was conducted on full-thickness clad beam specimens fabricated using material from the reactor pressure vessel of a canceled nuclear power plant to generate fracture toughness data for both deep and shallow cracks in prototypic RPV material. The beam specimens incorporated weld, base, and overlay cladding materials. In the first testing phase, five full-thickness clad beam specimens were fabricated with through-thickness cracks that ranged in depth from 10 to 114 mm ($0.05 \leq a/W \leq 0.5$) in the weld material. These specimens were tested in three-point bending at temperatures in the transition region of the weld metal fracture toughness curve. Fracture toughness estimates were obtained from load versus load-line displacement and crack-mouth opening displacement data using finite-element techniques and estimation schemes based on the η-factor method. Effects of precleavage ductile tearing on fracture toughness were investigated using a continuum damage model based on the Gurson-Tvergaard formulation. The cleavage toughness data were compared with other shallow- and deep-crack uniaxial beam data generated previously from plate material that conformed to SA533, Grade B material specification requirements. The range in scatter for data obtained from the clad beam specimens is consistent with that from the laboratory-scale single-edge-notched-bend specimens tested at the same temperature.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

A533 Grade B Class 1 steel
Full-thickness clad beam
Shallow-crack beam
Constraint analysis
Furson-Tvergaard Model
Elastic-plastic fracture mechanics
Fracture toughness data
Precleavage ductile tearing
J-Q Methodology
Compact tension specimen

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/CR-6471, Vol. 3
PNNL-11143

2. TITLE AND SUBTITLE

Characterization of Flaws in U.S. Reactor Pressure Vessels

Density and Distribution of Flaw Indications in the Shoreham Vessel

3. DATE REPORT PUBLISHED

MONTH	YEAR
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4. FIN OR GRANT NUMBER

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5. AUTHOR(S)

G.J. Schuster, S.R. Doctor, S.L. Crawford, A. F. Pardini

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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Pacific Northwest National Laboratory
Richland, WA 99352

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

D.A. Jackson, NRC Project Manager

11. ABSTRACT *(200 words or less)*

Characterization of Flaws in U.S. Reactor Pressure Vessels is a multi-volume report. Volume 3, this document, contains the density and distribution of flaw indications in material removed from the non-irradiated Shoreham Nuclear Reactor Pressure Vessel. The flaw indications were obtained from nondestructive evaluation (NDE) of weldment specimens. The first volume gives the density and distribution of flaw indications in the Pressure Vessel Research User Facility (PVRUF) vessel. Volume 2 contains a description of the removal of material from the PVRUF vessel, the conduct of confirmatory NDE techniques and metallographic analysis, and the confirmation of flaw rates for the vessel.

This volume provides the characteristics of the flaw indications in the Shoreham vessel and their density and distribution. This report also gives a description of the Shoreham vessel weldments and the approach to the research. The performance of the inspection system and the measurements made on the reactor pressure vessel (RPV) material are described.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Ultrasonic Testing, Inservice Inspection, Nondestructive Testing, Nondestructive Examination, Nondestructive Evaluation, Reactor Pressure Vessels, Fabrication Flaw

170

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/CR-6505, Vol. 2
ORNL/TM-13323/V2

2. TITLE AND SUBTITLE

The Potential for Criticality Following Disposal of Uranium at Low-Level-Waste Facilities

Containerized Disposal

3. DATE REPORT PUBLISHED

MONTH	YEAR
June	1999

4. FIN OR GRANT NUMBER

L1376

5. AUTHOR(S)

L. E. Toran, C. M. Hopper, C. V. Parks, ORNL
V. A. Colten-Bradley, NRC

6. TYPE OF REPORT

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8. PERFORMING ORGANIZATION – NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

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P. O. Box 2008
Oak Ridge, Tennessee 37831-6370

9. SPONSORING ORGANIZATION – NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

Division of Waste Management
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

The purpose of this study was to evaluate whether or not fissile uranium in low-level-waste (LLW) facilities can be concentrated by hydrogeochemical processes to permit nuclear criticality. A team of experts in hydrology, geology, geochemistry, soil chemistry, and criticality safety was formed to develop and test some reasonable scenarios for hydrogeochemical increases in concentration of special nuclear material (SNM), and to use these scenarios to aid in evaluating the potential for nuclear criticality. The team's approach was to perform simultaneous hydrogeochemical and nuclear criticality studies to (1) identify some possible scenarios for uranium migration and concentration increase at LLW disposal facilities, (2) model groundwater transport and subsequent concentration increase via precipitation of uranium, and (3) evaluate the potential for nuclear criticality resulting from potential increases in uranium concentration over disposal limits. The analysis of SNM was restricted to ^{235}U in the present scope of work. The work documented in this report indicates that the potential for a criticality safety concern to arise in an LLW facility is extremely remote, but not impossible. Theoretically, conditions that lead to a potential criticality safety concern might arise. However, study of the hydrogeochemical mechanisms, the associated time frames, and the factors required for an actual criticality event indicate that proper emplacement of the SNM at the site can eliminate practical concerns relative to the occurrence and possible consequences of a criticality event.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

uranium, low-level waste (LLW), special nuclear material (SNM), nuclear criticality, uranium migration, hydrogeochemical modeling, Barnwell

13. AVAILABILITY STATEMENT

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14. SECURITY CLASSIFICATION

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/CR-6510, Vol. 1
PU NE-96/2

2. TITLE AND SUBTITLE

Corium Dispersion in Direct Containment Heating

Separate Effect Experiments With Water and Woods Metal Simulating Core Melt for Zion Reactor Conditions

3. DATE REPORT PUBLISHED

MONTH	YEAR
September	1999

4. FIN OR GRANT NUMBER

L1990

5. AUTHOR(S)

M. Ishii, Q. Wu, S.T. Revankar, S. Kim, G. Zhang, Purdue University
R.Y. Lee, C.G. Tinkler, U.S. Nuclear Regulatory Commission

6. TYPE OF REPORT

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7. PERIOD COVERED *(Inclusive Dates)*

3/92 - 10/95

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

School of Engineering
Purdue University
West Lafayette, IN 47907

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

R.Y. Lee, NRC Project Manager

11. ABSTRACT *(200 words or less)*

The research at Purdue University addresses corium dispersion during the Direct Containment Heating scenario in a severe nuclear reactor accident. The degree of corium dispersion has not only the strongest parametric effects on the containment pressurization, but also has the highest uncertainty in predicting it. In view of this, a separate effect testing program on the corium dispersion mechanisms in the reactor cavity and the subcompartment trapping mechanisms was initiated at the Purdue University. The four major objectives of this study are: (1) to perform a detailed study using a step-by-step integral scaling method, and to evaluate existing models and correlations for droplet entrainment, particle size distribution and particle trapping, (2) to design and construct a 1/10 scale Zion reactor model, and to perform carefully scaled experiments using air-water and air-woods metal to simulate the prototypic steam and core melt, (3) to develop reliable mechanistic models for the corium dispersion and transport in the accident scenario, which can be used to predict the liquid and gas blowdown, entrainment droplet size, liquid carryover to the containment, and the subcompartment trapping, and (4) to use the models to perform stand alone calculations for the prototypic conditions. In this report (volume 1), efforts are focused on the first two objectives, whereas the modeling study is documented in volume 2.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

direct containment heating
corium dispersion
scaling
severe accident
separate effect experiments

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/CR-6510, Vol. 2
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2. TITLE AND SUBTITLE

