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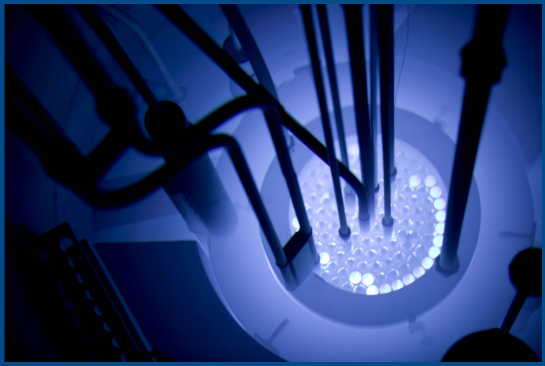

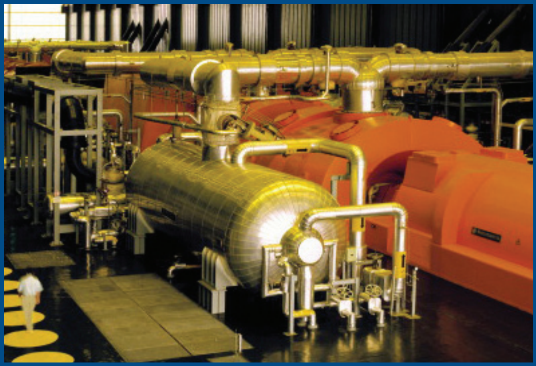

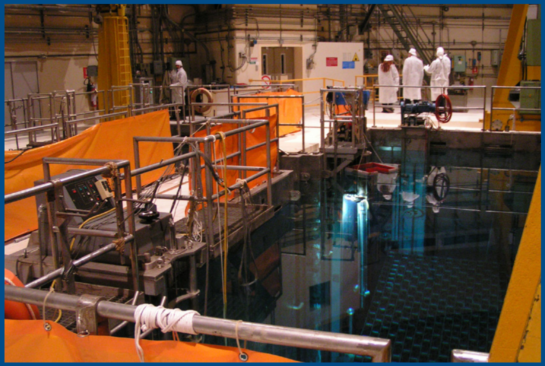

RESEARCH ACTIVITIES 2009

Office of Nuclear Regulatory Research (RES)

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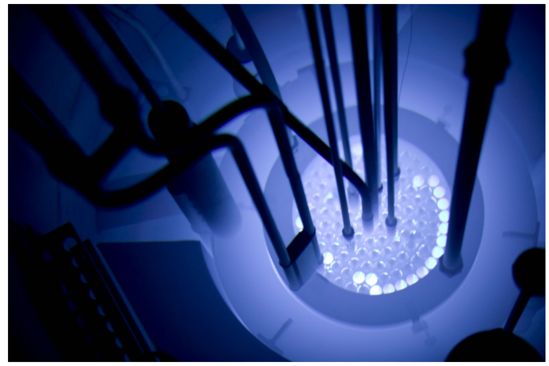
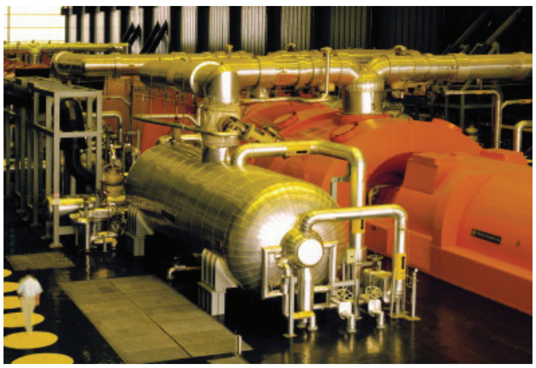
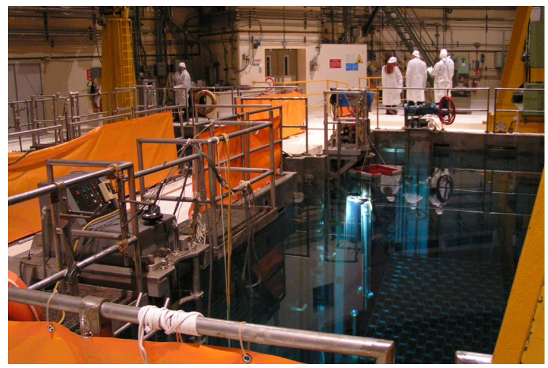

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ABSTRACT

The Office of Nuclear Regulatory Research (RES) develops technical tools, analytical models, and experimental data with which the agency assesses safety and regulatory issues for operating reactors as well as for new and advanced reactor designs. RES staff develops these tools, models, and data through contracts with commercial entities, national laboratories, and universities, or in collaboration with international organizations.

RES conducts research across a wide variety of disciplines, ranging from fuel behavior under accident conditions to seismology to health physics. This research at times also provides the technical bases for regulatory decisions and confirms licensee analyses. RES works closely with the NRC's licensing offices in the review and analysis of high-risk events and provides its expertise to support licensing. RES also develops regulatory guides and is responsible for resolving generic safety issues.

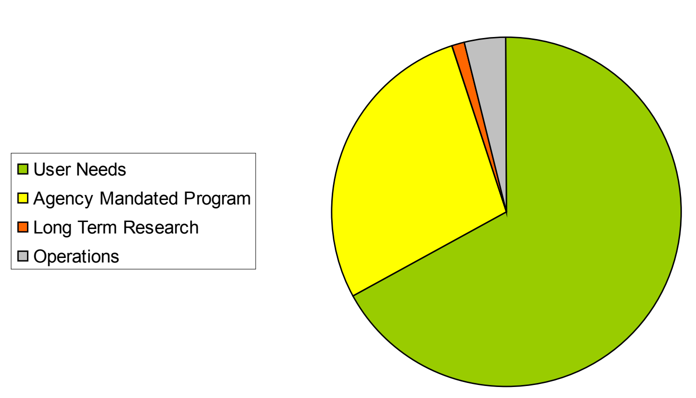
This NUREG provides a collection of information sheets, organized by topical areas and specific projects, that summarize programs currently in progress. If you need additional information, each sheet provides the staff contact and RES Division that can be contacted.

FOREWORD

A Message from the Director

The Office of Nuclear Regulatory Research (RES) is a major U.S. Nuclear Regulatory Commission (NRC) program office, mandated by Congress. Title 10 of the *Code of Federal Regulations* defines the Office's functions. Specifically, the Office plans, recommends, and implements a program of nuclear regulatory research, standards development, and resolution of generic safety issues for nuclear power plants and other facilities regulated by the NRC. The Office coordinates research activities within and outside the agency, including appointment of staff to committees and conferences. In addition, the Office coordinates NRC participation in activities related to international standards.

RES FY09 Activities



The Office's annual budget is around $70–$80 million; the accompanying chart illustrates the funding allocation.

- Two-thirds of activities are driven by the needs of regulatory offices (User Needs).
- One-third of activities are driven by the Commission (Agency-Mandated Programs).
- A small amount of long-term research focuses on subjects expected to be critical 5 to 10 years in the future.

Currently, the Office has about 250 staff members. This staff continues to reflect diversity in academic degrees, demographics, and technical disciplines. The wide range of engineering and scientific disciplines includes expertise in nuclear, materials, human factors and human reliability, health physics, fire protection, seismology, and probabilistic risk assessment, to name a few. It is this diversity in highly technical and specialized disciplines that allows RES to support the licensing offices as they carry out their licensing and regulatory tasks. In each of these disciplines, RES supports the licensing offices by producing technical studies, methods and tools, and regulatory guidance. This year, RES also initiated a management initiative entitled RES FOCUS AREAS 2009 (see page vi) that identified office improvement priorities and created groups to implement specific activities that would improve RES operations and benefit its staff and customers. These focus areas are intertwined in the office and its activities.

To summarize, in this NUREG, individual information sheets contain summaries for each research activity. The summaries are meant only to provide an overview of the activity discussed. Any questions or comments on the content should be directed to the technical staff or the Division noted in the specific information sheet.

We appreciate your interest in these RES activities, and the Office will continue to issue annual updates of this NUREG for your information

Brian W. Sheron, Director
Office of Nuclear Regulatory Research


RES
Focus Areas 2009
Five Focus Areas
Maintain High Technical Quality
Emphasize Project Management & Fiscal Awareness
Optimize Performance of Corporate Support
Stay Connected & Maintain Relationships w/Stakeholders
Promote Self-Development & Well-Being
Stay Connected & Maintain Relationships w/Stakeholders
Reach Out to Customers to Overcome Distance
Utilize Technology to Maintain Relationships
Share Success Stories & Best Practices
Lead: Chris Hoxie
Executive Sponsor: Doug Coe
Maintain High Technical Quality
Develop Standardized Research Plans
Develop Life-Cycle Project Coordination Process
Collect Feed Back on Research Products
Lead: Ben Beasley
Executive Sponsor: Kathy Gibson
Promote Self-Development & Well-Being
Develop and Maintain Knowledge Management Notebook
Form an Employee Network Group Church Street Information, Living and Learning (CHILL)
Lead: Tim Lupold
Executive Sponsor: Stu Richards
Emphasize Project Management & Fiscal Awareness
Optimize ROMA Capabilities
Maintain Current Contracting Templates on the Web
Apply Lean Six Sigma to Contracting Processes
Lead: Jim Danna
Executive Sponsor: Mary Muessle
Optimize Performance of Corporate Support
Adapt Electronic Document and Action Tracking System to RES Tracking Systems
Establish Central Location for Up-To-Date Guidance For Processes and Procedures
Lead: Gina Thompson
Executive Sponsor: Mike Case

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FIGURES

ABBREVIATIONS AND ACRONYMS

NUMERALS

ΔCDP	change in core damage probability
4S	Toshiba Super Safe, Small and Simple reactor

A

ABWR	Advanced Boiling-Water Reactor
ac	alternating current
ACRS	Advisory Committee on Reactor Safeguards
ADAMS	Agencywide (NRC) Documents Access and Management System
AES	Advanced Environmental Solutions, LLC
ALARA	as low as reasonably achievable
AMP	Aging Management Program
ANL	Argonne National Laboratory
AO	abnormal occurrence
AP1000	Advanced Passive 1000 Megawatt
APEX	Advanced Power Extraction
API	application programming interface
APWR	U.S. Advanced Pressurized Water Reactor (Mitsubishi)
ARRP	Advanced Reactor Research Program
ARTIST	AeRosol Trapping In STeam generator
ASCII	American Standard Code for Information Interchange
ASME	American Society of Mechanical Engineers
ASP	accident sequence precursor

B

BAM	German Federal Institute for Materials Research and Testing
BFBT	BWR full-size fine-mesh bundle test
BIP	Behavior of Iodine Project (CSNI)
BNL	Brookhaven National Laboratories
BRIIE	Baseline Risk Index for Initiating Event
BWR	boiling-water reactor

C

Cal/g	calorie per gram
CAMP	Code Application and Maintenance Program
CAROLFIRE	Cable Response to Live Fire
CCDP	conditional core damage probability
CCF	common-cause failure
CCI	core-concrete interaction
CFD	computational fluid dynamics
CFR	Code of Federal Regulations
CHRISTI-FIRE	Cable Heat Release, Ignition, and Spread in Tray Installations during FIRE
COL	combined license
CONTAIN	Containment Analysis Code
CP	computerized procedure
CRDM	control rod drive mechanism
CSARP	Cooperative Severe Accident Research Program (U.S. NRC)
CSAU	Code Scaling, Applicability, and Uncertainty
CSM	conceptual site model
CSNI	Committee on the Safety of Nuclear Installations

D

D3	diversity and defense-in-depth
DBA	design-basis accident
DC	design certification
dc	direct current
DESIREE-FIRE	Direct Current Electrical Shorting In Response to Exposure-Fire
DFWCS	digital feedwater control system
DOE	U.S. Department of Energy

E

ECI	exterior communications interface
EPA	U.S. Environmental Protection Agency
EPAct	Energy Policy Act of 2005
EPICUR	Experimental Program for Iodine Chemistry Under Irradiation
EPIX	Equipment Performance and Information Exchange System
EPR	U.S. Evolutionary Power Reactor
EPRI	Electric Power Research Institute
ESBWR	Economic Simplified Boiling-Water Reactor (General Electric)

F

FAQ	frequently-asked-questions
FDT	fire dynamics tools
Fe	iron
FFD	fitness for duty
FRB	Fire Research Branch
FLECHT	Full Length Emergency Cooling Heat Transfer
FPGA	field programmable gate array
FPT	fission product transport
FR	Federal Register
FRAPCON3	Fuel Rod Analysis Program (FRAPCON3 is the Constant (steady portion and FRAPTRAN is the Transient version)
FY	fiscal year

G

GDC	general design criterion
GI	generic issue
GIP	Generic Issues Program
GSI	generic safety issue
GUI	graphical user interface
GWD/t	gigawatt day per ton

H

HAMMLAB	Halden Man-Machine Laboratory
HBWR	Halden Boiling-Water Reactor
HEP	human error probability
HERA	Human Event Repository and Analysis
HFE	human factors engineering
HGL	hot gas layer
HRA	human reliability analysis
HRP	Halden Reactor Project
HRR	heat release rate
HSI	human-system interface
HTGR	high-temperature gas-cooled reactor

I

IAEA	International Atomic Energy Agency
I&C	instrumentation and control
ICAP	International Code Assessment and Application Program
ICRP	International Commission on Radiological
Protection	
IFE	Institutt for Energiteknikk
INL	Idaho National Laboratory
IPEEE	individual plant examinations of external
events	
IRIS	International Reactor Innovative and Secure Light Water Reactor (Westinghouse)
IRSN	Institut de Radioprotection et de Sûreté Nucléaire (French Institute for Radiological Protection and Nuclear Safety)
ISG	interim staff guidance
ISI	inservice inspection
ISL	in situ leach
ISTP	International Source Term Program
ITP	Industry Trends Program

J

JAERI	Japan Atomic Energy Research Institute

K

KM	knowledge management

L

LANL	Los Alamos National Laboratory
LBB	leak-before-break
LER	licensee event report
LERF	large early-release frequency
LLW	low-level waste
LOCA	loss-of-coolant accident
LOFW	loss of feedwater
LWR	light-water reactor

M

MACCS	MELCOR Accident Consequence Code System
MARSAME	Multi-Agency Radiation Survey and Assessment of Materials and Equipment Manual
MARSSIM	Multi-Agency Radiological Survey and Site Investigation Manual
MASLWR	Multi-Application Light Water Reactor
MATLAB	MATrix LABoratory
MCAP	MELCOR Code Assessment Program
MCCI	Melt Coolability and Concrete Interaction
MCNP	Monte Carlo N-Particle Transport Code
MD	monitoring device
MD	management directive
MELCOR	
MOX	mixed oxide
MOX FFF	Mixed-Oxide Fuel Fabrication Facility
MP	monitoring point
MSPI	Mitigating Systems Performance Index
MTO	Man-Technology-Organization
MW	megawatt

N

NCI	U.S. National Cancer Institute
NDE	nondestructive examination
NEA	Nuclear Energy Agency
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NGA	next generation attenuation
NGNP	Next Generation Nuclear Plant
NIST	National Institute of Standards and Technology
NGNP	Next Generation Nuclear Plant
NMSS	Office of Nuclear Material Safety and Safeguards
NPP	nuclear power plant
NRC	U.S. Nuclear Regulatory Commission
NRO	Office of New Reactors
NRR	Office of Nuclear Reactor Regulation
NUREG	NRC technical report designation
NUREG/CR	NRC technical report designation/contractor report
NUREG/IA	NRC technical report designation/international agreement

O

OECD	Organization for Economic Cooperation and Development
ORNL	Oak Ridge National Laboratory

P

PA	performance assessment
PANDA	PAssive Non-Destructive Assay of Nuclear Materials
PARCS	Purdue's Advanced Reactor Core Simulator
PA-UT	phased array ultrasonic
PBMR	pebble bed modular reactor
PBP	paper-based procedure
PI	performance indicator
PINC	Program for the Inspection of Nickel-Alloy Components
PIRT	phenomena identification and ranking table
PKL	Primärkreislauf-Versuchsanlage (German for primary coolant loop test facility)
PMMD	Proactive Management of Materials Degradation
PNNL	Pacific Northwest National Laboratory
PPS	Package Performance Study
PRA	probabilistic risk assessment
PSA8	Probabilistic Safety Conference 2008
PSHA	probabilistic seismic hazard assessment
PTS	pressurized thermal shock
PUMA	Purdue University Multi-Dimensional Integral Test Assembly
PWR	pressurized-water reactor
PWSCC	primary water stress-corrosion cracking

Q

R

RADS	Reliability and Availability Data System
RADTRAD	RADionuclide Transport, Removal, and Dose code
RASP	Risk Assessment Standardization Project
RCS	reactor coolant system
R&D	research and development
RG	regulatory guide
REIRS	Radiation Exposure Information and Reporting System
RELAP5	Reactor Excursion and Leak Analysis Program
REMIX	Regional Mixing Model
RES	Office of Nuclear Regulatory Research
RIDM	risk-informed decisionmaking
ROE	red oil excursion
ROP	Reactor Oversight Process
ROSA-IV and V	Rig of Safety Assessment
RPV	reactor pressure vessel
RSICC	Radiation Safety Information Computational Center
RV	reactor vessel

S

SAPHIRE	Systems Analysis Programs for Hands-on Integrated Reliability Evaluation
SDP	Significance Determination Process
SEASET	Separate Effects And Systems Effects Tests
SECY	Office of the Secretary (NRC)
SFR	sodium-cooled fast reactor
SFP	spent fuel pool
SG	steam generator
SGTR	steam generator tube rupture
SKC	susceptibility, knowledge, and confidence
SI Units	International System of Units (abbreviated SI from the French Ie Systeme International)
SNAP	Symbolic Nuclear Analysis Package
SNF	spent nuclear fuel
SNFT	spent nuclear fuel transportation
SOARCA	State-of-the-Art Reactor Consequence Analysis
SPAR	Standardized Plant Assessment of Risk
SRM	staff requirements memorandum
SRP	Standard Review Plan
SSC	structure, system, and component
SSHAC	Senior Seismic Hazard Analysis Committee
SSWICS	Small-Scale Water Ingression and Crust Strength
S/U	sensitivity/uncertainty

T

T/H	thermal-hydraulic
THIEF	Thermally-Induced Electrical Failure model
TRACE	TRAC/RELAP Advanced Computational Engine (TRAC stands for Transient Reactor Analysis Code)
TRISO	TRistructual-ISOtropic
TWG	task working group

U

U	uranium
U.S.-APWR	U.S. Advanced Pressurized-Water Reactor
USEGC	U.S. east and gulf coasts
USGS	U.S. Geological Survey

V

VARSKIN	code used to model and calculate skin dose
VERCORS	French test program
VHTR	very-high-temperature gas-cooled reactor
V&V	verification and validation

W

WIR	waste-incidental-to-reprocessing

X

xLPR	extremely low probability of rupture

CHAPTER 1: DIGITAL INSTRUMENTATION AND CONTROL RESEARCH

Digital Instrumentation and Control

Digital Instrumentation and Control

Probabilistic Risk Assessment

DIGITAL INSTRUMENTATION AND CONTROL

Background

The digital instrumentation and control (I&C) area continues to evolve as the technology changes and the NRC continues to refine its regulatory approach. Current control rooms are dominated by analog equipment such as electromechanical switches, annunciators, chart recorders, and panel-mounted meters. However, as nuclear power plants upgrade their control rooms, they are replacing analog equipment with modern digital equipment including flat screen operator interfaces and soft controls. Future plants will have highly integrated control rooms similar to that shown in Figure 1.1. The NRC expects a substantial increase in the use of digital systems for both new reactors and retrofits in operating reactors. As a result, the agency is updating applicable licensing criteria and regulatory guidance and performing research to support licensing of these new digital I&C systems.

In the 1990s, the NRC developed guidance to support the review of digital systems in nuclear power plants. Since then, the agency has been effectively using the current licensing guidance for review of applications of digital technology in operating reactors and in certification of new reactor designs. In an effort to continue improving the licensing process, the NRC commissioned the National Academy of Sciences National Research Council to review issues associated with the use of digital systems. The National Research Council issued the report "Digital Instrumentation and Control Systems in Nuclear Power Plants," in 1997, and made several recommendations, including the suggestion that the NRC update its research program to balance short-term regulatory needs and long-term anticipated research needs. The Advisory Committee on Reactor Safeguards has also encouraged research in the digital I&C area to keep pace with the ever-changing technology.

Overview

RES has developed a comprehensive Digital System Research Program Plan, which defines the I&C research programs to support the regulatory needs of the agency. RES developed the research plan with input from several sources, including the National Research Council report on digital I&C systems at nuclear power plants, the Advisory Committee on Reactor Safeguards, external stakeholders, and the NRC staff. User offices have approved the research plan, which currently consists of many projects. The products of these research programs include technical review guidance, information to support regulatory-based acceptance criteria, assessment tools and methods, and reviews that address key technical issues that affect the licensing of operating and new reactors.

Figure 1.1 Highly integrated control room

In 2007, the NRC formed a Digital I&C Steering Committee and seven task working groups (TWGs) to work with the nuclear industry in improving regulatory guidance for digital I&C system upgrades in operating reactors, support design certification submittals for new reactors, and support review of digital I&C systems in fuel cycle facilities. The research projects under the Digital System Research Program also support the development of improved regulatory guidance by the TWGs.

Research Program

RES is currently conducting research in several key technical areas that affect licensing of operating reactors and new reactors.

The NRC applies its diversity and defense-in-depth (D3) policy as a means to address common-cause failures in digital safety systems. However, knowledge of digital technology has increased significantly, and the technology itself has evolved considerably since the agency established its D3 policy in 1993. The current NRC D3 guidance considers six categories of diversity attributes that can be used in the design of digital systems, including hardware, software, and operator action. The complexity of determining digital system failure modes and assessing the adequacy of D3 of the system can make this guidance difficult to use. This research project will develop combinations of diversity attributes and associated criteria that provide acceptable D3 strategies for addressing common-cause failure vulnerabilities. The agency will use these improved methods for D3 analysis to enhance and further refine regulatory guidance and acceptance criteria for licensing activities.

The NRC and the industry are interested in risk-informing digital safety system licensing reviews. One of the major challenges to risk-informing digital system reviews is developing an acceptable method for modeling digital system reliability. The staff examined reliability and risk methods developed in other industries, such as aerospace, defense, and telecommunications. These digital system risk modeling methods may be adaptable for use in the nuclear industry. Based on its review of these techniques and available failure data, the staff is evaluating digital system modeling methods including traditional event-tree/fault-tree and dynamic methods, with the intent of establishing the best practice for modeling digital systems in nuclear power plants.

New reactor control room designs will use highly integrated "glass" or "cockpit style" controls and displays and advanced control strategies, such as touch screen video display devices and semi-autonomous controls. The NRC must enhance its understanding of these control room designs and develop guidance to ensure compliance with regulatory requirements. In 2004, the staff published the results of a study to review international experience with digital control rooms. This report identified potential issues associated with digital system architecture and communications, information and data management, and system performance. A research project is in progress to further evaluate these items, and regulatory guidance will be established. As part of this research effort, the staff will continue to collaborate with other industries, foreign regulators, and Government agencies, including Naval Reactors, to benefit from others' experiences in licensing highly integrated control room designs. This research will be useful in developing licensing review and acceptance criteria for issues such as electrical and communication separation and independence between (1) safety-related and nonsafety-related displays and controls and (2) redundant safety channels (interchannel communications).

The staff is actively engaged in ongoing cyber research to explore cyber vulnerabilities in digital systems that are expected to be deployed in nuclear power plants. This research will ultimately provide regulatory guidance and tools for evaluating digital systems for cyber vulnerabilities, including potential vulnerabilities arising from interconnections between safety and nonsafety systems. The staff has already initiated cooperative agreements with a licensee and microprocessor vendor to perform cyber assessments of their digital systems. Also, the staff is currently developing a new Regulatory Guide 5.71, "Cyber Security Programs for Nuclear Facilities," in support of the rulemaking for Title 10 of the Code of Federal Regulations, 73.54, "Protection of Digital Computer and Communication Systems and Networks."

Looking to the future, the agency is interacting with the industry in other current research projects that will support licensing of new technologies in nuclear power plants. Research to assess alternatives to traditional software-driven microprocessor technology is ongoing. These devices are referred to as field programmable gate arrays, or FPGAs, which can be programmed one time to perform the basic function of logic gates. New reactor vendors and current licensees have announced their intention to implement safety functions using FPGAs. This research project will build on regulatory approaches used in other countries and other agencies.

Finally, new research is planned in several areas in preparation for the U.S. Department of Energy advanced reactor design programs. For the digital I&C area, research plans for advanced instrumentation, advanced control schemes, and advanced diagnostics and prognostics are being formulated.

For More Information

Contact Russell Sydnor, RES/DE, at 301-251-7405 or Russell.Sydnor@nrc.gov

DIGITAL INSTRUMENTATION AND CONTROL PROBABILISTIC RISK ASSESSMENT

Background

Nuclear power plants (NPPs) have traditionally relied on analog systems for their monitoring, control, and protection functions. With a shift in technology to digital systems because of analog obsolescence and digital functional advantages, existing plants have begun to replace current analog systems, while new plant designs fully incorporate digital systems. Since digital instrumentation and control (I&C) systems are expected to play an increasingly important role in nuclear power plant safety, the NRC has devised a digital I&C research plan that defines a coherent set of research programs to support its regulatory needs.

The current licensing process for digital I&C systems is based on deterministic engineering criteria. In its 1995 policy statement on probabilistic risk assessment (PRA), the Commission encouraged the use of PRA technology in all regulatory matters to the extent supported by the state of the art in PRA methods and data. Although many activities have been completed in the area of risk-informed regulation, the risk-informed analysis process for digital I&C systems has not yet been satisfactorily developed. Since, at present, no consensus methods exist for quantifying the reliability of digital I&C systems, one of the programs included in the NRC digital I&C research plan addresses risk assessment methods and data for digital I&C systems. The objective of this research is to identify and develop methods, analytical tools, and regulatory guidance to support (1) NPP licensing decisions using information on the risks of digital systems and (2) inclusion of models of digital systems in NPP PRAs.

Approach

Previous and current RES projects have identified a set of desirable characteristics for reliability models of digital systems and have applied various probabilistic reliability modeling methods to an example digital system (i.e., a digital feedwater control system [DFWCS]). Several NUREG/CR reports, which have received extensive internal and external stakeholder review, document this work. The results of these "benchmark" studies have been compared to the set of desirable characteristics to identify areas where additional research might improve the capabilities of the methods. One specific area currently being pursued by RES is the quantification of software reliability. To examine the substantial differences in PRA modeling of software (versus conventional NPP components), in May 2009, RES convened a workshop involving experts with knowledge of software reliability and/or NPP PRA. At the workshop, the experts established a philosophical basis for modeling software failures in a reliability model. RES is now reviewing quantitative software reliability methods and plans to develop one or two technically sound approaches to modeling and quantifying software failures in terms of failure rates and probabilities. Assuming such approaches can be developed, they will then be applied to an example software-based protection system in a proof-of-concept study.

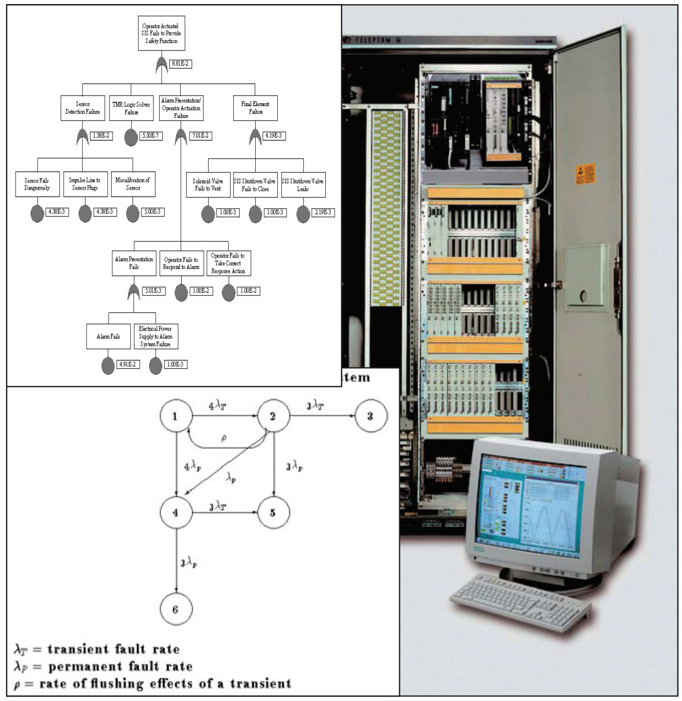

Figure 1.2 PRA Modeling Methods and Digital Instrumentation and Control Cabinet

The results of the benchmark studies have also highlighted the following areas where enhancement in the state of the art for PRA modeling of digital systems is needed:

- approaches for defining and identifying failure modes of digital systems and determining the effects of their combinations on the system
- methods and parameter data for modeling self-diagnostics, reconfiguration, and surveillance, including using other components to detect failures
- better data on hardware failures of digital components, including addressing the potential issue of double-crediting fault-tolerant features, such as self-diagnostics
- better data on the common-cause failures (CCFs) of digital components
- methods for modeling software CCF across system boundaries (e.g., when there is common support software)

- methods for addressing modeling uncertainties in modeling digital systems
- methods for human reliability analysis associated with digital systems
- determining if and when a model of controlled processes is necessary in developing a reliability model of a digital system

Even if an acceptable method is established for modeling digital systems in a PRA and progress is made in the above areas, (1) the level of effort and expertise required to develop and quantify the models will need to be practical for vendors and licensees and (2) the level of uncertainty associated with the quantitative results will need to be sufficiently constrained so that the results are useful for regulatory applications.

