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Computer Codes and Mathematical Models

January–December 1994

August 1995

End–User Support Services Branch
Office of Information Resources Management

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Energy Science and Technology Software Center
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End-User Support Services Branch
Office of Information Resources Management
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

ABSTRACT

This report contains citations of NUREG-series documents issued in calendar year 1994 relating to computer software and mathematical models for scientific, engineering, or technology-related programs performed or sponsored by the U.S. Nuclear Regulatory Commission (NRC). It is intended as a reference tool to assist the scientific and technical analyst in obtaining information on NRC computer-related activities.

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FOREWORD

The citations in this document appear in NUREG-series document order. Citations of NRC staff-generated reports designated NUREG--xxxx are listed first, followed by any conference proceedings identified as NUREG/CP--xxxx, contractor-generated reports published as NUREG/CR--xxxx documents, grant reports published as NUREG/GR--xxxx documents, and International Agreement reports issued as NUREG/IA--xxxx publications. Each citation contains the following: NUREG-series report number; software identification (where applicable); contractor report number; report title; a description of the report contents; publication date; names of the individuals responsible for preparing, compiling, or editing the report; contractor name and location; sponsoring NRC organization; and keywords or descriptors. Indexes by NUREG-series report number, software identification, contractor report number, and keyword are included in the Appendixes.

Specific code names and software identification appear in the heading of those citations with primary emphasis on specific mathematical models, computer codes, or databases. The term "General" is used in the heading of those citations which contain significant information on many models, computer codes, or databases.

Title: Assessment of Databases and Modeling Capabilities for the CANDU 3 Design

Description: The NRC staff has been conducting a preliminary review of the Canadian Deuterium Uranium Model 3 (CANDU 3) reactor design, a new heavy-water design developed by Atomic Energy of Canada Limited through its U.S. affiliate, AECL Technologies. The review has been aimed at identifying key technical areas and policy issues that will have to be addressed for standard design certification. As part of the research program associated with the preliminary review, the NRC Office of Nuclear Regulatory Research (RES) has completed an assessment of databases and modeling capabilities that might be needed to support the CANDU 3 design. To ensure full coverage of the design, a detailed assessment methodology was developed by the RES staff and was implemented with help from research projects at three national laboratories. This report integrates and summarizes the database and modeling assessments, including major contributions from these laboratories.

Publication Date: July 1994

Prepared by: Carlson, D.E.; Meyer, R.O.

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Systems Research

Keywords: ARROTTA, CANDU type reactors, CASMO, CATHENA, CERBERUS, CONTAIN, CORCON, CPM-2, CSAU, design, DIF3D, evaluation, fission product release, FRAPCON, FRAP-T, HELIOS, hydraulics, information needs, information systems, MCNP, MELCOR, NESTLE, nuclear fuels, reactor accidents, reactor kinetics, RELAP5, SCDAP, thermodynamics, TRAC-P, WIMS

General

Title: Proceedings of the CSNI Specialists Meeting on Fuel–Coolant Interactions

Description: A specialists meeting on fuel–coolant interactions was held in Santa Barbara, California, from January 5–7, 1993. The meeting was sponsored by the United States Nuclear Regulatory Commission in collaboration with the Committee on the Safety of Nuclear Installation (CSNI) of the Organization for Economic Cooperation and Development (OECD) Nuclear Energy Agency (NEA) and the University of California at Santa Barbara. The objectives of the meeting are to cross-fertilize ongoing work, provide opportunities for mutual check points, seek to focus the technical issues on matters of practical significance, and reevaluate both the objectives as well as path of future research.

Publication Date: March 1994

Prepared by:

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Systems Research

Keywords: BWR type reactors, CHYMES, corium, CULDESAC, ESPROSE, explosions, FEAT, FLOW3D, fluid flow, fragmentation, fuel–coolant interactions, heat transfer, hydraulics, IDEMO, IFCI, IVA-3, management, meetings, meltdown, mixing, PM-ALPHA, PRECURSOR, PWR type reactors, quenching, reactor safety, RELAP5/MOD2, ROAAM, steam, TEXAS-III, THIRMAL-1, TIGER, TRIO-MC, vapors

Title: Twenty-First Water Reaction Safety Information Meeting. Volume 2, Severe Accident Research

Description: This three-volume report contains 90 papers out of the 102 that were presented at the Twenty-First Water Reactor Safety Information Meeting held at the Bethesda Marriott Hotel, Bethesda, Maryland, during the week of October 25–27, 1993. 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, Japan, Russia, Switzerland, Taiwan, and United Kingdom. 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. Individual papers have been cataloged separately. This document, Volume 2, presents papers on severe accident research.

Publication Date: April 1994

Prepared by: Monteleone, S. [comp.] [Brookhaven National Lab., Upton, NY (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Office of Nuclear Regulatory Research

Keywords: ADINA, ALPHA, APRIL, BASSIM, BWR type reactors, CLCH model, combustion, COMMIX, COMPACT, CONTAIN, containment, CORCON, corium, CORMLT, CORSOR, DEBRIS, detonations, EVNTRE, FLUTAN, GOTHIC, heat transfer, hydraulics, hydrogen, IFCI, LOWCORV, low-Reynolds number k-e models, MAAP, MAPHY-BURN, meetings, MELCOR, MELPROG, NARAL-5M, nuclear power plants, PWR type reactors, RALOC, RASPLAV, reactor accidents, reactor components, reactor safety, research programs, RETRAN, ROAAM, SABRE, SCANAIR, SCDAP/RELAP5, SPARC, STANJAN, TAC-2D, TCE model, TRACG, VICTORIA, WAVCO, ZND model

General

Title: Proceedings of Workshop I in Advanced Topics in Risk and Reliability Analysis. Model Uncertainty: Its Characterization and Quantification

Description: The purpose of the workshop series is to provide a forum for in-depth discussion of key problems in risk and reliability analysis and for the development of solution strategies to attack these problems. The workshop topics are selected by an international organizing committee. Special emphasis is placed on risk and reliability problems that are important, timely, difficult to resolve without further research, and in need of expert input to formulate structured research agendas to assist timely and efficient resolution. The topic of model uncertainty fits all of these criteria. Model uncertainties have been acknowledged as being extremely important in a wide variety of risk assessment application areas, including nuclear power plant risk assessments, radioactive waste repository performance assessments, human health risk assessments, and environmental risk assessments. Clearly, a risk study that neglects to provide a careful treatment of model uncertainties can provide decision makers with a distorted picture of the uncertainties in the study's results. The papers and working group reports contained in these proceedings are divided into three sections. The first section contains papers discussing the appropriate formalism for dealing with model uncertainty. The second section of the proceedings contains papers discussing problems in coping with model uncertainties and approaches for dealing with these problems. The third section of the proceedings contains the summaries of the three working groups. Group 1 deals with the implications of model uncertainty on decision making (including regulatory applications), Group 2 deals with the formal definition of model uncertainty, and Group 3 deals with approaches to quantify model uncertainty. Selected papers were indexed separately for inclusion in the Energy Science and Technology Database.

Publication Date: October 1994

Prepared by: Mosleh, A.; Smidts, C. [eds.] [Maryland Univ., College Park, MD (United States)]; Siu, N. [ed.] [Idaho National Engineering Lab., Idaho Falls, ID (United States)]; Lui, C. [ed.] [Nuclear Regulatory Commission, Washington, DC (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Safety Issue Resolution

Keywords: accuracy, environmental impacts, fires, health hazards, meetings, nuclear facilities, probabilistic estimation, probability, reliability, risk assessment

Title: Transactions of the Twenty-Second Water Reactor Safety Information Meeting

Description: This report contains summaries of papers on reactor safety research to be presented at the 22nd Water Reactor Safety Information Meeting at the Bethesda Marriott Hotel, Bethesda, Maryland, October 24–26, 1994. The summaries briefly describe the programs and results of nuclear safety research sponsored by the Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission. 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.

Publication Date: October 1994

Prepared by:

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Office of Nuclear Regulatory Research

Keywords: aging, ALPHA, APRIL, BWR type reactors, COMETA, CONTAIN, CORCON, engineered safety systems, FAVOR, fuel elements, heat transfer, human factors, hydraulics, IFCI, MAAP, meetings, MELCOR, PWR type reactors, reactor accidents, reactor components, reactor cooling systems, reactor safety, regulations, RELAP/MOD3, research programs, risk assessment, SAPHIRE, SCDAP/RELAP5, seismic effects

General

Title: Workshop on Developing Safe Software

Description: The Workshop on Developing Safe Software was held July 22–23, 1992, at the Hotel del Coronado, San Diego, California. The purpose of the workshop was to have four world experts discuss among themselves software safety issues that are of interest to the U.S. Nuclear Regulatory Commission. These issues concern the development of software systems for use in nuclear power plant protection systems. The workshop comprised four sessions. Wednesday morning, July 22, consisted of presentations from each of the four panel members. On Wednesday afternoon, the panel members went through a list of possible software development techniques and commented on them. The Thursday morning, July 23, session consisted of an extended discussion among the panel members and the observers from the NRC. A final session on Thursday afternoon consisted of a discussion among the NRC observers as to what was learned from the workshop.

Publication Date: November 1994

Prepared by: Lawrence, J.D. [Lawrence Livermore National Lab., CA (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Reactor Controls and Human Factors

Keywords: engineered safety systems, failure mode analysis, meetings, nuclear power plants, reactor safety, reliability, security, testing

Title: Data Base on Dose Reduction Research Projects for Nuclear Power Plants. Volume 5

Description:

This is the fifth volume in a series of reports that provide information on dose reduction research and health physics technology for nuclear power plants. The information is taken from two of several data bases maintained by Brookhaven National Laboratory's ALARA Center for the Nuclear Regulatory Commission. The research section of the report covers dose reduction projects that are in the experimental or developmental phase. It includes such topics as steam generator degradation, decontamination, robotics, improvements in reactor materials, and inspection techniques. The section on health physics technology discusses dose reduction efforts that are in place or in the process of being implemented at nuclear power plants. A total of 105 new or updated projects are described. All project abstracts from this report are available to nuclear industry professionals with access to a fax machine through the ACEFAX system or with access to a computer with a modem and the proper communications software through the ACE system. Detailed descriptions of how to access all the data bases electronically are in the appendices of the report.

Publication Date: May 1994

Prepared by: Khan, T.A.; Yu, C.K.; Roecklein, A.K. [Brookhaven National Lab., Upton, NY (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Regulatory Applications

Keywords: accidents, ACE, ACEFAX, BWR type reactors, data base management, data compilation, nuclear power plants, PWR type reactors, radiation doses, radiation hazards, radiation protection

NUCLARR

Title: Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR): Data Manual. Part 2: Human Error Probability (HEP) Data; Volume 5, Revision 4

Description: This data manual contains a hard copy of the information in the Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR) Version 3.5 data base, which is sponsored by the U.S. Nuclear Regulatory Commission. NUCLARR was designed as a tool for risk analysis. Many of the nuclear reactors in the United States and several outside the United States are represented in the NUCLARR data base. NUCLARR includes both human error probability estimates for workers at the plants and hardware failure data for nuclear reactor equipment. Aggregations of these data yield valuable reliability estimates for probabilistic risk assessments and human reliability analyses. The data manual is organized to permit manual searches of the information if the computerized version is not available. Originally, the manual was published in three parts. In this revision the introductory material located in the original Part 1 has been incorporated into the text of Parts 2 and 3. The user can now find introductory material either in the original Part 1, or in Parts 2 and 3 as revised. Part 2 contains the human error probability data, and Part 3, the hardware component reliability data.

Publication Date: September 1994

Prepared by: Reece, W.J.; Gilbert, B.G.; Richards, R.E. [EG and G Idaho, Inc., Idaho Falls, ID (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Systems Research

Keywords: compiled data, information systems, NUCLARR, personnel, reactor operators, reactors, reliability, risk assessment, US NRC

Title: Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR): Data Manual. Part 3: Hardware Component Failure Data; Volume 5, Revision 4

Description: This data manual contains a hard copy of the information in the Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR) Version 3.5 data base, which is sponsored by the U.S. Nuclear Regulatory Commission. NUCLARR was designed as a tool for risk analysis. Many of the nuclear reactors in the United States and several outside the United States are represented in the NUCLARR data base. NUCLARR includes both human error probability estimates for workers at the plants and hardware failure data for nuclear reactor equipment. Aggregations of these data yield valuable reliability estimates for probabilistic risk assessments and human reliability analyses. The data manual is organized to permit manual searches of the information if the computerized version is not available. Originally, the manual was published in three parts. In this revision the introductory material located in the original Part 1 has been incorporated into the text of Parts 2 and 3. The user can now find introductory material either in the original Part 1, or in Parts 2 and 3 as revised. Part 2 contains the human error probability data, and Part 3, the hardware component reliability data.