Corium Dispersion in Direct Containment Heating

Theoretical Analysis of the Hydrodynamic Characteristics

3. DATE REPORT PUBLISHED

MONTH	YEAR
September	1999

4. FIN OR GRANT NUMBER

L1990

5. AUTHOR(S)

M. Ishii, Q. Wu, G. Zhang, Purdue University
R.Y. Lee, C.G. Tinkler, U.S. Nuclear Regulatory Commission

6. TYPE OF REPORT

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7. PERIOD COVERED *(Inclusive Dates)*

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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

School of Engineering
Purdue University
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9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

R.Y. Lee, NRC Project Manager

11. ABSTRACT *(200 words or less)*

The research at Purdue University addresses corium dispersion during the Direct Containment Heating scenario in a severe nuclear reactor accident. The degree of corium dispersion has not only the strongest parametric effects on the containment pressurization, but also has the highest uncertainty in predicting it. In view of this, a separate effect testing program on the corium dispersion mechanisms in the reactor cavity and the subcompartment trapping mechanisms was initiated at the Purdue University. The four major objectives of this study are: (1) to perform a detailed study using a step-by-step integral scaling method, and to evaluate existing models and correlations for droplet entrainment, particle size and size distribution and particle trapping, (2) to design and construct a 1/10 scale Zion reactor model, and to perform carefully scaled experiments using air-water and air-woods metal to simulate the prototypic steam and core melt, (3) to develop reliable mechanistic models for the corium dispersion and transport in the accident scenario, which can be used to predict the liquid and gas blowdown, entrainment, droplet size, liquid carryover to the containment, and the subcompartment trapping, and (4) to use the models to perform stand alone calculations for the prototypic conditions. In this report (volume 2), efforts are focused on the last two objectives, whereas the scaling and experiments are documented in volume 1.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

direct containment heating
corium dispersion
scaling
severe accident
separate effect experiments

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NUREG/CR-6511, Vol. 4
ANL-98/15

2. TITLE AND SUBTITLE

Steam Generator Tube Integrity Program

Annual Report
October 1996 - September 1997

3. DATE REPORT PUBLISHED

MONTH	YEAR
January	1999

4. FIN OR GRANT NUMBER

W6487

5. AUTHOR(S)

D.R. Diercks, S. Bakhtiari, K.E. Kasza, D.S. Kupperman,
S. Majumdar, J.Y. Park, W.J. Shack

6. TYPE OF REPORT

Technical; Annual

7. PERIOD COVERED *(Inclusive Dates)*

Oct. 1996 - Sept. 1997

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Argonne National Laboratory
9700 South Cass Avenue
Argonne, IL 60439

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

J. Muscara, NRC Project Manager

11. ABSTRACT *(200 words or less)*

This report summarizes work performed by Argonne National Laboratory on the Steam Generator Tube Integrity Program during the period October 1996 through September 1997. The program is divided into five tasks: (1) assessment of Inspection Reliability, (2) Research on ISI (in-service-inspection) Technology, (3) Research on Degradation Modes and Integrity, (4) Tube Removals from Steam Generators, and (5) Program Management. Under Task 1, progress is reported on the assembly of a steam generator tube mockup for round-robin studies and on the evaluation of NDE techniques for characterizing the tubes going into the mockup. Inspection data from the Duke Power Company's McGuire Nuclear Station were evaluated to optimize the selection of tube samples for removal from two retired steam generators. Under Task 2, results are reported on the application of signal processing, visualization, and data analysis schemes to improve the NDE of service-degraded tubing. Results are also presented on implementation of multivariate linear and nonlinear models to study potential correlations between eddy current measurements and flaw size and tube failure pressure. In Task 3, a model boiler multitube corrosion cracking facility has been designed to simulate steam generator thermal-hydraulic and chemistry conditions for the tube/crevice chemistry. A Pressure and Leak-Rate Test Facility is being built to determine failure pressures and leak rates for flawed tubing under normal operating and design-basis accident conditions. An autoclave system and a room-temperature cracking facility are being used to produce cracked specimens for pressure and leak-rate tests and NDE studies. The results of 15 severe-accident pressure tests on Alloy 600 tubing with machined circumferential part-throughwall flaws are also reported. Under Task 4, the selection and removal of service degraded tubes, tube sheet samples, and tube support plate samples from the McGuire Nuclear Station are described.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Steam Generator
Tubes
Stress Corrosion Cracking
Eddy Current Testing
Nondestructive Evaluation
in-service inspection
Pressure Testing
Tube Burst
Leak Rate
Alloy 600, Inconel 600

13. AVAILABILITY STATEMENT

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/CR-6511, Vol. 5
ANL-98/23

2. TITLE AND SUBTITLE

Steam Generator Tube Integrity Program

Semiannual Report
October 1997 - March 1998

3. DATE REPORT PUBLISHED

MONTH	YEAR
April	1999

4. FIN OR GRANT NUMBER

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5. AUTHOR(S)

D.R. Diercks, S. Bakhtiari, K.E. Kasza, D.S. Kupperman,
S. Majumdar, J.Y. Park, W.J. Shack

6. TYPE OF REPORT

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7. PERIOD COVERED *(Inclusive Dates)*

Oct. 1997 - March 1998

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Argonne National Laboratory
9700 South Cass Avenue
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9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

J. Muscara, NRC Project Manager

11. ABSTRACT *(200 words or less)*

This report summarizes work performed by Argonne National Laboratory on the Steam Generator Tube Integrity Program during the period from October 1997 through March 1998. The program is divided into five task, namely: (1) assessment of Inspection Reliability, (2) Research on ISI Technology, (3) Research on Degradation Modes and Integrity, (4) Development of Methodology and Technical Requirements for Current and Emerging Regulatory Issues, and (5) Program Management. Under Task 1, progress is reported on the assembly of the steam generator tube mockup, the characterization of the mockup tubes, the development of the round robin protocol and procedures, and the further analysis of data from the McGuire steam generators. Activities in Task 2 focused on analytical methods for the prediction of EC response, development of effective signal analysis procedures, development of flaw imaging and display methods, and evaluation of improved probe designs using directional arrays. Under Task 3, additional laboratory-degraded tubes were produced, the Pressure and Leak-Rate Test Facility neared completion of shakedown and performance qualification testing, a series of finite element analyses was conducted for multiple colinear cracks with various ligament widths, a literature survey on the mechanisms of crack initiation and arrest neared completion, and decontamination of the McGuire tubes was initiated. Under Task 4, progress is reported on the NDE of electrosleeved tubes.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Steam Generator
Tubes
Stress Corrosion Cracking
Eddy Current Testing
Nondestructive Evaluation
In-service Inspection
Pressure Testing
Tube Burst
Leak Rate
Alloy 600, Inconel 600

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1. REPORT NUMBER *(Assigned by NRC. Add Vol., Supp., Rev., and Addendum Numbers, if any.)*
NUREG/CR-6511, Vol. 6
ANL-99/8

2. TITLE AND SUBTITLE
Steam Generator Tube Integrity Program:
Annual Report, October 1997—September 1998

3. DATE REPORT PUBLISHED

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October	1999

4. FIN OR GRANT NUMBER
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5. AUTHOR(S)
D. R. Diercks, S. Bakhtiari, K. E. Kasza, D. S. Kupperman, S. Majumdar, J. Y. Park, and W. J. Shack

6. TYPE OF REPORT
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7. PERIOD COVERED *(Inclusive Dates)*
Oct. 1997—Sept. 1998

8. PERFORMING ORGANIZATION – NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*
Argonne National Laboratory
9700 South Cass Avenue
Argonne, IL 60439