International Collaboration

In October 2008, RES staff led a technical meeting on digital I&C risk modeling for the working group on risk of the Organization for Economic Cooperation and Development, Nuclear Energy Agency, Committee on the Safety of Nuclear Installations. The objectives of this meeting were to make recommendations regarding current methods and information sources used for quantitative evaluation of the reliability of digital I&C systems for PRAs of NPPs, and identify, where appropriate, the near- and long-term developments necessary to improve modeling and evaluation of the reliability of these systems. The meeting included discussion of many of the areas of needed enhancement identified above. During the meeting, it became apparent that although many studies have been performed in various countries, the models of digital I&C systems developed so far vary widely in scope and level of detail, and there is no consensus on an acceptable method for modeling digital systems. The participants agreed that probabilistic data are scarce, so the need to address this shortcoming is urgent. While the meeting did not produce specific recommendations of the methods or information sources that should be used for quantitative evaluation of the reliability of digital I&C systems for PRAs of NPPs, it did provide a useful forum for the participants to share and discuss their experience with modeling these systems. The staff is currently considering the possibility of pursuing bilateral agreements with one or more of the countries that provided information at the meeting.

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CHAPTER 2: FIRE SAFETY RESEARCH

Red Oil Hazard Risk Analysis

Fire Modeling Activities

Methods Development and Stakeholder Interaction for Fire Probabilistic Risk Assessment and Fire Human Reliability Analysis

DESIREE-Fire—Direct Current Electrical Shorting In Response to Exposure-Fire

Cable Heat Release Rate, Ignition, and Flame Spread

Electrical Cable Fire Damage

RED OIL HAZARD RISK ANALYSIS

Background

In SECY-04-0182, dated October 7, 2004, the NRC staff reported to the Commission on the status of risk-informed regulation in the Office of Nuclear Material Safety and Safeguards (NMSS). The SECY proposed the trial use of a guidance document, “Risk-Informed Decision-Making for Nuclear Material and Waste Applications” (RIDM) (ML080720238), to assist staff in applying risk information. The staff requirements memorandum to SECY-04-0182, dated January 18, 2005, approved this proposed use with caveats and directed specific changes to be made to the document. The document (with the changes and caveats from the Commission incorporated) was issued for trial use on May 11, 2005.

Since issuance, the guidance in the RIDM document has been used to provide risk insights in specific regulatory areas within NMSS. The NRC has been using the RIDM methodology in reviewing the license application for the mixed-oxide fuel fabrication facility (MOX FFF).

During the construction authorization request stage of the NRC’s review, staff expressed concern about the effectiveness of the process safety features to prevent and mitigate the formation and consequences of a red oil excursion (ROE). For the license application and integrated safety analysis, the licensee updated the design of the facility to increase protection against the red oil hazard.

Red oil is formed when an organic constituent reacts with nitric acid in specific conditions of temperature, concentration, and residence time. Previous studies have shown that red oil decomposition is exothermic and could generate a relatively large amount of gas. Therefore, a risk exists of runaway reaction(s) and overpressurization.

Approach

Brookhaven National Laboratories (BNL) was given the task of conducting a risk analysis study on the red oil hazard of the MOX FFF. This study considered all processes that are vulnerable to ROEs. BNL performed a qualitative assessment of the possibility of red oil reactions for the various units composing the Aqueous Polishing Unit. To supplement the qualitative analyses, BNL conducted a quantitative evaluation, which used failure modes and effects analyses consisting of event trees and fault trees, to gain further insights into possible combinations of failures that could lead to ROEs. The study analyzed active and passive engineered controls and administrative controls of the MOX FFF used to prevent an ROE. The results of the BNL assessment show that the point estimates of ROE in the various process units are low. The low values reflect the robustness and defense-in-depth character of the multiple strategies employed in the facility.

BNL performed the analysis using probabilistic risk analysis techniques. These techniques can be considered as risk-informing the qualitative analysis to help the NRC staff focus attention on areas of higher risk significance.

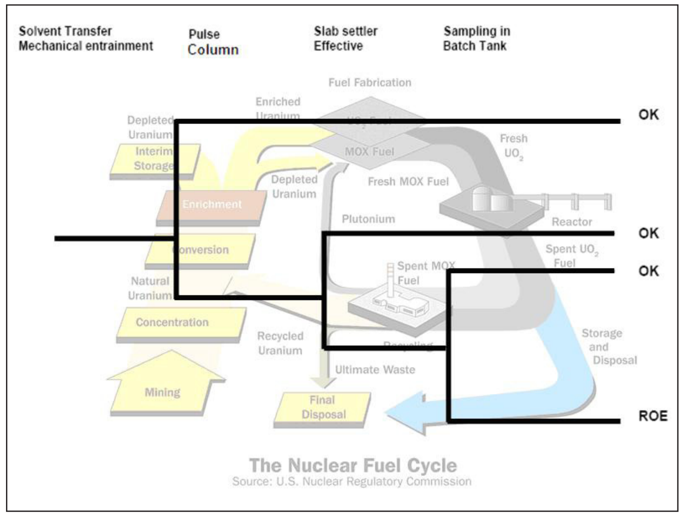

Figure 2.1 Simplified event tree with the nuclear fuel cycle in the background. The event tree shows the conditions for a possible ROE, given certain failures in the equipment. These equipment failures can cause the organic constituent to be transferred to equipment with the conditions necessary for red oil formation and potentially an ROE.

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FIRE MODELING ACTIVITIES

Background

The results of the Individual Plant Examinations of External Events Program and actual fire events indicate that fire can contribute significantly to nuclear power plant risk, depending on design and operational conditions. Risk assessments often use fire models to evaluate fire scenarios. The models are used to determine damage to cables and other systems and components important to safety. They are also used to characterize the progression of fire beyond initial targets. Used in these ways, fire models are important tools in determining the contribution of fire to the overall risk in nuclear power plants (NPPs).

The NRC recently amended its fire protection requirements to allow existing reactor licensees to voluntarily adopt the fire protection requirements contained in National Fire Protection Association (NFPA) 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants." NFPA 805 allows licensees to use fire models to evaluate their fire protection program. However, the fire models that are used must be verified and validated and acceptable to the NRC. To this end, RES, along with the Electric Power Research Institute (EPRI) and the National Institute of Standards and Technology (NIST), conducted an extensive verification and validation study of fire models used to analyze NPP fire scenarios. This study has resulted in the seven-volume report, NUREG-1824, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications," issued May 2007.

A need exists in fire risk assessments to determine when cables fail during a fire in NPPs. In the past, cable damage models have been crude and have not been validated. Recently, as part of the Cable Response to Live Fire (CAROLFIRE) program, the NRC and NIST have developed a simple cable damage model called Thermally-Induced Electrical Failure (THIEF). This model uses empirical information about cable failure temperatures and calculations of the thermal response of a cable to predict the time to cable damage. The THIEF model was benchmarked and validated against real cable failure and thermal data acquired during the CAROLFIRE program.

Approach

The results in NUREG-1824 are designed to give licensees and the NRC insights into the predictive capabilities of the various models evaluated. For example, although engineering calculations have limited capabilities, they provide reasonable estimates of certain phenomena when used within limitations. These insights are valuable to fire model users who are developing analyses to support the transition to NFPA 805, to justify exemptions from existing prescriptive regulatory requirements, and to conduct reviews under the Reactor Oversight Process.

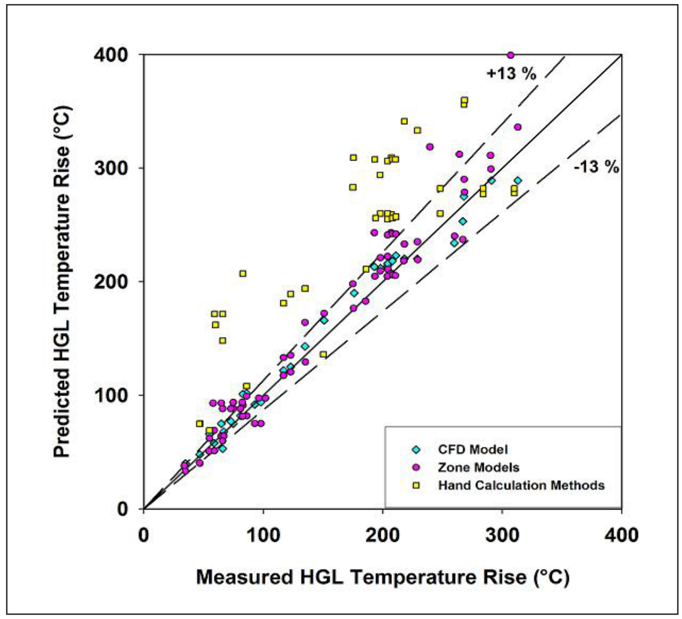

Figure 2.2 Measured vs. predicted hot gas layer (HGL) temperature rise. The models evaluated provide reasonable estimates of actual temperature rise.

The THIEF model will be implemented into both two zone and computational fluids dynamics models at NIST. Additionally, the NRC has implemented the THIEF model into its Fire Dynamics Tools spreadsheets (NUREG-1805, "Fire Dynamics Tools (FDTs) Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program," issued December 2004). The THIEF spreadsheet is a useful tool for inspectors and licensees, as it can quickly determine the likelihood of cable damage given a fire or indicate the need for further analysis.

The NRC has completed a phenomena identification and ranking table study of fire modeling (NUREG/CR 6978, issued November 2008). This effort identified important fire modeling capabilities needed to increase confidence in the results. This study is being used to help define future research priorities in fire modeling.

The NRC is currently working again with EPRI and NIST to develop technical guidance for those who conduct fire modeling analyses of NPPs. This guidance will continue to expand on the effort of NUREG-1824 by providing users with best practices from experts in fire modeling and NPP fire safety.

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METHODS DEVELOPMENT AND STAKEHOLDER INTERACTION FOR FIRE PROBABILISTIC RISK ASSESSMENT AND FIRE HUMAN RELIABILITY ANALYSIS

Background

The results of the Individual Plant Examination of External Events Program and actual fire events indicate that fire can contribute significantly to nuclear power plant risk, depending on design and operational conditions. In particular, failures of fire protection defense in depth (i.e., failure to prevent fires, failure to rapidly suppress fires, or failure to protect plant systems to provide stable, safe shutdown) can lead to risk-significant conditions. Fire probabilistic risk assessment (PRA) provides a structured, integrated approach to evaluate the impact of failures in the fire protection defense-in-depth strategy on safety. Human reliability analysis (HRA) is the tool used to assess the implications of various aspects of human performance for risk.

In 1995, the NRC adopted a policy statement on PRA with the intent to increase the use of PRA technology in all regulatory matters to the extent supported by the state of the art in PRA methods and data. Through the use of PRA, safety is enhanced by gaining insights that supplement the NRC's traditional approach of maintaining defense in depth and safety margin, as well as the staff's overall engineering judgment. In 2004, the NRC amended its fire protection requirements to allow existing reactor licensees to voluntarily adopt the risk-informed, performance-based rule in Title 10 of the Code of Federal Regulations (10 CFR) 50.48c, which endorses NFPA 805, "Performance Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," as an alternative to the existing prescriptive fire protection requirements. To realize the full benefits of the transition to the risk-informed, performance-based standard, plants will need to conduct a fire PRA, which should include quantitative HRA for postfire mitigative human actions modeled in a fire PRA.

Approach

In 2001, the Electric Power Research Institute (EPRI) and RES embarked on a cooperative project to improve the state of the art in fire risk assessment to support this new risk-informed environment in fire protection. This project produced a consensus document, NUREG/CR-6850 (EPRI 1011989), entitled "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," issued September 2005, which addresses fire risk for at-power operations.

This report provides high-level qualitative guidance and quantitative screening guidance for conducting a fire HRA. However, this document does not provide a detailed quantitative methodology to develop best estimate human error probabilities. To fulfill this need, NRC-RES is collaborating with EPRI to develop a methodology and associated guidance for performing quantitative HRA for postfire mitigative human actions modeled in a fire PRA.

The NRC and EPRI jointly conducted well-attended general fire PRA workshops based on NUREG/CR 6850 in both 2005 and 2006. They offered detailed training in 2007–2009 and will offer additional detailed training in 2009. Pilot plants in transition to the rule in 10 CFR 50.48c are relying on NUREG/CR-6850 in upgrading their fire PRA, while the NRC uses the document to support reviews. RES and EPRI are currently working to resolve fire PRA issues related to NUREG/CR-6850 implementation, beyond those they have already addressed, in the NFPA 805 frequently-asked-questions (FAQ) program. Overall, this joint work is producing a significant convergence of technical approaches.

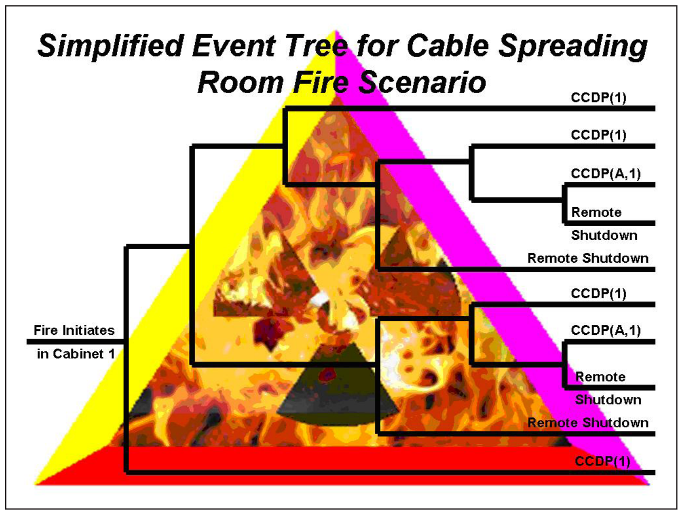

Figure 2.3 Simplified fire event tree representing different sets of fire damage and plant response. The conditional core damage probability (CCDP) represents failure of only the cabinet in which the fire initiates, the additional fire-induced failure of train A, and the fire-induced failure of both trains A and B leading to remote shutdown operations.

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DESIREE-FIRE—DIRECT CURRENT ELECTRICAL SHORTING IN RESPONSE TO EXPOSURE-FIRE

Background

The Individual Plant Examination of External Events Program results and actual fire events indicate that fire can contribute significantly to nuclear power plant risk. The question of how to determine risk resulting from fire damage to electrical cables in nuclear power plants has been of concern since the Browns Ferry fire in 1975. In earlier years, it was generally believed that any system that depended on electric cables passing through a compartment damaged by fire would be unavailable to perform its intended safety function. The Browns Ferry fire and recent testing have prompted wider realization that short circuits involving an energized conductor can pose considerably greater risk by creating "hot shorts" which can cause systems to malfunction (e.g., by inadvertently repositioning motor-operated valves and starting or stopping plant equipment). Plant safety analyses should consider this risk.

A consensus regarding the likelihood of hot shorts given fire-damaged cables did not exist in the late 1990s. The Nuclear Energy Institute and the Electric Power Research Institute conducted a testing program in 2001, and the NRC conducted its CAROLFIRE program in 2006. Volumes 1 through 3 of NUREG/CR-6931, issued April 2008, document the CAROLFIRE results. These programs produced a wealth of data and knowledge related to fire induced circuit failures of alternating current (ac) circuits. However, none of the previous testing explicitly explored the fire-induced circuit failure phenomena for direct current (dc). Currently operating plants and the proposed new reactor designs both use dc circuits to operate many safety related systems.

Some recent testing performed by industry has indicated that the results for ac circuits may not be fully representative of the results of fire-induced damage to dc circuits. Because of the differences in the operating voltages and circuit design between ac and dc, the previous data gathered for ac circuits are unlikely to apply to dc circuits. The DESIREE-FIRE testing of risk-significant dc circuits will allow the fire protection community to better understand dc circuit failure characteristics.

Approach

The NRC staff decided to perform fire testing of dc circuits using configurations that are representative of safety-significant circuits and components used in nuclear power plants to gain a better understanding of the probability of spurious actuations and the duration of those actuations in dc circuits.

The DESIREE-FIRE program will use intermediate- and small-scale fire testing to evaluate the response of dc circuits to fire conditions. The program will test several different circuits, including the following:

- dc motor starters
- pilot solenoid-operated valve coils
- medium-voltage circuit breaker control
- instrumentation circuit

The DESIREE-FIRE project is unique to the Fire Research Branch, because of the collaborative research agreement with EPRI. This agreement has provided various components and cabling to the DESIREE-FIRE testing program at little to no cost to the NRC. It has also provided expert advice on the various aspects of the dc power system and circuit design.

Figure 2.4 Electrical cable fire damage

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CABLE HEAT RELEASE RATE, IGNITION, AND FLAME SPREAD

Background

The results of the Individual Plant Examination of External Events Program and actual fire events show that fire can contribute significantly to nuclear power plant (NPP) risk, depending on design and operational conditions. Electrical cables have been responsible for a number of fires in NPPs over the years. In 1975, a serious fire involving electrical cables occurred at the Browns Ferry Nuclear Power Plant operated by the Tennessee Valley Authority (see NUREG-0050, "Recommendations Related to Browns Ferry Fire," issued February 1976). Electrical cables perform many functions in NPPs. Power cables supply electricity to equipment, control cables connect plant equipment to remote initiating devices, and instrumentation cables transmit low-voltage signals between input devices and readout display panels. NPPs typically contain hundreds of miles of electrical cables. The in situ fire fuel load in an NPP is clearly dominated by electrical cable insulating materials. In a postulated fire scenario, these materials can be an ignition source, intervening combustible, and/or part of the target set.

Cables are made up of a variety of thermoplastic and thermoset materials. The burning behavior of cables in a fire depends on several factors, including their constituent materials and construction, as well as their location and installation geometry. Burning cables can propagate flames from one area to another, or they can add to the amount of fuel available for combustion. Burning cables also produce smoke containing toxic and corrosive gases. The lower the heat exposure required to ignite the electrical cables, the greater the fire hazard in terms of ignition and flame spread. Electrical cables exposed to fire can lose physical integrity (because of melting of the insulation) and insulation resistance, leading to electrical breakdown or short-circuiting or spread of fire to other cables or combustibles.

The experimental evidence and analytical tools available to calculate the effects of cable tray fires are relatively limited when compared to the vast number of possible fire scenarios. Many of the large-scale fire tests conducted with cables are qualification tests in which the materials are tested in a realistic configuration and qualitatively ranked on a comparative basis. This type of test typically does not address the details of fire growth and spread and does not provide any useful data for model calculations.

Approach

The CHRISTI-FIRE (Cable Heat Release, Ignition, and Spread in Tray Installations during FIRE) experimental program is an effort to quantify the mass and energy released from burning electrical cables. The objective of this program is to perform fire tests on grouped electrical cables to gain a better understanding of the fire hazard characteristics, including heat release rate and flame spread. This type of quantitative information will contribute to the development of more realistic models of cable fires for use in fire probabilistic risk assessment (PRA) analyses, such as those performed using the methods of NUREG/CR-6850, "Fire PRA Methodology for Nuclear Power Facilities," issued September 2005, in applications of National Fire Protection Association 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants."

The experimental program has two main thrusts—bench-scale measurements of small samples of burning cables and full-scale measurements of the heat release and fire spread rates of cables burning within typical ladder-type trays. CHRISTI-FIRE involves mainly horizontal tray configurations. Fires do not spread as rapidly over horizontal trays, but the rates can vary greatly depend on the proximity of a given tray to other trays or surrounding walls or ceiling.

Figure 2.5 Cable tray fire test

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CHAPTER 3: HUMAN FACTORS AND RELIABILITY RESEARCH

Dr. Brian Sheron discusses with Dr. Klein (former Chairman), Commissioner Lyons, and R. William Borchardt, Executive Director for Operations, a Davis-Besse nuclear reactor head model that shows the football-sized hole discovered in the 6-inch carbon steel head from boric acid corrosion in 2002. Although there were many indications that this corrosion was taking place, plant personnel missed or ignored them, and the NRC failed to fully follow up. The model serves as a reminder of the continuous need to avoid complacency and recognize the vulnerability of technology to degradation.

HUMAN EVENT REPOSITORY AND ANALYSIS

Background

Consistent with the Commission's policy statements on the use of probabilistic risk assessment (PRA) and to promote an appropriate PRA quality for risk-informed regulatory decisionmaking, the NRC has ongoing activities to improve the quality of human reliability analysis (HRA). The adequacy of data available for HRA is essential to the credibility and consistency of human error probability estimates. To address this need, RES is developing, with the support of the Idaho National Laboratory, the Human Event Repository and Analysis (HERA) system, which supports both human factors and HRA.

The development and use of HERA is a key component of the NRC's efforts to improve HRA and has been recommended by the Advisory Committee on Reactor Safeguards (ACRS) in its response to the commission's staff requirements memorandum on HRA models (SRM-M061020). The ACRS, in its letter to the Commission (ADAMS Accession No. ML071140297), states: "Additional evidence should be collected from operating experience, especially the Augmented Inspection Team reports on past incidents. The staff is already evaluating the operating experience in the Human Event Repository and Analysis System (NUREG/CR-6903). These sources of information should be used to enhance the insights gained from the Empirical Study." This activity is included as an item in the "Action Plan—Stabilizing the PRA Quality Expectation and Requirements," an attachment to SECY 04-0118, "Plan for the Implementation of the Commission's Phased Approach to Probabilistic Risk Assessment Quality," dated July 13, 2004, and SECY 07-0042, a status update of the plan, dated March 7, 2007. NRC, national, and international experts are collaborating in this activity.

Approach

The approach to addressing HRA data needs via HERA is to make available empirical and experimental human performance data with emphasis on commercial nuclear power plant operations in a content and format suitable to HRA.

The development of HERA has three aspects: (1) develop a data taxonomy for collecting information on human performance during abnormal conditions suitable for HRA, (2) populate HERA with information from real and simulated events, and (3) develop mathematical structures and tools enabling the use of HERA to inform HRA.

The NRC staff published NUREG/CR-6903, Volume 1, "Human Event Repository and Analysis (HERA) System, Overview," in July 2006 (ML062700593). This volume discusses the need for a systematic collection of human performance data on the basis of current regulatory HRA needs, describes the taxonomy and structure of the data in HERA, and presents examples of information extraction and coding from nuclear power plant operational and simulator experience.

The NRC staff also developed Volume 2 of NUREG/CR-6903, "A Human Event Repository and Analysis (HERA): The HERA Coding Manual and Quality Assurance," issued November 2007 (ML073130034). Volume 2 describes an effective process for event coding and quality control, as well as a refined data taxonomy. In parallel, HERA is being populated with human events from selected operational occurrences and from simulator experiments. In addition, software has been developed to store and analyze human performance insights.

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PILOT TESTING OF HUMAN RELIABILITY ANALYSIS-INFORMED TRAINING AND JOB AID FOR NRC STAFF INVOLVED WITH MEDICAL APPLICATIONS OF BYPRODUCT MATERIALS

Background

In 2003, the Office of Nuclear Material Safety and Safeguards (NMSS) provided RES with a user need request to develop a human reliability analysis (HRA) capability specific to materials and waste applications (NMSS-2003-003). In this user need memo, NMSS requested two phases of work. RES staff completed both phases in December 2008.

The Phase 1 work consisted of feasibility studies for developing NMSS capability in HRA. The feasibility study for materials applications addressed both medical and industrial applications.

The Phase 2 work focused on the recommendations from the feasibility study, namely, the development of job aids (e.g., HRA-informed decisionmaking aids) and associated training for NRC staff on HRA informed issues on human performance in medical applications.

The final products of the Phase 2 work, a prototype HRA-informed job aid (i.e., a database of risk relevant human performance issues and historical errors, related to treatment steps) and associated training materials for medical applications (gamma-knife based), were presented to staff of the Office of Federal and State Materials and Environmental Management Programs and delivered to the NRC in December 2008.

The staff is planning follow-on work for the pilot of the HRA-informed job aid and training materials.

Approach

The overall objective is to develop an HRA-informed job aid and associated training for NRC staff involved with medical applications of byproduct materials. While prototypes of the HRA-informed job aid and training materials have been developed, methods for using these tools for specific NRC tasks (e.g., inspections, license reviews) have not been developed. Consequently, this pilot testing phase of development requires interaction with NRC staff from the regions, as well as the continued involvement of NRC Headquarters staff.

RES is currently planning for pilot testing of both products in NRC Region I.

The expected tasks for the pilot testing of the HRA-informed job aid and associated training include:

- initial interactions with NRC Region I staff
- onsite HRA-informed training
- onsite demonstration of HRA-informed job aid
- selection of candidates for trial use of HRA informed job aid
- trial use of HRA-informed job aid
- feedback on trial use
- updates to HRA-informed job aid and associated training

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HUMAN PERFORMANCE FOR ADVANCED CONTROL ROOM DESIGN

Background

Plans to begin constructing new plants within the next several years indicate the renewed interest in nuclear energy. The new generation of plants will differ from the existing fleet in several important ways, including the design of the reactors, the instrumentation and control (I&C), and the human-system interface (HSI). Taken together, these advances in technology will lead to concepts of operation that are different from those found in currently operating nuclear power plants (NPPs). The potential benefits of the new NPP technologies should result in more efficient operations and maintenance. However, if the technologies are poorly designed and implemented, there is a potential to reduce human reliability, increase errors, and negatively impact human performance, which could result in a detrimental effect on safety. To address these concerns, the NRC sponsored a study to identify human performance research that may be needed to support the review of licensees' implementation of new technology in new and advanced NPPs.

Approach

To identify the research issues, the staff evaluated current industry trends and developments in the areas of reactor technology, I&C technology, HSI integration technology, and human factors engineering (HFE) methods and tools. These four research issues were then organized into seven HFE topic areas: Role of Personnel and Automation, Staffing and Training, Normal Operations Management, Disturbance and Emergency Management, Maintenance and Change Management, Plant Design and Construction, and HFE Methods and Tools. Next, a panel of independent subject matter experts, representing various disciplines (e.g., HFE, I&C) and backgrounds (e.g., vendors, utilities, research organizations), prioritized the issues. This effort distributed a total of 64 issues among four categories, with 20 research issues placed in the top priority category. NUREG/CR-6947, "Human Factors Considerations with Respect to Emerging Technology in Nuclear Power Plants," issued October 2008, documents the results of the study. The report contains a summary of the high-priority topic areas, the research issues in each topic area, the priorities for each issue, and a human performance rationale that explains the relevance of each research issue. RES is using the findings from this study to develop a long-term research plan for addressing human performance within these technology areas, for the purpose of establishing a technical basis for the generation of regulatory review guidance.

Of the 20 research projects identified as having a top priority, three are currently underway, and an additional two projects are scheduled to begin this year. The following briefly describes these projects.

HUMAN-SYSTEM INTERFACES TO AUTOMATIC SYSTEMS

In light of the increasing use and importance of automation in new and future plants, guidance is needed to enable the NRC staff to conduct safety reviews of the HFE aspects of modern automation. This research study is aimed at developing guidance for reviewing the operator's interface with automation. The guidance will address automation displays, interaction and control, automation modes, automation levels, adaptive automation, error tolerance and failure management, and HSI integration.

ADVANCES IN HUMAN FACTORS ENGINEERING METHODS AND TOOLS

The methods and tools used to design, analyze and evaluate the HFE aspects of NPPs are rapidly changing. This study identified the current trends in the use of HFE methodologies and tools, their applicability to NPP design and evaluation, and their role in safety reviews conducted by the NRC. The study found seven categories of methods and tools for which additional review guidance may be needed: application of human performance models, use of virtual environments and visualizations, analysis of cognitive tasks, rapid development engineering, integration of HFE methods and tools, computer-aided design, and computer applications for performing traditional analyses.

EFFECTS OF DEGRADED I&C ON HUMAN PERFORMANCE

The I&C system is the primary means by which personnel obtain information about the plant. Degradation of the I&C system will have a significant impact on the operator's ability to monitor the plant, detect disturbances, assess the plant status, and respond to unfolding conditions. Failure or degradation of I&C systems can pose additional challenges by causing abnormal operating conditions as the result of erroneous automatic action. This study will address how degraded or failed I&C systems affect operator situational awareness and performance and, consequently, impact plant operations and safety.

IMPACT OF HUMAN FACTORS ISSUES IN COMPUTERIZED PROCEDURES ON HUMAN PERFORMANCE UNDER I&C DEGRADATION

Computerized procedures (CPs) will be replacing traditional paper-based procedures (PBPs) in new and advanced NPPs. Experience with operating CP systems and findings from research studies show that CP systems offer performance benefits. However, various challenges have been identified with CP systems that may have an impact on plant safety. These challenges include a serial access to HSIs through "keyholes," difficulties with crew performance and communication, and problems in transitioning to PBPs in the event of CP system malfunction. While these issues affect operators' performance in general, the effects can be more harmful under degraded I&C conditions. This study will assess the impacts of human factors issues in CPs on human performance under degraded I&C conditions and will identify and evaluate new guidance that can be used to conduct regulatory review activities for the use of CPs in NPPs.

UPDATE EXISTING HUMAN FACTORS ENGINEERING REGULATORY GUIDANCE

The NRC staff reviews the HFE aspects of NPPs in accordance with the guidance presented in NUREG 0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants." Detailed design review procedures for the HFE programs of applicants for construction permits, operating licenses, standard design certifications, combined operating licenses, and license amendments appear in NUREG-0711, Revision 2, "Human Factors Engineering Program Review Model," last updated in 2004. As part of the review process, the staff evaluates the interfaces between plant personnel and plant systems and components for conformance with the guidance in NUREG-0700, Revision 2, "Human-System Interface Design Review Guidelines," last updated in 2002. This study will update these two NUREGs with HFE criteria developed from the most recent and best available technical bases. The availability of up-to-date HFE review guidance will help to ensure that the NRC staff has the latest knowledge, information, and tools to safely and efficiently perform its regulatory tasks.