Publication Date: September 1994

Prepared by: Reece, W.J.; Gilbert, B.G.; Richards, R.E. [EG and G Idaho, Inc., Idaho Falls, ID (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Systems Research

Keywords: compiled data, equipment, failures, information systems, NUCLARR, probabilistic estimation, reactor components, reactors, reliability, risk assessment, US NRC

PR-EDB

Title: PR-EDB: Power Reactor Embrittlement Data Base, Version 2. Revision 2, Program Description

Description: Investigations of regulatory issues, such as vessel integrity over plant life, vessel failure, sufficiency of current codes, Standard Review Plans (SRP's), and Guides for license renewal, can be greatly expedited by the use of a well-designed computerized data base. Also, such a data base is essential for the validation of embrittlement prediction models by researchers. The Power Reactor Embrittlement Data Base (PR-EDB) is a comprehensive collection of data for U.S. commercial nuclear reactors. The current version of the PR-EDB contains the Charpy test data that were irradiated in 252 capsules of 96 reactors and consist of 207 data points for heat-affected-zone (HAZ) materials (9[illegible] different HAZ), 227 data points for weld materials (105 different welds), 524 data points for base materials (136 different base materials), including 297 plate data points (85 different plates), 119 forging data points (31) different forging), and 108 correlation monitor materials data points (3 different plates). The data files are given in dBASE format and can be accessed with any computer using the DOS operating system. User-friendly utility programs are used to retrieve and select specific data, manipulate data, display data to the screen or printer, and to fit and plot Charpy impact data. The results of several studies investigated are presented in Appendix D.

Publication Date: January 1994

Prepared by: Stallmann, F.W.; Wang, J.A.; Kam, F.B.K.; Taylor, B.J. [Oak Ridge National Lab., TN (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Engineering

Keywords: Charpy test, data base management, embrittlement, power reactors, PR-EDB, reactor components, welded joints

Title: Microcomputer Applications of, and Modifications to, the Modular Fault Trees

Description: The La Salle Probabilistic Risk Assessment was the first major application of the modular logic fault trees after the IREP program. In the process of performing the analysis, many errors were discovered in the fault tree modules that led to difficulties in combining the modules to form the final system fault trees. These errors are corrected in the revised modules listed in this report. In addition, the application of the modules in terms of editing them and forming them into the system fault trees was inefficient. Originally, the editing had to be done line by line and no error checking was performed by the computer. This led to many typos and logic errors in the construction of the modular fault tree files. Two programs were written to help alleviate this problem: (1) MODEDIT—this program allows an operator to retrieve a file for editing, edit the file for the plant-specific application, perform some general error checking while the file is being modified, and store the file for later use; and (2) INDEX —this program checks that the modules that are supposed to form one fault tree all link up appropriately before the files are loaded onto the mainframe computer. Lastly, the modules were not designed for relay-type logic common in Boiling Water Reactor (BWR) designs but for solid state type logic. Some additional modules were defined for modeling relay logic, and an explanation and example of their use are included in this report.

Publication Date: October 1994

Prepared by: Zimmerman, T.L.; Graves, N.L.; Payne, A.C., Jr.; Whitehead, D.W. [Sandia National Labs., Albuquerque, NM (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Safety Issue Resolution

Keywords: diagrams, fault tree analysis, La Salle County-1 reactor, La Salle County-2 reactor, modifications, reactor safety, risk assessment

SQUIRT

Title: Evaluation and Refinement of Leak-Rate Estimation Models. Revision 1

Description: Leak-rate estimation models are important elements in developing a leak-before-break methodology in piping integrity and safety analyses. Existing thermal hydraulic and crack-opening-area models used in current leak-rate estimations have been incorporated into a single computer code for leak-rate estimation. The code is called SQUIRT, which stands for Seepage Quantification of Upsets In Reactor Tubes. The SQUIRT program has been validated by comparing its thermal hydraulic predictions with the limited experimental data that have been published on two-phase flow through slits and cracks and by comparing its crack-opening-area predictions with data from the Degraded Piping Program. In addition, leak-rate experiments were conducted to obtain validation data for a circumferential fatigue crack in a carbon steel pipe girth weld.

Publication Date: June 1994

Prepared by: Paul, D.D.; Ahmad, J.; Scott, P.M.; Flanigan, L.F.; Wilkowski, G.M. [Battelle, Columbus, OH (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Engineering

Keywords: carbon steels, cracks, elasticity, fracture mechanics, leaks, leak testing, nuclear power plants, pipes, plasticity, SQUIRT

Title: Field Lysimeter Investigations: Low-Level Waste Data Base Development Program for Fiscal Year 1993. Annual Report: Volume 6

Description: The March 28, 1979, accident at Three Mile Island Unit 2 released approximately 560,000 gal of contaminated water to the auxiliary and fuel handling buildings. The water was decontaminated using a three-stage demineralization system called EPICOR-II that contained organic and inorganic ion-exchange media. The first stage of the system was designated the prefilter, and the second and third stages were called demineralizers. Research is being conducted at the Idaho National Engineering Laboratory on materials from four of those EPICOR-II prefilters. The Field Lysimeter Investigations: Low-Level Waste Data Base Development Program, funded by the U.S. Nuclear Regulatory Commission, is studying the degradation effects in EPICOR-II organic ion-exchange resins caused by radiation, examining the adequacy of test procedures recommended in the Branch Technical Position on Waste Form to meet the requirements of 10 CFR 61 using solidified EPICOR-II resins, obtaining performance information on solidified EPICOR-II ion-exchange resins in a disposal environment, and determining the condition of EPICOR-II liners. Results of the eighth year of data acquisition from the field testing are presented and discussed. During the continuing field testing, both Portland type I–II cement and Dow vinyl ester-styrene waste forms are being tested in lysimeter arrays located at Argonne National Laboratory–East in Illinois and at Oak Ridge National Laboratory. The study is designed to provide continuous data on nuclide release and movement, as well as environment conditions, over a 20-year period.

Publication Date: May 1994

Prepared by: McConnell, J.W., Jr.; Rogers, R.D.; Jastrow, J.D.; Sanford, W.E.; Sullivan, T.M. [EG and G Idaho, Inc., Idaho Falls, ID (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Regulatory Applications

Keywords: antimony 125, calcium, cesium 137, chlorides, cobalt 60, EPICOR-II, ionic composition, low-level waste data base, magnesium, measuring methods, moisture, monitoring, nitrates, organic ion exchangers, performance testing, phosphates, potassium, progress report, radioactive waste disposal, radiolysis, radionuclide migration, sodium, soil chemistry, soils, strontium 90, sulfates, temperature measurement, waste forms, weather

FM-DOSE, RASCAL, ST-DOSE

Title: RASCAL Version 2.1, User's Guide

Description: The Radiological Assessment System for Consequence Analysis, Version 2.1 (RASCAL 2.1) has been developed for use during response to radiological emergencies. The system supplements assessments based on plant conditions and quick estimates based on hand-calculational methods. The model is designed to provide a comparison to Environmental Protection Agency (EPA) Protective Action Guidance (PAG) and thresholds for acute health effects. RASCAL will be used by NRC personnel who report to the site of a nuclear accident to conduct an independent evaluation of dose-and-consequence projections and for training and drills. The model was developed to allow consideration of the dominant aspects of source term, transport, dose, and consequences. The model is a DOS application that can be run under Windows, and the results can be displayed as text or graphics. Revisions to RASCAL 2.0 have required the release of this new version of the system. Three new source term calculations have been added to ST-DOSE in RASCAL 2.1. They are (1) a source term based on the reactor containment monitor reading, (2) a source term for a spent fuel pool accident, and (3) an isotopic concentration source term. Field Measurements to Dose (FM-DOSE) calculations have been modified to include consideration of the effect of delay for re-entry on first-year and second-year dose, to incorporate a variable resuspension rate, and to compute a factor used to estimate first-year dose from R/h measurements on the ground. The tabular output of ST-DOSE and FM-DOSE have been changed to include EPA PAGs. Also, RASCAL 2.1 supports the saving of cases for later display or modification and the use of a mouse for user input. References to the technical details are included. A RASCAL 2.1 workbook is available.

Publication Date: December 1994

Prepared by: Sjoreen, A.L. [Oak Ridge National Lab., TN (United States)]; Athey, G.F. [Athey Consulting, Charles Town, WV (United States)]; Ramsdell, J.V. [Pacific Northwest Lab., Richland, WA (United States)]; McKenna, T. [Nuclear Regulatory Commission, Washington, DC (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Office for Analysis and Evaluation of Operational Data

Keywords: computerized simulation, computer program documentation, FM-DOSE, radiation transport, RASCAL, reactor accidents, source terms, ST-DOSE

DECAY, FM-DOSE, RASCAL, ST-DOSE

Title: RASCAL Version 2.1 Workbook. Volume 2, Revision 2

Description: The Radiological Assessment System for Consequence Analysis, Version 2.1 (RASCAL 2.1) was developed for use by the NRC personnel who respond to radiological emergencies. This workbook complements the RASCAL 2.1 User's guide (NUREG/CR-5247, Vol. 1, Rev. 2). The workbook contains exercises designed to familiarize the user with the computer-based tools of RASCAL through hands-on problem solving. The workbook contains four major sections. The first is a RASCAL familiarization exercise to acquaint the user with the operation of the forms, menus, online help, and documentation. The latter three sections contain exercises in using the three tools of RASCAL Version 2.1: DECAY, FM-DOSE, and ST-DOSE. A discussion section describing how the tools could be used to solve the problems follows each set of exercises.

Publication Date: December 1994

Prepared by: Athey, G.F. [Athey Consulting, Charles Town, WV (United States)]; Sjoreen, A.L. [Oak Ridge National Lab., TN (United States)]; McKenna, T.J. [Nuclear Regulatory Commission, Washington, DC (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Office for Analysis and Evaluation of Operational Data

Keywords: biological radiation effects, computer program documentation, DECAY, emergency plans, environmental exposure, environmental transport, FM-DOSE, half-life, manuals, nuclear decay, professional personnel, radiation accidents, radiation doses, radioisotopes, RASCAL, reactor accidents, ST-DOSE, training, US NRC

RECAP

Title: Replacement Energy Cost Analysis Package (RECAP): User's Guide. Revision 1

Description: A microcomputer program called the Replacement Energy Cost Analysis Package (RECAP) has been developed to assist the U.S. Nuclear Regulatory Commission (NRC) in determining the replacement energy costs associated with short-term shutdowns or deratings of one or more nuclear reactors. The calculations are based on the seasonal, unit-specific cost estimates for 1993–1996 previously published in NRC Report NUREG/CR--4012, Vol. 3 (1992), for all 112 U.S. reactors. Because the RECAP program is menu-driven, the user can define specific case studies in terms of such parameters as the units to be included, the length and timing of the shutdown or derating period, the unit capacity factors, and the reference year for reporting cost results. In addition to simultaneous shutdown cases, more complicated situations, such as overlapping shutdown periods or shutdowns that occur in different years, can be examined through the use of a present-worth calculation option.

Publication Date: July 1994

Prepared by: VanKuiken, J.C.; Willing, D.L. [Argonne National Lab., IL (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Regulatory Applications

Keywords: cost, economic analysis, economics, electric utilities, energy expenses, nuclear power plants, reactor shutdown, RECAP

Title: Predicting the Pressure-Driven Flow of Gases through Microcapillaries and Micro-Orifices

Description: A large body of experimentally measured gas flow rates were obtained from the literature and then compared to the predictions obtained with constitutive flow equations. This was done to determine whether the equations apply to the predictions of gas flow rates from leaking containment vessels used to transport radioactive materials. The experiments consisted of measuring the volumetric pressure-driven flow of gases through micro-capillaries and micro-orifices. The experimental results were compared to the predictions obtained with the equations given in ANSI N14.5, the American National Standard for Radioactive Materials-Leakage Tests on Package for Shipment. The equations were applied to both (1) the data set according to the recommendations given in ANSI N14.5 and (2) globally to the complete data set. It was found that the continuum and molecular flow equation provided good agreement between the experimental and calculated flow rates for flow rates less than about 1 atm · cm^3/s. The choked flow equation resulted in over-prediction of the flow rates for flow rates less than about 1 atm · cm^3/s. For flow rates higher than 1 atm · cm^3/s, the molecular and continuum flow equation over-predicted the measured flow rates and the predictions obtained with the choked flow equation agreed well with the experimental values. Because the flow rates of interest for packages used to transport radioactive materials are almost always less than 1 atm · cm^3/s, it is suggested that the continuum and molecular flow equation be used for gas flow rate predictions related to these applications.