9. SPONSORING ORGANIZATION – NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*
Division of Engineering Technology
Office of Nuclear Regulatory Research
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES
J. Muscara, NRC Project Manager

11. ABSTRACT (200 words or less)
This report summarizes work performed by Argonne National Laboratory on the Steam Generator Tube Integrity Program during the period October 1997-September 1998. Under Task 1, progress is reported on the assembly of a steam generator tube mock-up for round-robin studies on nondestructive evaluation (NDE) procedures, on the evaluation of NDE techniques for characterizing the tubes going into the mock-up, and on the development of protocols and procedures for the round-robin. In addition, results are reported on the EC inspection of deplugged tubes obtained from the McGuire Nuclear Station. Under Task 2, results are reported on numerical and experimental results on the response of a bobbin coil probe to axial notches in calibration-standard tubes. Multivariate linear and nonlinear models for the correlation of eddy current data with tube structural integrity are also being evaluated. In Task 3, cracked tubes are being produced for the steam generator tube mock-up. Checkout, shakedown, and performance qualification of the Pressure and Leak-Rate Test Facility are described, and the results of initial tests are reported. In addition, a series of finite-element analyses was conducted for multiple collinear cracks with various ligament widths, a test matrix was developed for the testing of tubes with axial machined notches, a literature survey is being completed on the mechanisms of stress corrosion cracking, and the tubes removed from the McGuire steam generators are being decontaminated. Under Task 4, the results of eddy current and ultrasonic examinations of electrosleeved tubes are reported, and residual stresses in these tubes have been measured by neutron diffraction techniques. The results of a critical review of the corrosion resistance of electrosleeved material are also summarized.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating this report.)*
Steam Generator
Tubes
Stress Corrosion Cracking
Eddy Current Testing
Nondestructive Evaluation
In-service Inspection
Pressure Testing
Tube Burst
Leak Rate
Alloy 600, Inconel 600

176

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NUREG/CR-6512
ORNL/TM-13342
NSWCCD-TR-61-CR-96/01

2. TITLE AND SUBTITLE

Dynamic Fracture Initiation Toughness of ASTM A533, Grade B Steel Plate

3. DATE REPORT PUBLISHED

MONTH	YEAR
May	1999

4. FIN OR GRANT NUMBER

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5. AUTHOR(S)

R.E. Link*, S.M. Graham**

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8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

*U.S. Naval Academy, Annapolis, MD 21402
**Vector Research, Rockville, MD 20852

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Oak Ridge, TN 37831

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Division of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

S.N. Malik, NRC Project Manager

11. ABSTRACT (200 words or less)

The dynamic fracture toughness of an ASTM A533, Grade B steel plate was determined at several temperatures in the ductile-brittle transition region. Crack-tip loading rates ranged from approximately 10^3 to 10^5 MPa$\sqrt{m}$/s. The fracture toughness was shown to decrease with increased loading rate. The dynamic fracture toughness was compared with results from previous investigations and it was shown that the decrease in toughness due to increased loading rate at the highest test temperature was not as severe as reported in previous investigations. It was also shown that the reference temperature, T_0, was a better index of the fracture toughness vs. temperature relationship than the nil-ductility temperature, RT_{NDT} for this material.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

fracture toughness, dynamic loading, reactor pressure vessel steel

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NUREG/CR-6516
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2. TITLE AND SUBTITLE

Pretest Analyses of the Steel Containment Vessel Model

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MONTH	YEAR
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4. FIN OR GRANT NUMBER

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5. AUTHOR(S)

V.L. Porter, P.A. Carter, S.W. Key

6. TYPE OF REPORT

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June 1993-November 1996

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Sandia National Laboratories
Albuquerque, NM 87185-0443

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

J.F. Costello, NRC Project Manager

11. ABSTRACT *(200 words or less)*

As part of the containment integrity program jointly sponsored by the Nuclear Power Engineering Corporation (NUPEC) of Japan and the United States Nuclear Regulatory Commission (NRC), Sandia National Laboratories (SNL) conducted a high pressure test of a steel containment vessel (SCV) model, nominally based on a Japanese Improved MK-II boiling water reactor containment. The test included an external contact structure (CS), a steel shell that covered most of the SCV model with a gap between the two structures. One of the program objectives is to validate analytical methods used to predict the response of containment buildings subjected to severe accident pressures. This report describes the finite element analyses conducted by Sandia in support of the test program and for pretest prediction of model behavior. Preliminary calculations were performed to support model design, such as the effects of mixed scaling and the effects of including a contact structure in the test. Global response of the SCV model was predicted using both axisymmetric and three-dimensional shell models. An axisymmetric continuum analysis of the top head and a three-dimensional shell analysis of the equipment hatch region were developed to provide detailed mappings of local model responses.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Finite element analysis
Reactor containments
Steel containment vessel
Severe accident
Structural response
Model validation

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NUREG/CR-6532
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2. TITLE AND SUBTITLE
Systems Analysis Programs for Hands-on Reliability Evaluations (SAPHIRE) Version 6.0 – System Overview Manual

3. DATE REPORT PUBLISHED

MONTH	YEAR
May	1999

4. FIN OR GRANT NUMBER
L1429

5. AUTHOR(S)
K.D. Russell, C. L. Hoffman, K.J, Kvardfordt, E. Lois*, C. L. Smith, S. T. Wood

* U. S. Nuclear Regulatory Commision

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Idaho National Engineering and Environmental Laboratory
Lockheed Martin Idaho Technologies Co.
P.O. Box 1625
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Division of Risk Analysis and Applications
Office for Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES
Project Manager: E. Lois

11. ABSTRACT (200 words or less)

The Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) is a software application for performing probabilistic risk assessments (PRAs) for industrial facilities and in particular for nuclear power plants, using a personal computer. This is a summary report of the functions and capabilities available in SAPHIRE. It provides instructions for installing and using the code and discusses SAPHIRE's database structure and concepts: PRA data generation and manipulation, quantification, and analysis; functions for performing end-state analysis; and SAPHIRE's quality assurance process. This report also gives an overview of the separate module called the Graphical Evaluation Module (GEM), which automates the process for evaluating operational events at commercial nuclear plants. Detailed documentation of SAPHIRE is available in the on-line reference manual.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)
Computer code, probabilistic risk analysis, PRA, SAPHIRE, GEM, probability, frequency, initiating event assessments, fault tree analysis, event tree analysis, operational event assessments, safety assessments

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2. TITLE AND SUBTITLE Severe Accident Risks for VVER Reactors: The Kalinin PRA Program Volume 3: Procedure Guides	3. DATE REPORT PUBLISHED MONTH: September — YEAR: 1999 4. FIN OR GRANT NUMBER R9608
5. AUTHOR(S)	6. TYPE OF REPORT Technical 7. PERIOD COVERED *(Inclusive Dates)*

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Brookhaven National Laboratory
Upton, NY 11973-5000

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Risk Analysis and Applications
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

In order to facilitate the probabilistic risk assessment (PRA) of a VVER-1000 nuclear power plant, a set of procedure guides has been written. These procedure guides, along with training supplied by experts and supplementary material from the literature, were used to advance the PRA carried out for the Kalinin Nuclear Power Station in the Russian Federation. Although written for a specific project, these guides have general applicability. For a Level 1 PRA (determination of core damage frequency for different scenarios), the guides are written for all of the technical tasks involved for internal events, including internal fires and floods and seismic events. Guides are also provided for a Level 2 PRA (probabilistic accident progression and source term analysis) and a Level 3 PRA (consequence analysis and integrated risk assessment). In addition, introductory material is provided to explain the rationale and approach for a PRA. Procedure guides are also provided on the quality assurance and documentation requirements.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*	13. AVAILABILITY STATEMENT unlimited
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NUREG/CR-6578
ORNL-6932