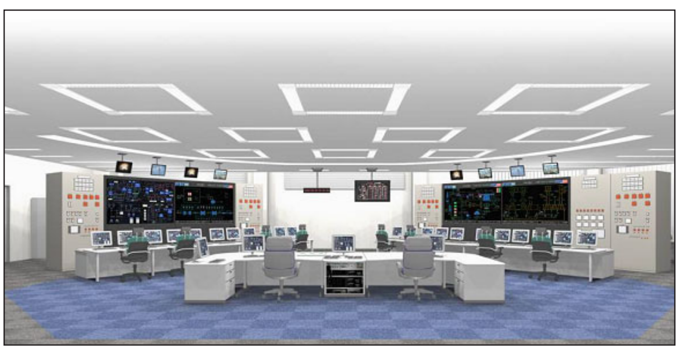

Figure 3.1 Candidate Generation III Nuclear Power Plant Control Room

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FITNESS-FOR-DUTY PROGRAMS 10 CFR PART 26

Background

To ensure the safety and security of nuclear facilities, the NRC has developed regulations to standardize and ensure effective implementation of fitness-for-duty (FFD) programs that apply to personnel who engage in certain safety- and security-related activities. For example, certain personnel at commercial nuclear power plants who have unescorted access to the plant's protected areas and those who transport strategic special nuclear materials must be subject to an FFD program. The NRC requires FFD programs to provide reasonable assurance that nuclear facility personnel are trustworthy and will perform their tasks reliably.

Title 10 of the Code of Federal Regulations (10 CFR) Part 26, "Fitness for Duty Programs," describes the scientific and technical requirements for FFD programs that address illegal drug use, alcohol abuse, misuse of legal drugs, impairment from fatigue, and any other mental or physical conditions that could impair job performance. At the time Part 26 was first published in the Federal Register (54 FR 24468, June 7, 1989) and subsequently, the Commission directed the NRC staff to continue to analyze FFD programs, assess the effectiveness and efficiency of the rule, and recommend appropriate improvements or changes.

Most recently, the NRC, with extensive stakeholder input, published an amended, reorganized, and updated rule. The amended Part 26 was published in the Federal Register on March 31, 2008, and is organized into 12 subparts that group together related requirements. The NRC permitted licensees and other entities to defer implementation of most of the rule's requirements until March 31, 2009, and granted an additional 6 months to implement the rule's new fatigue management requirements. The fatigue management requirements must be implemented by October 1, 2009.

Approach

RES participates in a multidisciplinary team of NRC staff that is supporting a myriad of agency initiatives and efforts to facilitate educate about the rule and its implementation.

Fatigue Regulatory Guide

RES worked closely with other NRC staff and stakeholders to publish guidance for implementing the fatigue management requirements of 10 CFR Part 26. Specific requirements for nuclear power plant licensees to manage worker fatigue are a new addition to Part 26. The NRC published Regulatory Guide 5.73, "Fatigue Management for Nuclear Power Plant Personnel," in March 2009.

TRAINING DEVELOPMENT

To ensure that implementation efforts among the regions and various offices are coordinated and consistent, RES staff and contractors have developed training materials for inspectors and other NRC staff involved in implementation of Part 26. The training has been developed and pilot tested and will be supplemented with computer-based training specifically focused on the fatigue management requirements.

FITNESS-FOR-DUTY WEB SITE UPDATE

Transparency is an important NRC goal. As such, the NRC staff maintains a public Web site to provide one location for stakeholders to access information and submit questions regarding the rule and any implementation concerns. The Web site includes the history of the 10 CFR Part 26 rulemaking, frequently asked questions regarding Part 26 and its implementation, FFD program reports from licensees, and related documents and resources.

INSPECTION PROCEDURES

RES is supporting other NRC offices in developing inspection procedures that will be used to evaluate the effectiveness of FFD programs and verify licensee compliance with the rule's requirements.

TECHNICAL BASES FOR ALTERNATE SPECIMENS AND FATIGUE TECHNOLOGIES

The science and technologies for ensuring personnel's fitness for duty continue to advance. Consistent with the Commission's direction to continue assessing the effectiveness and efficiency of FFD programs, RES is identifying scientific and technological advances that may enhance FFD programs. For example, 10 CFR Part 26 currently requires the use of urine, breath, and saliva testing for drugs and alcohol. However, new drug-testing technologies are being developed that rely on alternative specimens, including hair and sweat. New methods to manage fatigue in the workplace and technologies for assessing fatigue and other types of possible impairment are also of interest. Finally, other readiness-to-perform technologies are under evaluation, as these tests have implications for effective job and task performance.

FUTURE UPDATES TO 10 CFR PART 26

The Commission directed the NRC staff to initiate a new Part 26 rulemaking after publication of the March 31, 2008, amended and revised rule. The Commission requested that the NRC staff review specific elements of the rule related to the technical basis and evaluate the inclusion of quality control, quality verification, and/or quality assurance licensee personnel to the fatigue provisions of Part 26. The RES staff is continuing to provide its technical expertise to staff engaged in the new rulemaking.

For More Information
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QUALITATIVE HUMAN RELIABILITY ANALYSIS FOR SPENT FUEL HANDLING

Background

In 2003, the Office of Nuclear Material Safety and Safeguards (NMSS) provided RES with a user need request (NMSS-2003-003) to develop a human reliability analysis (HRA) capability specifically for materials and waste applications. In this memo, NMSS requested two phases of work, the first of which is complete.

Phase 1 work consisted of feasibility studies for developing NMSS capability in HRA. The feasibility study for waste applications (performed by NRC staff) addressed high-level waste, spent fuel storage, fuel cycle, and decommissioning applications. In summary, this study identified the following needs for potential NMSS-specific HRA development that were common to more than one waste application:

1. development of HRA methods specific to NMSS needs
2. guidance for evaluating the effectiveness of administrative controls
3. guidance on "good practices" for implementing HRA
4. guidance for reviewing HRAs
5. assistance in incident significance assessments

Initial Phase 2 work on this project began with an investigation of items 1 and 3 above.

Additionally, NMSS and RES identified new priorities, resulting in project efforts focused on the development of HRA insights for spent fuel handling. Such activities include both misloading and drops during spent fuel handling activities.

Approach

The first step in developing HRA capability for NMSS, as described above, was to develop a qualitative understanding of the important human performance issues in spent fuel handling that need to be addressed by HRA.

To this end, this project has completed the following efforts:

- identification and review of literature relevant to understanding human performance in spent fuel handling
- interviews with experts in spent fuel handling
- evaluation and use of relevant literature and interviews with experts to perform qualitative HRA tasks for spent fuel handling

The results of this work consist of a July 2006 Sandia National Laboratories letter report, "Final Scoping: Human Reliability Analysis for Spent Fuel Handling" describing potential vulnerabilities and possible scenarios that could lead to misloads and cask drops.

Currently, work is proceeding to develop additional HRA-informed insights on cask drops. Such work is expected to provide useful input to future NRC inspections and reviews. The following is the current schedule for deliverables for this effort:

- preliminary report—completed May 2008
- expanded report—completed April 2009
- presentation to NMSS staff—completed May 2009
- draft NUREG/CR—January 2010
- preparation of final NUREG/CR—September 2010

In addition, RES staff plans to continue its interactions with NMSS as these deliverables are completed.

For More Information

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HUMAN RELIABILITY ANALYSIS METHOD BENCHMARKING USING SIMULATOR DATA

Background

Regulatory Guide (RG) 1.200 describes an acceptable approach for determining the technical adequacy of probabilistic risk assessment (PRA) results for risk-informed activities. However, RG 1.200 (including the PRA standards that it reflects and endorses) addresses what to do but not how to do it. Consequently, several approaches to addressing certain analytical elements may meet the intent of RG 1.200 and its associated standards. However, these approaches may make distinct assumptions and approximations and, therefore, may yield different results. This variability in results is particularly true for human reliability analysis (HRA), for which many methods are available to model mitigative human actions in PRAs.

Given the differences in the scope of the methods and their underlying models, there is a need to assess and ultimately to validate the methods. The NRC staff is addressing this issue by developing lower level guidance documents to support the implementation of RG 1.200. In particular, the NRC staff is participating in and supporting the international effort referred to as the International HRA Empirical Study.

Objective

The objective of this study is to benchmark HRA models by comparing results produced by HRA analysts to experimental results produced by collecting crew performance data at a simulator.

Approach

The International HRA Empirical Study is a multi-method and multi-team study for which the Halden Reactor Project provides facilities, crews, and expertise to perform and analyze simulator crew performance. The study involves a variety of experts with different roles. Licensed operator crews respond to a series of scenarios in Halden's simulator facility; Halden experimental staff design, collect, and analyze crew performance data; HRA teams apply an HRA method to predict the human failure events defined for the study; and a group of independent experts, called the Assessment and Evaluation Group, have the overall responsibility for the study, including its design and implementation process, the comparison of analytical outcomes to simulator outcomes, the evaluation of HRA methods on the basis of the comparison, and the communication and documentation of the results. The NRC, Halden, Swiss Federal Nuclear Inspectorate, and Electric Power Research Institute support this group.

The pilot phase of this project has been completed and documented in draft NUREG/IA-0216/HWR-844, "International HRA Empirical Study Description of Overall Approach and First Pilot Results from Comparing HRA Methods to Simulator Data,," expected to be issued in November 2009. The study has also generated many scientific papers, presented at the annual Institute of Electrical and Electronics Engineers Conference on Human Factors in August 2007, the American Nuclear Society International Probabilistic Safety Conference 2008 (PSA8) in September 2008, and the Ninth International Conference on Probabilistic Safety Assessment and Management in May 2009.

The "actual" phase of an experimental work typically consists of several iterations of experiments and analysis. However, the scope of this study is limited to the use of simulator data collected from two (one easy and one complicated) steam generator tube rupture scenarios and two loss-of-feedwater scenarios performed by 14 crews at Halden in fall 2006. The collection and analysis of the data from these simulator runs and their comparison with HRA results are currently being completed. The NRC staff expects to have documented the results related to steam generator tube rupture in a draft report in September 2009 and the results related to loss of feedwater in April 2010. These publications will go through peer and public review and will be completed in 2010.

For More Information
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SAFETY CULTURE

Background

The culture of an organization affects the performance of the people within the organization. Weaknesses in an organization's safety culture may set the stage for equipment failures and human errors that can have an adverse impact on safe performance.

Goal of Safety Culture Activities

The initial goal of the 2006 safety culture initiative at the NRC was to enhance the Reactor Oversight Process (ROP) to more fully consider safety culture in the NRC's assessments of inspection findings and overall nuclear power plant performance. More recently, the Commission directed the NRC staff to (1) consider the need for an agencywide safety culture policy statement that would apply to all entities regulated by the NRC and (2) recommend whether and how to better integrate security culture considerations into the NRC's safety and security oversight activities.

RES is providing technical expertise related to human and organizational performance to support the agency's safety culture activities. The RES staff participates in the Safety Culture Working Group, the Safety Culture Policy Statement Task Force, and the Safety Culture Policy Statement Steering Committee.

Industry Safety Culture Assessment Initiative

Concurrent with the NRC staff's activities, the nuclear power industry, led by the Nuclear Energy Institute (NEI), is developing a standardized safety culture assessment methodology and performance indicators. NEI has indicated that nuclear power plant licensees will use the assessment methodology for biennial self-assessments and, with modifications, to reply to NRC requests for independent or third-party safety culture assessments under the ROP. The NRC will use the performance indicators to provide ongoing monitoring of safety culture trends. RES staff will assist the Office of Nuclear Reactor Regulation in evaluating the industry's new approach.

For More Information

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HUMAN RELIABILITY ANALYSIS MODEL DIFFERENCES

Background

RES is supporting the Advisory Committee on Reactor Safeguards (ACRS) in addressing the Staff Requirements Memorandum (SRM-M061020), dated November 8, 2006, concerning differences in human reliability analysis (HRA) models. In this SRM, the Commission directed ACRS to "work with the staff and other stakeholders to evaluate different human reliability models in an effort to propose a single model for the agency to use or guidance on which model(s) should be used in specific circumstances." ACRS is addressing this issue through collaborative work with the Electric Power Research Institute (EPRI), initiated under the RES memorandum of understanding with EPRI on probabilistic risk assessment (PRA).

Approach

The focus of this collaborative work is to identify areas where HRA has a significant impact on regulatory decisionmaking. The following steps outline the approach:

1. Determine current and future regulatory applications for which HRA results may play a significant role in the decision.

2. For those applications in which HRA plays an important role, identify which HRA methods are, or are planned to be, used, and evaluate the adequacy and suitability of these methods to support the respective applications.

3. On the basis of the results from steps 1 and 2, as well as the results and insights from the International HRA Empirical Study, determine whether the agency should use a single model, or whether specific methods should be used for specific applications. In either case, improve the method(s) as needed.

4. Develop guidance (and associated training materials) for how (and which) model(s) should be used in specific circumstances.

The collaborative efforts will contribute to the NRC's goals of efficiency and effectiveness. The NRC's cost represents only a fraction of the actual costs for both the international empirical study and the collaborative work with EPRI to address SRM M061020 on HRA model differences. In addition, through this collaboration, the NRC has the advantage of extensive domestic and internationally recognized PRA and HRA expertise from academia and practitioners.

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CHAPTER 4: MATERIALS PERFORMANCE RESEARCH

Reactor Pressure Vessel Integrity

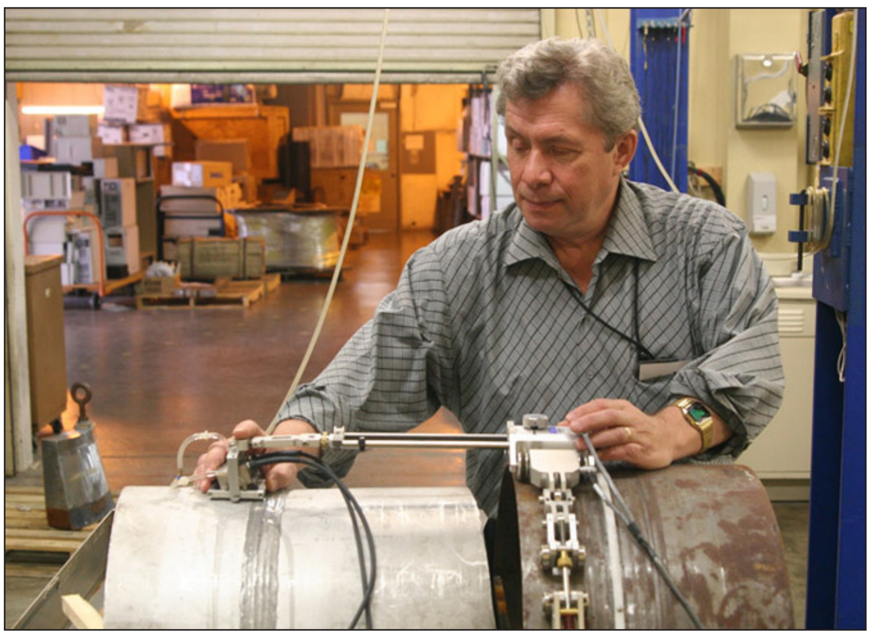
Steam Generator Tube Integrity

Nondestructive Examination

Aging Management Research ("Life Beyond 60")

Extremely Low Probability of Rupture

Proactive Management of Materials Degradation



Dr. Iouri Prokofiev scans a primary loop piping weld segment using a state-of-the-art nondestructive ultrasonic probe in order to determine its effectiveness in detection and sizing of flaws contained in and around the weld. The Office of Nuclear Regulatory Research evaluates nondestructive examination (NDE) state-of-the-art technology and the current state of practice in NDE so that the NRC can make better-informed decisions regarding NDE requirements.

REACTOR PRESSURE VESSEL INTEGRITY

Background

One key to the safe operation of a nuclear power plant is maintaining the structural integrity of the reactor pressure vessel (RPV) during both routine operations (i.e., heatup, cooldown, and hydro test) and during postulated accident scenarios (e.g., pressurized thermal shock [PTS]). To do this, plants need procedures to estimate and compare the structural driving force for failure to the resistance of the structure to this driving force (and the effect of radiation on this resistance). Current statutory procedures for making these estimates appear in Title 10 of the Code of Federal Regulations (10 CFR) Section 50.61, "Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events" (the PTS rule); 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements"; and Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials." These procedures generally depend on empirically based engineering methods. While these methods have proven to be effective components of safety-focused regulations, they are known to incorporate large implicit conservatisms adopted to address deficiencies in the state of knowledge at the time of their promulgation. When coupled with the deterministic basis of current regulations, these conservatisms create artificial and unnecessary impediments to continued operation and license renewal. Moreover, the largely empirical basis of these regulations makes their confident and accurate application to new- and advanced-design reactors problematic.

Objectives

1. Integrate the advances in the state of knowledge, empirical data, and computational power that have occurred in the 20+ years since the adoption of the current regulatory requirements to develop the technical bases for state-of-the-science and risk-informed revisions to the four statutory procedures that regulate the structural integrity of the currently operational RPV fleet.

2. Use the advances in the state of knowledge and empirical data that have accumulated over 20+ years of structural materials research by the nuclear community to develop, validate, and refine physically based predictive models of material deformation and failure behavior.

3. Apply the insights into probabilistic structural integrity assessment gained from Objective 1 and the predictive material models developed in Objective 2 to develop and validate a modular probabilistic computer code that can be used to evaluate the structural integrity assessment of any pressurized structure in a nuclear power plant.

Approach

RES has recently completed a multiyear study to develop the technical basis for a risk-informed revision to 10 CFR 50.61 (the PTS rule). RES performed this study in cooperation with the Oak Ridge National Laboratory, other national laboratories, Government contractors, and the domestic nuclear industry working under the auspices of the Electric Power Research Institute's Materials Reliability Project. The NRC's Office of Nuclear Reactor Regulation (NRR) has used this technical basis to develop a voluntary alternative to 10 CFR 50.61 that relaxes many of the conservatisms in the current rule without impacting public safety. This voluntary alternative rule is now in rulemaking, which is expected to be completed in 2009.

Also in the coming years, RES will publish and make available for public comment a revised version of Regulatory Guide 1.99—along with its technical basis. This revision is based on over five times the empirical database of the current regulatory guide. Additionally, the physically motivated basis for the embrittlement trend curves contained within the guide is expected to improve the accuracy and reliability of embrittlement predictions of yet-to-be-experienced conditions of radiation exposure.

The insights gained from these activities provide much of the work needed as the technical bases supporting revisions to 10 CFR Part 50, Appendices G and H, both of which are scheduled for revision in 2009–2010.

In the next 5–10 years, RES will conduct two major initiatives to ensure the structural integrity of the pressurized nuclear power plant components in the existing fleet during the period of license extension and in the new reactor fleet:

1. development and validation of a method capable of identifying embrittlement mechanisms in reactor materials before they occur in commercial reactor service

2. development and validation of a modular computational tool to perform probabilistic structural integrity assessments of any passive structural component on the RPV pressure boundary

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STEAM GENERATOR TUBE INTEGRITY

Background

Steam generator (SG) tubes are an integral part of the reactor coolant system (RCS) pressure boundary. They serve as a barrier to isolate the radiological fission products in the primary coolant from the secondary coolant and the environment. The understanding of SG tube degradation phenomena is continually evolving to keep pace with advances in SG designs and materials. To date, many modes of degradation have been observed in SG tubes, including bulk corrosion and wastage, crevice corrosion, pitting, denting, stress-corrosion cracking, and intergranular corrosion attack. Flaws have developed on both the primary and the secondary side of SG tubes. If such flaws go undetected or unmitigated, they can lead to tube rupture and possible radiological release to the environment.

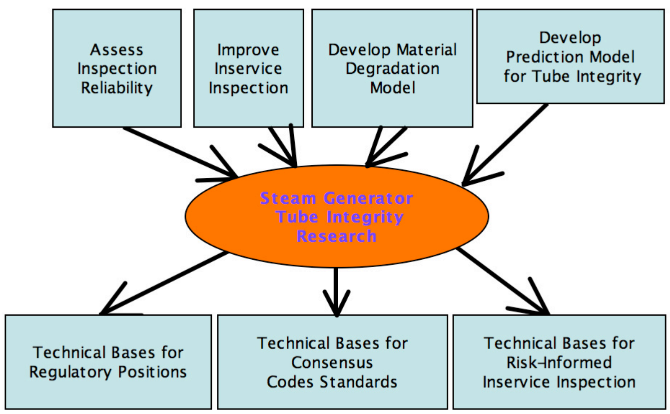

Figure 4.1 Schematic of the elements of steam generator tube integrity research

Overview

The main objective of this research program is to develop a technical basis for SG tube integrity evaluations. This basis is needed to ensure that SG tubes continue to be inspected appropriately, flaw evaluations continue to be conducted correctly, and repair or plugging criteria are implemented appropriately. The areas addressed in this research program include:

- assessment of inspection reliability
- evaluation of inservice inspection technology
- evaluation and experimental validation of tube integrity and integrity prediction modeling
- evaluation and experimental validation of degradation modes

Approach

This ongoing research at Argonne National Laboratory is intended to formulate and document a comprehensive technical basis that will contribute directly to the safety, openness, and effectiveness of the NRC's regulatory actions related to SGs. The key elements of the program are best described by technical area:

ASSESSING INSPECTION RELIABILITY

In this area, research aims to determine the reliability of current inspection methods based on the flaws observed in the field and to evaluate any new and emerging inspection methods as they arise. For example, one task in this area involves the properties and limitations of automated eddy current analysis. The task will utilize the Argonne National Laboratory SG tube flaw mockup facility, which contains a variety of flaws typically found in the field. Results of automated eddy current analysis will be compared to a previous eddy current round-robin test, which studied the reliability of human analysts. In this way, the staff can evaluate the reliability of automated eddy current analysis techniques.

RESEARCH ON INSERVICE INSPECTION TECHNOLOGY

Advanced non-destructive examination (NDE) techniques are used to evaluate SG tube integrity. During inservice inspections, NDE is used to detect and characterize tube flaws. Research in this area aims to evaluate the reliability of NDE techniques for both original and repaired SG tubes. For eddy current inspection, this research will evaluate correlations of signal voltage to flaw morphology and structural integrity. A technical report on this research will present an evaluation of the differences and limitations between various eddy current methods including bobbin coil, rotating pancake, and x-probe.

Research will also continue to examine the leak rate from postulated tube flaws in the region of the tubesheet under postulated severe accident conditions. Experimental tests will be conducted to calibrate and validate the leak models.

RESEARCH ON TUBE INTEGRITY AND PERFORMANCE MODELING

When a flaw is detected in a steam generator tube, its must be assessed to determine if it should remain in service during future operating cycles. Tube integrity is assessed using models that predict leak rates and burst pressures that a particular flaw might exhibit during normal operation or design-basis accidents. While models exist to predict flaw behavior, they require that complex flaw morphology be simplified. One means of simplifying a complex crack is to use a rectangular crack profile. Ongoing research will continue to assess the use of the rectangular crack method for estimating failure pressure and leak rate for complex crack geometries.

RESEARCH ON DEGRADATION MODES

Analytical models exist to predict potential degradation behavior in SG tubes during normal operating conditions. Research in this area seeks to evaluate and experimentally validate those models, as well as to develop further guidance for conducting operational assessments. This will require a better understanding of crevice conditions and stress-corrosion crack initiation, evolution, and growth. The NRC has already conducted considerable research in these areas, which has established a better understanding of the nature of crevice behavior. A topical report will describe the research in this area.

For more information

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NON-DESTRUCTIVE EXAMINATION

Background

Title 10 of the Code of Federal Regulations, Part 50.55a, “Codes and Standards,” requires that licensees inspect structures, systems, and components to ensure that they can perform their safety function and that they meet the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. Research on non-destructive examination (NDE) of light-water reactor (LWR) components and structures provides the technical basis for regulatory decisionmaking related to these requirements. Results from the NDE of these components and structures are also used to assess models developed to predict the effects of material degradation mechanisms and as initial conditions for component-specific fracture mechanics calculations. The Pacific Northwest National Laboratory (PNNL) conducts this work.

Regulatory Needs

Areas of concern addressed by NDE research are the following:

- quantification of the accuracy and reliability of NDE techniques used for inservice inspection (ISI) of LWR systems and components
- support for NRC rulemaking efforts in materials reliability such as the pressurized thermal shock rulemaking
- improvement of the effectiveness and adequacy of the ISI requirements in the ASME Boiler and Pressure Vessel Code
- development of a technical basis for the evaluation of proposed NDE methods and ISI programs for new and advanced reactor licensing

The four specific project areas highlighted below address these regulatory needs.

Approach

EVALUATION OF NDE RELIABILITY AND ISI TECHNIQUES

Research activities include NDE of fabrication flaws and destructive verification. The research objectives are to (1) determine the relationships among preservice inspection methods, inservice degradation (cracking, aging), and ISI practice and results and (2) evaluate the effectiveness, accuracy, and reliability of new techniques expected to be applied by licensees in current, new, and advanced reactors. Certain materials, degradation mechanisms, and locations are difficult to inspect in the current fleet of reactors and will most likely remain challenging in new reactors. The NRC is using fabricated mockups and components removed from reactors, including some canceled plants and some operating reactors, to determine the effectiveness of existing and emerging NDE techniques.

Figure 4.2 Sectioning of reactor vessel head penetrations from Washington Nuclear Power, Unit 1, a canceled plant

Figure 4.3 Non-destructive and destructive examination of salvaged control rod drive mechanism penetrations and J-groove welds from North Anna, Unit 2

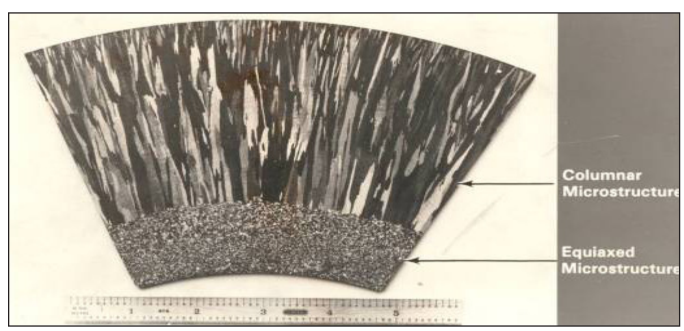

Figure 4.4 Sample illustrating the coarse grain microstructure of centrifugally cast stainless steel

The NRC performs some of this work under collaborative agreements to help defray costs and to gain access to the expertise of other organizations. For example, the ability to detect and characterize primary water stress-corrosion cracking in LWR components is being evaluated under the

NRC-initiated international Program for the Inspection of Nickel-Alloy Components (PINC). This program is linking NDE performance to crack morphology and is developing reliability data on advanced ultrasonics and other new NDE methods. Eight organizations participate in PINC and exchange information and test results from their related research.

The NRC is directing, under its current program at PNNL, research on inspection of coarse-grained austenitic alloys/welds (Figures 4.4 and 4.5). NDE of these components is difficult because of signal attenuation and reflections. In these materials, grain boundaries and other microstructural features appear similar to cracks. Research findings will support appropriate inspection requirements for these components to ensure safety.

ENHANCED SIGNAL PROCESSING AND ANALYSIS SYSTEMS

Modern NDE systems (Figure 4.5) produce voluminous data that must be examined during ISIs. Automated data analysis algorithms reduce the processing time for large amounts of NDE data and thus improve ISI reliability by allowing more extensive inspections. Computer-aided data analysis methods may further improve NDE reliability by reducing or eliminating operator-related errors. Advanced processing techniques also support the use of alternative NDE techniques (e.g., high-resolution eddy current and phased array inspections). The research is focused on determining the accuracy and reliability of advanced NDE for complicated defects in comparison with conventional techniques confirmed by destructive examination.

ADVANCED INSPECTION FOR FABRICATED COMPONENTS

Proposals to increase the use of nonmetallic piping with special welds/joints present a significant challenge to the nuclear industry and the NRC because there is little experience with using these materials in nuclear power plants, and the application of NDE to these joints presents new technical issues. The initial efforts of this research focus on evaluation of relevant inspection techniques deployed in other industries and the review of research results on these techniques. The NRC will use this information in developing licensing requirements for licensee ISI programs for such materials.

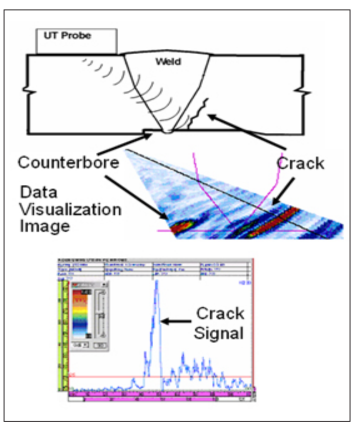

Figure 4.5 Schematic view of flaw detection at the far side of a weld, using a phased array ultrasonic (PA-UT) technique. PA-UT improves flaw detection in coarse-grained metals and welds.

IN-SITU MATERIAL AND STRESS-STATE CHARACTERIZATION

Material characterization using NDE is being developed to produce more accurate in-situ evaluation of the structural integrity of degraded components and radiation damage. This is promising because many NDE methods have been found to be sufficiently sensitive to the presence of residual stress, while they are also sensitive to the microstructural material variations that usually accompany residual stresses and aging. The NRC will perform research to determine the effectiveness of the various techniques as they are developed in the industry.

Summary

The NRC is conducting research to determine the accuracy and reliability of NDE techniques used to identify and characterize flaws in LWR structures and components stemming from aging-related degradation or induced during fabrication or repair processes. International cooperative programs help to defray the cost of this research and allow the NRC to learn from other organizations.