Publication Date: November 1994

Prepared by: Anderson, B.L.; Carlson, R.W.; Fischer, L.E. [Lawrence Livermore National Lab., CA (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Industrial and Medical Nuclear Safety

Keywords: capillary flow, comparative evaluations, containers, flow models, flow rate, gas flow, leaks, radioactive materials, transport

RELAP5/MOD3

Title: RELAP5/MOD3 Code Manual. Volume 6: Validation of Numerical Techniques in RELAP5/MOD3

Description: The RELAP code has been developed for best-estimate transient simulation of light-water reactor coolant systems during large and small break loss-of-coolant accidents and as well as operational transients. The code models the coupled behavior of the reactor coolant system and the core during a severe accident transient and models large- and small-break loss-of-coolant accidents and operational transients, such as anticipated transient without scram, loss of offsite power, loss of feedwater, and loss of flow. A generic modeling approach is used that permits as much of a particular system to be modeled as necessary. Control system and secondary system components are included to permit modeling of plant controls, turbines, condensers, and secondary feedwater conditioning systems. RELAP5/MOD3 code documentation is divided into five volumes: Volume 1 provides modeling theory and associated numerical schemes; Volume 2 contains detailed instructions for code application and input data preparation; Volume 3 provides the results of developmental assessment cases that demonstrate and verify the models used in the code; Volume 4 presents a detailed discussion of RELAP5 models and correlations; Volume 5 contains guidelines that have evolved over the past several years through use of the RELAP5 code; and Volume 6 contains descriptions of numerical modeling of two-phase flow used in RELAP5 and discussions on stability, accuracy, and convergence of the numerical techniques in RELAP5.

Publication Date: October 1994

Prepared by: Shieh, A.S.; Ransom, V.H. [EG and G Idaho, Inc., Idaho Falls, ID (United States)]; Krishnamurthy, R. [Pacific-Nuclear Co., Westmont, IL (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Systems Research

Keywords: comparative evaluations, computer program documentation, flow models, manuals, numerical solution, reactor accidents, reactor control systems, reactor cooling systems, reactor cores, RELAP5/MOD3, two-phase flow, water cooled reactors

RELAP5/MOD3

Title: RELAP5/MOD3 Code Manual

Description: Summaries of RELAP5/MOD3 code assessments, a listing of the assessment matrix, and a chronology of the various versions of the code are given. Results from these code assessments have been used to formulate a compilation of some of the strengths and weaknesses of the code. These results are documented in the report. Volume 7 was designed to be updated periodically and to include the results of the latest code assessments as they become available. Consequently, users of Volume 7 should ensure that the latest revision is available.

Publication Date: June 1994

Prepared by: Sloan, S.M.; Schultz, R.R.; Wilson, G.E.

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Systems Research.

Keywords: computerized simulation, fluid flow, reactors, RELAP5/MOD3, thermal analysis

HPPOS

Title: Health Physics Positions Data Base: Revision 1

Description: The Health Physics Positions (HPPOS) Data Base of the Nuclear Regulatory Commission (NRC) is a collection of NRC staff positions on a wide range of topics involving radiation protection (health physics). It consists of 328 documents in the form of letters, memoranda, and excerpts from technical reports. The HPPOS Data Base was developed by the NRC Headquarters and Regional Offices to help ensure uniformity in inspections, enforcement, and licensing actions. Staff members of the Oak Ridge National Laboratory (ORNL) have assisted the NRC staff in summarizing the documents during the preparation of this NUREG report. These summaries are also being made available as a stand-alone software package for IBM and IBM-compatible personal computers. The software package for this report is called HPPOS Version 2.0. A variety of indexing schemes were used to increase the usefulness of the NUREG report and its associated software. The software package and the summaries in the report are written in the context of the new 10 CFR Part 20 (§§20.1001–20.2401). The purpose of this NUREG report is to allow interested individuals to familiarize themselves with the contents of the HPPOS Data Base and with the basis of many NRC decisions and regulations. The HPPOS summaries and original documents are intended to serve as a source of information for radiation protection programs at nuclear research and power reactors, nuclear medicine, and other industries that either process or use nuclear materials.

Publication Date: February 1994

Prepared by: Kerr, G.D.; Borges, T.; Stafford, R.S.; Lu, P.Y. [Oak Ridge National Lab., TN (United States)]; Carter, D. [US Nuclear Regulatory Commission, Washington, DC (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Regulatory Applications

Keywords: data base management, HPPOS, radiation protection, recommendations, US NRC

MELCOR, STCP

Title: Analysis of Long-Term Station Blackout Without Automatic Depressurization at Peach Bottom Using MELCOR (Version 1.8)

Description: This report documents the results from MELCOR calculations of the Long-Term Station Blackout Accident Sequence, with failure to depressurize the reactor vessel, at the Peach Bottom (BWR Mark I) plant, and presents compari:ons with Source Term Code Package (STCP) calculations of the same sequence. With STCP the transient has been calculated out to 13.5 hours after core uncovery. Most of the MELCOR calculations presented have been carried out to between 15 and 16.7 hours after core uncovery. The results include the release of source terms to the environment. The results of several sensitivity calculations with MELCOR are also presented, which explore the impact of varying user-input modeling and timestep control parameters on the accident progression and release of source terms to the environment. Most of the calculations documented here were performed in FY1990 using MELCOR Version 1.8BC. However, the appendices also document the results of more recent calculations performed in FY1991 using MELCOR versions 1.8CZ and 1.8DNX.

Publication Date: May 1994

Prepared by: Madni, I.K. [Brookhaven National Lab., Uptoı., ˇY (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Systems Research

Keywords: blackouts, computer calculations, containment, depressurization, fission product release, heat transfer, hydraulics, MELCOR, Peach Bottom-1 reactor, Peach Bottom-2 reactor, reactor accidents, reactor core disruption, reactor safety, reactor vessels, source terms, STCP, transients

General

Title: Modeling Field-Scale Unsaturated Flow and Transport Processes

Description: The scales of concern in subsurface transport of contaminants from low-level radioactive waste disposal facilities are in the range of 1 to 1000 m. Natural geologic materials generally show very suostantial spatial variability in hydraulic properties over this range of scales. Such heterogeneity can significantly influence the migration of contaminants. It is also envisioned that complex earth structures will be constructed to isolate the waste and minimize infiltration of water into the facility. The flow of water and gases through such facilities must also be a concern. A stochastic theory describing unsaturated flow and contamination transport in naturally heterogeneous soils has been enhanced by adopting a more realistic characterization of soil variability. The enhanced theory is used to predict field-scale effective properties and variances of tension and moisture content. Applications illustrate the important effects of small-scale heterogeneity on large-scale anisotropy and hysteresis and demonstrate the feasibility of simulating two-dimensional flow systems at time and space scales of interest in radioactive waste disposal investigations. Numerical algorithms for predicting field-scale unsaturated flow and contaminant transport have been improved by requiring them to respect fundamental physical principles such as mass conservation. These algorithms are able to provide realistic simulations of systems with very dry initial conditions and high degrees of heterogeneity. Numerical simulation of the movement of water and air in unsaturated soils has demonstrated the importance of air pathways for contaminant transport. The stochastic flow and transport theory has been used to develop a systematic approach to performance assessment and site characterization. Hypothesis-testing techniques have been used to determine whether model predictions are consistent with observed data.

Publication Date: August 1994

Prepared by: Gelhar, L.W.; Celia, M.A.; McLaughlin, D. [Massachusetts Inst. of Tech., Cambridge, MA (United States). Dept. of Civil and Environmental Engineering]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Regulatory Applications

Keywords: environmental transport, flow models, gas flow, ground water, liquid flow, low-level radioactive wastes, radioactive waste facilities, Richards equation, site characterization, soils, two-dimensional calculations

LSARG

Title: Development and Application of Degradation Modeling To Define Maintenance Practices

Description: This report presents the development and application of component degradation modeling to analyze degradation effects on reliability and to identify aspects of maintenance practices that mitigate degradation and aging effects. Using continuous time Markov approaches, a component degradation model is discussed that includes information about degradation and maintenance. The component model commonly used in probabilistic risk assessments is a simple case of this general model. The parameters used in the general model have engineering interpretations and can be estimated using data and engineering experience. The generation of equations for specific models, the solution of these equations, and a methodology for estimating the needed parameters are all discussed. Applications in this report show how these models can be used to quantitatively assess the benefits that are expected from maintaining a component, the effects of different maintenance efficiencies, the merits of different maintenance policies, and the interaction of surveillance test intervals with maintenance practices.

Publication Date: June 1994

Prepared by: Stock, D.; Samanta, P. [Brookhaven National Lab., Upton, NY (United States)]; Vesely, W. [Science Applications International Corp., Dublin, OH (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Engineering

Keywords: aging, BWR type reactors, LSARG, Markov models, Markov process, prediction equations, PWR type reactors, reactor components, reactor maintenance, reliability

EMTP

Title: The Effects of Solar-Geomagnetically Induced Currents on Electrical Systems in Nuclear Power Stations

Description: This report presents the results of a study to evaluate the potential effects of geomagnetically induced currents (GICs) caused by the solar disturbances on the in-plant electrical distribution system and equipment in nuclear power stations. The plant-specific electrical distribution system for a typical nuclear plant is modeled using the ElectroMagnetic Transient Program (EMTP). The computer model simulates online equipment and loads from the station transformer in the switchyard of the power station to the safety-buses at 120 volts to which all electronic devices are connected for plant monitoring. The analytical model of the plant's electrical distribution system is studied to identify the transient effects caused by the half-cycle saturation of the station transformers due to GIC. This study provides results of the voltage harmonics levels that have been noted at various electrical buses inside the plant. The emergency circuits appear to be more susceptible to high harmonics due to the normally light load conditions. In addition to steady-state analysis, this model was further analyzed simulating various plant transient conditions (e.g., loss of load or large motor start-up) occurring during GIC events. Detail models of the plant's protective relaying system employed in bus transfer application were included in this model to study the effects of the harmonic distortion of the voltage input. Potential harmonic effects on the uninterruptible power system (UPS) are qualitatively discussed as well.

Publication Date: January 1994

Prepared by: Subudhi, M. [Brookhaven National Lab., Upton, NY (United States)]; Carroll, D.P. [Florida Univ., Gainesville, FL (United States)]; Kasturi, S. [MOS, Inc., Melville, NY (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Engineering

Keywords: Bruce site, computerized simulation, electrical equipment, EMTP, frequency analysis, geomagnetic field, harmonics, heating, Hope Creek-1 reactor, Hope Creek-2 reactor, magnetic storms, nuclear power plants, power systems, relays, Salem-1 reactor, Salem-2 reactor, Three Mile Island-1 reactor, Three Mile Island-2 reactor, transformers

CONTAIN

Title: Experiments To Investigate Direct Containment Heating Phenomena with Scaled Models of the Zion Nuclear Power Plant in the Surtsey Test Facility

Description: The Surtsey Test Facility at Sandia National Laboratories (SNL) is used to perform scaled experiments that simulate hypothetical high-pressure melt ejection (HPME) accidents in a nuclear power plant (NPP). These experiments are designed to investigate the effect of specific phenomena associated with direct containment heating (DCH) on the containment load, such as the effect of physical scale, prototypic subcompartment structures, water in the cavity, and hydrogen generation and combustion. In the Integral Effects Test (IET) series, 1:10 linear scale models of the Zion NPP structures were constructed in the Surtsey vessel. The reactor pressure vessel (RPV) was modeled with a steel pressure vessel that had a hemispherical bottom head with a 4-cm hole that simulated the final ablated hole that would be formed by ejection of an instrument guide tube in a severe NPP accident. Iron/alumina/chromium thermite was used to simulate molten corium that would accumulate on the bottom head of an actual RPV. The chemically reactive melt simulant was ejected by high-pressure steam from the RPV model into the scaled reactor cavity. Debris was then entrained through the instrument tunnel into the subcompartment structures and the upper dome of the simulated reactor containment building. The results of the IET experiments are given in this report.