2. TITLE AND SUBTITLE

A Methodology for Evaluation of Inservice Test Intervals for Pumps and Motor-Operated Valves

3. DATE REPORT PUBLISHED

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4. FIN OR GRANT NUMBER

W6324

5. AUTHOR(S)

K.L. McElhaney, D.C. Fox, H.D. Haynes
P.J. Otaduy, R.H. Staunton, W.E. Vesely*

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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Oak Ridge National Laboratory
Oak Ridge, TN 37831-8038

*Consultant
4805 Lakeview Drive
Powell, OH 43065

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

J. Jackson, NRC Project Manager

11. ABSTRACT *(200 words or less)*

Recent industry reevaluation of component inservice testing (IST) requirements has resulted in requests for IST interval extension and changes to traditional IST programs. To evaluate these requests, long-term component performance and the methods for mitigating degradation need to be understood. Determining the appropriate IST intervals, along with component testing, monitoring, trending, and maintenance effects, has become necessary. This study provides guidelines to support the evaluation of IST intervals for pumps and motor-operated valves (MOVs). It presents specific engineering information pertinent to the performance and monitoring/testing of pumps and MOVs, provides an analytical methodology for assessing the bounding effects of aging on component margin behavior, and identifies basic elements of an overall program to help ensure component operability. guidance for assessing probabilistic methods and the risk importance of safety consequences of the performance of pumps and MOVs has not been specifically included within the scope of this report, but these elements may be included in licensee change requests.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

inservice testing (IST), pumps, motor-operated valves (MOVs), component performance, test interval, condition monitoring, risk-informed testing, margin, trending, maintenance, component operability

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)
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2. TITLE AND SUBTITLE
Environmental Dynamics of Carbon-14 Near a Low-Level Radioactive Wasre Burial Ground

3. DATE REPORT PUBLISHED

MONTH	YEAR
May	1999

4. FIN OR GRANT NUMBER
L-1935

5. AUTHOR(S)
S.O. Link, WSU Tri Cities; K.J. King, Chalk River Laboratories;
M. Benz, Chalk River Laboratories; W.G. Evenden, Chalk River Laboratories;
D.E. Robertson, Pacific Northwest National Laboratory

6. TYPE OF REPORT
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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*
Washington State University - Tri Cities, 2710 University Drive, Richland, WA 99352
Chalk River Laboratories, Chalk River, Ontario, Canada KOJIJO
Pacific Northwest National Laboratory, P.O. Box 999, Richland, WA 99352

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*
Division of Risk Analysis and Application
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES
E.O'Donnell, NRC Project Manager

11. ABSTRACT *(200 words or less)*

To better understand the environmental pathways for transport of $^{14}CO_2$ from low-level radioactive waste management (LLW) sites, a field study was conducted over a two-year period at a waste management area at Chalk River Laboratories, Chalk River, Ontario, Canada. The transport rates and mechanisms of $^{14}CO_2$ as it moves from groundwater into the atmosphere and then into the ambient vegetation was investigated near the Area C solid LLW disposal site. At this location, a slightly contaminated groundwater plume extends from the near-surface waste burial site and emerges in a wetlands area (Duke Swamp) about 300 meters away. Measurements of $^{14}CO_2$ in groundwater, surface water, air, vegetation, and soil (peat) during the major growing season (spring to fall) showed that the dominant pathway for transport of the $^{14}CO_2$ from the groundwater to the vegetation was via emission to the atmosphere followed by photosynthetic uptake. Experiments with native and transplanted vegetation at the study site showed that the transpirational pathway of $^{14}CO_2$ into the vegetation was essentially negligible compared to the atmospheric transport route. The uptake of $^{14}CO_2$ by vegetation was highly dependent on air temperature and the height of the vegetation above ground surface.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*
Carbon-14
radionuclide transport
vegetative uptake

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/CR-6592

2. TITLE AND SUBTITLE

Evaluation of Terminated Nuclear Material Licenses

A Report of Identified Sites and Sealed Source Licenses

3. DATE REPORT PUBLISHED

MONTH	YEAR
February	1999

4. FIN OR GRANT NUMBER

L1596

5. AUTHOR(S)

E.A. Zeighami, K.M. Spencer

6. TYPE OF REPORT

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Sept. 1990 - Jan. 1998

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Oak Ridge National Laboratory
Oak Ridge, TN 37831-6151

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Waste Management
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This report presents the results of a six-year project that reviewed material licenses that had been terminated during the period from inception of licensing until approximately late 1994. The material licenses covered in the review project were Part 30, or byproduct material licenses, Part 40, or source material licenses and Part 70, or special nuclear material licenses. The report describes the methodology developed for the project, summarizes the findings of the license file inventory process, and describes the findings of the reviews or evaluations of the license files. The review identified sites of nuclear material use for which either review of the licensing material or more direct follow-up of some type was judged to be needed. The review process also identified licenses authorized to possess sealed sources for which there was incomplete or missing documentation of the fate of the sources.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

byproduct materials, source materials, SNM, material licensing, contamination, contaminated sites, sealed sources, licensing, expert system

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13. AVAILABILITY STATEMENT

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NRC FORM 335 (2-89) NRCM 1102, 3201, 3202	U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET** *(See instructions on the reverse)*	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any) NUREG/CR-6604 SAND98-0272 Supplement 1
2. TITLE AND SUBTITLE RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation		3. DATE REPORT PUBLISHED MONTH: June YEAR: 1999
		4. FIN OR GRANT NUMBER Y6062
5. AUTHOR(S) N.E. Bixler and C.M. Erickson (SNL)		6. TYPE OF REPORT Technical
		7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION – NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

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9. SPONSORING ORGANIZATION – NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

C. G. Gingrich, NRC Project Manager

11. ABSTRACT *(200 words or less)*

This report is a supplement to the original RADTRAD user's manual. It describes modifications that have been made to the graphical user interface (GUI) and to the numerical engine used to solve coupled ordinary differential equations. Other improvements to the code are also described.

A major portion of this report is a replacement to the original user's guide, which describes how to install and use the current version (3.01). The GUI is now based on Visual Basic and operates quite differently than the GUI used in earlier code versions. The original numerical engine, which was based on the Laplace transform technique, has also been replaced with a new method that is both faster and more accurate. One new test case has been added to the standard test suite. Updated results for the entire suite of test problems are presented. Finally, a description of the new input format is provided.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*	13. AVAILABILITY STATEMENT Unlimited
nuclear power reactor accident, design basis accident, severe accident, source terms, removal mechanisms, environment dose, control room dose	14. SECURITY CLASSIFICATION *(This Page)* Unclassified *(This Report)* Unclassified
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1. REPORT NUMBER
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NUREG/CR-6618

2. TITLE AND SUBTITLE

Development and Findings of the Performance Trending Methodology

3. DATE REPORT PUBLISHED

MONTH	YEAR
April	1999

4. FIN OR GRANT NUMBER

J8229

5. AUTHOR(S)

6. TYPE OF REPORT

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7. PERIOD COVERED *(Inclusive Dates)*

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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Arthur Andersen LLP, Office of Government Services, 1150 17th Street, N.W., Suite 901, Washington, D.C. 20036-4613