For more information

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AGING MANAGEMENT RESEARCH ("LIFE BEYOND 60")

Background

In carrying out its mission to protect the safety and health of the public and the environment, the NRC evaluates the consequences of aging on commercial nuclear power plants (NPPs). U.S. NPPs were designed with significant engineering margins to ensure that they are able to safely operate for their licensed life. In addition, in accordance with Title 10 of the Code of Federal Regulations (10 CFR) Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," those plants that enter into a period of extended operation are required to have in place aging management programs (AMPs) that mitigate aging degradation effects during the period of continued operations. As a result of these engineering margins and AMPs, the plants' structures, systems, and components (SSCs) inherently afford a measure of protection against the potentially deleterious effects of aging; however, for those plants that may elect to pursue subsequent license renewal periods (beyond the extended 60 year operating period), the staff is investigating whether additional research into aging effects is needed.

Therefore, the NRC staff, in collaboration with the U.S. Department of Energy (DOE), the domestic industry, and international partners, is developing an integrated aging management research plan ("Life Beyond 60"), which will focus on those areas covered by 10 CFR Part 54 that may need additional technical information to provide regulatory assurance of the capabilities of the SSCs and materials to maintain their safety-related functionality in subsequent license renewal periods.

Approach

The NRC and DOE jointly sponsored a public workshop on February 19–21, 2008, in Bethesda, MD, to discuss potential research and development issues related to ensuring that, if licensees elect to pursue subsequent license renewal periods, continued long-term operation can be conducted safely. Panel and public discussions were held on: systems structures and components, materials degradation issues, diagnostics and prognostic technologies, and the future needs (beyond the scope of 10 CFR Part 54) of the nuclear industry to continue long-term operation. Participants included representatives from the NRC, DOE, industry, national laboratories, academia, the International Atomic Energy Agency, the public, and international organizations.

The next steps have included focused discussions with DOE, the domestic industry (e.g., the Nuclear Energy Institute and the Electric Power Research Institute), and potential international partners to begin development of an integrated research plan in order to better leverage resources and prevent unnecessary duplication of efforts. In addition, public outreach continues to ensure appropriate stakeholder participation.

In the longer term (fiscal year 2010 and beyond), the NRC, in collaboration with DOE, the industry, and international partners, will begin work on the priority areas identified in the integrated "Life Beyond 60" research plan to ensure that adequate technical information is available for regulatory decisions if licensees pursue subsequent license renewal terms.

For more information
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EXTREMELY LOW PROBABILITY OF RUPTURE

Background

Section 3.6.3 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (the SRP), describes leak-before-break assessment methodologies that are acceptable to the NRC staff. Specifically, the SRP describes a deterministic assessment procedure that can be used to demonstrate compliance with the requirement in General Design Criterion (GDC) 4, "Environmental and Dynamic Effects Design Bases," of Appendix A, "General Design Criteria," to Title 10 of the Code of Federal Regulations, Part 50, "Domestic Licensing of Production and Utilization Facilities," that primary system pressure piping exhibit an extremely low probability of rupture. SRP Section 3.6.3 does not allow for assessment of piping systems with active degradation mechanisms. However, it is known that primary water stress-corrosion cracking (PWSCC) is occurring in systems that have been granted leak-before-break exemptions to remove pipe-whip restraints and jet impingement shields. Strictly speaking, these systems no longer satisfy the SRP Section 3.6.3 requirements. Thus, in its present form, SRP Section 3.6.3 is impractically restrictive because it does not allow or account for active degradation mechanisms.

Recent activities have been undertaken to demonstrate that public safety is maintained despite violation of the SRP Section 3.6.3 prohibition against active degradation mechanisms. These activities include the following:

- Qualitative arguments have been made that the great majority of observed cracking is of limited extent and of shallow depth. These factors tend to mitigate the risk of piping rupture.
- PWSCC mitigation activities have been implemented (e.g., reduction of mechanical stresses via the application of weld overlays or inlays over the PWSCC-susceptible welds)
- Redefinition and reduction of the design-basis break size from the so-called double-ended guillotine break recognize the extremely low likelihood of this type of break.

While such actions are prudent, timely, and warranted, they fail to resolve clear deficiencies in the SRP Section 3.6.3 assessment paradigm, which reveal the continued need for a new and comprehensive piping system assessment methodology. To address this need, a program has been proposed with the long term goal of developing an assessment tool that can be used to directly demonstrate compliance with the probabilistic acceptance criterion of GDC 4. This tool would properly model the effects of both active degradation mechanisms and the associated mitigation activities. It would be comprehensive with respect to known challenges, vetted for the scientific adequacy of models and inputs, flexible enough to permit analysis of a variety of inservice situations, and adaptable enough to accommodate evolving and improving knowledge.

Approach

As part of the effort to quantitatively ensure the long-term extremely low probability of rupture, as required by GDC 4, RES is beginning to develop a modular-based computer code for the determination of the probability of failure for reactor coolant system (RCS) components. In this effort, RES has the support of national laboratories and commercial contractors and communicates with the domestic nuclear industry working under the auspices of the Electric Power Research Institute. The computer code being developed will be capable of considering all degradation mechanisms that may contribute to low-probability failure events, while properly handling the uncertainty in the failure process. The code must be structured in a modular fashion so that as additional situations arise, additions or modifications can be easily incorporated without code restructuring. The first arm of the modular code to be developed deals directly with primary piping integrity and is coined "xLPR" for extremely low probability of rupture.

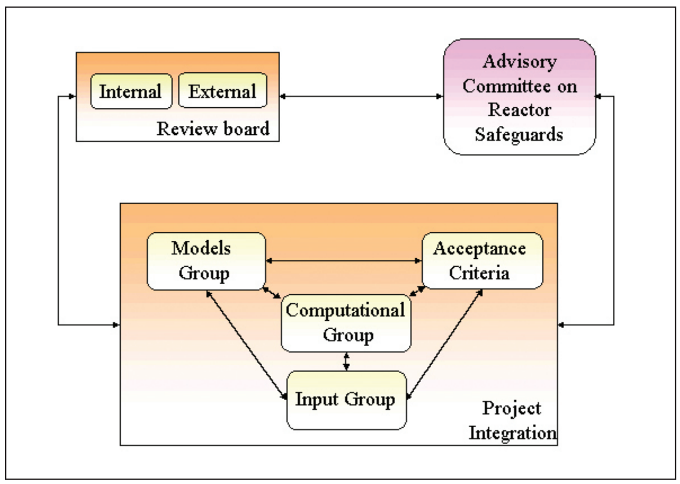

Figure 4.6 xLPR organizational development structure

As shown in Figure 4.6, the vision in RES is to create a group of teams, each with specific long-term and short-term technical objectives. These teams will develop the quantification of extremely low probability of rupture. In the short term, the team effort will focus on a particular problem (i.e., the failure of a pressurizer surge nozzle dissimilar metal weld [See Figure 4.7] with a circumferential crack).

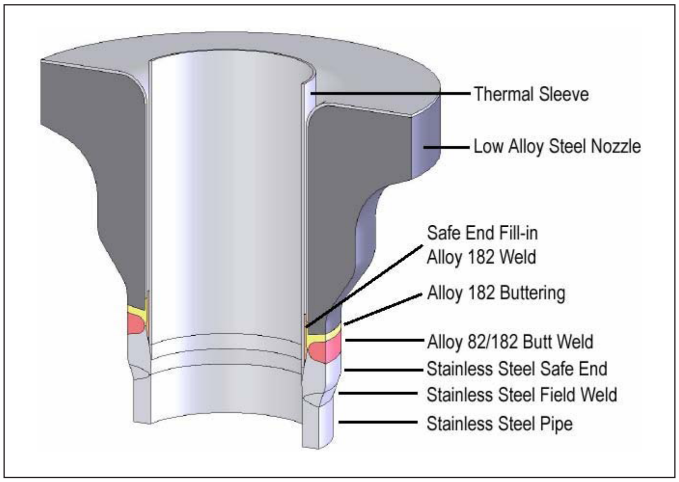

Figure 4.7 Pressurizer surge nozzle

The outcome will be an understanding of the tools available to solve this problem, the limitations associated with these codes, and a firm basis for developing a more robust modular-type code. The short-term deliverable will be a working modular-type code focused on the primary piping integrity issue. In the long term, focus shifts to the more generic problems associated with reactor coolant system integrity. The long-term outcome will be a modular computer code based on verified and validated methodologies for predicting events with a low probability of failure.

Schedule

The planned schedule for the xLPR program is as follows:

- pilot study: surge nozzle problem—April 2010
- short term: xLPR modular code—April 2012
- long term: generic modular code—April 2015

For more information
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PROACTIVE MANAGEMENT OF MATERIALS DEGRADATION

Background

Both the NRC and the nuclear industry have concluded that a proactive approach to materials degradation assessment and management is desirable. The proactive approach will develop a foundation for appropriate actions to (1) minimize the likelihood that materials degradation will adversely impact reactor component integrity and safety and (2) mitigate future safety-significant issues caused by materials degradation.

Overview

The intent of proactive management of materials degradation (PMMD) is to plan and implement timely mitigation actions for components that can potentially degrade before significant challenges to structural integrity and safety arise (Figure 4.8).

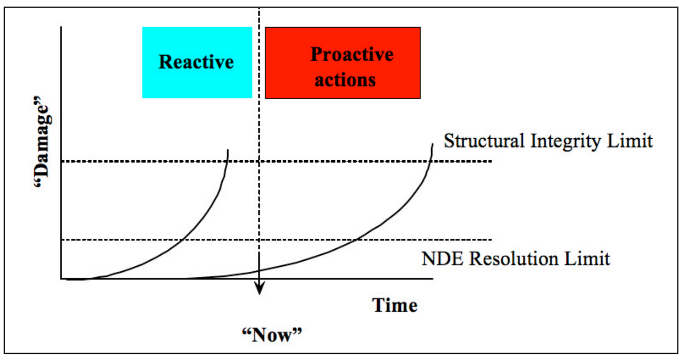

Figure 4.8 Differentiation between reactive and proactive approaches towards managing materials degradation. Degradation process versus time is rarely linear. Proactive actions allow a much larger relative time between the non-destructive examination (NDE) resolution limit and the structural integrity limit.

The NRC's materials degradation assessment identified materials and components where future degradation may occur in specific light-water reactor systems. This work, described in NUREG/CR-6923, "Expert Panel Report on Proactive Materials Degradation Assessment," issued February 2007, is based on a structured phenomena identification and ranking table developed by a world-class panel of experts.

NUREG/CR-6923 is the first publication of an ongoing three-part NRC research program to (1) identify reactor components that could experience future degradation, (2) assess the effectiveness of inservice inspection programs for degradation-susceptible components, and (3) assess the risk significance associated with failure of susceptible components.

The goal of this work is to develop, in collaboration with the domestic nuclear industry and international organizations (including regulatory bodies), a PMMD program. High-priority areas are expected to be addressed initially and include but are not limited to the following:

- materials aging and degradation mechanisms
- mitigation, repair, and replacement
- NDE and continuous monitoring of component status
- risk and prognostics (prediction of remaining life)
- PMMD information tool development (currently derived from NUREG/CR-6923) to include NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," issued July 2001, and other publicly available resources on materials degradations in operating reactors
- PMMD international forum

Approach

A panel of international experts developed metrics to evaluate the potential susceptibility of specific components to different degradation mechanisms and assessed the level of understanding of the effect of these mechanisms for the analyzed components. The panel based its judgments of future behavior on an understanding of the efficacy of the prediction methodologies for the various degradation phenomena, calibrated by the past component failures encountered in light-water reactors worldwide. The evaluation also considered the successes and limitations of any mitigation or control approach from operating experience.

The expert panel conducted systematic assessments of various components subjected to varying operational stresses (e.g., temperature, pressure, residual stress level, fatigue cycles, irradiation, and water chemistry) and considered potential degradation mechanisms in terms of susceptibility, knowledge, and confidence (SKC) parameters. Color-coded graphical representations of data were presented in SKC diagrams, rainbow charts summarizing degradation evaluations, and flag tables for statistical information (Figure 4.9).

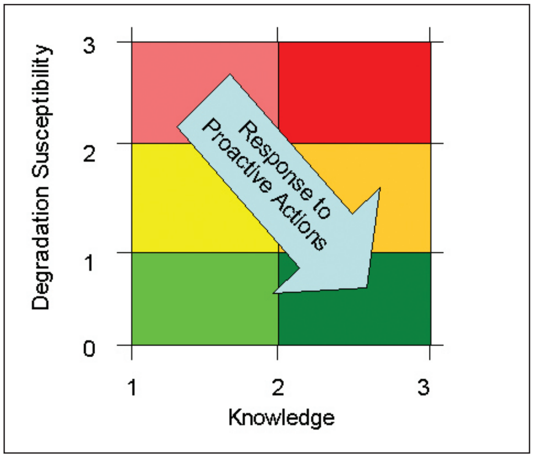

Figure 4.9 Generic SKC diagram used to relate degradation scoring attributes to specific PMMD actions for a given component/degradation mode combination. The arrow indicating the response to proactive actions proceeds diagonally downward from the upper left (representing a situation where degradation is highly likely with limited knowledge to mitigate it) toward the lower right (representing a situation where possible degradation is manageable by mitigation).

Given the global interest in extending the life and licenses of existing nuclear power plants to meet electrical supply needs and in developing comprehensive plant life management for new construction, the PMMD will engage the international nuclear community on a number of fronts (Figure 4.10). A near-term activity will be the establishment of an international forum through two workshops being planned for 2009. One workshop will be conducted in Asia and one in Europe. Emerging from these workshops will be a comprehensive international PMMD network that will address critical issues in PMMD, including:

- maximizing resources and funds to engage the talent of NPP PMMD worldwide
- increasing the effectiveness and communication of research results
- exchanging information and encouraging peer-to-peer interactions among scientists working on PMMD
- identifying critical issues and tapping global talent for developing PMMD solutions
- acquiring the data needed for the NRC and other international regulatory agencies to extend the regulatory framework needed for second and subsequent extensions of licenses

These activities and more will maximize the NRC's engagement within the global PMMD community.

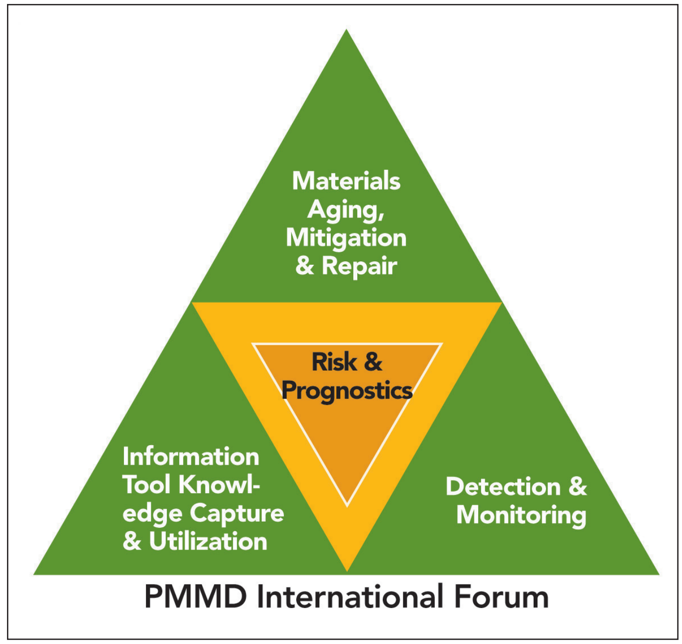

Figure 4.10 Thrust areas in the PMMD program

For more information
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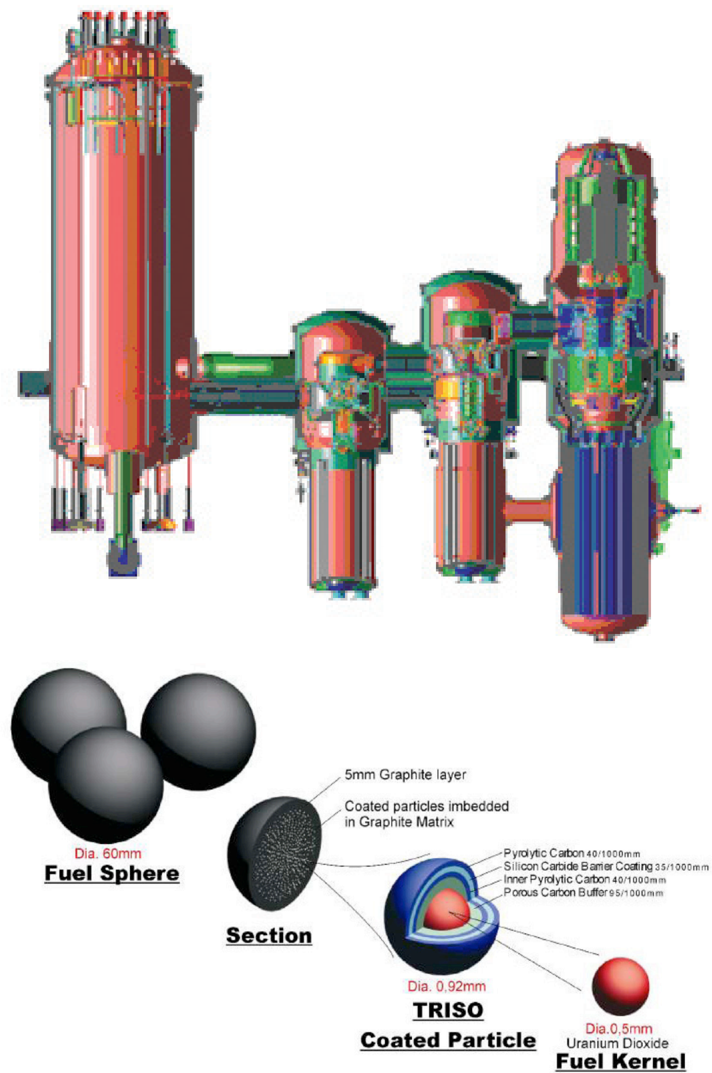
CHAPTER 5: NEW AND ADVANCED REACTORS

Thermal-Hydraulic Analyses of New Reactors

Computational Fluid Dynamics in Regulatory Applications

Next Generation Nuclear Plant

Advanced Reactor Research Program



One Next Generation Nuclear Plant Design: Pebble Bed Modular Reactor

THERMAL-HYDRAULIC ANALYSES OF NEW REACTORS

Background

The NRC uses the TRAC/RELAP Advanced Computational Engine (TRACE) code to perform confirmatory calculations in support of design certification and combined operating license reviews for all new reactors—the Advanced Passive 1000 Megawatt (AP1000), U.S. Advanced Pressurized-Water Reactor (U.S.-APWR), the U.S. Evolutionary Power Reactor (U.S. EPR), the Economic Simplified Boiling-Water Reactor (ESBWR), and the Advanced Boiling-Water Reactor (ABWR) designs. New reactor designs include evolutionary advances in light-water reactor technology and thus pose unique modeling challenges as a result of novel systems and operating conditions. Many of these modeling challenges are associated with passive systems that rely on phenomena such as gravity, pressure differentials, natural convection, or the inherent response of certain materials to temperature changes. Most developmental assessments conducted for currently operating light-water reactors cover the phenomenology necessary in thermal-hydraulic simulations for new reactor designs. However, the modeling of some of the novel systems and operating conditions of new reactors requires further code development and additional assessments against specific experimental data.

New Reactor Designs

AP1000

The AP1000 relies extensively on passive safety systems. Passive systems are used for core cooling, containment cooling, main control room emergency habitability, and containment isolation. These systems challenge system codes in predicting fluid flow induced by small driving heads. The applicability of TRACE to simulate AP1000 transients was demonstrated through comparisons with data from relevant integral and separate-effects test facilities.

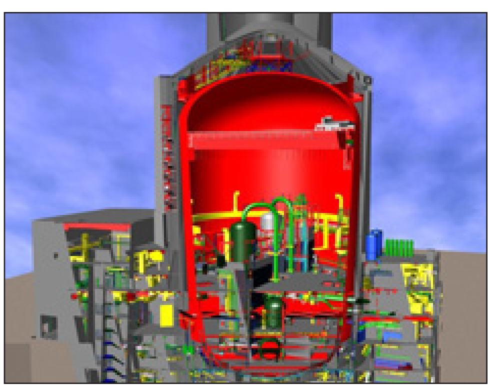


Figure 5.1 Advanced Passive 1000 Megawatt (AP1000)

U.S.-APWR

Most of the major components of the U.S.-APWR are very similar to those of existing pressurized-water reactors (PWRs). The major exception is the advanced accumulator that eliminates the need for pumped low-pressure safety injection. The ability of TRACE to predict the behavior of advanced accumulators has been demonstrated with separate-effects data. Furthermore, detailed three-dimensional phenomena such as cavitation, nitrogen ingress, and mass flow rate have been modeled using computational fluid dynamics tools, and the results were coupled as needed with system code simulations.

Figure 5.2 U.S. Advanced Pressurized-Water Reactor (U.S,-APWR)

U.S. EPR

The U.S. EPR is an evolutionary PWR design that uses rapid secondary-side depressurization for mitigation of loss-of-coolant accidents (LOCAs). This increases the emphasis on the ability of TRACE to predict reflux condensation in steam generator tubes. To demonstrate the applicability of TRACE to the U.S. EPR, code predictions were assessed against data acquired from separate and integral test facilities, such as APEX, FLECHT-SEASET, ROSA-IV, and ROSA-V.

Figure 5.3 U.S. Evolutionary Power Reactor (EPR)

ESBWR

The ESBWR has a passively driven containment cooling system and a gravity-driven cooling system. Both of these systems rely entirely on natural phenomena for the convection of mass and energy. The prediction of void distributions and two-phase natural circulation is very important for the ESBWR. Integral test data from the PUMA and PANDA facilities were used to assess the code for this application. In addition, proper modeling of film condensation in the presence of noncondensable gases at low power levels posed a significant challenge in the ESBWR analysis. Improved models in TRACE predicted these phenomena very well.

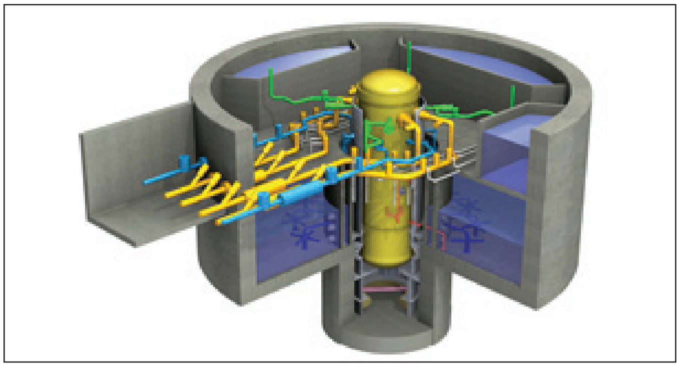


Figure 5.4 Economic Simplified Boiling-Water Reactor (ESBWR)

ABWR

The ABWR is an evolutionary boiling-water reactor that includes such design enhancements as recirculation pumps internal to the reactor vessel and digital controls. TRACE will be used to simulate the plant response to LOCAs, as well as to anticipated operational occurrences and other transients. Modeling internal pumps and incorporating the logic needed for digital controls will pose potential challenges to the code.

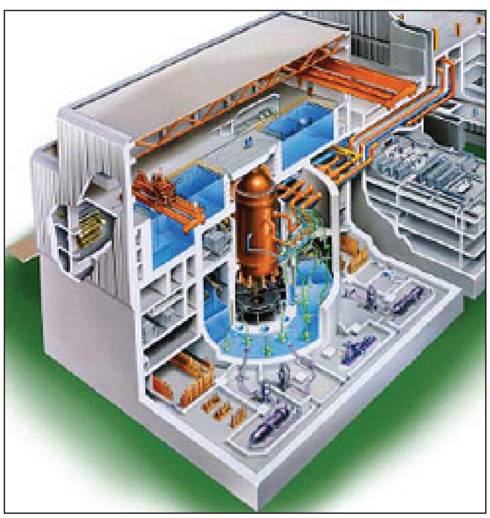


Figure 5.5 Advanced Boiling-Water Reactor (ABWR)

For more information

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COMPUTATIONAL FLUID DYNAMICS IN REGULATORY APPLICATIONS

Background

Computational fluid dynamics (CFD) has reached the maturity necessary to play an increased role in the nuclear power generation industry. CFD provides detailed three-dimensional fluid flow information not available from system code thermal-hydraulic simulations. These multi-dimensional details can enhance the understanding of certain phenomena and thus play a role in reducing uncertainty and improving the technical bases for licensing decisions.

RES has developed a state-of-the-art CFD capability that supports multiple offices within the NRC. RES uses the commercial CFD codes from ANSYS Inc. (FLUENT) and CD-adapco (STAR-CCM+) and has supported the development of multiphase modeling capabilities in research codes. The office maintains a Linux cluster with over 128 processors to provide the capability needed to solve the large-scale problems that are characteristic in the nuclear industry. RES staff is actively involved in national and international CFD programs and maintains a high level of expertise in the field. This state-of-the-art capability provides a robust infrastructure for both confirmatory and exploratory CFD computations.

Applications

SPENT FUEL TRANSPORTATION AND STORAGE

RES works closely with the Office of Nuclear Material Safety and Safeguards in areas concerning the analysis of spent fuel storage cask designs.

The CFD approach has been used to study cask designs under a variety of external conditions, such as fires, reduced ventilation, and hotter fuels. This work supports dry cask certification efforts by improving the agency's technical bases for licensing decisions.

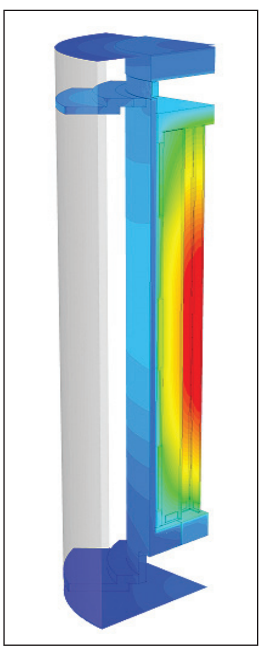

Figure 5.6 Temperature contours of a ventilated dry cask that uses ambient air to passively cool the spent fuel stored inside the canister surrounded by a concrete overpack.

OPERATING REACTORS

CFD predictions have also aided in understanding detailed fluid behavior for broad-scope analyses such as pressurized thermal shock, induced steam generator tube failures, boron dilution and transport, and spent fuel pool analyses. In most cases, CFD results are used iteratively with system code predictions, or they provide boundary or initial conditions for other simulations.

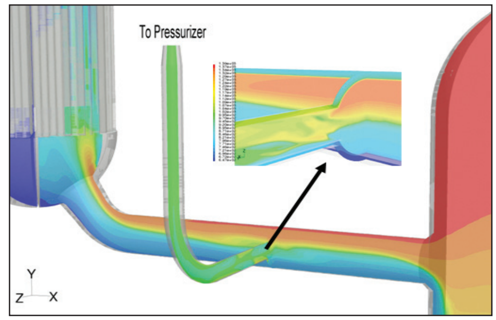

Figure 5.7 During a particular severe accident scenario, hot gases from the core circulate through the hot legs and steam generator in a counter-current flow pattern. The risk of induced failures is considered.

NEW AND ADVANCED REACTORS

The agency has used CFD to confirm the distribution of injected boron in the Economic Simplified Boiling-Water Reactor. In the design certification of the U.S.-Advanced Pressurized-Water Reactor, CFD was used to investigate the performance of an advanced accumulator. The phenomena of interest are cavitation and nitrogen ingress, which exceed typical system code capabilities. The validation of the CFD simulation against experimental data was particularly challenging for this application, especially because CFD results were also used to examine possible scale effects.

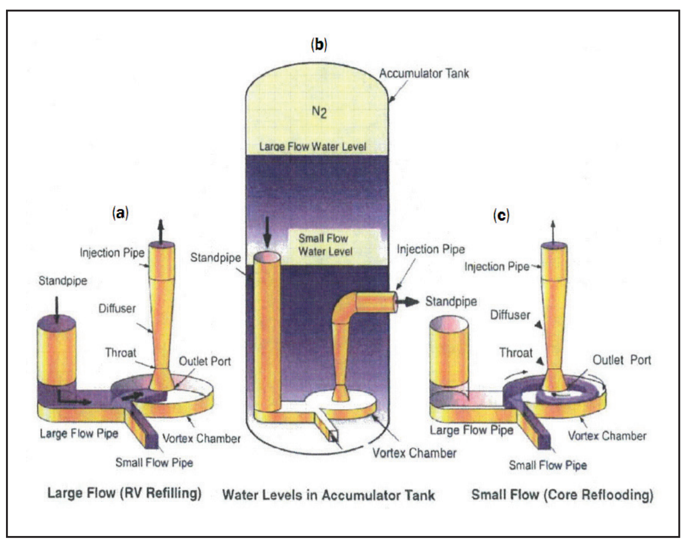

Figure 5.8 The advanced accumulator (b) is a water storage tank with a flow damper in it that switches the flow rate of cooling water injected into a reactor vessel from a large (a) to small (c) flow rate.

For More Information

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NEXT GENERATION NUCLEAR PLANT

Background

The Next Generation Nuclear Plant (NGNP) is an advanced reactor concept for generating electricity and producing hydrogen using the process heat from the reactor outlet. The Energy Policy Act of 2005 (Public Law 109 58) (EPAct), Title VI, Subtitle C, Section 641, directs the Secretary of the U.S. Department of Energy (DOE) to develop an NGNP prototype for operation by 2021. Furthermore, Title VI, Subtitle C, Section 644(a) of the EPAct provides the NRC with the licensing authority for the NGNP prototype, and Section 644(b) requires that the Secretary of DOE and the Chairman of the NRC jointly develop a licensing strategy for the NGNP to submit to the U.S. Congress by August 2008.