Publication Date: May 1994

Prepared by: Allen, M.D.; Pilch, M.M.; Blanchat, T.K.; Griffith, R.O. [Sandia National Labs., Albuquerque, NM (United States)]; Nichols, R.T. [Ktech Corp., Albuquerque, NM (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Systems Research

Keywords: CONTAIN, containment systems, corium, design, experimental data, measuring instruments, meltdown, pressure measurement, scale models, simulation, temperature measurement, thermal analysis, two-cell adiabatic equilibrium model, Zion-1 reactor, Zion-2 reactor

ARANO, CONDOR, COSYMA, LENA, MACCS, OSCAAR

Title: Comparison of MACCS Users Calculations for the International Comparison Exercise on Probabilistic Accident Consequence Assessment Code, October 1989–June 1993

Description: Over the past several years, the Organization for Economic Cooperation and Development/ Nuclear Energy Agency (OECD/NEA) and Commission of the European Community (CEC) sponsored an international program comparing a group of six probabilistic consequence assessment (PCA) codes designed to simulate health and economic consequences of radioactive releases into atmosphere of radioactive materials following severe accidents at nuclear power plants (NPPs): ARANO (Finland), CONDOR (UK), COSYMA (CEC), LENA (Sweden), MACCS (USA), and OSCAAR (Japan). In parallel with this effort, two separate groups performed similar calculations using the MACCS and COSYMA codes. Results produced in the MACCS Users Group's (Greece, Italy, Spain, and the United States) calculations and their comparison are contained in the present report. Version 1.5.11.1 of the MACCS code was used for the calculations. Good agreement between the results produced in the four participating calculations has been reached, with the exception of the results related to the ingestion pathway dose predictions. The main reason for the scatter in those particular results is attributed to the lack of a straightforward implementation of the specifications for agricultural production and counter-measures criteria provided for the exercise. A significantly smaller scatter in predictions of other consequences was successfully explained by differences in meteorological files and weather sampling, grids, rain distance intervals, dispersion model options, and population distributions.

Publication Date: April 1994

Prepared by: Neymotin, L. [Brookhaven National Lab., Upton, NY (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Safety Issue Resolution

Keywords: ARANO, biological radiation effects, computer calculations, computer program documentation, CONDOR, coordinated research programs, COSYMA, crops, economic analysis, fission product release, human populations, LENA, MACCS, nuclear power plants, OSCAAR, plants, public health, radiation doses, radiation hazards, radiation transport, reactor accidents, risk assessment, source terms

Title: INTRAVAL Phase 2: Modeling Testing at the Las Cruces Trench Site

Description: Several field experiments have been performed by scientists from the University of Arizona and New Mexico State University at the Las Cruces Trench Site to provide data to test deterministic and stochastic models for water flow and solute transport. These experiments were performed in collaboration with INTRAVAL, an international effort toward validation of geosphere models for the transport of radionuclides. During Phase I of INTRAVAL, qualitative comparisons between experimental data and model predictions were made using contour plots of water contents and solute concentrations. Detailed quantitative comparisons were not made. A third Las Cruces Trench experiment was designed by scientists from the University of Arizona and New Mexico State University to provide data for more rigorous model testing. Modelers from the Center for Nuclear Waste Regulatory Analysis, Massachusetts Institute of Technology, New Mexico State University, Pacific Northwest Laboratory, and the University of Texas provided predictions of water flow and tritium transport to New Mexico State University for analysis. The corresponding models assumed soil characterizations ranging from uniform to deterministically heterogeneous to stochastic. This report presents detailed quantitative comparisons to field data.

Publication Date: January 1994

Prepared by: Hills, R.G. [New Mexico State Univ., Las Cruces, NM (United States). Dept. of Mechanical Engineering]; Wierenga, P.J. [Arizona Univ., Tucson, AZ (United States). Dept. of Soil and Water Science]; Luis, S.; McLaughlin, D. [Massachusetts Inst. of Tech., Cambridge, MA (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Regulatory Applications

Keywords: BIGFLOW, computer calculations, computer codes, experimental data, field tests, fluid flow, leachates, leaching, numerical analysis, PORFLO-3, radionuclide migration, soils, solutes, TRACER3D, tritium, TRUST+TRUMP, UNSAT2, validation, VAM2D

General

Title: The Probability of Containment Failure by Direct Containment Heating in Zion

Description: This report is the first step in the resolution of the Direct Containment Heating (DCH) issue for the Zion Nuclear Power Plant using the Risk Oriented Accident Analysis Methodology (ROAAM). This report includes the definition of a probabilistic framework that decomposes the DCH problem into three probability density functions that reflect the most uncertain initial conditions (UO_2 mass, zirconium oxidation fraction, and steel mass). Uncertainties in the initial conditions are significant, but our quantification approach is based on establishing reasonable bounds that are not unnecessarily conservative. To this end, we also make use of the ROAAM ideas of enveloping scenarios and splintering. Two causal relations (CR) are used in this framework: CR1 is a model that calculates the peak pressure in the containment as a function of the initial conditions, and CR2 is a model that returns the frequency of containment failure as a function of pressure within the containment. Uncertainty in CR1 is accounted for by the use of two independently developed phenomenological models, the Convection Limited Containment Heating (CLCH) model and the Two-Cell Equilibrium (TCE) model, and by probabilistically distributing the key parameter in both, which is the ratio of the melt entrainment time to the system blowdown time constant. The two phenomenological models have been compared with an extensive database including recent integral simulations at two different physical scales. The containment load distributions do not intersect the containment strength (fragility) curve in any significant way, resulting in containment failure probabilities less than 10^{-3} for all scenarios considered. Sensitivity analyses did not show any areas of large sensitivity.

Publication Date: December 1994

Prepared by: Pilch, M.M. [Sandia National Labs., Albuquerque, NM (United States)]; Yan, H.; Theofanous, T.G. [California Univ., Santa Barbara, CA (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Systems Research

Keywords: ABAQUS, ALPHA, ANATECH, APRIL, blowdown, CLCH model, COMMIX, containment, CORCON, CORMLT, DIRHEAT, distribution functions, dynamic loads, failures, HARDCORE, KIVA, MAAP, MELCOR, MELPROG, MELTSPREAD, PARSEC, pressure dependence, pressurization, probability, radiation heating, reactor accidents, reactor safety, risk assessment, ROAAM, SASM, SCDAP/RELAP5, SENKIN, sensitivity analysis, TCE model, Zion-1 reactor, Zion-2 reactor

Title: The Probability of Containment Failure by Direct Containment Heating in Zion. Supplement 1

Description: Supplement 1 of NUREG/CR--6075 brings to closure the DCH issue for the Zion plant. It includes the documentation of the peer review process for NUREG/CR--6075, the assessments of four new splinter scenarios defined in working group meetings, and modeling enhancements recommended by the working groups. In the four new scenarios, consistency of the initial conditions has been implemented by using insights from systems-level codes. SCDAP/RELAP5 was used to analyze three short-term station blackout cases with different lead rates. In all three case, the hot leg or surge line failed well before the lower head and thus the primary system depressurized to a point where DCH was no longer considered a threat. However, these calculations were continued to lower head failure in order to gain insights that were useful in establishing the initial and boundary conditions. THE SCDAP/RELAP output was used as input to CONTAIN to assess the containment conditions at vessel breach. The containment-side conditions predicted by CONTAIN are similar to those originally specified in NUREG/CR--6075.

Publication Date: December 1994

Prepared by: Pilch, M.M.; Allen, M.D.; Stamps, D.W.; Tadios, E.L. [Sandia National Labs., Albuquerque, NM (United States)]; Knudson, D.L. [Idaho National Engineering Lab., Idaho Falls, ID (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Systems Research

Keywords: blackouts, CONTAIN, containment, documentation, evaluation, failures, meltdown, pressure dependence, pressurization, probability, radiation heating, reactor accidents, reactor safety, SCDAP/RELAP5, Zion-1 reactor, Zion-2 reactor

TR-EDB

Title: TR-EDB: Test Reactor Embrittlement Data Base, Version 1

Description: The Test Reactor Embrittlement Data Base (TR-EDB) is a collection of results from irradiation in materials test reactors. It complements the Power Reactor Embrittlement Data Base (PR-EDB), whose data are restricted to the results from the analysis of surveillance capsules in commercial power reactors. The rationale behind their restriction was the assumption that the results of test reactor experiments may not be applicable to power reactors and could therefore be challenged if such data were included. For this very reason the embrittlement predictions in the Reg. Guide 1.99, Rev. 2, were based exclusively on power reactor data. However, test reactor experiments are able to cover a much wider range of materials and irradiation conditions that are needed to explore more fully a variety of models for the prediction of irradiation embrittlement. These data are also needed for the study of effects of annealing for life extension of reactor pressure vessels that are difficult to obtain from surveillance capsule results.

Publication Date: January 1994

Prepared by: Stallmann, F.W.; Wang, J.A.; Kam, F.B.K. [Oak Ridge National Lab., TN (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Engineering

Keywords: annealing, Charpy test, data base management, embrittlement, materials testing, pressure vessels, radiation effects, reactor materials, tensile properties, test reactors, TR-EDB

Title: Validation of the SCALE Broad Structure 44-Group ENDF/B-V Cross-Section Library for Use in Criticality Safety Analyses

Description: This report documents the validation of the recently developed 44-group ENDF/B-V based cross-section library (44GROUPNDF5). This cross-section set has been developed for use in the SCALE code system in the analysis of fresh and spent fuel and radioactive waste systems. Collapsed from a 238-group fine-structure cross-section library (238GROUPNDF5), this broad-group library contains approximately 300 nuclides from the ENDF/B-V data files. Additionally, ENDF/B-VI oxygen data have been substituted for ENDF/B-V oxygen, because of discrepancies in ENDF/B-V data. The 44 GROUPNDF5 library was tested against its parent library using a set of 33 benchmark problems in order to demonstrate that the collapsed set was an acceptable representation of 238GROUPNDF5. Results show that the broad 44-group structure is an acceptable representation of its parent 238-group library for thermal as well as hard fast spectrum systems. Accurate broad-group analyses of intermediate spectrum systems will require either a more detailed group structure in this energy range or a more appropriate collapsing spectrum. Further, validation calculations indicate that the 44-group library is an accurate tool in the prediction of criticality for arrays of light-water-reactor-type fuel assemblies, as would be encountered in fresh or spent fuel transportation or storage environments. However, a bias caused by inadequate representation of plutonium cross sections was identified. Further, a possible bias exists with respect to uranium enrichment; however, experiments referenced in this report provide an inadequate sampling of uranium enrichments. Additional work will be required to quantify any bias that may be present.

Publication Date: September 1994

Prepared by: DeHart, M.D.; Bowman, S.M. [Oak Ridge National Lab., TN (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Regulatory Applications

Keywords: criticality, fuel assemblies, fuel storage pools, nuclear data collections, nuclear fuels, radioactive wastes, SCALE, spent fuel storage, storage, transport, validation, water cooled reactors

MELCOR, STCP

Title: Summary of MELCOR 1.8.2 Calculations for Three LOCA Sequences (AG, S2D, and S3D) at the Surry Plant

Description: Activities involving regulatory implementation of updated source term information were pursued. These activities include the identification of the source term, the identification of the chemical form of iodine in the source term, and the timing of the source term's entrance into containment. These activities are intended to support a more realistic source term for licensing nuclear power plants than the current TID-14844 source term and current licensing assumptions. MELCOR calculations were performed to support the technical basis for the updated source term. This report presents the results from three MELCOR calculations of nuclear power plant accident sequences and presents comparisons with Source Term Code Package (STCP) calculations for the same sequences. The three low-pressure sequences were analyzed to identify the materials that enter containment (source terms) and are available for release to the environment and to obtain timing of sequence events. The source terms include fission products and other materials such as those generated by core–concrete interactions. All three calculations, for both MELCOR and STCP, analyzed the Surry plant, a pressurized water reactor (PWR) with a subatmospheric containment design.