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Risk Analysis & Applications
Office of Nuclear Regulatory Research
US Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

M. Harper, NRC Project Manager

11. ABSTRACT *(200 words or less)*

This report summarizes the development of a performance trend methodology that has been used as one factor to evaluate plant performance and to identify plants for discussion at the 1998 and 1999 NRC Senior Management Meeting (SMM). The report outlines four methodologies that were analyzed for potential utilization in evaluating plant performance. A trending model, a regression model, a statistical process control (SPC) tool, and a cluster analysis tool are discussed to provide the uses and benefits of each model. The purpose of these methodologies was to assist the NRC in its regulatory effectiveness by offering an objective and scrutable tool designed to evaluate plant performance and to identify potential plants for discussion.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

performance indicators
data
nuclear power plant

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1. REPORT NUMBER
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NUREG/CR-6619

2. TITLE AND SUBTITLE

Paleoliquefaction Studies in the South Carolina Coastal Plain

3. DATE REPORT PUBLISHED

MONTH	YEAR
April	1999

4. FIN OR GRANT NUMBER

L2181

5. AUTHOR(S)

P. Talwani, D.C. Amick, W.T. Schaeffer

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7. PERIOD COVERED *(Inclusive Dates)*

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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

University of South Carolina
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Columbia, SC 29208

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Engineering Technology
Office of Nuclear Regulatory Research
US Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

E. G. Zurflueh, NRC Project Manager

11. ABSTRACT *(200 words or less)*

We conducted new paleoliquefaction studies in the South Carolina Costal Plain and reanalyzed similar studies over the past fifteen years. We discovered a new paleoliquefaction site at Gapway and recalibrated all radiocarbon ages using a computer program to obtain a uniform age date set. The results of this analysis reveal the existence of seven prehistoric episodes and two possible scenarios for their origin. In the first scenario, there are three seismic sources in the Coastal Plain, and the episodes, locations, and magnitudes are as follows: Episode **A**, 546 ± 17 YBP, Charleston, M 7+; **B**, 1001 ± 33 YBP, Charleston, M 7+; **C**, 1648 ± 74 YBP, Georgetown, M 6.0; **D**, 1966 ± 212 YBP, Bluffton, M 6.0; **E**, 3548 ± 66 YBP, Charleston, M 7+; **F**, 5038 ± 166 YBP, Georgetown, M 6+; and Episode **G**, 5800 ± 500 YBP, Charleston, M 7+. In the second scenario, Episodes C and D are combined into one episode, **C'** 1683 ± 70 YBP. In this scenario all earthquakes occurred at Charleston with M 7+. Statistical data favor the first scenario although the second one cannot be rejected. Episodes A to C seem to be more representative of the earthquake cycle and suggest a recurrence time of 500 to 600 years for the more recent M 7+ events at Charleston.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Paleoseismology
Earthquake Recurrence
Charleston, SC

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/CR-6622

2. TITLE AND SUBTITLE

Probabilistic Liquefaction Analysis

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4. FIN OR GRANT NUMBER

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5. AUTHOR(S)

M.E. Hynes

6. TYPE OF REPORT

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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

U.S. Army Corps of Engineers
Waterways Experiment Station
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9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

E.G. Zurflueh, NRC Project Manager

11. ABSTRACT *(200 words or less)*

This document provides technical bases for formulating probabilistic approaches to liquefaction evaluation. The three basic elements of probabilistic liquefaction analysis are: (1) uncertainty in the earthquake load, (2) uncertainty in the available resistance, and (3) uncertainty in the method of analysis. The probabilistic approach is built from the steps in a deterministic liquefaction analysis; however, the input parameters, such as penetration resistance, site stratigraphy, acceleration, and magnitude, are treated as random variables and the accuracy of the method of analysis is factored in as a part of a capacity-demand model. Uncertainty in the earthquake load is generally treated with a probabilistic seismic hazard analysis, which introduces time as a parameter. The site stratigraphy and engineering properties are generally treated as one-, two-, or three-dimensional random fields. Uncertainty in the method of analysis is generally estimated with logit regression analysis of the field performance data base. It is assumed that the reader has a working knowledge of probability theory, stochastic processes, liquefaction evaluation, and probabilistic seismic hazard analysis calculations.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Capacity-demand model
Uncertainty
Logit regression

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

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2. TITLE AND SUBTITLE

Vapor Explosions in a One-Dimensional Large Scale Geometry with Simulant Melts

3. DATE REPORT PUBLISHED

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4. FIN OR GRANT NUMBER

W6183

5. AUTHOR(S)

H.S. Park, R. Chapman, and M.L. Corradini

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Nuclear Safety Research Center
University of Wisconsin - Madison
Madison, Wisconsin 53706

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C 20555-0001

10. SUPPLEMENTARY NOTES

S. Basu, NRC Project Manager

11. ABSTRACT *(200 words or less)*

In light water reactors after a prolonged lack of cooling, vapor explosions could occur when molten fuel is generated and contacts residual water coolant within the reactor vessel or below in the containment reactor cavity. The objectives for this work were to obtain well-characterized experimental data for the explosion propagation/escalation phases using different melt simulants and to investigate the effects of a comprehensive set of initial and boundary conditions on the explosion energetics; i.e., trigger strength, fuel mass, composition and temperature, coolant mass, viscosity and temperature, and system constraint.

This experimental work has yielded a number of results that have potentially important safety implications. First, it has provided evidence of the reproducibility of vapor explosion energetics for a controlled set of initial and boundary conditions. Second, the experiments have demonstrated that if geometric scaling is properly specified, it is possible to extrapolate the results of laboratory scale experiments to reactor scale predictions. Finally and most importantly, the experimental data suggests that once the fuel-coolant initial conditions are within an envelope for triggered events, the energetics is much less than thermodynamic limit, apparently due to the small amount of fuel that participates in the explosion time scale. This envelope of triggerability is much smaller for a simulant molten oxide with low superheat, such as molten iron-oxide in the tests described in the report and corium in the KROTOS tests.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

fuel-coolant interactions (FCI), steam explosions, vapor explosions, simulant melts, molten tin, molten iron oxide, corium, zirconium-containing melt, chemical augmentation, explosion propagation/escalation, explosion energetics, explosivity, thermodynamic limit, triggering envelope, initial and boundary conditions trigger strength, fuel mass and composition, superheat, subcooling, viscosity, system constraint, severe accident, nuclear reactor safety

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2. TITLE AND SUBTITLE Recommendations for Revision of Regulatory Guide 1.78	3. DATE REPORT PUBLISHED MONTH: November YEAR: 1999 4. FIN OR GRANT NUMBER W6308
5. AUTHOR(S) L.B. Sasser, P.M. Daling, P. Pelto, M. Yurconic	6. TYPE OF REPORT Technical 7. PERIOD COVERED *(Inclusive Dates)* September 1998-September 1999

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Pacific Northwest National Laboratory
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Richland, WA 99352

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Systems Analysis and Regulatory Efectiveness
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

S. Basu, NRC Project Manager

11. ABSTRACT *(200 words or less)*

To ensure safe operation of commercial nuclear power plants, control room operators must be protected from dangers arising from possible exposure to hazardous chemicals that may be discharged as a result of equipment failure, operator errors, or events external to plant operation. Conditions must exist where accidental exposure to such materials still allows the operators to operate the plant safely. Regulatory Guide 1.78 provides guidance in assessing the control room habitability of the control room during and after a postulated external release of hazardous chemicals from mobile or stationary sources, offsite or onsite. This report provides recommendations for revising the Regulatory Guide 1.78 in two areas, namely, control room ventilation flow modeling and toxicity limit. Additionally, the report provides a value and impact analysis associated with the revision of Regulatory Guide 1.78.