Approach

The scope of the NGNP licensing strategy development project addresses all elements of the NGNP licensing strategy as described in Section 644(b) of the EPAct:

- NGNP licensing approach (i.e., a description of the ways in which current light-water reactor (LWR) licensing requirements need to be adapted for the types of reactors considered for the NGNP project)
- analytical tools needed by the NRC to independently verify the NGNP design and its safety performance in order to license an NGNP
- other research and development (R&D) that the NRC will need to conduct for the review of an NGNP license application
- resource requirements to implement the licensing strategy

DOE has determined that the NGNP nuclear reactor will be a very-high-temperature gas-cooled reactor (VHTR) for the production of electricity, process heat, and hydrogen. The VHTR can provide high-temperature process heat (up to 950 degrees Celsius that can be used as a substitute for the burning of fossil fuels for a wide range of commercial applications. Since the VHTR is a new and unproven reactor design, the NRC will need to adapt its licensing requirements and processes, which have historically evolved around LWR designs, for licensing the NGNP nuclear reactor.

NGNP REACTOR TECHNOLOGY

NGNP reactor technology differs from that of commercial LWRs. Hence, to develop a licensing approach, an NGNP technology envelope needs to be defined, considering key project assumptions and uncertainties that are relevant to evaluating licensing options and establishing technical requirements. These aspects may include, but are not limited to, technology options being considered; potential prototype plant parameter envelope (licensed power level, fuel type and performance characteristics, power conversion cycle, hydrogen co-generation technology, spent fuel management, safety and security issues, etc.); and plans and schedules for technology development, for design development, and for licensing.

The final design of a prototype NGNP will be realized some time in the future; however, the two concepts in the forefront of technology development are the pebble bed reactor and the prismatic core reactor.

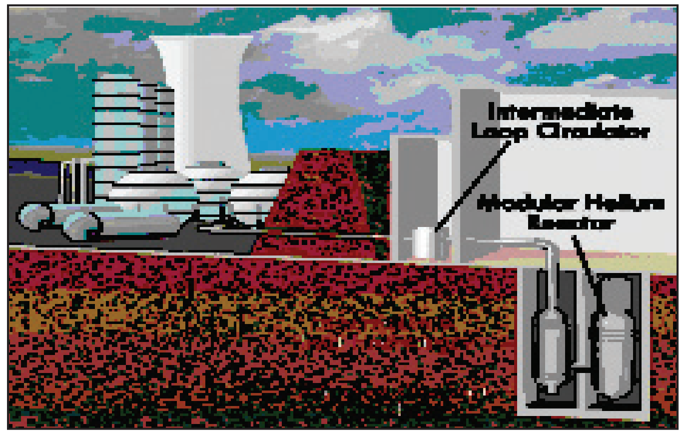

Figure 5.9 Artist's rendition of an NGNP Plant

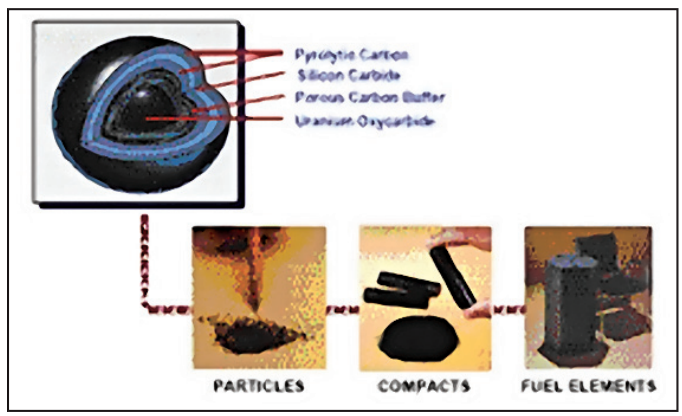

Figure 5.10 TRISO fuel for NGNP

NGNP LICENSING REQUIREMENTS

Many of the regulatory requirements and supporting review guidance for LWRs are technology neutral; that is, they are applicable to non-LWR designs as well as to LWR designs. However, certain LWR requirements may not apply to the unique aspects of an NGNP design. Accordingly, in developing the NGNP licensing strategy, the NRC and DOE considered the various options available to the NRC staff for adapting current NRC LWR licensing requirements for the NGNP VHTR. These options related to legal, process, technical, research, and regulatory infrastructure matters and included an examination of historical licensing activities. These considerations led to selection of a licensing strategy that would best comply with the considerations identified in the EPAct.

The licensing strategy developed jointly by the NRC and DOE has two distinct aspects. The first is a recommended approach for how the NRC will adapt the current LWR licensing technical requirements to apply to an NGNP. The second is a recommended licensing process alternative that identifies which of the procedural alternatives in the NRC regulations would be best for licensing the NGNP. To arrive at these recommendations, the NRC and DOE evaluated a number of options and alternatives.

ANALYTICAL TOOLS DEVELOPMENT AND OTHER R&D

Certain analytical tools will likely need to be developed or modified in different technical areas to enable the review of the NGNP license application, evaluate the safety case, and assess the safety margin. Given the early stage of the NGNP program, the development needs should be considered preliminary projections to be reevaluated on an ongoing basis.

To address regulatory and safety issues for an NGNP design in major technical areas, and in particular to identify important safety-relevant phenomena associated with these design concepts and to assess the knowledge base, a phenomena identification and ranking table (PIRT) exercise was conducted in 2007.

The PIRT process involved assembling groups of experts in each of the identified major areas, facilitating focused discussions among the experts in these areas, annotating expert deliberations and finally, assessing the knowledge gaps in these areas based on expert deliberations.

The PIRT exercise was conducted in the following major topical areas associated with the NGNP:

- thermal fluids and accident analysis
- high-temperature materials including graphite
- process heat and hydrogen co-generation
- fission product transport (FPT) and dose
- tristructural isotropic (TRISO)-coated fuel particles

The NRC plans to use existing analytical tools to the extent feasible, with appropriate modifications for the intended purpose. For LWR safety analysis, the NRC traditionally uses its system-level MELCOR code, which is capable of performing thermal-fluid and accident analysis, including FPT and release. This code will be modified for the NGNP. Also, as needed, computational fluid dynamics models and associated tools will be developed to investigate certain thermal-fluids phenomena in greater detail so as to reduce uncertainties in predictive capability.

The NRC uses PARCS, among other codes, for neutronic calculations, which provide initial and boundary conditions to accident analysis codes such as MELCOR. The neutronic codes can be modified as appropriate for NGNP confirmatory analysis. The agency will need a fuel performance code to provide fuel-related initial and boundary conditions to accident analysis codes. DOE has ongoing R&D activities to support development of such a code. The NRC will explore inclusion of this code or, at a minimum, the fuel performance models in the code, in the agency's suite of codes.

In other technical areas (notably, high-temperature materials and graphite performance and fuel performance), the development strategy for confirmatory analysis tools will utilize various sources of information to the maximum extent feasible. Current R&D activities funded by DOE, as well as international cooperative R&D programs, are addressing many of these areas. To the extent that data and tools are available from these activities, the NRC will use this information in the development of its independent confirmatory analysis capability. The NRC will also make extensive use of experimental data generated by an applicant and provided to the agency as part of the license submittal, as well as data from domestic and international programs and other data available in the open literature.

Project Status

The NGNP Licensing Strategy report was submitted to the U.S. Congress in August 2008. Work is currently in progress to implement various elements of the licensing strategy.

For More Information

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ADVANCED REACTOR RESEARCH PROGRAM

Background

RES has updated the NRC's Advanced Reactor Research Program (ARRP). The original ARRP was forwarded to the Commission on April 18, 2003, as Enclosures 1 and 2 to SECY-03-0059, "NRC's Advanced Reactor Research Program." The revised ARRP focuses on advanced non-light-water reactor (non-LWR) designs involving high- (and very-high-) temperature, graphite-moderated, gas-cooled reactors. The high-temperature gas-cooled reactor (HTGR) and very-high-temperature gas-cooled reactor (VHTR) research infrastructure assessment and related NRC research and development (R&D) plans rebaseline the earlier HTGR research infrastructure assessment and R&D plans documented in SECY-03-0059.

Overview

The revised ARRP documents the NRC's current assessment of its research infrastructure needs and the agency's planned safety research to support its review of HTGR and VHTR licensing applications. These include a combined license (COL) application for a VHTR to be constructed at the Idaho National Laboratory (INL) in connection with the Next Generation Nuclear Plant (NGNP) Project, as directed by the Energy Policy Act of 2005, and a potential design certification application for the pebble bed modular reactor.

The update also includes a high-level survey of the technical infrastructure development and initial safety research that the NRC would need to conduct to prepare for its review of a potential sodium-cooled fast reactor (SFR) licensing application. Such licensing applications include a near-term application for design approval for the Toshiba Super Safe, Small and Simple (4S) reactor and a longer term licensing application for a commercial advanced fast-burner reactor being developed by the U.S. Department of Energy (DOE) for nuclear fuel recycling. The SFR research infrastructure survey was conducted at a higher level than the HTGR and VHTR reassessment. The survey identifies the key technical, safety, and research issues associated with SFR licensing. The survey provides a framework for a potential follow-on in-depth SFR research infrastructure assessment similar in scope to the HTGR and VHTR assessment. As an example, the NRC HTGR accident analysis evaluation model concept schematic shown in Figure 5.11 demonstrates the applicability of research results to reactor plant systems analysis.

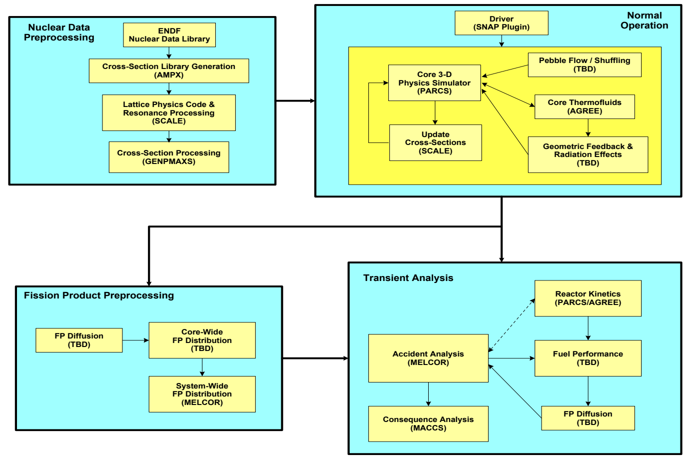

Figure 5.11: Schematic of NRC HTGR accident analysis evaluation model concept

The updated ARRP also includes technical infrastructure development and associated NRC safety research needs that are generically applicable to all advanced reactor designs. Generic advanced reactor arenas include human performance, advanced digital instrumentation and controls, and probabilistic risk assessment.

The revised ARRP reflects the results of phenomena identification and ranking table reviews conducted for the NGNP VHTR. The revision also reflects comments received from DOE and INL on a draft revision of the ARRP update, as well as technical information provided by DOE and INL on the R&D being conducted by the DOE national laboratories in support of design, development, and licensing of the NGNP VHTR. The ARRP also considers technical information received from other national and international organizations involved in HTGR safety R&D.

The current update recognizes that some of the technical infrastructure issues and NRC safety research plans documented in the 2003 ARRP were subsequently included in the R&D plans of selected foreign or domestic HTGR or VHTR design, development, or research organizations. The updated ARRP reflects completion of selected high-priority HTGR-specific and generic safety R&D described in the 2003 ARRP.

The scope of the reassessment does not include the technical infrastructure development and safety research that may be needed to support the review of licensing applications for advanced LWRs (e.g., AREVA Evolutionary Power Reactor [EPR], General Electric Economic Simplified Boiling-Water Reactor [ESBWR], Westinghouse International Reactor Innovative and Secure [IRIS] LWR, Mitsubishi U.S. Advanced Pressurized Water Reactor [APWR], and NuScale. The staff will document these R&D needs separately on an advanced LWR design-specific basis.

The NRC will assign priorities to R&D tasks for developing the agency's VHTR technical infrastructure development and safety research consistent with the NGNP VHTR technology selection and COL application schedule. Priorities will be similarly assigned to the generic NRC R&D tasks. NRC technical infrastructure development to support the agency's safety review of these designs will involve the development of staff expertise, analytic tools and methods, experimental facilities, and data. In the near term, the staff expects the highest priority NGNP VHTR-specific technical infrastructure development and safety research to be in the areas of materials analysis, fuel performance analysis, nuclear and thermal-fluid analysis, accident analysis, and technical review infrastructure.

The ARRP HTGR and VHTR infrastructure assessment and SFR infrastructure survey identify, respectively, the gaps in the NRC's technical information and data and independent technical capabilities for conducting licensing application reviews for HTGRs and SFRs.

Summary

The VHTR and HTGR infrastructure technical needs assessment activities are linked to the following nine key safety research arenas:

1. technical review infrastructure (including draft regulatory review guidance for applying probabilistic risk information in establishing licensing basis events; classification of systems, structures, and components; and defense in depth)
2. accident analysis (including probabilistic risk assessment methods and assessment guidance, human performance, and instrumentation and control)
3. reactor/plant systems analysis (including thermal-fluid analysis, nuclear analysis, mechanistic source-term analysis, and fission product transport analysis)
4. fuel performance analysis (including fuel performance mechanistic analysis and fuel fission product transport analysis)
5. materials analysis (including nuclear graphite component and metallic component performance)
6. structural analysis (including reactor building civil structure and reactor core internals structural performance) and reactor safety hazards posed by a connected nearby hydrogen production or process heat facility
7. consequence analysis (including dose calculations and environmental impact studies)
8. nuclear materials safety (including enrichment, fabrication, and transport) and waste safety (including storage, transport, and disposal)
9. nuclear safeguards and security

Human performance and instrumentation and controls are considered generic arenas applicable to all advanced reactor designs and technologies. The SFR infrastructure survey addressed reactor/plant systems analysis (including thermal-fluid dynamics, nuclear analysis, and severe accident and source-term analysis), fuels analysis, materials analysis, and structural analysis.

For More Information
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CHAPTER 6: RADIATION PROTECTION

In Situ Bioremediation of Uranium in Ground Water

Health Effects Program

Report to Congress on Abnormal Occurrences

Radiation Exposure Information and Reporting System

An Analysis of Cancer Risk in Populations
Living Near Nuclear-Power Facilities

VARSKIN Skin Dose Computer Code

Radiological Toolbox

Integrated Ground Water Monitoring

Environmental Transport Research Program

Region IV inspector Linda Gersey surveys for gamma radiation at the Smith Ranch in situ leach uranium recovery facility in Wyoming.

IN SITU BIOREMEDIATION OF URANIUM IN GROUND WATER

Background

The NRC has received license applications and decommissioning plans referring to the use of in situ bioremediation of ground water at two types of sites: (1) areas where shallow plumes of uranium (U) originated with waste disposal operations, and (2) in situ leach (ISL) U recovery sites that have been depleted and require ground water remediation according to Title 10 of the Code of Federal Regulations, Part 40, "Domestic Licensing of Source Material," Appendix A, "Criteria Relating to the Operation of Uranium Mills and the Disposition of Tailings or Wastes Produced by the Extraction or Concentration of Source Material from Ores Processed Primarily for Their Source Material Content." In both cases, the original remediation methods have not reduced aqueous U concentrations to acceptable levels. As a result, a new approach, using in situ manipulation of native bacterial populations to alter geochemical conditions, has been proposed.

In this remediation technique, electron donors (e.g., acetate, lactate) are injected through wells into the contaminated aquifer where bacterial activity is expected to increase and generate reducing conditions. In this process, iron (Fe [III]) and uranium [U [VI] will be reduced, and U precipitated from solution (Figure 6.1). This is a relatively new remediation approach with potential applications at many other nuclear facilities (for a review of the technology, see LBNL-42995, 2003). However, it is important to note that the U is left in place. Eventually, many sites that have been bioremediated will likely be exposed to oxidizing conditions. This is especially true for shallow sites. The NRC staff needs to be able to assess the long-term ability of bioremediation to sequester U from ground water. Consequently, it is important to have a technically defensible understanding of the long-term behavior of the residual uranium in remediated systems as they return to oxidizing conditions.

Approach

To assess the behavior of bioremediated systems, two approaches are being used for each of the two types of sites (shallow U plumes and ISL units). The two approaches, laboratory-scale experimental work and advanced modeling, complement each other. The U.S. Geological Survey is conducting the laboratory work in a project entitled "Uranium Sequestration and Solid Phase Behavior during and after Bioremediation." The modeling project, being done at the Pacific Northwest National Laboratory, is called "Modeling the Long-Term Behavior of Uranium during and after Bioremediation."

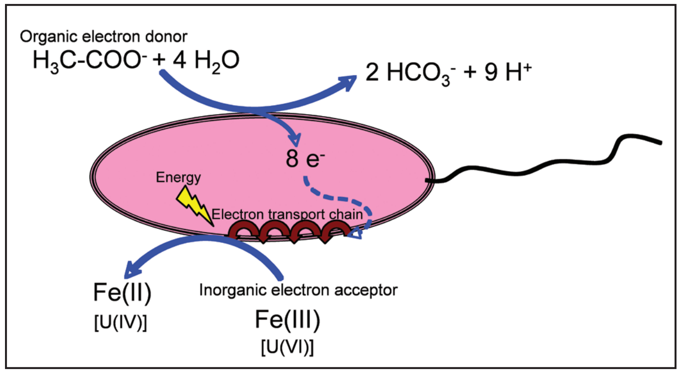

Figure 6.1: Microbial mediation of Fe(III) reduction. U(VI) is the mobile valence state of U, whereas reduced U, U(IV), is insoluble as uraninite. Addition of acetate as an electron donor stimulates dissimilatory metal-reducing microorganisms. U(VI) is reduced concurrently with Fe(III). Original concept from Lovley et al. (1991). Figure from NUREG/CR-6973.

EXPERIMENTAL APPROACH

For the experimental program, sediment representing the two kinds of sites is placed in columns, and reducing conditions are established by biostimulation. Especially important to long-term performance is the stability of solid phase U and Fe minerals generated by bioremediation as they are leached by various site-specific ground waters. The behavior of U and other elements is followed in both the aqueous phase and the solid phase during reduced conditions and then as oxygen containing water is introduced into the columns. Solid phase analysis will include synchrotron-based methods such as X-ray absorption spectroscopy to determine the oxidation state of U and Fe, as well as their microscale distributions under the reduced and oxidized conditions of the columns. For sediment from an ISL site, changes in the microbial community and important community members will be assessed by two DNA-based methods over the course of the column experiments.

Other processes will control the distribution of U between the solid and liquid phases after oxidizing conditions begin to return to the sites and potentially remobilize the U. Licensees contend that U will readily adsorb on Fe oxides that form in place after the remediated soil is oxidized. It is unknown if this is a viable process, or if the chemistry of reoxidation (i.e., acid generation) will inhibit adsorption.

A series of questions define this research. While it is well known that U precipitates under reducing conditions, does it always precipitate as a discrete mineral or is some U distributed as amorphous or very small (nanoscale) particles? During bioremediation, does U also co-precipitate with minerals such as mackinawite, siderite, and calcite? If so, in what oxidation state is the U? If new aqueous U enters the system, how does it react with these new solids? The experimental program will answer these questions and give the details needed to estimate the long-term behavior of U.

MODELING APPROACH

The objective of the modeling work is to identify, assess, and model short- and long-term chemical processes, caused by in situ bioremediation. The modeling will focus on processes controlling U sequestration and changes in U mobility, during and after bioremediation. The approach is to use coupled models of biological, geochemical, and transport processes to determine how the chemistry in these systems changes and what the effects will be on parameters that can be monitored in the field. (Yabusaki et al., 2007).

The modeling effort will iterate through key parameters such as flow rates, U concentrations, mass of Fe available, carbonate concentrations, biological kinetics, alkalinity, and O2 and U input. Data from experimental field sites of the U.S. Department of Energy and the experimental work described earlier will inform the modeling. The modeling will explore the response of U to transient and dynamic processes and events that influence geochemical conditions (e.g., incursion of lixiviant [the oxygen enriched solution used to extract U from the ore] from other operating cells and floods). The results will include expected responses of monitored parameters (e.g., U concentrations, pH, oxidation-reduction potential) to these events, as determined by model calculations.

Products of the modeling work will include a guidance document that describes approaches, criteria, and methods to predict the stability of biorestored ISL sites and shallow plumes. The guidance, designed to help staff perform licensing reviews, will provide modeling-based information on changes in monitorable parameters that can be expected as a result of changing conditions in the subsurface system.

Ultimately, both the experimental and modeling approaches will allow the NRC to do the following:

1. assess the geochemical, microbial and ground water conditions and processes that affect U transport and its potential long-term sequestration
2. provide the technical basis to predict long-term performance (e.g., considering a performance period of 1,000 to 10,000 years) for decommissioning, particularly during reoxidation following bioremediation treatments
3. evaluate biorestoration design, performance, and stability for U recovery and related financial surety costs

REFERENCES

Lawrence Berkeley National Laboratory, LBNL-42595, “Bioremediation of Metals and Radionuclides…What it is and how it works,” 2nd Edition, 2003.

Lovly, D.R., et al., Microbial Reduction of Uranium, Nature 350(6317): 413–416, 1991.

Long, P.E. et al., NUREG/CR-6973, “Technical Basis for Assessing Uranium Bioremediation Performance,” 2008.

Yabusaki, S.B., et al., “Uranium Removal from Groundwater via In Situ Biostimulation: Field-Scale Modeling of Transport and Biological Processes.” Journal of Contaminant Hydrology 93(1-4): 216–235, 2007.

For More Information

Contact Dr. Mark Fuhrmann, RES/DRA, at 301-251-7472 or Mark.Fuhrmann@nrc.gov

HEALTH EFFECTS PROGRAM

Background

The Health Effects Program is an agencywide resource that provides technical support in the areas of radiation protection, dose assessment, and assessment of human health effects for reactor and nuclear materials licensing, emergency preparedness, and nuclear security activities.

The scope of this program includes the technical basis for radiation protection regulations, exposure and abnormal occurrence reports, computer codes and database development, and health effects and dosimetry research.

Figure 6.2: Foundation and Basis for Regulatory Program

Technical/Regulatory Programs

- internal dosimetry research
- monitoring national and international scientific
- organizations related to radiation protection
- radiation protection regulatory guides
- VARSKIN
- report to Congress on abnormal occurrences
- Radiation Exposure Information and Reporting System
- risk communication outreach
- update to cancer study
- radiation security support

Ongoing Projects

INTERNAL DOSIMETRY RESEARCH

The Health Effects Branch provides technical resources to the agency by conducting radiation dosimetry research for regulatory applications. This research improves the agency's capability in modeling radiation interactions within humans and evaluating internal dosimetry codes for estimating radiation exposures and assessment of worker or public exposures from licensed activities or incidents.

NATIONAL AND INTERNATIONAL ACTIVITIES

One of the benefits of the Health Effects Program is the promotion of consistency in regulatory applications of radiation protection and health effects research among NRC programs, as well as those of other Federal and State regulatory agencies. The Health Effects Program staff collaborates with national and international experts in health physics at national laboratories, universities, and other organizations, including those listed below:

- Interagency Steering Committee on Radiation Standards
- National Council on Radiation Protection and Measurements
- National Academies
- Nuclear Energy Agency Information System on Occupational Exposure
- International Commission on Radiological Protection
- United Nations/International Atomic Energy Agency
- L'institut de Radioprotection et de Sûreté Nucléaire (French Institute for Radiological Protection and Nuclear Safety)

RADIATION PROTECTION REGULATORY GUIDES

Development and updating of regulatory guides on occupational health and other topics related to radiation protection provide licensees with better methods for complying with NRC regulations. Regulatory guides describe approved NRC methods of meeting the requirements of the regulations.

VARSKIN

The NRC funded the development of the VARSKIN computer code in the 1980s to facilitate skin-dose calculations. Since then, the code has been upgraded to make it more efficient and easier to use. The NRC is currently developing the code to replace the existing photon dose algorithm with a more sophisticated one and adding further enhancements to the code's functionality.

REPORT TO CONGRESS ON ABNORMAL OCCURRENCES

The NRC publishes an annual report to Congress on abnormal occurrences (AOs). An AO is defined as an unscheduled incident or event that the NRC determines to be significant from the standpoint of public health or safety. The AO report contains event details from both NRC and Agreement State-licensed facilities that meet the AO criteria published by the Commission.

RADIATION EXPOSURE INFORMATION AND REPORTING SYSTEM

The NRC's Radiation Exposure Information and Reporting System (REIRS) collects information on occupational radiation exposures to workers from certain NRC-licensed activities. The agency uses the data collected in the REIRS database to evaluate licensees' as low as reasonably achievable (ALARA) programs and shares data with national and international research counterparts. The NRC also uses the REIRS database to compile the annual report NUREG-0713, "Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities."

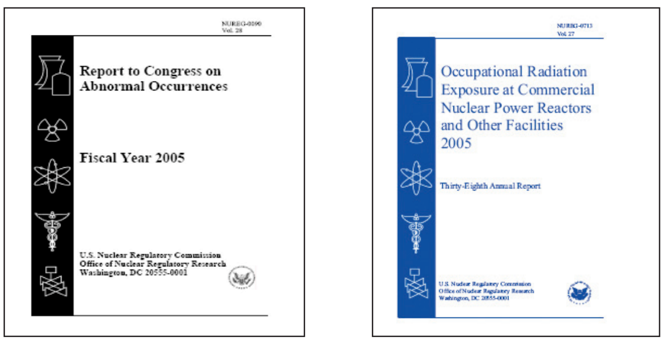

Figure 6.3: 2005 AO and REIRS NUREGs

RISK COMMUNICATION OUTREACH

The NRC held radiation risk assessment and communication workshops. The purpose of these workshops was to provide training in effectively communicating risk with stakeholders.

UPDATE TO "ANALYSIS OF CANCER RISK IN POPULATIONS LIVING NEAR NUCLEAR-POWER FACILITIES"

The NRC is conducting a study to complement the 1990 National Cancer Institute (NCI) report, "Cancer in Populations Living Near Nuclear Facilities." The NCI report found that cancer risk was generally not elevated in the populations that lived near nuclear-power facilities. Today, stakeholder interest continues about perceived elevated cancer rates in populations near reactors, including cancer incidence (i.e., being diagnosed with cancer, but not necessarily dying from the disease).

RADIATION SECURITY SUPPORT

The NRC supports the Energy Policy Act's Radiation Source Protection and Security Task Force, which is overseen by the NRC with participation by 14 Federal agencies. The NRC worked with the National Academies to prepare a report on possible technological alternatives to existing high-activity radiation sources that are regulated by the NRC and the Agreement States. This 2-year study assessed a range of alternatives, including X-ray, accelerator, and ultrasonic devices, and recommended options for their implementation.

Health Effects Program Products

- VARSKIN computer code
- NUREG-0713, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities"
- MARSSIM and MARSAME radiological survey manuals
- Fact Sheet on Tritium, Radiation Protection Limits, and Drinking Water Standards
- NUREG-0090, "Report to Congress on Abnormal Occurrences"
- radiation risk assessment and risk communication workshops

Future Goals: Strengthen the NRC's Health Physics Capabilities

The goal of the Health Effects Program is to be an agencywide resource for technical and regulatory health physics information, including development of implementation tools for state-of-the-art techniques in radiation protection and recommendations on health physics policy.

For More Information

Contact Stephanie Bush-Goddard, RES/DSA, at 301-251-7528 or Stephanie.Bush-Goddard@nrc.gov

REPORT TO CONGRESS ON ABNORMAL OCCURRENCES

Background

Section 208 of the Energy Reorganization Act of 1974 defines an "abnormal occurrence" (AO) as an unscheduled incident or event that the NRC determines to be significant from the standpoint of public health or safety.

The Federal Reports Elimination and Sunset Act of 1995 (Public Law 104-66) requires that the NRC report AOs to Congress annually. The NRC initially promulgated the AO criteria in a policy statement published in the Federal Register on February 24, 1977 (42 FR 10950), followed by several revisions in subsequent years.

The most recent revision to the AO criteria, published in the Federal Register on October 12, 2006 (71 FR 60198), took effect on October 1, 2007.

Approach

The AO process aids in identifying deficiencies and ensuring that corrective actions are taken to prevent recurrence. An accident or event will be considered an AO if it involves a major reduction in the degree of protection of public health or safety. This type of incident or event would have a moderate or more severe impact on public health or safety and could include, but need not be limited to, the following:

1. moderate exposure to, or release of, radioactive material licensed by or otherwise regulated by the Commission
2. major degradation of essential safety-related equipment
3. major deficiencies in design, construction, use of, or management controls for facilities or radioactive material licensed by or otherwise regulated by the Commission

APPLICATION

When an incident or event occurs, the NRC uses a generic event assessment process illustrated by the diagram to the right. This generic event assessment process includes the following:

- internal coordination with NRC offices
- systematic review of cause(s) of the event
- followup with the reporting licensee
- appropriate outreach to external stakeholders
- communication of lessons learned

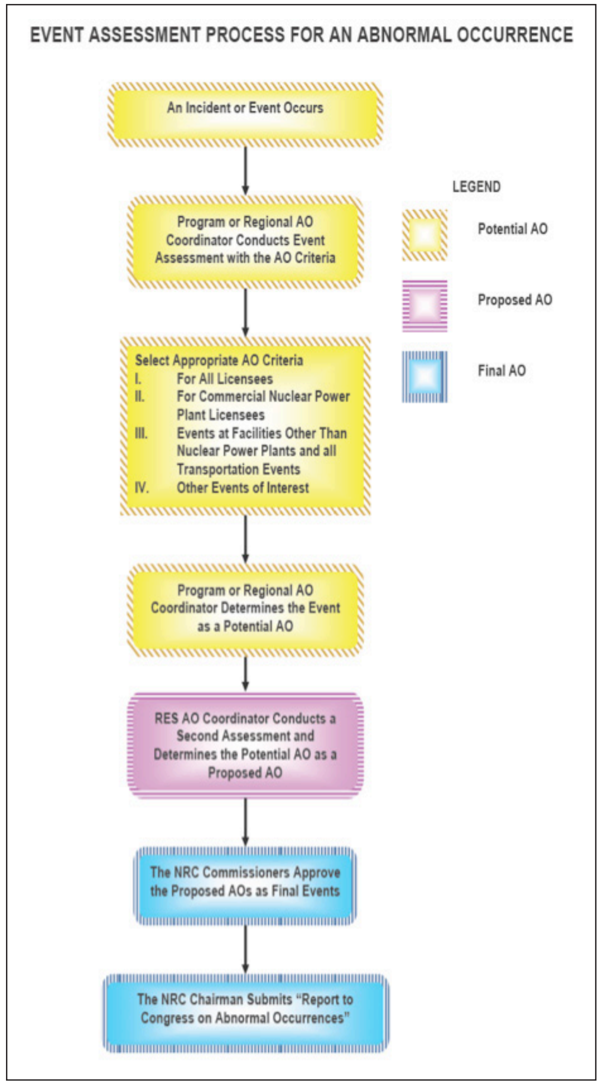

Figure 6.4 Event Assessment Process for an Abnormal Occurrence

Examples of AO Events

MEDICAL EVENT AT HACKLEY HOSPITAL INVOLVING BRACHYTHERAPY TREATMENT FOR PROSTATE CANCER

Hackley Hospital notified the NRC of a medical event that occurred during a brachytherapy seed implant procedure to treat prostate cancer. The procedure prescribed the use of radiation from permanent implanted iodine-125 seeds to treat the patient's prostate. Movement of the patient caused only 7 of the prescribed 41 seeds to be implanted in the prostate, and the other 34 seeds were implanted in the area surrounding the prostate.