Publication Date: March 1994

Prepared by: Kmetyk, L. [Sandia National Labs., Albuquerque, NM (United States)]; Smith, L. [Geo-Centers Inc., Albuquerque, NM (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Safety Issue Resolution

Keywords: availability, biosphere, comparative evaluations, computer calculations, containment systems, fission product release, loss of coolant, MELCOR, reactor protection systems, source terms, STCP, Surry-1 reactor, theoretical data

SUTRA MAC

Title: Performance Assessment of a Hypothetical Low-Level Waste Facility: Groundwater Flow and Transport Simulation. Volume 3

Description: Stochastic subsurface hydrologic theory is applied to data for a hypothetical low-level radioactive waste site to demonstrate the features of the hydraulic parameter estimation process, as developed by Gelhar and others. Effective values of hydraulic conductivity, macrodispersivity, and macrodispersivity enhancement are estimated from the data in this manner. A two-dimensional saturated flow and transport finite-element computer code is used to model the site. Four different isotope inputs and two types of input configurations contribute to an evaluation of model sensitivities. These sensitivities of the mean concentrations and the uncertainties around the mean are explored using an analytical model as an example. Results indicate that the spatial heterogeneity of isotope sorption, through its contribution to longitudinal dispersivity enhancement, has a large effect on the magnitude of concentration predictions, especially for isotopes with short half-lives in comparison to their retarded mean travel times. This observation emphasizes the need for accurate site data measurements that compliment the parameter estimation process. A comparison of simplified analytical screening models with the numerical model predictions shows that the analytical models tend to underestimate concentration levels at low times, potentially as a result of oversimplification of the flow field. Future models could address aspects that are neglected in this report, such as three-dimensionality or unsaturated flow and transport.

Publication Date: May 1994

Prepared by: Talbott, M.E.; Gelhar, L.W. [Massachusetts Inst. of Tech., Cambridge, MA (United States). Ralph M. Parsons Lab.]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Regulatory Applications

Keywords: adsorption, environmental transport, experimental data, flow models, geolcgy, ground water, hydrology, low-level radioactive wastes, radioactive waste disposal, radioactive waste facilities, sensitivity analysis, site characterization, strontium 90, SUTRA MAC, technetium 99, two-dimensional calculations, underground disposal, uranium 238

IRRAS, SAPHIRE, SARA

Title: Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) Version 5.0, Technical Reference Manual

Description: The Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) refers to a set of several microcomputer programs that were developed to create and analyze probabilistic risk assessments (PRAs), primarily for nuclear power plants. This volume provides information on the principles used in the construction and operation of Version 5.0 of the Integrated Reliability and Risk Analysis System (IRRAS) and the System Analysis and Risk Assessment (SARA) system. It summarizes the fundamental mathematical concepts of sets and logic, fault trees, and probability. This volume then describes the algorithms that these programs use to construct a fault tree and to obtain the minimal cut sets. It gives the formulas used to obtain the probability of the top event from the minimal cut sets and the formulas for probabilities that are appropriate under various assumptions concerning repairability and mission time. It defines the measures of basic event importance that these programs can calculate. This volume gives an overview of uncertainty analysis using simple Monte Carlo sampling or Latin Hypercube sampling and states the algorithms used by these programs to generate random basic event probabilities from various distributions. Further references are given, and a detailed example of the reduction and quantification of a simple fault tree is provided in an appendix.

Publication Date: July 1994

Prepared by: Russell, K.D.; Atwood, C.L.; Galyean, W.J.; Sattison, M.B. [EG&G Idaho, Inc., Idaho Falls, ID (United States)]; Rasmuson, D.M. [Nuclear Regulatory Commission, Washington, DC (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Safety Issue Resolution

Keywords: fault tree analysis, IRRAS, Latin Hypercube sampling, Monte Carlo method, nuclear power plants, risk assessment, safety analysis, SAPHIRE, SARA

IRRAS, SAPHIRE

Title: Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE), Version 5.0: Integrated Reliability and Risk Analysis System (IRRAS) Reference Manual. Volume 2

Description: The Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) refers to a set of several microcomputer programs that were developed to create and analyze probabilistic risk assessments (PRAs), primarily for nuclear power plants. The Integrated Reliability and Risk Analysis System (IRRAS) is a state-of-the-art, microcomputer-based probabilistic risk assessment (PRA) model development and analysis tool to address key nuclear plant safety issues. IRRAS is an integrated software tool that gives the user the ability to create and analyze fault trees and accident sequences using a microcomputer. This program provides functions that range from graphical fault tree construction to cut set generation and quantification to report generation. Version 1.0 of the IRRAS program was released in February of 1987. Since then, many user comments and enhancements have been incorporated into the program providing a much more powerful and user-friendly system. This version has been designated IRRAS 5.0 and is the subject of this Reference Manual. Version 5.0 of IRRAS provides the same capabilities as earlier versions and adds the ability to perform location transformations and seismic analysis and provides enhancements to the user interface as well as improved algorithm performance. Additionally, version 5.0 contains new alphanumeric fault tree and event used for event tree rules, recovery rules, and end state partitioning.

Publication Date: July 1994

Prepared by: Russell, K.D.; Kvarfordt, K.J.; Skinner, N.L.; Wood, S.T. [EG and G Idaho, Inc., Idaho Falls, ID (United States)]; Rasmuson, D.M. [Nuclear Regulatory Commission, Washington, DC (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Safety Issue Resolution

Keywords: algorithms, computer program documentation, failure mode analysis, fault tree analysis, IRRAS, manuals, nuclear power plants, reactor safety, reliability, risk assessment, SAPHIRE, systems analysis

Title: Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE), Version 5.0: Integrated Reliability and Risk Analysis System (IRRAS) Tutorial Manual. Volume 3

Description: The Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) refers to a set of several microcomputer programs that were developed to create and analyze probabilistic risk assessments (PRAs), primarily for nuclear power plants. This volume is the tutorial manual for the Integrated Reliability and Risk Analysis System (IRRAS) Version 5.0, a state-of-the-art, microcomputer-based probabilistic risk assessment (PRA) model development and analysis tool to address key nuclear plant safety issues. IRRAS is an integrated software tool that gives the user the ability to create and analyze fault trees and accident sequences using a microcomputer. A series of lessons is provided that guides the user through basic steps common to most analyses performed with IRRAS. The tutorial is divided into two major sections: basic and additional features. The basic section contains lessons that lead the student through development of a very simple problem in IRRAS, highlighting the program's most basic features. The additional features section contains lessons that expand on basic analysis features of IRRAS 5.0.

Publication Date: July 1994

Prepared by: VanHorn, R.L.; Russell, K.D.; Skinner, N.L. [EG and G Idaho, Inc., Idaho Falls, ID (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Safety Issue Resolution

Keywords: automation, computer-aided design, computer program documentation, IRRAS, man–machine systems, manuals, nuclear power plants, risk assessment, SAPHIRE

SAPHIRE, SARA

Title: Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE), Version 5.0. Volume 5, Systems Analysis and Risk Assessment (SARA) Tutorial Manual

Description: The Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) refers to a set of several microcomputer programs that were developed to create and analyze probabilistic risk assessments (PRAs) primarily for nuclear power plants. This volume is the tutorial manual for the Systems Analysis and Risk Assessment (SARA) System Version 5.0, a microcomputer-based system used to analyze the safety issues of a family [i.e., a power plant, a manufacturing facility, any facility on which a probabilistic risk assessment (PRA) might be performed]. A series of lessons is provided that guides the user through some basic steps common to most analyses performed with SARA. The example problems presented in the lessons build on one another, and in combination, lead the user through all aspects of SARA sensitivity analysis capabilities.

Publication Date: July 1994

Prepared by: Sattison, M.B.; Russell, K.D.; Skinner, N.L. [EG and G Idaho, Inc., Idaho Falls, ID (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Safety Issue Resolution

Keywords: manuals, probabilistic estimation, risk assessment, safety engineering, SAPHIRE, SARA

FEP, IRRAS, SAPHIRE

Title: Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) Version 5.0. Fault Tree, Event Tree, and Piping and Instrumentation Diagram (FEP) Editors Reference Manual: Volume 7

Description: The Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) refers to a set of several microcomputer programs that were developed to create and analyze probabilistic risk assessments (PRAs), primarily for nuclear power plants. The Fault Tree, Event Tree, and Piping and Instrumentation Diagram (FEP) editors allow the user to graphically build and edit fault trees, event trees, and piping and instrumentation diagrams (P and IDs). The software is designed to enable the independent use of the graphical-based editors found in the Integrated Reliability and Risk Assessment System (IRRAS). FEP is comprised of three separate editors (Fault Tree, Event Tree, and Piping and Instrumentation Diagram) and a utility module. This reference manual provides a screen-by-screen guide of the entire FEP System.

Publication Date: July 1994

Prepared by: McKay, M.K.; Skinner, N.L.; Wood, S.T. [EG and G Idaho, Inc., Idaho Falls, ID (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Safety Issue Resolution

Keywords: computer graphics, fault tree analysis, FEP, IRRAS, manuals, nuclear power plants, risk assessment, SAPHIRE, text editors

Title: Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE), Version 5.0: Models and Results Data Base (MAR-D) Reference Manual. Volume 8

Description: The Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) refers to a set of several microcomputer programs that were developed to create and analyze probabilistic risk assessments (PRAs), primarily for nuclear power plants. The primary function of MAR-D is to create a data repository for completed PRAs and Individual Plant Examinations (IPEs) by providing input, conversion, and output capabilities for data used by IRRAS, SARA, SETS, and FRANTIC software. As probabilistic risk assessments and individual plant examinations are submitted to the NRC for review, MAR-D can be used to convert the models and results from the study for use with IRRAS and SARA. Then, these data can be easily accessed by future studies and will be in a form that will enhance the analysis process. This reference manual provides an overview of the functions available within MAR-D and step-by-step operating instructions.

Publication Date: July 1994

Prepared by: Russell, K.D.; Skinner, N.L. [EG and G Idaho, Inc., Idaho Falls, ID (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Safety Issue Resolution

Keywords: data base management, FRANTIC, IRRAS, manuals, MAR-D, nuclear power plants, reactor safety, reliability, risk assessment, SAPHIRE, SARA, SETS

NUREG/CR--6120

General

Title: Controlled Field Study for Validation of Vadose Zone Transport Models

Description: Prediction of radionuclide migration through soil and ground water requires models that have been tested under a variety of conditions. Unfortunately, many of the existing models have not been tested in the field, partly because such testing requires accurate and representative data. This report provides the design of a large-scale field experiment representative, in terms of surface area and depth of vadose zone, of an actual disposal area. Experiments are proposed that will yield documented data of sufficient scale to allow testing of a variety of models including effective media stochastic models and deterministic models. Details of the methodology and procedures to be used in the experiment are presented.

Publication Date: August 1994

Prepared by: Wierenga, P.J.; Warrick, A.W.; Yeh, T.C. [Arizona Univ., Tucson, AZ (United States)]; Hills, R.G. [New Mexico State Univ., Las Cruces, NM (United States). Dept. of Mechanical Engineering]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Regulatory Applications

Keywords: geology, ground water, low-level radioactive wastes, model validation, Monte Carlo method, radioactive waste disposal, radionuclide migration, saturation, soils, statistical models

Title: Piping Benchmark Problems for the ABB/CE System 80+ Standardized Plant

Description: To satisfy the need for verification of the computer programs and modeling techniques that will be used to perform the final piping analyses for the ABB/Combustion Engineering System 80+ Standardized Plant, three benchmark problems were developed. The problems are representative of piping systems subjected to representative dynamic loads with solutions developed using the methods being proposed for analysis for the System 80+ standard design. It will be required that the combined licensees demonstrate that their solution to these problems are in agreement with the benchmark problem set. The first System 80+ piping benchmark is a uniform support motion response spectrum solution for one section of the feedwater piping subjected to safe shutdown seismic loads. The second System 80+ piping benchmark is a time–history solution for the feedwater piping subjected to the transient loading induced by a water hammer. The third System 80+ piping benchmark is a time–history solution of the pressurizer surge line subjected to the accelerations induced by a main steam line pipe break. The System 80+ reactor is an advanced PWR type.