In the area of ventilation flow modeling, the report recommends the use of the HABIT code, in particular, the EXTRAN module of the code. EXTRAN represents an improvement in atmospheric dispersion modeling. In the area of toxicity limits, the report recommends the use of National Institute for Occupational Safety and Health (NIOSH) Immediately Dangerous to Life and Health (IDLH) concentration values. The IDLH values, based on a 30-minute exposure level, is defined as one that is likely to cause death or immediate delayed permanent adverse health effects if no protection is afforded within 30 minutes. Control room operators are expected to use protective measures within 2 minutes after the detection of hazardous chemicals so that they will not be subjected to prolonged exposure at the IDLH concentration levels. Thus, the IDLH limits represent reasonable values to provide adequate margin of safety in protecting control room operators.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*	13. AVAILABILITY STATEMENT unlimited
Regulatory Guide 1.78, Regulatory Guide 1.95, control room habitability, hazardous chemical release, toxic chemical release, chlorine, toxicity limits, immediately dangerous to life and health (IDLH) odor threshold, control room ventilation flow, air dispersion modeling, Gaussian plume model, meteorology, frequent shipment criterion, transportation accident statistics, value/impact assessment, regulatory efficiency	14. SECURITY CLASSIFICATION *(This Page)* unclassified *(This Report)* unclassified
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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/CR-6631
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2. TITLE AND SUBTITLE

Fragility Modeling of Aging Containment Metallic Pressure Boundaries

3. DATE REPORT PUBLISHED

MONTH	YEAR
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4. FIN OR GRANT NUMBER

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5. AUTHOR(S)

B.R. Ellingwood, The Johns Hopkins University
J.L. Cherry, Sandia National Laboratories

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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

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Subcontractor:
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Sandia National Laboratories
Albuquerque, NM 87185-0744

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

H.L. Graves III, NRC Project Manager

11. ABSTRACT *(200 words or less)*

A general framework for probabilistic modeling of containment structural performance, with specific emphasis on steel containments subjected to corrosion is presented. Mathematical foundations for fragility modeling are presented; sources of uncertainty are identified and quantified; and an experimental sampling plan to minimize the finite-element analyses are required to generate the fragility models is presented. Fragilities are developed for a PWR ice condenser containment in uncorroded and corroded conditions. Corrosion patterns were postulated based on what has been observed by location, areal extent, and mean loss of shell thickness. A discussion of insights and perspectives that might be drawn from such fragility analyses is provided.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

containment
corrosion
design
finite-element analysis
fragility
limit states
probability
reliability
statistics
structural engineering

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1. REPORT NUMBER *(Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)*

NUREG/CR-6639
SAND99-1464

2. TITLE AND SUBTITLE

Seismic Analysis of a Prestressed Concrete Containment Vessel Model

3. DATE REPORT PUBLISHED

MONTH	YEAR
August	1999

4. FIN OR GRANT NUMBER

W6251

5. AUTHOR(S)

R.J. James, L. Zhang, Y.R. Rashid, ANATECH Corporation
J.L. Cherry, Sandia National Laboratories

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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

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San Diego, CA 92121

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

N.C. Chokshi, NRC Project Manager

11. ABSTRACT *(200 words or less)*

A 1:10 scale Prestressed Concrete Containment Vessel (PCCV) model was constructed by the Nuclear Power Engineering Corporation (NUPEC) of Japan and subjected to seismic simulation tests using the high-performance shaking table at the Tadotsu Engineering Laboratory. A series of tests representing design-level seismic ground motions was initially conducted. These were followed by a series of tests in which progressively larger base motions were applied. Sandia National Laboratories and ANATECH Corp. conducted independent analyses to predict the seismic behavior of the scaled model PCCV structure. Dynamic 3D finite element analyses were performed before and after the tests in a sequence that corresponded to the test series. The pretest analyses, which relied on target seismic accelerations. gave good predictions of the model's behavior for the design-level earthquake simulations. However, for higher level motions, the pretest analyses significantly underestimated the structural response. Input accelerations for the post-test calculations used actual basemat accelerations that were measured during the test. In addition, the concrete constitutive shear-degradation model was modified. With these improvements, the post-test analyses showed good agreement with test data throughout the design-level and failure-level series of tests.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Concrete Constitutive Material Model, Damping, Earthquake, Finite Element Analysis, Failure Prediction, Prestressed Concrete Containment Vessel (PCCV), Scaled Model Test, Seismic

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/CR-6641

2. TITLE AND SUBTITLE

Final Report of NRC AP600 Research Conducted at Oregon State University

3. DATE REPORT PUBLISHED

MONTH	YEAR
August	1999

4. FIN OR GRANT NUMBER

W6641

5. AUTHOR(S)

J.N. Reyes, Jr., G.T. Groome, A.Y. Lafi, S.C. Franz, C. Rusher,
M. Strohecker, D. Wachs, S. Colpo, S. Binney

6. TYPE OF REPORT

Technical

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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Department of Nuclear Engineering
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9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Systems Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

D.E. Bessette, NRC Project Manager

11. ABSTRACT *(200 words or less)*

This report summarizes a major research effort performed using the Advanced Plant Experiment (APEX) at Oregon State University (OSU). This research program was sponsored by the U.S. Nuclear Regulatory Commission (NRC). The effort was started in January 1995 and the final test was concluded in February 1998. A total of forty-six (46) tests were successfully completed in the APEX facility during this period. The purpose of this program was to obtain experimental data to benchmark NRC's AP600 thermal hydraulic computer codes; to perform confirmatory tests to evaluate the Westinghouse AP600 test series; and to assess the performance of the AP600 passive safety systems for a wide range of accident scenarios; including Beyond Design Basis Accidents. The accident scenarios investigated in APEX include Small Break Loss-of-Coolant Accidents (SBLOCA) simulating ½-inch diameter breaks up to double-ended guillotine breaks of a Direct Vessel Injection (DVI) Line. Parametric tests examining core uncovering, return to saturation oscillations, and nitrogen transport have also been performed. This report includes a description of the APEX facility, summarizes the results of each test or test series and describes the key phenomena observed during the test program.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Advanced Plant Experiment (APEX)
thermal hydraulic computer code
SBLOCA
AP600

13. AVAILABILITY STATEMENT

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NRC FORM 335 (2-89) NRCM 1102, 3201, 3202 — U.S. NUCLEAR REGULATORY COMMISSION — **BIBLIOGRAPHIC DATA SHEET** *(See instructions on the reverse)*	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.) NUREG/CR-6642 SCIE-NRC-379-99
2. TITLE AND SUBTITLE Risk Analysis and Evaluation of Regulatory Options for Nuclear Byproduct Material Systems Draft for Comment	3. DATE REPORT PUBLISHED — MONTH: July, YEAR: 1999 4. FIN OR GRANT NUMBER: J5215
5. AUTHOR(S) E.R. Schmidt, J.F.Meyer, R.W. Youngblood, K.J. Green, W.C. Arcieri	6. TYPE OF REPORT: Draft 7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