Figure 6.5 depicts the typical size of radioactive seeds used for cancer treatment.

Figure 6.6 depicts normally implanted radioactive seeds in the prostate.

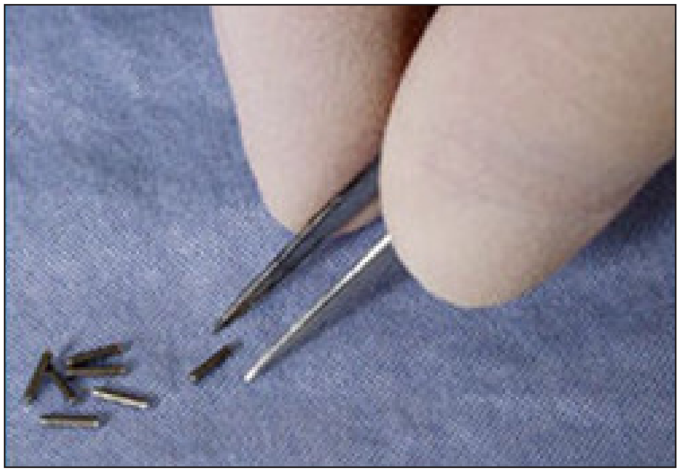

Figure 6.5 Photo of radioactive implantable seeds

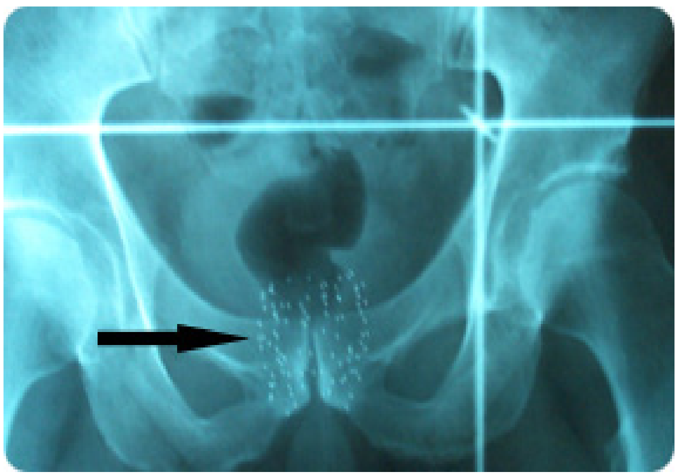

Figure 6.6 X-ray of normal brachytherapy seeds implanted in the prostate

MEDICAL EVENT AT MEMORIAL MISSION HOSPITAL INVOLVING TREATMENT FOR ASSESSING THYROID HEALTH

Memorial Mission Hospital reported that a patient was prescribed a dose of radiation using iodine-131 to assess the health of her thyroid (see Figure 6.7). However, she was administered a dose about 100 times greater. The hospital discovered this event when the patient returned the next day for another scan.

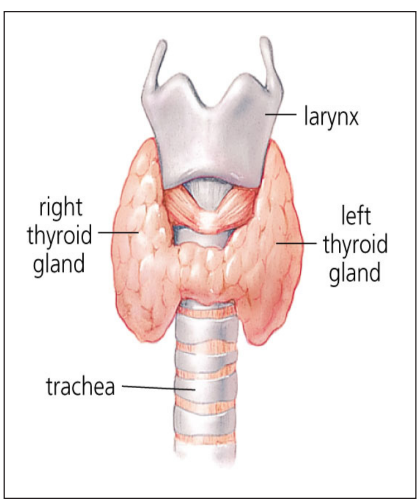

Figure 6.7: Diagram of thyroid

For More Information

Contact Doris Lewis, RES/DSA, at 301-251-7559 or Doris.Lewis@nrc.gov

RADIATION EXPOSURE INFORMATION AND REPORTING SYSTEM

Background

The Radiation Exposure Information and Reporting System (REIRS) project involves the collection and analysis of the radiation exposure records reported by NRC licensees under Title 10 of the Code of Federal Regulations, Section 20.2206, "Reports of Individual Monitoring."

Each year, over 200,000 exposure records are submitted by five of the seven categories of NRC licensees:

1. industrial radiography
2. manufacturers and distributors of byproduct material
3. commercial nuclear power reactors
4. independent spent fuel storage installations
5. fuel processors, fabricators, and reprocessors

Currently, the NRC does not receive exposure records from the other two categories (low-level waste disposal facilities and high-level waste geologic repositories), because they are not under NRC jurisdiction or currently in operation. Most records are submitted on electronic media, while some licensees (primarily nonreactor licensees) continue to submit records on paper.

Approach

To comply with this regulation, licensees can electronically submit to REIRS or send paper records of their exposure data to the NRC.

The objective of the REIRS database is to provide the NRC with data for evaluating licensee performance and further research studies. The data in the REIRS database provides facts regarding routine occupational exposures to radiation and radioactive material that can occur in connection with certain NRC-licensed activities.

The data analyzed in REIRS is published annually in NUREG-0713, "Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities."

APPLICATION

The REIRS project involves the collection and analysis of the radiation exposure records reported by NRC licensees. The agency uses these records to meet its regulatory goals in the following ways:

- for commercial nuclear power plants, to evaluate the effectiveness of the licensee's as low as reasonably achievable (ALARA) program
- to evaluate the radiological risk associated with certain categories of NRC-licensed activities
- to compare occupational radiation risks with potential public risks
- to establish priorities for the use of NRC health physics resources (research, standards development, and regulatory program development)
- to answer congressional and public inquiries
- to provide radiation exposure histories to individuals who were exposed to radiation at NRC-licensed facilities
- to plan epidemiological studies

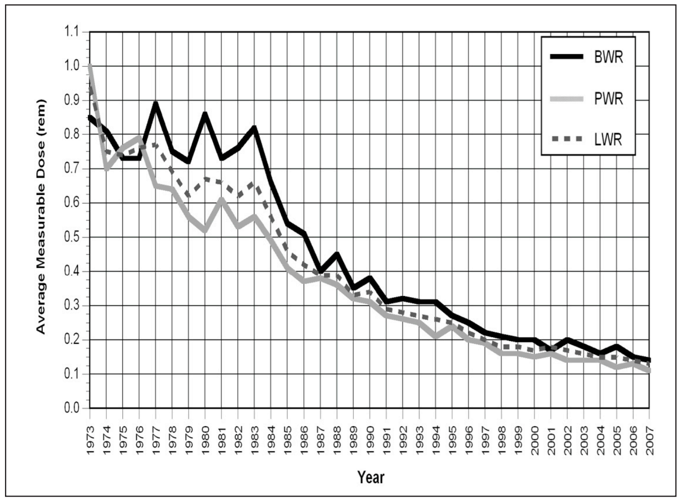

Figure 6.8 Sample data from REIRS database
Average measurable dose at power reactors 1973–2007

WEB SITE

The annual NUREG-0713 reports are available on the NRC public Web page at www.nrc.gov and can also be found on the REIRS Web site at www.reirs.com.

REIRS SOFTWARE

REMIT is a software package that allows licensees to maintain and report their exposure records to the REIRS database. REMIT allows for the electronic exchange of records between one licensee and another and the importing of records from the licensee's dosimetry processor. REIRView is another NRC-developed software package that allows licensees to validate their annual electronic submittal to the REIRS database. This saves licensees and the NRC considerable processing time since the licensee can identify and correct problems before submitting the information to the REIRS database.

For More Information

Contact Doris Lewis, RES/DSA, at 301-251-7559 or Doris.Lewis@nrc.gov

ANALYSIS OF CANCER RISK IN POPULATIONS LIVING NEAR NUCLEAR-POWER FACILITIES

Background

The NRC is conducting a new study to update a 1990 U.S. National Cancer Institute (NCI) report, "Cancer in Populations Living Near Nuclear Facilities." NCI performed this study in response to concerns about elevated risk of childhood leukemia to persons near a British nuclear facility (Sellafield). NCI researchers studied more than 900,000 cancer deaths using county mortality records collected from 1950–1984. Changes in mortality rates for 16 types of cancer were evaluated. The NCI report concluded that cancer mortality rates are generally not elevated for people living in the 107 U.S. counties containing or closely adjacent to 62 nuclear facilities. However, the population data used in the NCI report are more than 20 years old.

Today, stakeholder interest persists in the perceived elevated cancer rates in populations near reactors, including cancer incidence (i.e., being diagnosed with cancer, but not necessarily dying from the disease). The NRC is conducting the update to provide contemporary information on potential elevated risks of cancer near nuclear power facilities.

Approach

CANCER MORTALITY STUDY

An external peer review committee will review a protocol for selecting study and control populations in the vicinity of past, present, and future nuclear power facilities. Also, this study will explore the use of advanced geographical information systems to refine the populations examined from the county level to a smaller geographical unit. The peer review committee will include academic, industry, and government experts to ensure a high quality and technically robust study. The study's draft report, including an overview of its findings, will be submitted to the peer review committee and NRC staff for review and comment. Following resolution of comments, the committee will issue a final report, which is scheduled for publication in 2011.

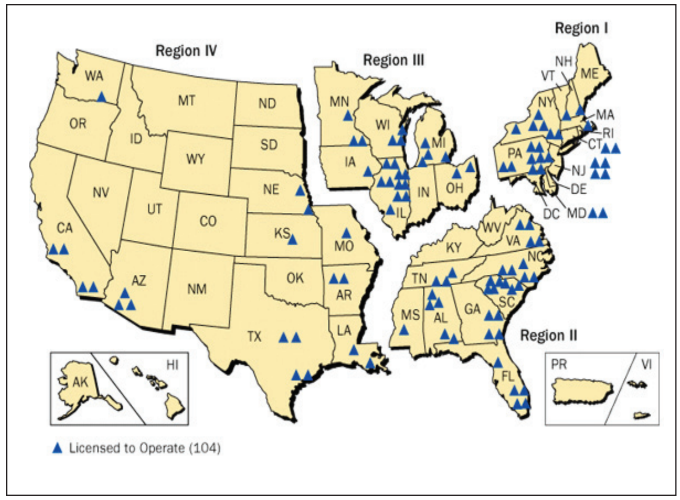

Figure 6.9: Locations of operating nuclear power facilities

CANCER INCIDENCE FEASIBILITY STUDY

The update to the 1990 NCI study will include development of a protocol for examining cancer incidence in the vicinity of nuclear power facilities. This part of the study is intended to provide the NRC staff with information on the feasibility of conducting a future study of cancer incidence in 2011.

Study Status

The NRC began this study in October 2008 and is currently assembling the external peer review committee to review the draft study protocol for performing the cancer mortality study. The peer review committee currently has commitments for participation of scientists from NCI, the Centers for Disease Control, the U.S. Department of Energy, and the French Institut de Radioprotection et de Sûreté Nucléaire.

Biographies of committee members will be available when the committee membership is complete in 2009.

For More Information

Contact Terry Brock, RES/DSA, at 301-251-7487 or Terry.Brock@nrc.gov

VARSKIN SKIN DOSE COMPUTER CODE

Background

The computer code VARSKIN 3 is currently used to model and calculate skin dose from skin or protective clothing contamination for regulatory requirements under Title 10 (10 CFR) of the Code of Federal Regulations Part 20, "Standards for Protection against Radiation."

The NRC sponsored the development of the VARSKIN code to assist licensees in demonstrating compliance with 10 CFR 20.1201(c). Specifically, this regulation requires licensees to have an approved radiation protection program that includes established protocols for calculating and documenting the dose attributable to radioactive contamination of the skin.

Approach

ORIGINAL VARSKIN CODE

The initial version of the code, developed in the 1980s, fulfilled the regulatory requirement but was limited to point sources or infinitely thin disk sources directly on the skin. Soon after the initial release of VARSKIN, however, the industry encountered a "new" type of skin contaminant, which consisted of discrete microscopic radioactive particles, called "hot" particles.

These particles differ radically from uniform skin contamination, in that hot particles have a thickness and many of the exposures result from particles on the outside of protective clothing. Therefore, the code required further modifications.

VARSKIN MOD 2

VARSKIN Mod 2, developed in the early 1990s, significantly enhanced the code by adding the ability to model three-dimensional sources (cylinders, spheres, and slabs) with materials placed between the source and skin (including air gaps that attenuate the beta particles).

The code also modeled hot particle photon doses in certain cases. In addition, VARSKIN Mod 2 incorporated a user interface that greatly simplified data entry and increased efficiency in calculating skin dose.

VARSKIN 3

VARSKIN 3, released in 2004, operates in a Microsoft Windows environment and is designed to be significantly easier to learn and use than VARSKIN Mod 2.

In addition, this release enables users to calculate the skin dose (from both beta and gamma sources) attributable to radioactive contamination of skin or protective clothing.

The code also offers the ability to compute the dose at any skin depth or skin volume, with point, disk, cylindrical, spherical, or slab (rectangular) sources, and even enables users to compute doses from multiple sources.

Figure 6.11 shows a typical VARSKIN 3 input screen for point source geometry.

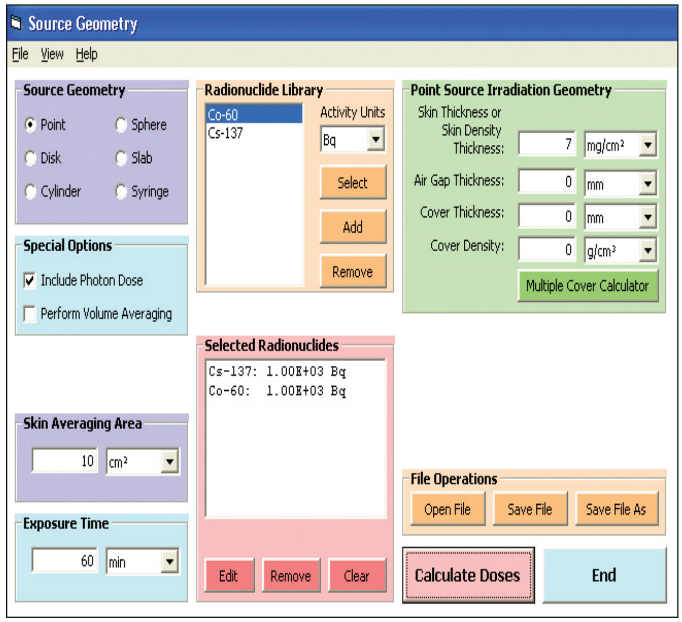

Figure 6.10: Point source geometry screen

The input data file was also modified for VARSKIN 3 to reflect current physical data, include the dose contribution from internal conversion and Auger electrons, and allow a correction for low-energy electrons.

CURRENT STATUS

Since the release of VARSKIN 3, the NRC staff has compared its dose calculations, for various energies and at various skin depths, with doses calculated by the Monte Carlo N-Particle Transport Code System (MCNP) developed by the Los Alamos National Laboratory. The comparison shows that VARSKIN 3 overestimates the dose with increasing photon energy.

For that reason, the NRC is currently sponsoring further enhancement of the code to replace the existing photon dose algorithm and develop quality assurance methods for this model.

Upgrades to the updated VARSKIN will include the following:

- an enhanced photon dosimetry model that is based on Monte Carlo simulations of hot-particle contamination
- mathematical formulations rather than look-up tables to drive the estimation of dose

- dose averaging to provide efficient convergence of the solution
- incorporation of parameters of energy, attenuation, dose-averaging area, and air gap
- protective clothing thickness, as well as simple volumetric sources

Code developers have also addressed deficiencies in the current code by creating the capability to calculate dose while accounting for attenuation and correcting the assumption of using the same effective-Z for all materials.

FUTURE UPDATES

- Correct technical issues with the beta dose model reported by the code users.
- Develop a quality assurance program for beta dose model.
- Develop a training module for using the code.
- The Web site link for the code is as follows:

http://www.nrc.gov/about-nrc/regulatory/research/comp-codes.html#health

VARSKIN 3 is available from the Radiation Safety Information Computational Center. For additional information, see NUREG/CR-6918, "VARSKIN 3: A Computer Code for Assessing Skin Dose from Skin Contamination," issued October 6, 2006.

For More Information
Contact Mohammad Saba, RES/DSA, at 301-251-7558 or Mohammad.Saba@NRC.gov

RADIOLOGICAL TOOLBOX

Background

The NRC, in conjunction with Oak Ridge National Laboratory, developed the Radiological Toolbox as a means to quickly access databases needed for radiation protection, shielding, and dosimetry calculations.

The toolbox is essentially an electronic handbook. It contains data of interest to the health physicist, radiological engineer, and others working in fields involving radiation. Examples of data contained in the toolbox include the following:

- radioactive decay data
- biokinetic data
- internal and external dose coefficients
- elemental composition of many materials
- radiation interaction coefficients
- kerma coefficients
- other tabular data of interest

The toolbox includes a means to export the tabular data to an Excel worksheet for use in other calculations. It operates in a Windows environment.

Approach

The Radiological Toolbox, hereafter referred to as the Rad Toolbox or simply the toolbox, is a computer application that provides access to physical, chemical, anatomical, physiological, and mathematical data (and models) relevant to the protection of workers and the public from exposures to ionizing radiation.

A graphical interface enables viewing of the data and the means to extract data for further use in computations and analysis. The numerical data, for the most part, are stored in International System (SI) units. However the user can display and extract the data using non-SI units.

The data are stored in Microsoft Access databases and in flat ASCII files. The toolbox features additional computational capabilities and numerical data of interest.

The following data elements are included:

- nuclear decay data—International Commission on Radiological Protection (ICRP) 38 (ICRP 1983) and the Japan Atomic Energy Research Institute (Endo 1999, 2001)
- dose coefficients for photon and neutron fields—ICRP Publication 74 (ICRP 1996b)
- organ masses values (ICRP 72) and reference values (ICRP 89)
- radiation workers—ICRP Publications 30 and 68 (ICRP 1978, 1994)
- members of the public—ICRP Publication 72 (ICRP 1996a)
- external irradiation—Federal Guidance Report 12 (U.S. Environmental Protection Agency, 1993)

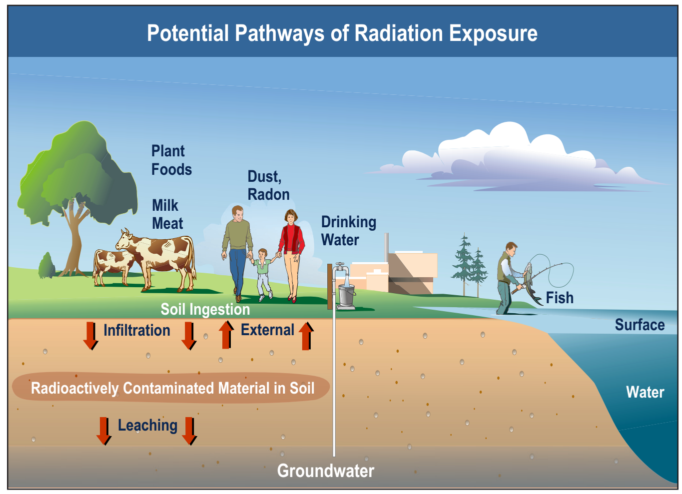

Figure 6.11: Radiological Toolbox Poster Board Presentation

For the most part, the Rad Toolbox accesses numerical databases and converts the requested values to the units specified by the users.

Computational modules are included to calculate inhalation dose coefficients for deterministic effects over the time period specified by the user and to compute radiation interaction coefficients for materials based on their elemental composition.

The software's help files provide access to textual information on topics ranging from those of a general nature to the details of models describing the fate of radionuclides in the body.

TOOLBOX CONTENT

When the toolbox is initiated, a user screen appears.

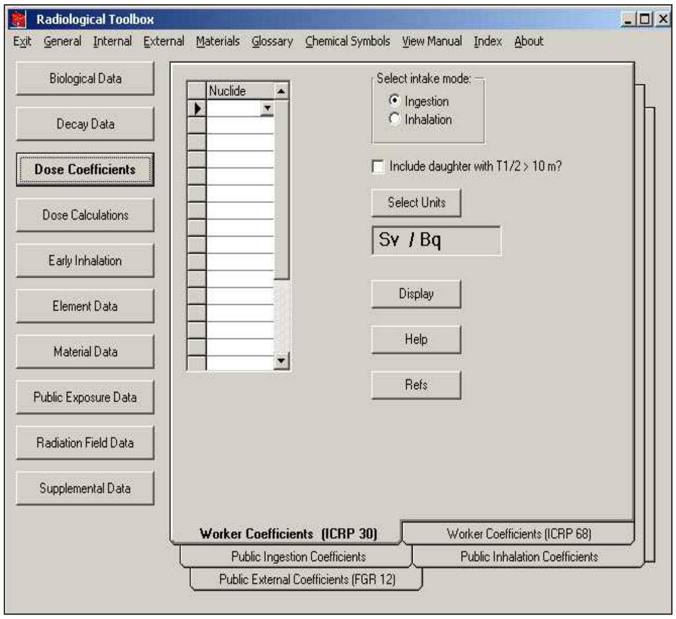

Figure 6.12 Radiological Toolbox Graphical User Interface

The menu bar at the top of the screen allows access to the software help files in addition to other standard functions.

The menu bar at the left of the screen allows access to all data elements that are included in the toolbox.

For example, the Dose Coefficients section of the toolbox provides access to the following sets of nuclide-specific dose coefficients:

- external dose rate coefficients for 826 radionuclides from Federal Guidance Report 12 (EPA 1993)
- committed dose coefficients for inhalation and ingestion intakes of 738 radionuclides by workers in ICRP Publications 30 and 68 (ICRP 1978, 1994)
- age-dependent committed dose coefficients for the inhalation and ingestion intakes of 738 radionuclides by members of the public (six ages at intake) in ICRP Publication 72 (ICRP 1996a)

For each set of coefficients, it is possible to display up to 20 nuclides at a time for a chosen route of exposure or intake.

Future Updates

Further revisions of the toolbox are planned as the NRC staff and other users identify the need for additional data.

The program and user manual can be downloaded from the NRC public web site at:
http://www.nrc.gov/about-nrc/regulatory/research/radiological-toolbox.html

For More Information

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References:

Endo, A., T. Tamura, and Y. Yamaguchi. 1999. *Compilation of Nuclear Decay Data Used for Dose Calculation Revised Data for Radionuclides Not Listed in ICRP Publication 38,* JAERI-Data/Code 99-035.

Endo, A., and Y. Yamaguchi. 2001. *Compilation of Nuclear Decay Data Used for Dose Calculation Revised Data for Radionuclides Listed in ICRP Publication 38,* JAERI-Data/Code 2001-004.

Environmental Protection Agency (EPA). 1993. *External Exposure to Radionuclides in Air, Water, and Soil. Federal Guidance Report 12*, U.S. Environmental Protection Agency, Washington, DC.

International Commission on Radiological Protection (ICRP). 1978. *Limits for Intakes of Radionuclides by Workers.* ICRP Publication 30. Annals of the ICRP Vol. 2. International Commission on Radiological Protection, Pergamon Press, New York.

International Commission on Radiological Protection (ICRP). 1994. *Dose Coefficients for Intakes of Radionuclides by Workers.* ICRP Publication 68. Annals of the ICRP 24(4). International Commission on Radiological Protection, Pergamon Press, New York.

International Commission on Radiological Protection (ICRP). 1996a. *Age-dependent Doses to the Members of the Public from Intake of Radionuclides: Part 5 Compilation of Ingestion and Inhalation Coefficients.* ICRP Publication 72. International Commission on Radiological Protection, Pergamon Press, New York.

International Commission on Radiological Protection (ICRP). 1996b. *Conversion Coefficients for Use in Radiological Protection Against External Radiation.* ICRP Publication 74. International Commission on Radiological Protection, Pergamon Press, New York.

INTEGRATED GROUND WATER MONITORING

Background

RES is working with NRC licensing offices (the Office of Federal and State Materials and Environmental Management Programs, Office of New Reactors, and the Office of Nuclear Reactor Regulation) and the regions to develop guidance for reviewing ground water monitoring programs, as required in Title 10 of the Code of Federal Regulations Section 20.1406(a) and (b), Section 50.75(g), Appendix A to Part 40, Section 61.53, and Section 63.131. In November 2007, RES issued NUREG/CR-6948, "Integrated Ground-Water Monitoring Strategy for NRC-Licensed Facilities and Sites," which provides the technical bases for this guidance. Also in 2007, the Nuclear Energy Institute (NEI) issued its industry initiative on ground water protection that includes onsite ground water monitoring at all nuclear reactor sites. NEI funded the Electric Power Research Institute (EPRI) in the development of guidelines for ground water protection that were issued in January 2008. NRC regional inspectors working with NRR and RES staff will use the RES-developed information to review these new programs in conjunction with existing offsite radiological environmental monitoring programs. This monitoring is needed to detect radionuclide releases and to evaluate the need for and selection of remediation approaches.

Overview

NUREG/CR-6948 documents the development and testing of an integrated ground water monitoring strategy. It integrates conceptual site model (CSM) confirmation with ground water monitoring through the use of performance indicators (PIs) (e.g., concentrations, water fluxes in the unsaturated and saturated zones). It outlines procedures for selecting, locating, and calibrating field instruments and methods to detect radionuclide releases in the subsurface and to determine the need and effective approaches for remediation.

Approach

The strategy provides an integrated and systematic approach for monitoring subsurface flow and transport beginning at the land surface and extending through the unsaturated zone to the underlying water-table aquifer. The strategy is robust and useful for reviewing site- and facility-specific ground water monitoring programs to do the following:

- Assess the effectiveness of contaminant isolation systems and remediation activities.
- Communicate to decisionmakers and stakeholders the monitored PIs through effective data management, analysis, and visualization techniques.
- Detect and identify the presence of contaminant plumes and preferential ground water transport pathways.
- Test alternative conceptual and numerical flow and transport models.
- Aid in the confirmation of the assumptions of the CSM and, hence, the performance of the facility through monitoring of PIs.

The documented strategy provides the technical bases, along with identified guidance and analytical tools, for assessing the completeness and efficacy of an integrated ground water monitoring program. It focuses on quantifying uncertainties in the hydrologic features, events, and processes using "real-time, near-continuous" monitoring data to confirm the CSM. The strategy links the ground water monitoring program to the detection level required for early warning of releases.

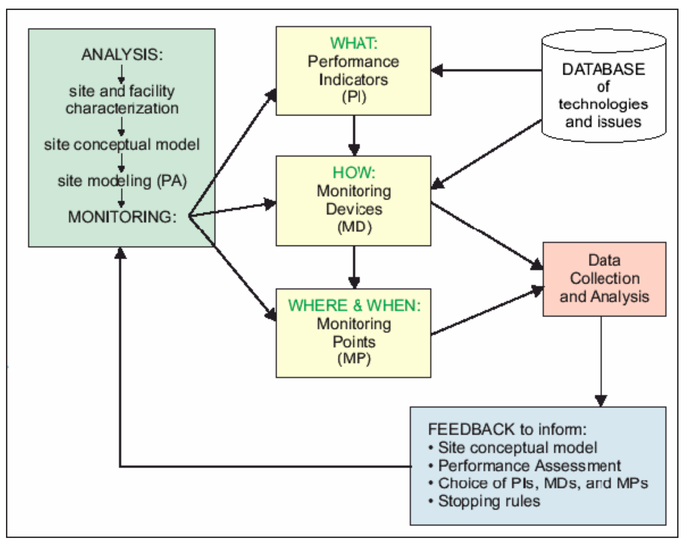

Figure 6.13: Flow chart of the integrated monitoring strategy from NUREG/CR-6948

RES and its contractor, Advanced Environmental Solutions, LLC (AES), developed and tested the strategy to demonstrate how to identify and monitor PIs of the subsurface flow and transport system behavior. Using field case studies, the strategy illustrates how these methods coupled to the CSM models can provide early detection of releases. The strategy was tested for a range of complex hydrogeologic settings using field monitoring datasets to demonstrate its validity and usefulness for reviewing nuclear facility issues.