Publication Date: July 1994

Prepared by: Bezler, P.; DeGrassi, G.; Braverman, J.; Wang, Y.K. [Brookhaven National Lab., Upton, NY (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Engineering

Keywords: ADLPIPE, auxiliary water systems, compiled data, computer-aided design, NUPIPE, pipes, PIPSYS, PISTAR, PISYS, primary coolant circuits, PSAFE2, PWR type reactors, SAPIV, standardization, SUPERPIPE, supports, validation, water hammer, WECAN

THIRMAL

Title: Fragmentation and Quench Behavior of Corium Melt Streams in Water

Description: The interaction of molten core materials with water has been investigated for the pour stream mixing mode. This interaction plays a crucial role during the later stages of in-vessel core melt progression inside a light water reactor such as during the TMI-2 accident. The key issues that arise during the molten core relocation include the thermal attack and possible damage to the reactor pressure vessel (RPV) lower head from the impinging molten fuel stream and/or the debris bed; the molten fuel relocation pathways including the effects of redistribution due to core support structure and the reactor lower internals; the quench rate of the molten fuel through the water in the lower plenum; the steam generation and hydrogen generation during the interaction; the transient pressurization of the primary system; and the possibility of a steam explosion. In order to understand these issues, a series of six experiments (designated CCM-1 through 6) was performed in which molten corium passed through a deep pool of water in a long, slender pour stream mode. Results discussed include the transient temperatures and pressures, the rate and magnitude of steam/hydrogen generation, and the posttest debris characteristics.

Publication Date: February 1994

Prepared by: Spencer, B.W.; Wang, K.; Blomquist, C.A.; McUmber, L.M. [Argonne National Lab., IL (United States)]; Schneider, J.P. [Illinois Univ., Urbana, IL (United States). Dept. of Nuclear Engineering]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Systems Research

Keywords: cooling, corium, flow models, fluid flow, fragmentation, heat transfer, hydraulics, hydrogen, interactions, meltdown, mixing, nuclear power plants, quenching, reactor safety, steam, testing, THIRMAL, water

Title: User's Guide for Simplified Computer Models for the Estimation of Long-Term Performance of Cement-Based Materials

Description: This report documents user instructions for several simplified subroutines and driver programs that can be used to estimate various aspects of the long-term performance of cement-based barriers used in low-level radioactive waste disposal facilities. The subroutines are prepared in a modular fashion to allow flexibility for a variety of applications. Three levels of codes are provided: the individual subroutines, interactive drivers for each of the subroutines, and an interactive main driver, CEMENT, that calls each of the individual drivers. The individual subroutines for the different models can be taken independently and used in larger programs, or the driver modules can be used to execute the subroutines separately or as part of the main driver routine. A brief program description is included and user-interface instructions for the individual subroutines are documented in the main report. These are intended to be used when the subroutines are used as subroutines in a larger computer code.

Publication Date: February 1994

Prepared by: Plansky, L.E.; Seitz, R.R. [EG and G Idaho, Inc., Idaho Falls, ID (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Regulatory Applications

Keywords: CEMENT, cements, computerized simulation, containment shells, low-level radioactive wastes, radioactive waste storage

NUREG/CR--6145 **EGG--2713**

FEP, IRRAS, MAR-D, SAPHIRE, SARA

Title: Verification and Validation of the SAPHIRE Version 4.0 PRA Software Package

Description: A verification and validation (V&V) process has been performed for the System Analysis Programs for Hands-on Integrated Reliability Evaluation (SAPHIRE). SAPHIRE is a set of four computer programs that the Nuclear Regulatory Commission (NRC) developed to perform probabilistic risk assessments (PRAs). These programs allow an analyst to create, quantify, and evaluate the risk associated with a facility or process being analyzed. The programs included in this set are Integrated Reliability and Risk Analysis System (IRRAS), System Analysis and Risk Assessment (SARA), Models and Results Database (MAR-D), and Fault Tree/Event Tree/Piping and Instrumentation Diagram (FEP) graphical editor. The V&V steps included a V&V plan to describe the process and criteria by which the V&V would be performed; a software requirements documentation review to determine the correctness, completeness, and traceability of the requirements; a user survey to determine the usefulness of the user documentation; identification and testing of vital and non-vital features; and documentation of the test results.

Publication Date: February 1994

Prepared by: Bolander, T.W.; Calley, M.B.; Capps, E.L. [EG and G Idaho, Inc., Idaho Falls, ID (United States)] [and others]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Safety Issue Resolution

Keywords: documentation, fault tree analysis, FEP, IRRAS, MAR-D, nuclear power plants, performance testing, reactor safety, risk assessment, SAPHIRE, SARA, systems analysis, validation

General

Title: Characterization of Class-A Low-Level Radioactive Waste 1986–1990. Volume 1: Executive Summary

Description: Under contract to the U.S. Nuclear Regulatory Commission, office of Nuclear Regulatory Research, the firms of S. Cohen & Associates, Inc. (SC&A) and Eastern Research Group (ERG) have compiled a report that describes the physical, chemical, and radiological properties of Class-A low-level radioactive waste. The report also presents information characterizing various methods and facilities used to treat and dispose of non-radioactive waste. A database management program was developed for use in accessing, sorting, analyzing, and displaying the electronic data provided by EG&G. The program was used to present and aggregate data characterizing the radiological, physical, and chemical properties of the waste from descriptions contained in shipping manifests. The data thus retrieved are summarized in tables, histograms, and cumulative distribution curves presenting radionuclide concentration distributions in Class-A waste as a function of waste streams, by category of waste generators, and by regions of the United States. The report also provides information characterizing methods and facilities used to treat and dispose of non-radioactive waste, including industrial, municipal, and hazardous waste regulated under Subparts C and D of the Resource Conservation and Recovery Act (RCRA). The information includes a list of disposal options, the geographical locations of the processing and disposal facilities, and a description of the characteristics of such processing and disposal facilities. Volume 1 contains the Executive Summary, Volume 2 presents the Class-A waste database, Volume 3 presents the information characterizing non-radioactive waste management practices and facilities, and Volumes 4 through 7 contain Appendices A through P with supporting information.

Publication Date: January 1994

Prepared by: Dehmel, J.C.; Loomis, D.; Mauro, J. [S. Cohen & Associates, Inc., McLean, VA (United States)]; Kaplan, M. [Eastern Research Group, Inc., Lexington, MA (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Regulatory Applications

Keywords: chemical properties, compiled data, low-level radioactive wastes, physical properties, radioactive waste management, radiochemical analysis

Title: Summary of Important Results and SCDAP/RELAP5 Analysis for OECD LOFT Experiment LP-FP-2

Description: This report summarizes significant technical findings from the LP-FP-2 Experiment sponsored by the Organization of Economic Cooperation and Development (OECD). It was the second, and final, fission product experiment conducted in the Loss-of-Fluid Test (LOFT) facility at the Idaho National Engineering Laboratory. The overall technical objective of the test was to contribute to the understanding of fuel rod behavior, hydrogen generation, and fission product release, transport, and deposition during a V-sequence accident scenario that resulted in severe core damage. An 11 by 11 test bundle, comprised of 100 prepressurized fuel rods, 11 control rods, and 10 instrumented guide tubes, was surrounded by an insulating shroud and contained in a specially designed central fuel module that was inserted into the LOFT reactor. The simulated transient was a V-sequence loss-of-coolant accident scenario featuring a pipe break in the low-pressure injection system line attached to the hot leg of the LOFT broken loop piping. The transient was terminated by reflood of the reactor vessel when the outer wall shroud temperature reached 1517 K. With sustained fission power and heat from oxidation and metal–water reactions, elevated temperatures resulted in zircaloy melting, fuel liquefaction, material relocation, and the release of hydrogen, aerosols, and fission products. A description and evaluation of the major phenomena, based upon the response of online instrumentation, analysis of fission product data, postirradiation examination of the fuel bundle, and calculations using the SCDAP/RELAP5 computer code, are presented.

Publication Date: April 1994

Prepared by: Coryell, E.W. [EG and G Idaho, Inc., Idaho Falls, ID (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Systems Research

Keywords: computerized simulation, ECCS, loss of coolant, PWR type reactors, reactor accidents, SCDAP/RELAP5, thermal analysis

ABAQUS, BOSOR4, BOSOR5

Title: Buckling Evaluation of System 80+™ Containment

Description: The System 80+™ containment may be subjected to compressive forces that could cause it to become unstable. The stability of the containment shell under prescribed loading combinations was investigated with two analysis levels: axisymmetric and three dimensional. An axisymmetric shell model, including additional mass to account for penetrations and the spray header system, was analyzed using BOSOR4 and BOSOR5 finite difference codes. Loading combinations with pressure, temperature, self weight, and seismic data satisfied the American Society of Mechanical Engineers (ASME) stress allowables. The buckling assessment was performed using the worst meridian assumption, including material nonlinearities and a sinusoidal axisymmetric imperfection. The minimum factor of safety for Service Level C was 2.35. A Safe Shutdown Earthquake (SSE) seismic margin of 2.91 was calculated. The ABAQUS finite element code was selected for the three-dimensional analysis and tested with classical and BOSOR solutions. The maximum structural response was computed using response spectrum analysis and six potential buckling regions were identified. A set of equivalent static loads was determined for each of the six regions to regenerate the maximum SRSS stress resultants. For each region, combined loads were increased until an instability was detected. A minimum factor of safety of 1.91 was predicted, which does not satisfy ASME Section NE3222.1 or Regulatory Guide 1.57. Code Case N-284 is satisfied. The analysis is conservative primarily because the SRSS 10% method provides a conservative estimate of model coupling.

Publication Date: August 1994

Prepared by: Greimann, L.; Fanous, F.; Safar, S.; Challa, R.; Bluhm, D. [Ames Lab., IA (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Engineering

Keywords: ABAQUS, BOSOR4, BOSOR5, containment shells, deformation, pressure dependence, response functions, seismic effects, stability, steels, stress analysis, temperature dependence, three-dimensional calculations

Title: Numerical Modeling of Ductile Tearing Effects on Cleavage Fracture Toughness

Description: Experimental studies demonstrate a significant effect of specimen size, a/W ratio, and prior ductile tearing on cleavage fracture toughness values (J_c) measured in the ductile-to-brittle transition region of ferritic materials. In the lower-transition region, cleavage fracture often occurs under conditions of large-scale yielding but without prior ductile crack extension. The increased toughness develops when plastic zones formed at the crack tip interact with nearby specimen surfaces which relaxes crack-tip constraint (stress triaxiality). In the mid-to-upper transition region, small amounts of ductile crack extension (often < 1–2 mm) routinely precede termination of the J-Δa curve by brittle fracture. Large-scale yielding, coupled with small amounts of ductile tearing, magnifies the impact of small variations in microscale material properties on the macroscopic fracture toughness, which contributes to the large amount scatter observed in measured J_c-values. Previous work by the authors described a micromechanics fracture model to correct measured J_c-values for the mechanistic effects of large-scale yielding. This new work extends the model to also include the influence of ductile crack extension prior to cleavage. The paper explores development of the new model, provides necessary graphs and procedures for its application, and demonstrates the effects of the model on fracture data sets for two pressure vessel steels (A533B and A515).

Publication Date: May 1994

Prepared by: Dodds, R.H., Jr.; Tang, M. [Univ. of Illinois, Urbana (United States)]; Anderson, T.L. [Texas A&M Univ., College Station, TX (United States)]

Prepared for: Illinois Univ., Urbana, IL (United States)

Keywords: ductility, ferritic steels, finite element method, tensile properties

HMS

Title: Hydrogen Mixing Studies (HMS) User's Manual

Description: Hydrogen Mixing Studies (HMS) is a best-estimate analysis tool for predicting the transport, mixing, and combustion of hydrogen and other gases in nuclear reactor containments and other facilities. It can model geometrically complex facilities that have multiple compartments and internal structures. The code can simulate the effects of steam condensation, heat transfer to walls and internal structures, chemical kinetics, and fluid turbulence. The gas mixture may consist of components included in a built-in library of 20 species. HMS is a finite-volume computer code that solves the time-dependent, three-dimensional (3-D) compressible Navier–Stokes equations. Both Cartesian and cylindrical coordinate systems are available. Transport equations for the fluid internal energy and for gas species densities are also solved. HMS was originally developed to run on Cray-type supercomputers with vector-processing units that greatly improve the computational speed, especially for large, complex problems. Recently the code has been converted to run on Sun workstations. Both the Cray and Sun versions have the same built-in graphics capabilities that allow 1-D, 2-D, 3-D, and time–history plots of all solution variables. Other code features include a restart capability and flexible definitions of initial and time-dependent boundary conditions. This manual describes how to use the code. It explains how to set up the model geometry, define walls and obstacles, and specify gas species and material properties. Definitions of initial and boundary conditions are also described. The manual also describes various physical model and numerical procedure options, as well as how to turn them on. The reader also learns how to specify different outputs, especially graphical display of solution variables. Finally, sample problems are included to illustrate some applications of the code. An input deck that illustrates the minimum required data to run HMS is given at the end of this manual.