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9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Industrial and Medical Nuclear Safety
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This project responds to NRC's Direction Setting Issue 12, Risk-Informed, Performance-Based Regulation. Its scope is limited to nuclear byproduct materials as defined in Section 11.e(1) of the Atomic Energy Act of 1954 and Title 10 off the Code of Federal Regulations (CFR), Section 30.4. 10 CFR Parts 30 through 36 and 39 address regulation of those materials. The goal is to identify risk-informed regulatory options for nuclear byproduct materials. The process involves (1) organization of nuclear byproduct material, as used, into 40 systems, (2) identification of existing and potential physical and procedural barriers that limit dose to workers and the public, (3) risk analysis of each system under normal operation and off-normal (accident) conditions, (4) consideration of regulatory options for each system, and (5) comparison of regulatory options to current regulatory barriers. A graded approach, based on comparison of risk analysis results with dose screening guidelines, is used. Options considered ranged from those that would provide a high level of assurance that doses exceeding guidelines would be prevented, through those where performance-based approaches would assure prevention of doses near the guidelines, to those where little appears necesary to assure doses well below the guidelines.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*	13. AVAILABILITY STATEMENT: unlimited
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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/CR-6644
INEL-95/0550

2. TITLE AND SUBTITLE

Generic Issue 158: Performance of Safety-Related Power-Operated Valves Under Operating Conditions

3. DATE REPORT PUBLISHED

MONTH	YEAR
September	1999

4. FIN OR GRANT NUMBER

W6354

5. AUTHOR(S)

P.H. McCabe

6. TYPE OF REPORT

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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Idaho National Engineering and Environmental Laboratory
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9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

T-Y Chang, NRC Project Manager

11. ABSTRACT *(200 words or less)*

This report documents an analysis of the operating performance of air-operated, hydraulic-operated, and solenoid-operated valves (referred to collectively as power-operated valves or POVs). The failure probabilities used in 44 Individual Plant Examinations/Probabilistic Risk Analyses (IPEs/PRAs) were listed, and upper bound failure probabilities were estimated for each valve type. Four years of Nuclear Plant Reliability Data System (NPRDS) failure records were examined and trends were identified. Seven plant-specific PRA models [using Integrated Reliability and Risk Analysis System (IRRAS)] were used to determine the sensitivity of core damage frequencies (CDFs) to postulated increases in POV failure probabilities. The uncertainties of the CDF calculations were calculated. The risk importance of POVs was estimated and listed.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

power-operated valves; failure probabilities; individual plant examination; probabilistic risk analysis; nuclear plant reliability data system; IRRAS; core damage frequencies

13. AVAILABILITY STATEMENT

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14. SECURITY CLASSIFICATION

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1. REPORT NUMBER
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NUREG/CR-6645
BNL-NUREG-52576

2. TITLE AND SUBTITLE

Reevaluation of Regulatory Guidance on Modal Response Combination Methods for Seismic Response Spectrum Analysis

3. DATE REPORT PUBLISHED

MONTH	YEAR
December	1999

4. FIN OR GRANT NUMBER
W-6521

6. TYPE OF REPORT

7. PERIOD COVERED *(Inclusive Dates)*

5. AUTHOR(S)

R. Morante, Y. Wang

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(if NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

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9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(if NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

R.M. Kenneally, Project Manager

11. ABSTRACT *(200 words or less)*

Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," was last revised in 1976. The objectives of this project were to re-evaluate the current regulatory guidance for combining modal responses in response spectrum analysis; evaluate recent technical developments; and recommend revisions to the regulatory guidance. In addition, Standard Review Plan Section 3.7.2, "Seismic System Analysis," was reviewed to identify related sections which may need to be revised. The objectives were addressed through a literature review of past studies, supplemented by analysis of a piping system model previously utilized in NUREG/CR-5627, "Alternate Modal Combination Methods in Response Spectrum Analysis."

This project evaluated (1) methods for separation of the in-phase and out-of-phase modal response components; (2) methods for combination of the out-of-phase modal response components; (3) the contribution of "missing mass"; and (4) the combination of the three elements of response to produce the total response. Numerical results from response spectrum analyses were compared to corresponding time history analysis results to assess the accuracy of the various combination methods tested.

During the course of the project, several insights relating to potential improvements in the methodology for seismic analysis were identified and documented. These include (1) improvements in correlation between mode superposition time history and direct integration time history; (2) use of response spectrum generation single degree of freedom oscillator responses to define the frequency above which modal responses are in-phase with the input time history; and (3) evaluation of the effects of potential differences in mass distribution used in static and dynamic analyses of a piping system.

12 KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Seismic effects
Regulatory Guides
Frequency analysis
Frequency measurement
Systems analysis
Spectral response

13. AVAILABILITY STATEMENT

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14. SECURITY CLASSIFICATION
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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/CR-6648
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2. TITLE AND SUBTITLE

Environmental Assessment

San Bernardino National Wildlife Refuge Well 10

3. DATE REPORT PUBLISHED

MONTH	YEAR
December	1999

4. FIN OR GRANT NUMBER

J5084

5. AUTHOR(S)

J.T. Ensminger, C.E. Easterly, R.H. Ketelle, H. Quarles, M.C. Wade

6. TYPE OF REPORT

Environmental Assessment

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

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9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Industrial and Medical Nuclear Safety
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

The U.S. Geological Survey (USGS) currently holds a license issued by NRC for a radioactive (Am-241-Be) well logging source that has been lost in an artesian well (#10) in the San Bernardino National Wildlife Refuge, Arizona, since 1986. The USGS has requested that the Am-Be source license be terminated and has attempted to seal the source in place with cement as required by NRC license termination regulations. This environmental assessment addresses the potential water quality, ecological, and human health impacts of three alternatives for final disposition of source: (1) the proposed action, abandonment in place; (2) source retrieval; and (3) the no-action alternative.

The proposed action would require additional sealing of the Am-Be source and emplacement of a permanent plaque at the well head. No impacts to water quality, aquatic biotia, or human-health would be expected. For the source retrieval alternative, the potential for an accidental release at or near the ground source is a negative factor for this alternative; and impacts to aquatic biota are possible for a worst-case scenario. Under the no-action alternative, no acute water quality, ecological, and human health effects would be expected. However, because the Am-Be source would not be sealed in the lower part of the well, continued monitoring would be necessary to ensure that unexpected contaminant concentrations do not occur.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

ecology
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San Bernardino National Wildlife Refuge
well logging

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14. SECURITY CLASSIFICATION

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

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ORNL/TM-1999/231

2. TITLE AND SUBTITLE

International Comparative Assessment Study of Pressurized Thermal Shock in Reactor Pressure Vessels

3. DATE REPORT PUBLISHED

MONTH	YEAR
December	1999

4. FIN OR GRANT NUMBER

W6631

5. AUTHOR(S)

B.R. Bass, C.E. Pugh, Oak Ridge National Laboratory
J. Sievers, H. Schulz, Gesellschaft fur Anlagen-und Reaktorsicherheit (GRS)

6. TYPE OF REPORT

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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Oak Ridge National Laboratory
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Oak Ridge, TN 37831-6370

Gesellschaft fur Anlagen-und Reaktorsicherheit (GRS)
Koln, Germany

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

C.G. Santos, NRC Project Manager

11. ABSTRACT *(200 words or less)*

A summary of the recently completed International Comparative Assessment Study of Pressurized-Thermal-Shock in Reactor Pressure Vessels (ICAS PTS RPV) is presented to record the results in actual and comparative fashions. The ICAS Project brought together an international group of experts from research, utility, and regulatory organizations to perform a comparative evaluation of analysis methodologies employed in the assessment of RPV integrity under PTS loading conditions. The Project was sponsored jointly by Gesellschaft fur Anlagen- und Reaktorsicherheit (GRS), Koln, Germany, and Oak Ridge National Laboratory (ORNL), with assistance from the Organization for Economic Co-operation and Development (OECD)/Nuclear Energy Agency (NEA)/Committee on the Safety of Nuclear Installations (CSNI)/Principal Working Group (PWG) No. 3 (Integrity of Components and Structures).