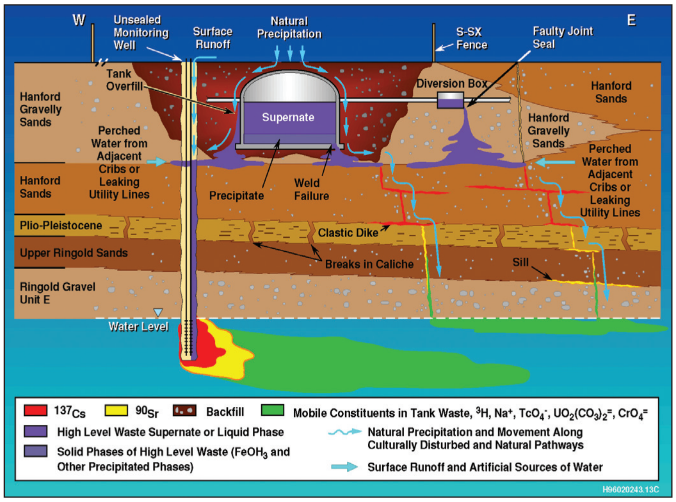

Figure 6.14: Illustration of a conceptual model of a complex field site for which monitoring can facilitate decisionmaking for remediation (Ward, A. L.; Gee, G. W.; White, M. D. "A Comprehensive Analysis of Contaminant Transport in the Vadose Zone Beneath Tank SX-109," PNNL-11463. Pacific Northwest National Laboratory, Richland, Washington, February 1997.)

Research Activities

AES examined the state-of-the-practice in ground water monitoring of radionuclides for confirming CSM models. Figure 6.14 shows the natural and engineered complexities that can affect subsurface flow and radionuclide transport. Monitoring strategies need to consider these complexities in the development and testing of conceptual models. Monitoring involves detection and sampling both above (i.e., in the unsaturated zone) and below the water table and, as illustrated in the figure, must not introduce inadvertent pathways. AES drew lessons learned from field case studies involving site-specific contaminant sources, release modes, and hydrogeologic and geochemical conditions affecting transport. AES tested these field studies using trend analyses and statistical methods to determine the frequency and duration of monitoring to confirm regulatory criteria. The testing also evaluated ground water flow and transport modeling assumptions in the CSMs for the various field examples. AES examined which technical bases in monitoring are useful for determining the need for remediation and ways to confirm the efficacy of remediation in interdicting and mitigating ground water contamination in the unsaturated and saturated zones.

In documenting the strategy, AES outlined the logic for selecting the appropriate sensors and geophysical technologies, monitoring locations and frequency, and analysis methods to confirm the CSMs and their assumptions. The tools and technical bases developed emphasize relevancy to decommissioning nuclear waste and reactor facilities. AES presented technology transfer seminars to NRC staff and Agreement State regulators on the strategy and case studies relevant to radionuclide transport assessments.

For More Information

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ENVIRONMENTAL TRANSPORT RESEARCH PROGRAM

Background

Many activities that are part of nuclear-material and nuclear-fuel cycles have the potential to expose the environment or the public to low levels of contamination from nuclear materials. Environmental assessment and protection address the vulnerability of environmental resources and public health to potential chronic exposure to radionuclides associated with nuclear facilities, including nuclear reactor, fuel cycle, waste disposal, and decommissioned facilities.

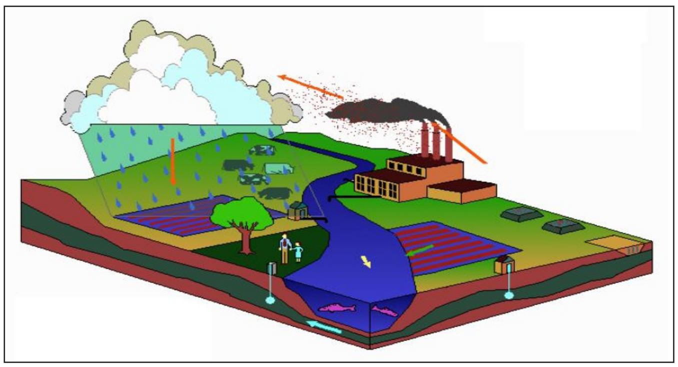

Figure 6.15: Conceptual visualization of contaminant pathways

Technical Issues

Monitoring and modeling of environmental systems at nuclear facilities are evolving in response to changing needs, increased understanding of environmental systems, and advances in technology. Issues associated with environmental monitoring include identification of potential sources and measurable indicators of system performance that can be coupled to regulatory requirements. Traditional analyses have involved conservative assumptions that often led to costly solutions. One goal of the research on environmental transport is to reduce conservatism and the associated overprediction of environmental impact implicit in current environmental assessments.

Specific Regulatory Needs

The program explicitly addresses needs imposed by risk-informed regulation. Individual research activities address needs identified by current regulatory programs, the new reactor licensing program, or the advanced reactors program. Research into the performance of reinforced-concrete or cement barriers supports assessment of reactor life extension and the performance of engineered disposal facilities.

The NRC licensing staff needs updated or new technical bases for reviewing site characterization, monitoring, modeling, and remediation programs submitted by current and prospective licensees. Regulatory guidance is needed on environmental assessments and performance monitoring associated with new reactors and the decommissioning of nuclear facilities.

Principal Research Activities

RELEASE OF RADIONUCLIDES FROM WASTES OR ENGINEERED STRUCTURES AND ADVANCED SAMPLING AND MONITORING OF RADIONUCLIDE RELEASES

The potential for chronic releases of radionuclides to the environment from nuclear facilities must be understood to ensure compliance with NRC regulations. Assessing long-term releases under varying chemical and physical conditions is a difficult but important aspect of ensuring that current or planned nuclear facilities conform to regulatory goals. Thus, research activities are planned to monitor, characterize, and model the behavior of radionuclide-containing materials in the environment including assessment of in situ sensors for real-time monitoring of radionuclides in the environment and field lysimeter studies at waste disposal and waste-incidental-to-reprocessing (WIR) sites using actual low-level waste (e.g., activated metals, ion exchange resins) and cement-solidified WIR samples for several years.

LONG-TERM BEHAVIOR OF ENGINEERED MATERIALS

The expectation of future use of engineered materials to isolate radioactive wastes or environmental contaminants results in a need for analytical tools to assess the design and performance of cement, concrete, and natural earth materials in engineered structures.

GEOTECHNICAL CONSIDERATIONS FOR NEW AND ADVANCED NUCLEAR POWER PLANTS

The recognition that design and construction of new or next-generation facilities can enhance or inhibit the release, migration, and isolation of materials in the geosphere is addressed by research activities to improve the understanding, modeling, and monitoring of the performance of engineered features of new facilities.

ADVANCED MODELING FOR ENVIRONMENTAL ASSESSMENT

Advances in computational facilities are enabling research to incorporate additional realism in the assessment of geochemical and biochemical processes that enhance or retard radionuclide transport. Additional realism significantly enhances the prospects for meaningful validation of system or subsystem models used for environmental assessment. Research on computational tools is focused on a generic framework for linking databases, models, and other analytic tools for flexible problem solving.

DECISION SUPPORT FOR GROUND WATER REMEDIATION

Technologies for the remediation of subsurface contamination have advanced significantly in recent years. Likewise, advances in understanding and manipulating subsurface biota are leading to advances in exploiting the ability of biota to remediate subsurface contamination. Research is planned to examine the efficacy of long-term performance of these remediation technologies and to provide tools to assist in remediation planning.

EVOLUTION OF FUTURE LAND USE IN THE VICINITY OF NUCLEAR FACILITIES

Over the past few decades, experience with remote sensing of the environment over large scales has led to advances in forecasting the evolution of land usage. Research activities are planned to exploit these advances for the purpose of forecasting how the use of land surrounding nuclear facilities might contribute to or limit the risk of chronic radionuclide exposures.

Collaborative Efforts and Opportunities

The environmental transport research program leverages resources through cooperative interactions and special research agreements (e.g., Memorandum of Understanding on Research and Development of Multimedia Environment Models; see http://sites.google.com/a/environmental-modeling.org/environmental-modeling/) with other national and international research organizations pursuing related work. The technical objective is to collaborate on or gain access to technologies, databases, computer software, lessons learned, and methods that support the NRC's regulatory activities. Collaborators include other Federal agencies (e.g., the U.S. Department of Agriculture's Agricultural Research Service, U.S. Geological Survey, National Institute of Standards and Technology, U.S. Environmental Protection Agency, U.S. Army Corps of Engineers), DOE national laboratories, universities, National Academy of Sciences, National Academy of Engineers, professional societies (e.g., American Nuclear Society, American Geophysical Union, International Association of Hydrological Sciences, National Ground Water Association, American Society for Testing and Materials) and international organizations (e.g., the Nuclear Energy Agency of the Organization for Economic Cooperation and Development and the International Atomic Energy Agency).

These cooperative ventures help to identify important research findings, datasets, and lessons learned for use in evaluating and testing multimedia environmental models, examining the role of engineered barrier systems in waste disposal, and evaluating the practicality of modeling chemical sorption in environmental systems. Interactions with professional societies assist in developing guidance and training programs. Knowledge management also profits from interactions with other Federal and professional organizations and from their information sources (e.g., technical journals, Web sites, and monographs).

For More Information
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CHAPTER 7: REACTOR SAFETY CODES AND ANALYSIS

Thermal-Hydraulic Simulations of Operating Reactors

Symbolic Nuclear Analysis Package Computer Code Applications

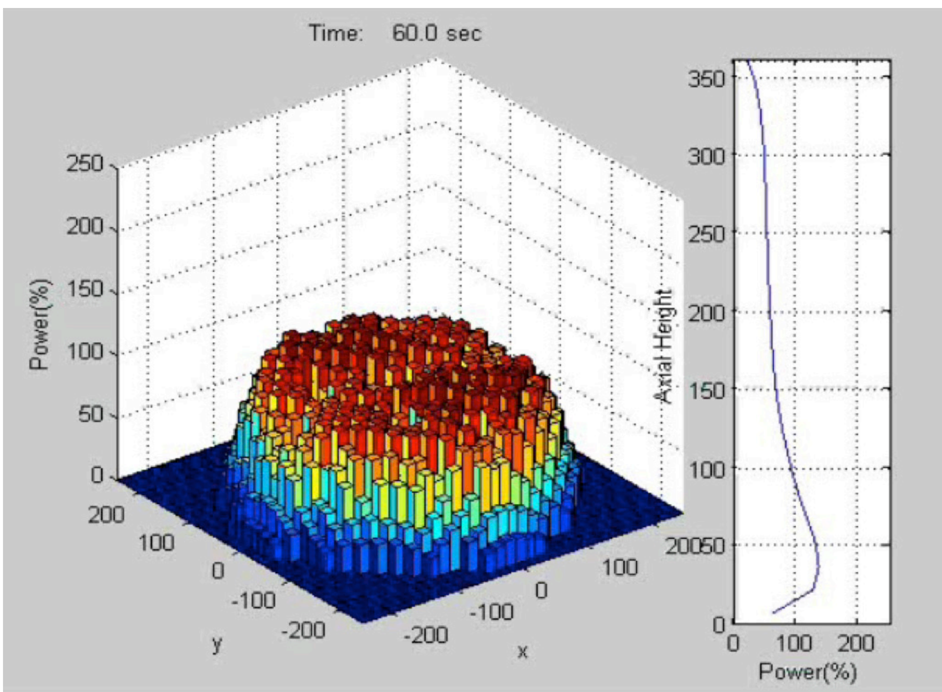
TRAC/RELAP Advanced Computational Engine
Thermal-Hydraulics Code

Code Application and Maintenance Program

Nuclear Analysis

High-Burnup Light-Water Reactor Fuel

Fission Products Burnup Credit



Boiling-Water Reactor Power Oscillation Map Modeled by the PARCS Computer Code for the Ringhals BWR Instability Project

THERMAL-HYDRAULIC SIMULATIONS OF OPERATING REACTORS

Background

RES provides the tools and methods used by program offices in the review of licensee submittals and the evaluation and resolution of safety issues. For thermal-hydraulic analyses, the TRAC/RELAP Advanced Computational Engine (TRACE) computer code is used to perform the following:

- confirmatory calculation reviews of licensee submissions, such as those for extended power uprates
- exploratory calculations to establish the technical bases for rule changes, such as the proposed redefinition of the emergency core cooling system rule in Title 10 of the Code of Federal Regulations 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors"
- exploratory calculations for the resolution of generic safety issues, such as Generic Safety Issue 191, "Assessment of Debris Accumulation on PWR Sump Performance"

RES is developing a library of TRACE input decks for simulating currently operating pressurized-water reactors (PWRs) and boiling-water reactors (BWRs).

Approach

TRACE plant input decks are prepared for specific simulations. These can be design-basis loss-of-coolant accidents (LOCAs), anticipated operational occurrences, anticipated transients without scram, and other transients. Depending on the simulations to be performed, the size and complexity of plant input decks can range from single system components up to the entire nuclear steam supply system. TRACE is able to simulate the multifaceted evolution of these events, capturing all of the major system operations and thermal-hydraulic processes that unfold.

Each physical piece of equipment in a plant can be represented as some type of TRACE component, and each component can be further nodalized into a number of physical volumes—also called cells—over which the fluid, conduction, and kinetics equations are averaged. TRACE input decks representing entire plants consist of an array of one-dimensional and three-dimensional TRACE components arranged and sized to match plant specifications.

Because of the modeling flexibility available to the user, best-practice modeling guidelines have been published in the "TRACE Users Guide" [1]. The user guide shows modelers the most effective methods of arranging generic one-dimensional components to depict particular systems and how to employ function-specific components, such as the PWR accumulator and pressurizer and the BWR jetpump and channel components, to achieve desired results.

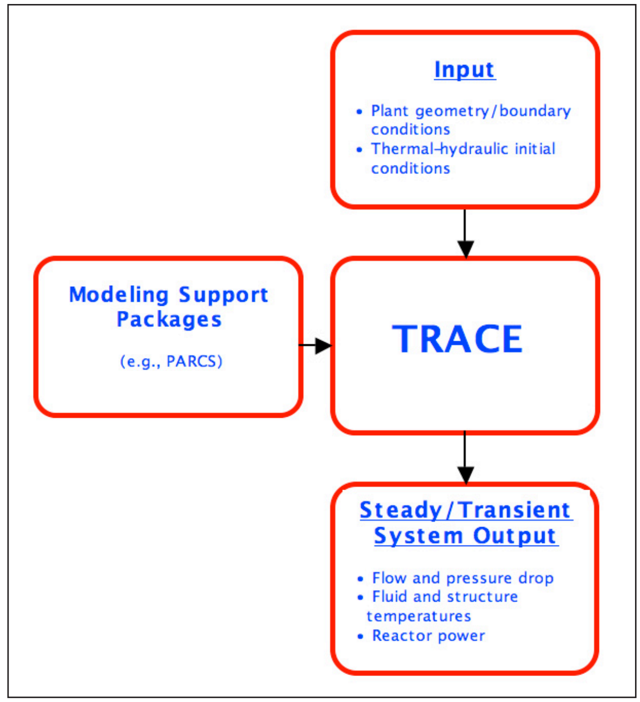


Figure 7.1: TRACE is an advanced, best-estimate reactor system code that is used to model the thermal-hydraulic performance of nuclear power plants.

User input includes plant geometry and process conditions (temperature, flow, etc.). The code supports integration with detailed modeling packages (e.g., the three-dimensional kinetics code, PARCS) that can be used to model specific performance issues, including neutronics.

Once the arrangement of the plant deck has been completed and each component has been set with initial values for normal operating pressures, temperatures, and flow conditions, TRACE is run in steady-state mode for a period of time to test the model and to develop appropriate steady-state initial conditions for the specified operating state and boundary conditions. TRACE models transients and accidents by simulating an initiating event after steady initial conditions have been reached. Developmental assessments support the applicability of TRACE in modeling these events [2].

Recently, plant input decks developed for other system codes have been updated and converted into TRACE to support the licensing reviews of extended power uprate applications. These models are being used to assess the effects of increased power on system behavior and safety margins.

BWR MODELS

Representative LOCA and design-basis accident input decks have been developed for most General Electric type BWRs, including the BWR3, BWR4, and BWR5 plants (see Figure 7.2). TRACE component specific features were significantly enhanced to improve the modeling of containment pressurization and feedback during design-basis events.

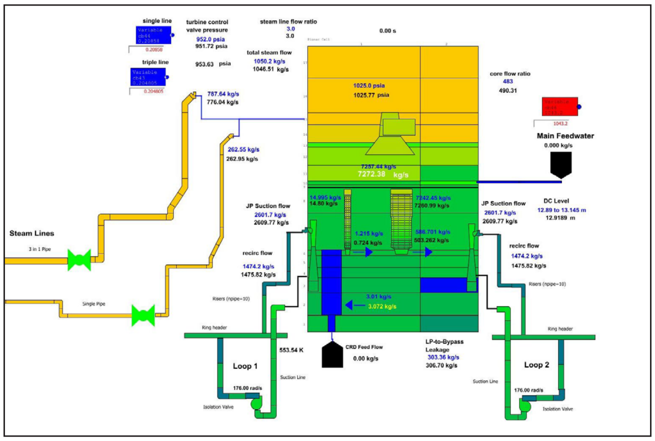


Figure 7.2: Steady-state conditions in a BWR (Symbolic Nuclear Analysis Package [SNAP] animation)

PWR MODELS

Representative LOCA and design-basis accident models exist for Westinghouse PWRs with two, three, and four loops, several Combustion Engineering plants, and two Babcock and Wilcox plants (see Figure 7.3).

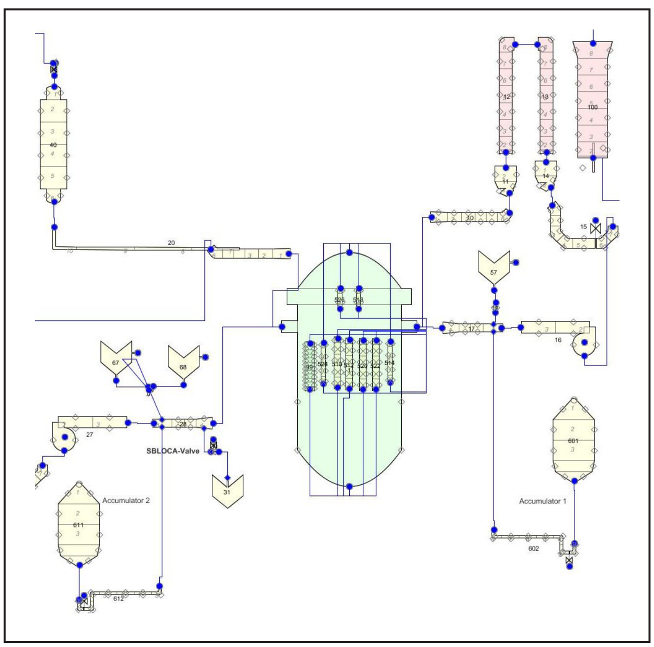


Figure 7.3: Key primary coolant thermal-hydraulic components, including reactor vessel, pumps, and steam generator, for a two-loop PWR depicted with SNAP

Building a comprehensive library of plant input decks will enhance the ability of the NRC staff to perform timely and defensible confirmatory analyses in support of regulatory decisions.

REFERENCES

1. TRACE Users Guide
2. TRACE 5.0 Assessment Manual

For More Information

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SYMBOLIC NUCLEAR ANALYSIS PACKAGE COMPUTER CODE APPLICATIONS

Background

The NRC recognizes that analytical capability and expertise are essential to ensuring design adequacy and safe operation of nuclear power plants. One part of accomplishing this mission is the analysis of operational and accident transients using thermal-hydraulic modeling software. The NRC has developed and uses several computer codes in its safety analyses of pressurized- and boiling-water reactor nuclear power plants. The input models for most of these codes are text based, requiring the user to write an input file (or deck) in a text editor and then run the analysis program. These input files are often very complex, difficult to read, and time consuming to prepare. Additionally, each computer code uses different input formats and variable names. This adds to the burden on the analysts, who usually use more than one of these modeling programs to perform a review. To lessen this burden, the NRC decided that it would be cost effective to develop a single, standardized graphical user interface (GUI), which could be extended for use with any analytical code.

An NRC analyst reviewing, for example, a power plant modification must perform several analyses using NRC thermal-hydraulic computer codes. The analyst needs to perform this analysis as efficiently and as error free as possible. Until the development of the Symbolic Nuclear Analysis Package (SNAP), the most efficient way for analysts to accomplish their work was to learn several cumbersome input formats. They also needed to use several different software packages for displaying and interpreting the results. The analyst was forced to spend much time preparing text-based input files and transferring information from one application to another. These efforts were very prone to errors, which could affect results.

SNAP eliminates the need for analysts to use the text-based entry methods and to transfer or replicate data between several different packages. It does this by providing a powerful, flexible, and easy-to-use GUI, both for preparation of analytical models and for interpretation of results. Since the core look and feel of SNAP is the same for several different programs, the analyst does not have to learn and remember several different interfaces and therefore is less likely to make an error because of differences in input formats. Currently, SNAP has interfaces for RELAP5, TRACE, CONTAIN, MELCOR, RADTRAD, and FRAPCON3.

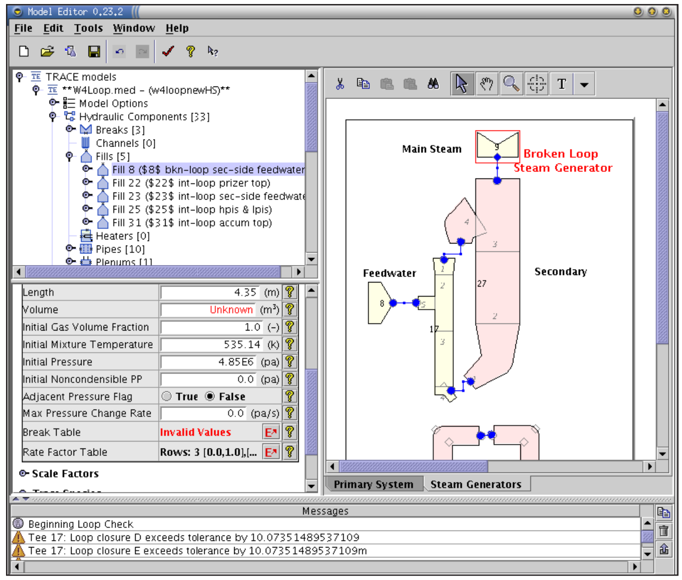

Figure 7.4: Creating input models using SNAP

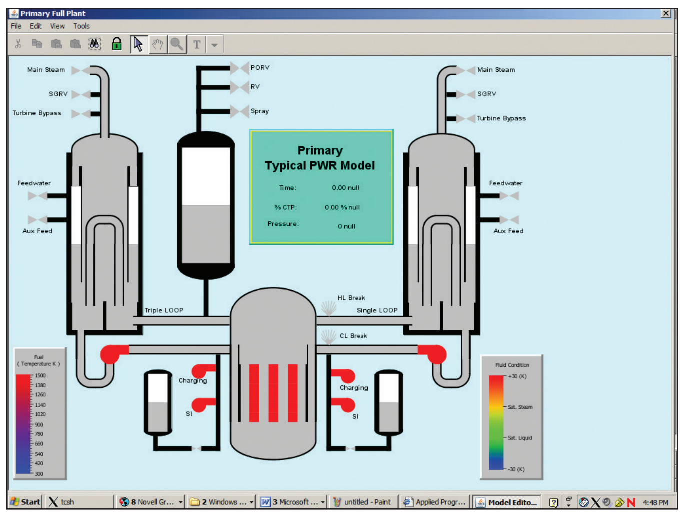

Figure 7.5: Animating analysis results using SNAP

Approach

DEVELOPMENT

The SNAP code has moved from "beta" development to its first full release version; Version 1.0 of SNAP was officially released in October 2008.

Over the past year, SNAP has greatly improved, in both available functionality and usability. Among the improvements of greatest note are the following:

- A RELAP5 to TRACE plug-in was developed. This plug-in allows a user to directly convert RELAP5 input models into TRACE input models. Direct conversion of RELAP5 models to TRACE models is extremely beneficial to TRACE analysts, as existing RELAP5 models for many plants and accident scenarios can now be used in TRACE with little or no additional modification.

- Compatibility with the current versions of TRACE was maintained during the recent TRACE development efforts. In addition, SNAP now supports a “developmental” option that allows users to select the developmental version of TRACE they want to use. This feature is very important to analysts who are assessing new TRACE models and features.

- NAP has a new installer that now allows users to select only the plug-ins that they need, rather then having to download the entire SNAP distribution.

- An active SNAP users group, both inside and outside the NRC, has been fostered and, based on user requests, many improvements have been made to SNAP.

- In June 2009, the NRC published programming documentation describing the SNAP application programming interface (API) as NUREG/CR-6974, “Symbolic Nuclear Analysis Package (SNAP): Common Application Framework for Engineering Analysis (CAFEAN) Preprocessor Plug-in Application Programming Interface.” Third-party developers may use this API documentation to develop their own SNAP-based plug-ins.

APPLICATION

Many analysts using TRACE have now adopted SNAP; to a lesser extent, analysts using RELAP5, CONTAIN, MELCOR, RADTRAD, and FRAPCON3 have also adopted the package. SNAP continues to gain acceptance and use throughout the agency, as well as in other organizations involved with nuclear analysis.

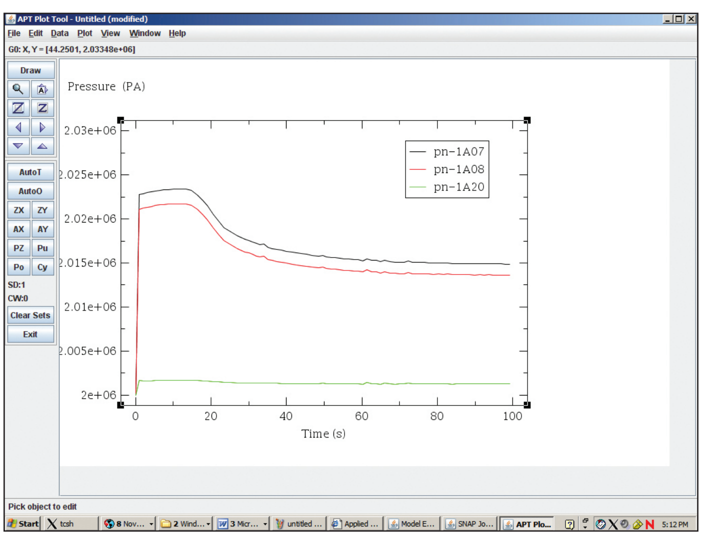

Figure 7.6: Plotting analysis results using SNAP

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TRAC/RELAP ADVANCED COMPUTATIONAL ENGINE THERMAL-HYDRAULICS CODE

Background

The NRC uses thermal-hydraulic codes to perform operational and accident transient analyses. Before the late 1990s, the NRC developed and used four system computer codes—RELAP5, TRAC-PWR, TRAC-BWR, and RAMONA—to perform independent safety analyses of pressurized- and boiling-water reactor (PWR and BWR) nuclear power plants. These computer codes used architecture and modeling methods developed in the 1970s. The NRC decided that it would be more cost effective to maintain a single modernized computer code that could be used to analyze all the reactor designs and operational conditions addressed by the four older computer codes.

To meet this goal, over the last 10 years, the NRC undertook a project to consolidate the above four analysis codes into a single, modernized computational platform. The code consolidation project began with the vision to have the capability to perform thermal-hydraulic safety analysis in the future that allows for solutions to the full spectrum of important nuclear safety problems in an efficient and effective manner, taking complete advantage of state-of-the-art modeling, hardware, and software capabilities. In other words, the agency must be able to do more with less:

LESS: The NRC must be able to reduce and consolidate personnel resources needed for solving any given problem and for maintaining code capability by developing and/or improving the following:

- ease-of-use
- speed
- robustness
- flexibility
- maintainability/upgradability

MORE: The NRC must be able to accommodate the new challenges and demands for best-estimate thermal-hydraulic analysis coupled to other related capabilities:

- accuracy
- flexibility
- maintainability/upgradability
- simplicity
- expanded scope of capabilities
- quality assurance

Version 5.0 of the TRAC/RELAP Advanced Computational Engine (TRACE) is the culmination of that effort. It is used to analyze operational and safety transients such as small- and large-break loss-of-coolant accidents (LOCAs), in PWRs and BWRs, as well as the interactions between the related neutronic and thermal-hydraulic systems.

The thermal-hydraulic and neutronic capabilities of TRACE v5.0 enable the NRC to perform independent evaluations of transients for existing and new reactor designs. The NRC uses these capabilities to perform sensitivity assessments of system hardware and phenomena, which can be modeled using different analytical or modeling approaches.

Approach

Development and assessment are an ongoing process. In 2008, work was completed to address modeling issues identified during (1) an independent peer review, (2) the development of input models used to support the licensing of new and operating reactors, and (3) code assessment activities leading to the release of Version 5.0. These efforts ultimately led to the release of TRACE v5.0 Patch 1 in October 2008.