Publication Date: December 1994

Prepared by: Lam, K.L.; Wilson, T.L.; Travis, J.R.

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Systems Research

Keywords: atom transport, combustion, computerized simulation, containment buildings, fission products, HMS, hydrogen, manuals, mixing, Navier–Stokes equations, reactor accidents

Title: OFFSCALE: A PC Input Processor for the SCALE Code System. The CSASIN Processor for the Criticality Sequences

Description: OFFSCALE is a suite of personal computer input processor programs developed at Oak Ridge National Laboratory to provide an easy-to-use interface for modules in the SCALE-4 code system. CSASIN (formerly known as OFFSCALE) is a program in the OFFSCALE suite that serves as a user-friendly interface for the Criticality Safety Analysis Sequences (CSAS) available in SCALE-4. It is designed to assist a SCALE-4 user in preparing an input file for execution of criticality safety problems. Output from CSASIN generates an input file that may be used to execute the CSAS control module in SCALE-4. CSASIN features a pulldown menu system that accesses sophisticated data entry screens. The program allows the user to quickly set up a CSAS input file and perform data checking. This capability increases productivity and decreases the chance of user error.

Publication Date: November 1994

Prepared by: Bowman, S.M. [Oak Ridge National Lab., TN (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Industrial and Medical Nuclear Safety

Keywords: computerized simulation, computer program documentation, criticality, CSASIN, OFFSCALE, SCALE, spent fuels

ORIGEN-S, ORIGNATE, SCALE

Title: OFFSCALE: A PC Input Processor for the SCALE Code System. The ORIGNATE Processor for ORIGEN-S

Description: OFFSCALE is a suite of personal computer input processor programs developed at Oak Ridge National Laboratory to provide an easy-to-use interface for modules in the SCALE-4 code system. ORIGNATE is a program in the OFFSCALE suite that serves as a user-friendly interface for the ORIGEN-S isotopic generation and depletion code. It is designed to assist an ORIGEN-S user in preparing an input file for execution of light water reactor (LWR) fuel depletion and decay cases. ORIGNATE generates an input file that may be used to execute ORIGEN-S in SCALE-4. ORIGNATE features a pulldown menu system that accesses sophisticated data entry screens. The program allows the user to quickly set up an ORIGEN-S input file and perform error checking. This capability increases productivity and decreases the chance of user error.

Publication Date: November 1994

Prepared by: Bowman, S.M. [Oak Ridge National Lab., TN (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Industrial and Medical Nuclear Safety

Keywords: computerized simulation, computer program documentation, decay, ORIGEN-S, ORIGNATE, SCALE, spent fuels, transmutation

CSAU

Title: Uncertainty Analysis of Suppression Pool Heating During an ATWS in a BWR-5 Plant. An Application of the CSAU Methodology Using the BNL Engineering Plant Analyzer

Description: The uncertainty of predicting the peak temperature in the suppression pool of a BWR power plant that undergoes an NRC-postulated Anticipated Transient Without Scram (ATWS) has been estimated. The ATWS is initiated by recirculation-pump trips, and then leads to power and flow oscillations as they had occurred at the La Salle-2 Power Station in March of 1988. After limit-cycle oscillations have been established, the turbines are tripped, but without MSIV closure, allowing steam discharge through the turbine bypass into the condenser. Postulated operator actions to lower the reactor vessel pressure and the level elevation in the downcomer are simulated by a robot model that accounts for operator uncertainty. All balance of plant and control systems modeling uncertainties were part of the statistical uncertainty analysis that was patterned after the Code Scaling, Applicability and Uncertainty (CSAU) evaluation methodology. The analysis showed that the predicted suppression-pool peak temperature of 329.3 K (133 °F) has a 95% uncertainty of 14.4 K (26 °F), and that the size of this uncertainty bracket is dominated by the experimental uncertainty of measuring safety and relief valve mass flow rates under critical-flow conditions. The analysis showed also that the probability of exceeding the suppression-pool temperature limit of 352.6 K (175 °F) is most likely zero (it is estimated as < 5-104). The square root of the sum of the squares of all the computed peak pool temperatures is 350.7 K (171.6 °F).

Publication Date: March 1994

Prepared by: Wulff, W.; Cheng, H.S.; Mallen, A.N. [Brookhaven National Lab., Upton, NY (United States)]; Johnsen, G.W. [Idaho National Engineering Lab., Idaho Falls, ID (United States)]; Lellouche, G.S. [Technical Data Services, Chicago, IL (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Systems Research

Keywords: ATWS, BWR type reactors, CSAU, evaluation, pressure suppression, transients

NUREG/CR--6203

NEFTRAN, NEWBALANCE, PHREEQE, VTOUGH

Title: Validation Studies for Assessing Unsaturated Flow and Transport Through Fractured Rock

Description: The objectives of this contract are to examine hypotheses and conceptual models concerning unsaturated flow and transport through heterogeneous fractured rock and to design and execute confirmatory field and laboratory experiments to test these hypotheses and conceptual models. Important new information is presented, such as the application and evaluation of procedures for estimating hydraulic, pneumatic, and solute transport coefficients for a range of thermal regimes. A field heater experiment was designed that focused on identifying the suitability of existing monitoring equipment to obtain required data. A reliable method was developed for conducting and interpreting tests for air permeability using a straddle-packer arrangement. Detailed studies of fracture flow from Queen Creek into the Magina Copper Company ore haulage tunnel have been initiated. These studies will provide data on travel time for transport of water and solute in unsaturated tuff. The collection of rainfall runoff and infiltration data at two small watersheds at the Apache Leap Tuff Site allowed evaluation of the quantity and rate of water infiltrating into the subsurface via either fractures or matrix. Characterization methods for hydraulic parameters relevant to weigh-level waste transport, including fracture apertures, transmissivity, matrix porosity, and fracture wetting front propagation velocities, were developed.

Publication Date: August 1994

Prepared by: Bassett, R.L.; Neuman, S.P.; Rasmussen, T.C.; Guzman, A.; Davidson, G.R.; Lohrstorfer, C.F. [Arizona Univ., Tucson, AZ (United States). Dept. of Hydrology and Water Resources]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Regulatory Applications

Keywords: carbon, fluid flow, fluid mechanics, geologic fractures, high-level radioactive wastes, hydraulics, NEFTRAN, NEWBALANCE, permeability, PHREEQE, porosity, radiation transport, radioactive waste disposal, radionuclide migration, saturation, tuff, vapors, VTOUGH

Title: Transport Calculations of Radiation Exposure to Vessel Support Structures in the Trojan Reactor

Description: Comparison of transport calculations of the dosimeter activities with the experimental measurements shows that the values obtained with ENDF/B-VI cross-section data overestimate the measured results for high-energy threshold reactions in the cavity by up to 41% and thermal reactions by up to a factor of 3.0. The transport calculations performed with the original SAILOR cross-section library (based on ENDF/B-VI data) overestimate measured threshold reactions by only 15% and the thermal reactions by about a factor of 2.50. These results are inconsistent with those obtained in earlier studies that compared transport calculations done with SAILOR vs. ENDF/B-VI, which indicate that SAILOR tends to underestimate cavity dosimeter activities for threshold reactions, whereas the ENDF/B-VI values usually agree better with experimental results. One factor that probably contributes to the rather large discrepancy between the computed and measured activities is the core power distribution used in the transport calculations. Because of the unavailability of plant-specific data, a generic power distribution provided by Westinghouse was used. Since the calculated cavity flux levels appear to be overestimated, the results estimated for the exposure to the support structure should be conservative.

Publication Date: July 1994

Prepared by: Asgari, M.; Williams, M.L. [Louisiana State Univ., Baton Rouge, LA (United States). Nuclear Science Center]; Kam, F.B.K. [Oak Ridge National Lab., TN (United States)]; McGarry, E.D. [National Inst. of Standards and Technology, Gaithersburg, MD (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Engineering

Keywords: calculation methods, cavities, computer calculations, dosimetry, ENDF/B-VI, neutron flux, nuclear data collections, power distribution, radiation transport, SAILOR, Trojan reactor

IFCI

Title: Integrated Fuel–Coolant Interaction (IFCI 6.0) Code. User's Manual

Description: The Integrated Fuel--Coolant Interaction (IFCI) computer code is being developed at Sandia National Laboratories to investigate the fuel–coolant interaction (FCI) problem at large scale using a two-dimensional, four-field hydrodynamic framework and physically based models. IFCI will be capable of treating all major FCI processes in an integrated manner. This document is a product of the effort to generate a stand-alone version of IFCI, IFCI 6.0. The User's Manual describes in detail the hydrodynamic method and physical models used in IFCI 6.0. Appendix A is an input manual, provided for the creation of working decks.

Publication Date: April 1994

Prepared by: Davis, F.J.; Young, M.F. [Sandia National Labs., Albuquerque, NM (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Systems Research

Keywords: computer program documentation, corium, explosions, fuel–coolant interactions, heat transfer, hydraulics, IFCI, jets, mixing, nuclear power plants, reactor accidents, reactor safety

Title: Evaluation of Rock Joint Models and Computer Code UDEC Against Experimental Results

Description: The Mohr–Coulomb, Barton–Bandis, and Continuously Yielding rock joint models and their numerical implementation in the UDEC code were evaluated for their ability to simulate joint behavior under cyclic pseudostatic and dynamic loading conditions. Some deficiencies of these joint models and their implementation in UDEC were identified. These deficiencies include that the rock joint models under evaluation may not be able to sufficiently predict the joint shear and dilation behavior during reverse joint shearing. Both joint forward and reverse shearing are important phenomena of a rock joint behavior. Reverse shearing can result from earthquakes or thermal load, both of which are expected to be experienced during the life of a high-level waste repository. These deficiencies could result in an overestimation of the stability of emplacement drifts and emplacement boreholes and prediction of incorrect near-field flow pattern (including preferential pathways for water and gas).

Publication Date: November 1994

Prepared by: Hsiung, S.M.; Ghosh, A.; Chowdhury, A.H.; Ahola, M.P. [Southwest Research Institute, San Antonio, TX (United States). Center for Nuclear Waste Regulatory Analyses]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Engineering

Keywords: computerized simulation, cracks, fracture mechanics, geologic fissures, geologic fractures, high-level radioactive wastes, radioactive waste disposal, radioactive waste storage, rocks, site characterization, UDEC, underground storage

MELCOR

Title: A Review of the Technical Issues of Air Ingression During Severe Reactor Accidents

Description: Severe reactor accident scenarios involving air ingression into the reactor coolant system are described. Evidence from modern reactor accident analyses and from the accident at Three Mile Island show that residual fuel will be present in the core region when air ingression is possible. This residual fuel can interact with the air. Exploratory calculations with the MELCOR code of station blackout accidents during shutdown conditions and during operations are used to examine clad oxidation by air and ruthenium release from fuel in air. Extensive ruthenium release is predicted when air ingression .ates exceed about 10 mol/s. Past studies of air interactions with irradiated reactor fuel are reviewed. Effects air ingression may have on fission product release, transport, deposition and revaporization are discussed. Perhaps the most important effects of air ingression are expected to be the enhanced release of ruthenium from the fuel and the formation of copious amounts of aerosol from uranium oxide vapors. Revaporization of iodine and tellurium retained in the reactor coolant system might be expected.

Publication Date: September 1994

Prepared by: Powers, D.A.; Kmetyk, L.N.; Schmidt, R.C.

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Systems Research

Keywords: computerized simulation, fission product release, fuel cans, MELCOR, meltdown, oxidation, reactor accidents, reactor cores

Title: Air–Water Simulation of Phenomena of Corium Dispersion in Direct Containment Heating

Description: The present research at Purdue addresses corium dispersion during the direct containment heating in reactor severe accidents. The degree of corium dispersion has not only the strongest parametric effect on containment pressurization but also has the highest uncertainty in predicting it. In view of this, a separate effect test program on corium dispersion mechanisms in the reactor cavity and the subcompartment trapping mechanisms was initiated in spring of 1992 at Purdue under the direction of the Nuclear Regulatory Commission. Four major objectives of this corium dispersion study are (1) to perform a detailed scaling study using the newly proposed step-by-step integral scaling method, then to evaluate existing models for entrainment, particle size, and trapping; (2) to perform carefully designed simulation experiments using water–air and Wood's metal–air in a 1/10 linear scale model; (3) to develop reliable mechanistic models and correlations for corium dispersions that can be used to predict corium jet disintegration, entrainment, drop size, liquid film carry over, and subcompartment trapping; and (4) to use the models to perform stand-along calculations for typical prototypic conditions. The combination of water–air and Wood's metal–air as working fluid will give a unique data base over broad parametric ranges that can be used together with the integral test results to develop reliable models and correlations. The results of the experiments that were conducted using air–water are presented.