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Reactor Pressure Vessel
Pressurized Thermal Shock
Fracture Mechnaics
Thermal Hydraulics
Probabilistic Fracture Mechanics
Integrity Analysis
International
Comparative

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NUREG/CR-6655, Vol. 1
ORNL/TM-13692/V1

2. TITLE AND SUBTITLE

Sensitivity and Uncertainty Analyses Applied to Criticality Safety Validation

Methods Development

3. DATE REPORT PUBLISHED

MONTH	YEAR
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4. FIN OR GRANT NUMBER

W6479

5. AUTHOR(S)

B.L. Broadhead, C.M. Hopper, R.L. Childs, C.V. Parks

6. TYPE OF REPORT

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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Oak Ridge National Laboratory
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Oak Ridge, TN 37831-6370

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

C.W. Nilsen, NRC Project Manager

11. ABSTRACT *(200 words or less)*

This report develops the methodology for application of Sensitivity and Uncertainty (S/U) Analysis Techniques to the data validation tasks of a criticality safety computational study. The S/U methods which are presented in this volume are designed to provide a formal means of establishing the range or area of applicability for criticality safety data validation studies. The development of two parameters that are analogous to the standard trending parameters form the key to the technique. These parameters represent the differences by group of S/U-generated sensitivity profiles and traditional correlation coefficients, each of which give information relative to the similarity between pairs of selected systems.

A Generalized Linear Least Squares Methodology (GLLSM) tool is also described, which is used largely to provide an understanding of the magnitude and quantity (i.e., number of systems) of these new parameters that can be used in a formal definition of applicability and bias estimation.

These methods and guidelines will be applied to a sample validation for uranium systems with enrichments greater then 5 wt % in Volume 2 of this document.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

criticality safety, data validation, sensitivity analysis, uncertainty analysis

13. AVAILABILITY STATEMENT

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14. SECURITY CLASSIFICATION

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BIBLIOGRAPHIC DATA SHEET (See instructions on the reverse)	NUREG/CR-6655, Vol. 2 ORNL/TM-13692/V2
2. TITLE AND SUBTITLE Sensitivity and Uncertainty Analyses Applied to Criticality Safety Validation Illustrative Applications and Initial Guidance	3. DATE REPORT PUBLISHED MONTH: November — YEAR: 1999 4. FIN OR GRANT NUMBER W6479
5. AUTHOR(S) B.L. Broadhead, C.M. Hopper, C.V. Parks	6. TYPE OF REPORT Technical 7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Oak Ridge National Laboratory
Managed by Lockheed Martin Energy Resarch Corporation
Oak Ridge, TN 37831-6370

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

C.W. Nilsen, NRC Project Manager

11. ABSTRACT (200 words or less)

This report presents the application of sensitivity and uncertainty (S/U) analysis methodologies developed in Volume 1 to the code/data validation tasks of a criticality safety computational study. Sensitivity and uncertainty analysis methods were first developed for application to fast reactor studies in the 1970s. This work has revitalized and updated the existing S/U computational capabilities such that they can be used as prototypic modules of the SCALE code system, which contains criticality analysis tools currently in use by criticality safety practitioners. After complete development, simplified tools are expected to be released for general use.

The methods for application of S/U and generalized linear-least squares methodology (GLLSM) tools to the criticality safety validation procedures were described in Volume 1 of this report. Volume 2 of this report presents the application of these procedures to the validation of criticality safety analyses supporting uranium operations where the enrichments are greater than 5 wt %.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)	13. AVAILABILITY STATEMENT
criticality safety, data validation, sensitivity analysis, uncertainty analysis	unlimited
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NUREG/CR-6656

2. TITLE AND SUBTITLE

Information on Hydrologic Conceptual Models, Parameters, Uncertainty Analysis, and Data Sources for Dose Assessments at Decommissioning Sites

3. DATE REPORT PUBLISHED

MONTH	YEAR
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4. FIN OR GRANT NUMBER

JCN W6933

5. AUTHOR(S)

P.D. Meyer and G.W. Gee

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Technical

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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Pacific Northwest National Laboratory
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9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Risk Analysis and Applications
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

T.J. Nicholson, NRC Project Manager

11. ABSTRACT *(200 words or less)*

This report addresses issues related to the analysis of uncertainty in dose assessments conducted as part of decommissioning analyses. The analysis is limited to the hydrologic aspects of the exposure pathway involving infiltration of water at the ground surface, leaching of contaminants, and transport of contaminants through the ground water to a point of exposure. The basic conceptual models and mathematical implementations of three dose assessment codes are outlined along with the site-specific conditions under which the codes may provide inaccurate, potentially nonconservative results. In addition, the hydrologic parameters of the codes are identified and compared. A methodology for parameter uncertainty assessment is outlined that considers the potential data limitations and modeling needs of decommissioning analyses. This methodology uses generic parameter distributions based on national or regional databases, sensitivity analysis, probabilistic modeling, and Bayesian updating to incorporate site-specific information. Data sources for best-estimate parameter values and parameter uncertainty information are also reviewed. A follow-on report will illustrate the uncertainty assessment methodology using decommissioning test cases.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

decommissioning
dose assessment
ground-water analysis
hydrology
parameter uncertainty
sensitivity
uncertainty

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1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/CR-6658
LA-UR-99-2312

2. TITLE AND SUBTITLE

TRAC-M
Programmer's Guide

Fortran 77, Version 5.5

3. DATE REPORT PUBLISHED

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4. FIN OR GRANT NUMBER

W6245

5. AUTHOR(S)

R.G. Steinke, S.J. Jolly-Woodruff, J.W. Spore

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8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

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Los Alamos, New Mexico 87545

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

F. Odar, NRC Project Manager

11. ABSTRACT *(200 words or less)*

The Transient Reactor Analysis Code (TRAC) was developed to provide advanced best-estimate predictions of postulated accidents in light-water reactors. The TRAC-M program provides this capacity for pressurized water reactors and for many thermal-hydraulic test facilities. The code features a one- (1-), two- (2-), and three-dimensional (3D) treatment of the pressure vessel and its associated internals. The code includes a two-fluid nonequilibrium hydrodynamics model with a noncondensable-gas field and solute tracking, flow-regime-dependent constitutive equation treatment, optional reflood tracking capability for bottom- and top-flood and falling-film quench fronts, and a consistent treatment of the entire accident sequences, including the generation of consistent initial conditions. The stability-enhancing two-step numerical algorithm is used in the 1-, 2-, and 3D hydrodynamics, and permits violation of the material Courant limit. This technique permits large timesteps, hence the running time for slow transients is reduced. TRAC-M has a heat -structure (HTSTR) component and a radiation heat-transfer model the allows the user to model heat transfer accurately for complicated geometries. An improved reflood model based on mechanistic and defensible models has been added. TRAC-M also contains improved constitutive models and additions and refinements for several components. This manual is the third volume of a four-volume set of documents on TRAC-M. This guide was developed to assist the TRAC-M programmer and contains information on the TRAC-M Version 1.10+ code and data structure, the TRAC-M calculational sequence, and memory.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

TRAC
TRAC-M
Fortran 77
thermal-hydraulics
Pressurized Water Reactor

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

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