Publication Date: October 1994

Prepared by: Ishii, M.; Revankar, S.T.; Zhang, G.; Wu, Q.; O'Brien, P. [Purdue Univ., Lafayette, IN (United States). School of Nuclear Engineering]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Systems Research

Keywords: blowdown, computerized simulation, containment systems, corium, dispersions, distribution, experimental data, heating, meltdown, parametric analysis, pressurization, primary coolant circuits, PWR type reactors, scale models, test facilities

CATHENA, TRAC-PF1/MOD2

Title: A Plan for the Modification and Assessment of TRAC-PF1/MOD2 for Use in Analyzing CANDU 3 Transient Thermal–Hydraulic Phenomena

Description: This report presents the results of the review and planning done for the United States Nuclear Regulatory Commission to identify the thermal–hydraulic phenomena that could occur in the CANDU 3 reactor design during transient conditions, plan modifications to the TRAC-PF1/MOD2 (TRAC) computer code needed to adequately predict CANDU 3 transient thermal–hydraulic phenomena, and identify an assessment program to verify the ability of TRAC, when modified, to predict these phenomena. This work builds on analyses and recommendations produced by the Idaho National Engineering Laboratory (INEL). To identify the thermal–hydraulic phenomena, a large-break loss-of-coolant accident simulation, performed as part of earlier work by INEL with an Atomic Energy of Canada, Ltd. (AECL) thermal–hydraulic computer code (CATHENA), was analyzed in detail. Other accident scenarios were examined for additional phenomena. A group of Los Alamos National Laboratory reactor thermal–hydraulics experts ranked the phenomena to produce a preliminary phenomena identification and ranking table (PIRT). The preliminary nature of the PIRT was a result of a lack of direct expertise with the unique processes and phenomena of the CANDU 3. Nonetheless, this PIRT provided an adequate foundation for planning a program of code modifications. We believe that this PIRT captured the most important phenomena and that refinements to the PIRT will mainly produce clarification of the relative importance (ranking) of phenomena. A plan for code modifications was developed based on this PIRT and on information about the modeling methodologies for CANDU-specific phenomena used in AECL codes. AECL thermal–hydraulic test facilities and programs were reviewed and the information used in developing an assessment plan to ensure that TRAC-PF1/MOD2, when modified, will adequately predict CANDU 3 phenomena.

Publication Date: November 1994

Prepared by: Siebe, D.A.; Boyack, B.E.; Giguere, P.T.

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Systems Research

Keywords: CANDU type reactors, CATHENA, computerized simulation, heat transfer, loss of coolant, performance, thermal analysis, TRAC-PF1/MOD2

Title: Survey of Industry Methods for Producing Highly Reliable Software

Description: The Nuclear Reactor Regulation Office of the U.S. Nuclear Regulatory Commission is charged with assessing the safety of new instrument and control designs for nuclear power plants that may use computer-based reactor protection systems. Lawrence Livermore National Laboratory has evaluated the latest techniques in software reliability for measurement, estimation, error detection, and prediction that can be used during the software life cycle as a means of risk assessment for reactor protection systems. One aspect of this task has been a survey of the software industry to collect information to help identify the design factors used to improve the reliability and safety of software. The intent was to discover what practices really work in industry and what design factors are used by industry to achieve highly reliable software. The results of the survey are documented in this report. Three companies participated in the survey: Computer Sciences Corporation, International Business Machines (Federal Systems Company), and TRW. Discussions were also held with NASA Software Engineering Lab/University of Maryland/CSC and the AIAA Software Reliability Project.

Publication Date: November 1994

Prepared by: Lawrence, J.D.; Persons, W.L. [Lawrence Livermore National Lab., CA (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Reactor Controls and Human Factors

Keywords: computerized control systems, programming, quality assurance, reactor control systems, reactor instrumentation, reactor protection systems, reliability, risk assessment

CRAS, GENII, ISOSHLD, TOXCHEM

Title: Reconcentration of Radioactive Material Released to Sanitary Sewers in Accordance with 10 CFR Part 20

Description: The U.S. Nuclear Regulatory Commission (NRC), in accordance with 10 CFR 20 and state regulations, regulates the discharge of radioactive materials into sanitary sewer systems. A one-year study was conducted by Pacific Northwest Laboratory (PNL) for the NRC to assess whether radioactive materials that are discharged to sanitary sewer systems undergo significant reconcentration within the wastewater treatment plants (WWTP) and to determine the physical and/or chemical processes that may result in radionuclide reconcentration within the WWTPs. The study objectives were addressed by collecting information and data on wastewater treatment, relevant geochemical processes, and individual radionuclide behavior in WWTPs from the open literature, NRC reports, EPA surveys, and interviews with NRC licensees and staff of WWTPs that may be affected by these discharges. Radionuclide mass balance and removal efficiencies were calculated for WWTPs at Oak Ridge, Tennessee, and Erwin, Tennessee, but were not shown to be reliable since the licensee release data generally underestimated the mass of radionuclide that was ultimately found in the sludge. This disparity may be due, in part, to the fact that data available for use in this study were collected to address regulatory concerns and not to perform mass balance calculations. A limited modeling study showed some promise for predicting radionuclide behavior in WWTPs, however, the general applicability of these empirical models remains uncertain. With the data and models currently available, it is not possible to quantitatively determine the physical and chemical processes that cause reconcentration or to calculate, a priori, reconcentration factors for specific WWTP unit processes or WWTPs in general.

Publication Date: December 1994

Prepared by: Ainsworth, C.C.; Hill, R.L.; Cantrell, K.J.; Kaplan, D.I.; Norton, M.V.; Aaberg, R.L. [Pacific Northwest Lab., Richland, WA (United States)]; Stetar, E.A. [Performance Technology Group, Inc., Nashville, TN (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Regulatory Applications

Keywords: abundance, biogeochemistry, CRAS, GENII, ground disposal, ISOSHLD, radiation monitoring, radioactive effluents, radioecological concentration, TOXCHEM, waste water, water treatment plants

Title: KEY Analysis System User's Guide. Version 2.0

Description: The KEY analysis system is a software program designed to process digital waveform data from the United States National Seismograph Network. The KEY system performs many data processing and scientific analysis functions. Detailed operating procedures for the KEY analysis system are provided in this User's Guide.

Publication Date: November 1994

Prepared by: Masse, R.P. [Geological Survey, Denver, CO (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Engineering

Keywords: data analysis, data processing, KEY, seismic waves, seismographs, statistical models

Title: Design Factors for Safety-Critical Software

Description: This report, the fourth of a series of reports prepared for the Nuclear Regulatory Commission Office of Reactor Regulation, provides the summary and conclusion for this task. It is widely believed in the software engineering community that almost anything can affect the ability of software to reliably perform its tasks, particularly when safety is at issue. Although this statement is true in both the abstract and specific instances, it is not particularly helpful. It remains necessary for auditors and other reviewers to assure themselves and the public that safety-critical software has a sufficiently low probability of failing in such a way as to cause death or injury to permit it to be used in safety-critical applications. Achieving this assurance is best done by using a well-planned, methodical approach. A possible approach is to concentrate on those attributes of the software and the development process (design factors) that are most influential in achieving dependable software. Seventy-four design factors are identified in this report, divided into nine categories. Seven categories relate to the development process, and one category relates to the products of that process. The remaining category contains negative factors whose presence should be regarded as cause for intense scrutiny of the development process. Seven of the design factors should be considered mandatory for any organization responsible for developing safety-critical software. An additional nine factors are considered essential to safety but not as important as the first seven. The remaining design factors can provide additional important indications of the quality of the development effort and the software resulting from that effort.

Publication Date: December 1994

Prepared by: Lawrence, J.D.; Preckshot, G.G. [Lawrence Livermore National Lab., CA (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Reactor Controls and Human Factors

Keywords: design, nuclear power plants, quality assurance, reactor safety, reliability

Title: Method for Performing Diversity and Defense-In-Depth Analyses of Reactor Protection Systems

Description: The purpose of this NUREG is to describe a method for analyzing computer-based nuclear reactor protection systems that discovers design vulnerabilities to common-mode failure. The potential for common-mode failure has become an important issue as the software content of protection systems has increased. This potential was not present in earlier analog protection systems because it could usually be assumed that common-mode failure, if it did occur, was the result of slow processes such as corrosion or premature wear-out. This assumption is no longer true for systems containing software. It is the purpose of the analysis method described here to determine points of a design for which credible common-mode failures are uncompensated either by diversity or defense-in-depth.

Publication Date: December 1994

Prepared by: Preckshot, G.G. [Lawrence Livermore National Lab., CA (United States)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Div. of Reactor Controls and Human Factors

Keywords: BWR type reactors, computerized control systems, computerized simulation, defects, ESFAS, failure mode analysis, implementation, PWR type reactors, reactor protection systems, recommendations

RELAP5/MOD3

Title: RELAP5/MOD3 Assessment for Calculation of Safety and Relief Valve Discharge Piping Hydrodynamic Loads. International Agreement Report

Description: This report presents an assessment study for the use of the code RELAP 5/MOD3/5M5 in the calculation of transient hydrodynamic loads on safety and relief discharge pipes. Its predecessor, RELAP 5/MOD1, was found adequate for this kind of calculations by Electric Power Research Institute (EPRI). The hydrodynamic loads are very important for the discharge piping design because of the fast opening of the valves and the presence of liquid in the upstream loop seals. The code results are compared to experimental load measurements performed at the Combustion Engineering Laboratory in Windsor, Connecticut, United States. Those measurements were part of the PWR Valve Test Program undertaken by EPRI after the TMI-2 accident. This particular kind of transient challenges the applicability of the following code models: two-phase choked discharge, interphase drag in conditions with large density gradients, heat transfer to metallic structures in fast changing conditions, and two-phase flow at abrupt expansions. The code applicability to this kind of transient is investigated. Some sensitivity analyses to different code and model options are performed. Finally, the suitability of the code and some modeling guidelines are discussed.

Publication Date: February 1994

Prepared by: Stubbe, E.J.; VanHoenacker, L.; Otero, R. [TRACTEBEL, Brussels (Belgium)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Office of Nuclear Regulatory Research

Keywords: computerized simulation, discharge canals, hydrodynamics, modifications, pipes, primary coolant circuits, PWR type reactors, reactor safety, RELAP5/MOD3, relief valves, test facilities, theoretical data, transients

Title: Assessment of RELAP5/MOD3 with the LOFT L9-1/L3-3 Experiment Simulating an Anticipated Transient with Multiple Failures

Description: The RELAP5/MOD3/5M5 code is assessed using the L9-1/L3-3 test carried out in the LOFT facility, a 1/60-scaled experimental reactor, simulating a loss of feedwater accident with multiple failures and the sequentially induced small-break loss-of-coolant accident. The code predictability is evaluated for the four separated sub-periods with respect to the system response: initial heatup phase, spray and power operated relief valve (PORV) cycling phase, blowdown phase, and recovery phase. On the basis of the comparisons of the results from the calculation with the experiment data, it is shown that the overall thermal–hydraulic behavior important to the scenario, such as a heat removal between the primary side and the secondary side and a system depressurization, can be well predicted and that the code could be applied to the full-scale nuclear power plant for an anticipated transient with multiple failures with a reasonable accuracy. The minor discrepancies between the prediction and the experiment are identified in reactor scram time, post-scram behavior in the initial heatup phase, excessive heatup rate in the cycling phase, insufficient energy convected out the PORV under the hot leg stratified condition in the saturated blowdown phase, and void distribution in secondary side in the recovery phase. This may come from the code uncertainties in predicting the spray mass flow rate, the associated condensation in pressurizer, and junction fluid density under stratified condition.

Publication Date: February 1994

Prepared by: Bang, Y.S.; Seul, K.W.; Kim, H.J. [Korea Inst. of Nuclear Safety, Taejon (Korea, Republic of)]

Prepared for: Nuclear Regulatory Commission, Washington, DC (United States). Office of Nuclear Regulatory Research

Keywords: computerized simulation, loss of coolant, PWR type reactors, reactor accidents, RELAP5/MOD3

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