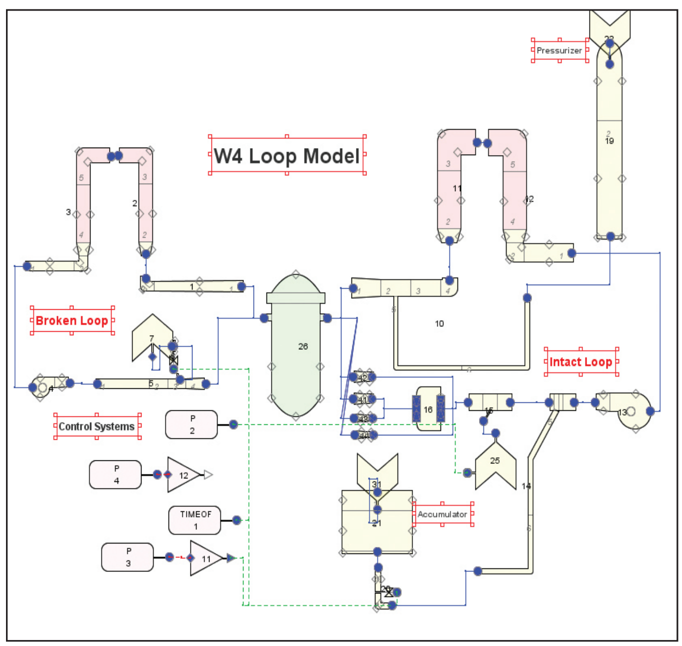

Figure 7.7: Typical plant model nodalization

MODELING CAPABILITIES

The code features a two-fluid, compressible, nonequilibrium hydrodynamics model that can be solved across a one-, two-, or three-dimensional mesh topology. It also features a three-dimensional reactor kinetics capability through coupling with Purdue's Advanced Reactor Core Simulator (PARCS). The code can perform any type of reactor analyses previously performed by any of the predecessor codes and has component models and mesh connectivity that allow a full reactor and containment system to be easily modeled (Figure 7.7 shows a typical reactor system nodalization for TRACE.)

The consolidation project has resulted in many new features being added to the code. The most notable achievements include the addition of a plethora of BWR-specific component types, a single junction component (to capture RELAP5-style mesh connectivity), three-dimensional kinetics (through coupling with PARCS), a new heat structure component, an improved set of constitutive models for reflood, condensation, and other basic phenomena, an improved level tracking model, numerous usability enhancements, and countless bug fixes.

A significant advance in the modeling capability of TRACE is the addition of a parallel processing capability that allows the code to communicate with itself or other codes. This feature is known as the exterior communications interface (ECI). The ECI is a request-driven interface that allows TRACE to communicate with any code that implements the ECI, without actually having to modify TRACE. The ECI has allowed TRACE to be easily coupled to codes such as SNAP, CONTAIN, REMIX, and MATLAB. The interface should allow TRACE to be coupled to computational fluid dynamics or other special-purpose codes in the future.

TRACE DEVELOPMENT

TRACE uses a modern code architecture that is portable, easy to maintain, and easy to extend to new models to address future safety issues (Figure 7.8 shows a graphical representation of this architecture). TRACE has been run successfully on multiple operating systems, including Windows NT/2000/XP, Linux, and Mac OSX.

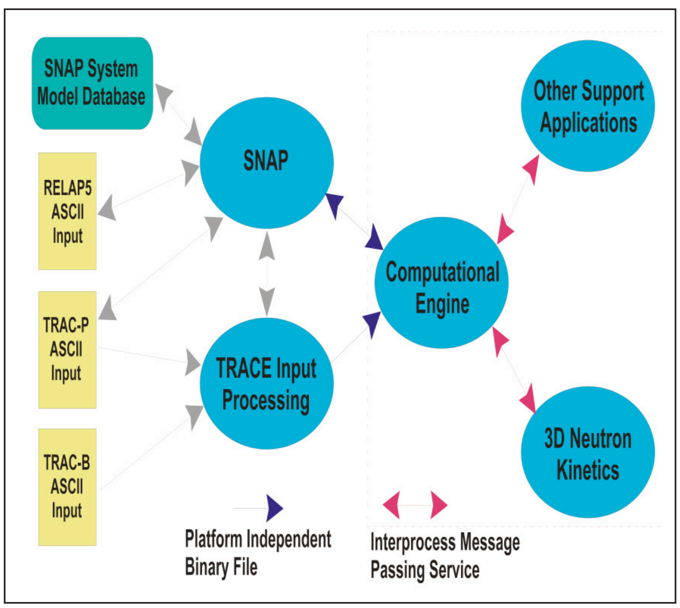

Figure 7.8: TRACE architecture

Code quality is the goal of a stringent development process. Some of the principal elements of this process include:

- configuration control
- establishment and strict enforcement of coding guidelines and development standards
- documented development process
    - software requirements document
    - software design and implementation
    - test plan
    - completion report
- three-tiered testing process
    - comprehensive regression set
    - automated robustness testing
    - automated code assessments
- multiplatform testing
- automated bug-tracking system

The final stage before any periodic official release of TRACE involves a thorough developmental assessment to identify any deficiencies in its physical models and correlations. New physical models may be developed as needs are identified.

The current assessment test matrix for TRACE contains more than 500 cases. The TRACE assessment test matrix contains a comprehensive set of fundamental, separate effects, and integral tests. These tests range from 1/1,000th scale to full scale and include new and advanced plant-specific experiments for both BWRs and PWRs. In addition to data from NRC-funded experiments, the assessment matrix also includes experimental data obtained through international collaboration, including experiments at the BETHSY, ROSA, and PANDA facilities. The set of experimental data against which TRACE has been validated is more comprehensive than that for any other NRC thermal-hydraulic code in terms of scope, quantity, and quality.

Improvements underway for future versions of TRACE focus on enhancing capabilities related to the simulation of advanced reactor designs, such as the U.S. Advanced Pressurized-Water Reactor, the Evolutionary Power Reactor, and the AP1000. Fixing bugs, addressing peer review findings, and improving code robustness and run-time performance are other areas receiving much attention. The TRACE development team recently released v5.0 Patch 1 to address some of the issues identified to date and plans additional patch releases. TRACE will provide a robust and extensible platform for safety analyses well into the future.

For More Information

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CODE APPLICATION AND MAINTENANCE PROGRAM

Background

In 1985, the International Code Assessment and Application Program (ICAP) was developed to assess and improve NRC thermal-hydraulic (T/H) transient computer codes. Approximately 14 nations signed bilateral cooperative agreements with the United States and provided contributions in the form of model development, code assessment, and information generated by application of the codes to operating nuclear power plants. Between 1985 and 1991, 14 ICAP management and specialist meetings were held. During this time, about 130 NUREG international agreement (NUREG/IA) reports were published on ICAP work in areas including core reflood, stratification in horizontal pipes, vertical stratification, postcritical heat flux, and blowdown and quench. A variety of test facilities were used to independently assess the codes. The information generated from this cooperative international work helped the NRC to improve the accuracy, reliability, and speed of its T/H codes. Input from the program also supported the development and application of the Code Scaling, Applicability, and Uncertainty (CSAU) code evaluation methodology in the late 1980s.

In the early 1990s, ICAP became the Code Application and Maintenance Program (CAMP). The CAMP agreement involved monetary contributions, in addition to in-kind technical contributions. The technical contributions include (1) sharing code experience and identifying areas for code and model improvements and (2) developing expertise in the use of the codes.

CAMP meetings are held twice each year, once in the United States and once abroad.

Approach

CAMP provides members with RELAP5, TRACE, PARCS, and SNAP codes. The RELAP5 and TRACE codes are the NRC's primary T/H reactor system analysis codes. PARCS is a multidimensional reactor kinetics code that can be coupled to TRACE and RELAP5. SNAP is a graphical user interface to the codes and provides pre-processing, runtime control, and post-processing capabilities. These codes are then used to perform analyses of accidents and transients in operating reactors, analyses to support resolution of generic issues, evaluation of emergency procedures and accident management strategies, confirmation of licensee's analyses, testing of the fidelity of NRC simulators, training exercises for NRC staff, and supporting analyses for the certification of advanced reactor designs.

During the biannual CAMP meetings, the members have an opportunity to present their technical findings. Specifically, the members (1) share experience with NRC T/H computer codes to identify code errors, perform code assessments, and identify areas for additional experiments, model development, and improvement, (2) maintain and improve user expertise, (3) develop and improve user application guidelines, (4) develop a well-documented T/H code assessment database, and (5) share experience in the use of the codes to resolve safety and other technical issues (e.g., scalability and uncertainty).

ACCOMPLISHMENTS

The CAMP and ICAP have generated more than 200 NUREG/IAs that contribute to the development, assessment, and application of the NRC T/H analysis codes. Technical areas span the entire range of accident and transient analysis, including low-pressure, low-power transients; advanced reactor design applications; coupling between the primary system and containment; operation of passive core cooling systems during accidents; boron dilution transients; neutronics coupling; reflood; condensation with noncondensables; and others. The reports document the contributions made to assessment, plant analysis, and physical model development.

In several recent cases, contributions to CAMP provided important code improvements and saved the NRC time and money. For example, analyses of proposed supercritical water reactor designs by CAMP members identified problems in the RELAP5 water properties near the critical point.

An effort is underway to improve the water properties near the critical point. (TRACE also uses the RELAP5 water properties.) Although the NRC is not currently analyzing supercritical water reactors, water properties near the critical point are important in calculations of anticipated transients without scram for pressurized-water reactors (PWRs). Another example of an efficiency gained resulted from the Republic of Korea's in-kind contributions on CANDU reactors, which were used during ACR-700 T/H code development. Korea's contribution allowed the NRC to start analyzing the ACR-700 during the preapplication review sooner than it could have without the Korean contributions. Korean modeling of the advanced accumulator in the AP1400 reactor design has helped guide the NRC efforts to model the advanced accumulator of the Mitsubishi Advanced Pressurized-Water Reactor, which has similar design features and is currently under design certification review by the Office of New Reactors.

FUTURE WORK

At the start of the CAMP agreement, the NRC used four primary T/H and reactor kinetics codes specifically designed for modeling transient and accident behavior in PWRs and boiling-water reactors (BWRs). The codes were developed using 1980-era computer languages and T/H modeling. In the late 1990s, the NRC began a code consolidation effort to merge the features of these codes into a new code using a modern software architecture that would more easily support the addition of modern T/H models and be easily portable to new computer hardware and operating systems. The new code would also reduce the personnel resources and money needed to maintain and improve multiple codes and the training costs to use multiple codes.

The new consolidated code, known as TRACE, is the primary T/H code used by the NRC for its reviews and audits of license amendments for operating reactors, advanced reactor license applications, generic safety issues, and power uprate requests.

CAMP members, who are experts in using and evaluating T/H codes, will continue to play a major role as an independent group with the necessary technical expertise to evaluate TRACE. Several CAMP members have started to use TRACE for in-kind technical contributions. CAMP members have shown good results in TRACE assessments of the ROSA and PKL integral test facilities, in separate effects condensation tests, and in the BWR Full-Size Fine-Mesh Bundle Test (BFBT) single-channel, steady-state, and transient tests. There have also been demonstrations of coupling TRACE to computational fluid dynamics. More TRACE contributions are expected in the future as the code matures. CAMP will become an important contributor to the future development and assessment of TRACE, as it will provide information to the NRC code development staff for use in improving TRACE's speed, accuracy, robustness, and usability. Ultimately, this will allow the NRC to better perform its reviews, analyses, and audits of licensee products and contribute to the protection of public health and safety.

For More Information

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NUCLEAR ANALYSIS

Background

As used here, the term "nuclear analysis" describes the use of analytical tools and experimental data to predict and understand the interactions of nuclear radiation and matter within various nuclear systems. Nuclear analysis thus encompasses the analyses of (1) fission reactor neutronics, both steady-state and dynamic, (2) nuclide generation and depletion as applied to predicting in-reactor and spent fuel decay heat power, fixed radiation sources, and radionuclide inventories potentially available for release, (3) radiation transport and attenuation as applied to the evaluation of material damage fluence, material dosimetry, material activation, radiation detection, and radiation protection, and (4) nuclear criticality safety (i.e., the prevention and mitigation of self-sustaining fission chain reactions outside reactors).

Approach

OVERVIEW

Nuclear analysis efforts support the staff's ongoing and anticipated nuclear safety evaluation activities for the licensing and oversight of existing reactors, front-end fuel cycle activities, and spent fuel storage, transport, and disposal systems, as well as proposed new and advanced reactors and their associated front-end and back-end fuel cycle activities. The primary nuclear analysis tools used for these activities are (1) the PARCS core neutronics simulator code, (2) the SCALE5.1 modular code system, and (3) the AMPX code for processing fundamental nuclear data in ENDF into code-usable libraries of continuous-energy or fine-group nuclear cross-sections and related nuclear data. When appropriate, planned nuclear analysis activities are integrated into larger NRC research plans for the respective applications.

IDENTIFICATION OF ISSUES AND NEEDS

An example of an area where additional data are needed for current and near-term activities is the burnup credit for the criticality safety analysis of spent fuel casks. For operating and new reactors, experimental data are needed for code validation and reduction of uncertainties. Such validation currently relies on limited data and/or code-to-code comparisons. The nuclear codes were recently validated for partial mixed-oxide fueling in pressurized-water reactors and are now being validated against plant operating and test data for use in steady-state and transient analysis of modern boiling-water reactor cores, including the economic simplified boiling-water reactor (ESBWR).

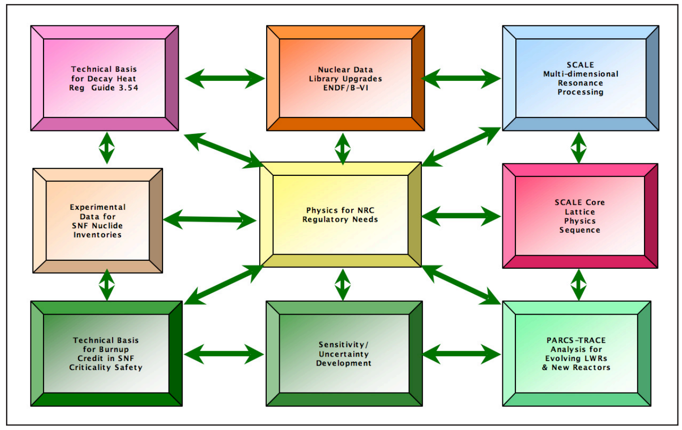

Figure 7.9: Coupled reactor and fuel cycle nuclear analyses

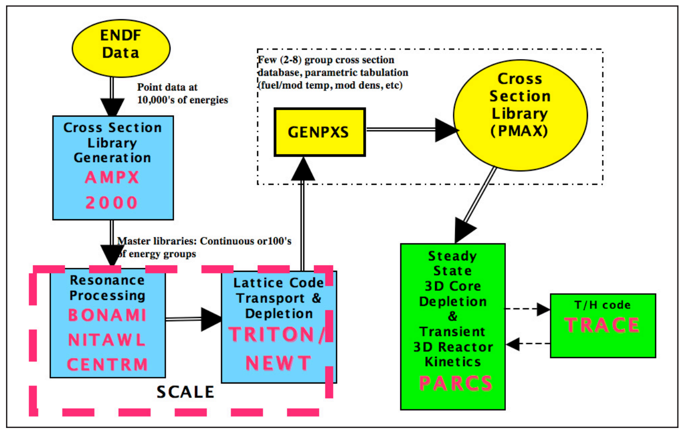

Figure 7.10: NRC nuclear analysis codes for reactor physics

The codes are also being modified and extended to accommodate different fuel, core, and control configurations and operating features of advanced non-light-water reactors. In addition, the radiation shielding codes are being updated for application to high-capacity spent fuel cask systems and advanced reactor systems.

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HIGH-BURNUP LIGHT-WATER REACTOR FUEL

Background

Structural integrity of the fuel rod cladding ensures a coolable geometry in the reactor and allows simplifying assumptions in calculations of spent fuel cask criticality. Regulations and regulatory guidance documents (examples are given below) offer numerous fuel damage criteria for use in analyzing fuel rod behavior in reactor operation and spent fuel transportation and storage.

The fuel damage criteria were originally developed from a database of mostly low-burnup fuel with Zircaloy cladding. Test data clearly showed that extrapolation from a low-burnup database was not satisfactory for regulatory purposes, and the NRC initiated a high-burnup fuel research program to address this issue. The Commission received an updated program plan in August 2003 (ADAMS Accession No. ML031810103), which discussed issues related to the use of high-burnup fuel and a strategy for assessing future requests for burnup extensions. The staff will again update this program plan in 2009.

Approach

The current research program is designed to provide information in the following areas:

- embrittlement criteria and oxidation correlations for loss-of-coolant accidents (Title 10 of the Code of Federal Regulations [10 CFR] 50.46(b); Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities"; and Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance")
- coolability criteria and threshold failure correlations for reactivity-initiated accidents (Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," and Standard Review Plan 4.2, "Fuel System Design")
- fuel rod properties for transportation and storage analysis (10 CFR 71.55, "General Requirements for Fissile Material Packages," and 10 CFR 72.122, "Overall Requirements")
- fuel rod computer codes used to audit licensees' evaluation models that demonstrate compliance with criteria and to analyze test data (Regulatory Guide 1.183)

The NRC conducts all of its fuel research in cooperation with other organizations. The U.S. nuclear fuel industry, including the Electric Power Research Institute, AREVA, Global Nuclear Fuel, and Westinghouse, is actively cooperating in a large experimental program at Argonne National Laboratory, Studsvik Nuclear AB hot cell laboratory, and Oak Ridge National Laboratory. The Pacific Northwest National Laboratory conducts a modest support program for the NRC's fuel rod computer codes, along with a code users' group consisting of 24 U.S. and international participants.

Other research receiving partial support from the NRC is conducted by the Halden Reactor Project (Norway), the Institute for Radiological and Nuclear Safety (France), the Japan Atomic Energy Agency, and Studsvik Nuclear AB (Sweden). The NRC has additional arrangements with Finland and Spain that provide a mechanism to exchange technical data and analytical results.

LOSS-OF-COOLANT ACCIDENTS

During a postulated loss-of-coolant accident (LOCA), the fuel rod cladding would experience very high temperatures and severe oxidation. The NRC's regulations specify limits for temperature and oxidation to preserve ductility and thereby ensure a coolable geometry following this postulated accident. However, additional phenomena occur with high-burnup fuel that are not addressed by the original embrittlement criteria. One of these phenomena is the change in an oxide structure that occurs at extended times, such as during a small-break LOCA, and dramatically accelerates the embrittlement process. Nevertheless, current plant operations provide adequate assurance of safety, largely as the result of the use of conservative methods. Figure 7.11 shows one cladding material that was tested in this program and experienced this change in oxide structure.

Figure 7.11 Oxidation of a foreign niobium-bearing cladding alloy under LOCA conditions

Based on NRC research (ADAMS Accession No. ML081350225), new performance-based criteria are being developed to account for all the high-burnup phenomena and to permit the use of new cladding materials without requiring license exemptions.

REACTIVITY-INITIATED ACCIDENTS

Following an accidental control rod ejection (in a pressurized-water reactor) or control blade drop (in a boiling-water reactor), the fuel rod cladding would experience very large stresses at relatively low temperatures. The NRC's requirements specify limits on the energy deposited in these events to avoid energetic dispersal of fuel particles with the potential for energetic fuel coolant interactions, core damage, and loss of coolability. However, additional phenomena occur with high-burnup fuel that lower the cladding's ductility and substantially reduce the amount of deposited energy that can be tolerated. Figure 7.12 shows a high-burnup fuel rod that was tested in Japan and expelled fuel at a small fraction of the NRC's current licensing limit. Although some confirmatory work is continuing in France using the Cabri reactor and in Japan using the Nuclear Safety Research Reactor, most of this research has been completed, and new criteria are being developed based on these results. Nevertheless, current plant operations provide adequate assurance of safety, largely because of the voluntary use of conservative methods.

TRANSPORTATION AND STORAGE

During transportation and storage of spent fuel, the fuel rod cladding experiences higher temperatures and pressure differences than during full-power operation, and the fuel rods experience large impact loads in postulated accidents. Because of the fuel rod cladding's reduced ductility at high burnup and its bonding to fuel pellets, its mechanical properties and failure conditions are substantially altered. Testing is being performed on high-burnup specimens of most commercial cladding types to provide the mechanical properties that are needed for safety analyses.

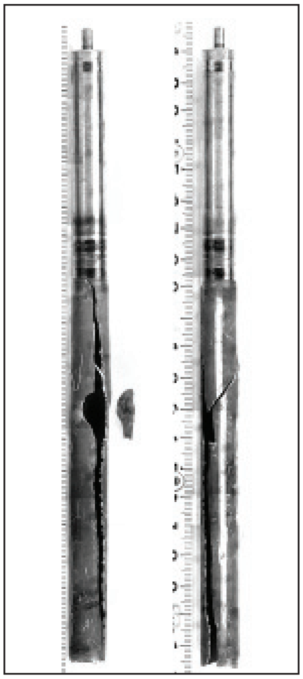

Figure 7.12 Fuel failure at moderate burnup (61 GWd/t) and low enthalpy (<100 cal/g)

FUEL ROD COMPUTER CODES

The NRC maintains one fuel rod code for steady-state analysis and one for transient analysis. These codes are used in the evaluation of experimental data and for auditing licensees' safety analyses. As new cladding alloys are introduced (e.g., AREVA's M5 and Westinghouse's Optimized ZIRLO), burnable poisons are changed (e.g., high concentrations of gadolinia and erbia), and higher burnups are sought (up to 75 gigawatt day/ton [GWd/t]), the materials properties and models in the codes must be revised. In-reactor testing must often be done to obtain data for these changes. Halden results are particularly valuable. The ability to perform quantitative analysis of fuel rod behavior is an essential part of the NRC's assessment of safety in reactor operations and spent fuel transportation and storage.

For More Information

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FISSION PRODUCTS BURNUP CREDIT

Background

The purpose of this research is to develop a technical basis to support the allowance of full (fission product and actinides) burnup credit for transportation and storage casks.

Spent nuclear fuel (SNF) refers to uranium-bearing fuel elements that have been used at commercial nuclear reactors and that no longer produce enough energy to sustain a nuclear reaction. The fission process has stopped once the spent fuel is removed from the reactor, but the spent fuel assemblies still generate significant amounts of radiation and heat. Because of the residual hazard, spent fuel must be stored or shipped in containers or casks that shield and contain the radioactivity and dissipate the heat. Further, the SNF storage or shipping system needs to support subcriticality (i.e., the neutron chain reactions cannot be maintained in the system), thereby preventing criticality accidents.

SNF is being stored at a variety of sites across the nation (e.g., in reactor spent fuel pools or in dry cask storage at reactor sites). Over the last 30 years, thousands of shipments of commercially generated SNF have been made over highways, through towns, and along railroads in the United States without causing any radiological releases to the environment or harm to the public. It is crucial that no criticality accidents occur during storage and transportation.

Most of these spent fuel shipments occur between reactors owned by the same utility to share storage space, or spent fuel may be shipped to a research facility for tests on the spent fuel itself. To minimize the number of such shipments, as much nuclear material as possible is put into each shipment without violating criticality safety.

Approach

REGULATORY NEED

The regulation for transportation and storage of spent fuel is delineated in Title 10 of the Code of Federal Regulations (10 CFR) Part 71, "Packaging and Transportation of Radioactive Material," and 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste." In reviewing transportation and storage packages for compliance with the regulation, the Office of Nuclear Material Safety and Safeguards (NMSS) issued interim staff guidance (ISG) concerning issues not currently addressed in a standard review plan (SRP) or issues where clarification of SRP text is necessary. This guidance is intended to ensure consistent reviews by the NMSS staff and will be incorporated into the next periodic update of the applicable SRP.

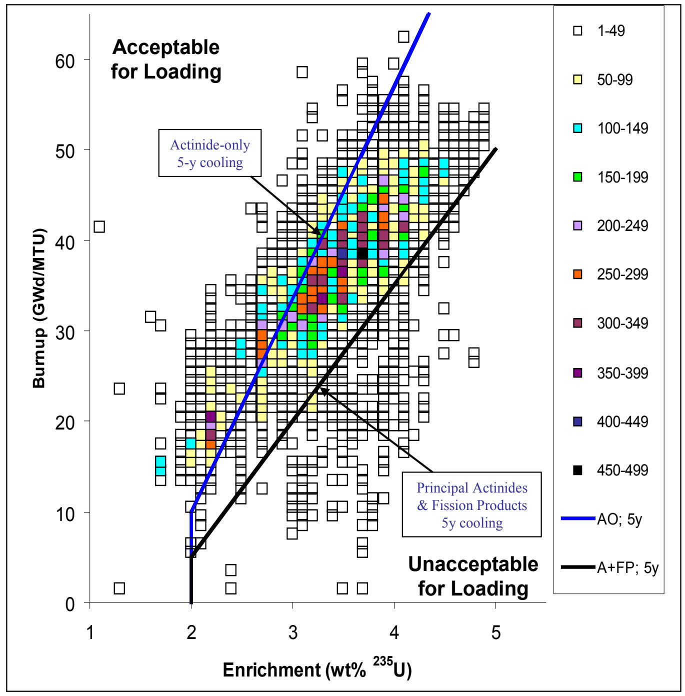

Figure 7.13 The acceptable loading inventory for a generic burnup credit rail cask design with 32 PWR assemblies is enhanced from 30% of the PWR SNF inventory to almost 90% of the inventory if credit for fission products can be obtained in the safety evaluation.

RES is conducting research to develop the technical basis to support revision of NMSS's ISG-8, "Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks," Revision 2, dated September 27, 2002, to include fission-product burnup credit.

RESEARCH ACTIVITIES

The development of the technical information and analysis approaches included (1) application of a sensitivity/uncertainty (S/U) method to support recommendations of appropriate critical experiments for use in the validation of criticality safety code, (2) recommendation of criteria for preshipment measurement, (3) SCALE-5 analysis of S/U-recommended critical experiments and recently acquired assay data to provide a generic estimate of bias and uncertainty for full burnup credit, (4) investigation to recommend modeling approaches for full burnup credit, including "best estimate" prediction of any additional reactivity margin, and (5) provision of technical support for the ISG-8 revision. This research, performed by Oak Ridge National Laboratory, supports the agency's goals for effectiveness and safety.

APPLICATIONS

The existing ISG-8, Revision 2, allows about 30 percent of the SNF assemblies in pressurized-water reactors (PWRs) to be loaded into high-capacity casks. Including fission products (based on appropriate experimental data for model validation) and allowing fission product burnup credit would allow 80–90 percent of the PWR SNF assemblies to be loaded into such casks. The potential savings to the industry (as a result of having fewer shipments) is conservatively estimated at $156 million.

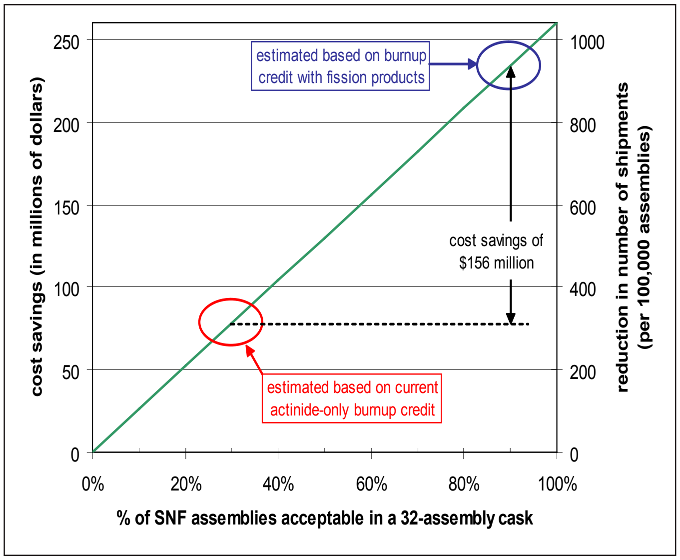

Figure 7.14 Increasing the inventory that can be put in high-capacity burnup credit casks will enable at least 625 fewer shipments and provide a savings of about $156 million.

For More Information

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CHAPTER 8: REGULATORY INFRASTRUCTURE AND INTERNATIONAL PROGRAMS

Halden Reactor Project

REGULATORY GUIDES

Scope

The NRC issues regulatory guides for public use to present approaches that the staff considers acceptable for use in implementing the agency's regulations.

The NRC's regulatory guides are grouped into 10 broad divisions to facilitate access to the information:

Division 1—Power Reactors

Division 2—Research and Test Reactors

Division 3—Fuels and Material Facilities

Division 4—Environmental and Siting

Division 5—Materials and Plant Production

Division 6—Products

Division 7—Transportation

Division 8—Occupational Health

Division 9—Antitrust and Financial Review

Division 10—General

Development Process

The NRC staff develops regulatory guides with input from external stakeholders. The guides are revised to incorporate new staff technical positions, revised industry standards, and lessons learned from practical experience. Each regulatory guide is initially issued as a draft guide for public comments for a specific period of time before its publication as a final guide. The NRC Web site at http://www.nrc.gov/reading-rm/doc-collections/reg-guides/ lists the agency's regulatory guides. Draft guides that are available for public comment can be accessed by following the links from this Web page.

The staff reviews and addresses public comments received and changes the draft guide as necessary. The NRC's Advisory Committee on Reactor Safeguards receives copies of proposed regulatory guides and may choose to meet and discuss the proposed regulatory guide before and after the public comment period.

Comments and suggestions are encouraged and welcomed in connection with improvements to published regulatory guides and development of new guides. The NRC staff revises existing guides, as appropriate, to accommodate comments and to reflect new information or experience.

Application

The NRC staff uses regulatory guides in its review of applications, while the nuclear industry uses them to understand the staff's expectations. Regulatory guides are not substitutes for regulations, and compliance with them is not required. Licensees may present alternative methods and solutions that differ from those set forth in regulatory guides. The staff will evaluate alternative methods and solutions and accept those that provide a basis of adequate safety and security for the staff's determinations.

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GENERIC ISSUES PROGRAM

Background

The Generic Issues Program (GIP) addresses those issues that have significant generic implications for risk or security and which cannot be more effectively handled by other regulatory programs and processes.

Approach

An update is underway that will improve GIP effectiveness and efficiency through the following measures:

- providing centralized leadership for GIP management and strengthening involvement of NRC regulatory offices
- ensuring consistent implementation of the generic issue (GI) assessment process across the NRC offices
- enhancing the issue screening process by consistently applying criteria; streamlining various stages of the GI process; and using enhanced risk-informed techniques, when feasible, for timely GI assessments
- using existing regulatory tools, programs, and processes, with early involvement of key stakeholders, as appropriate.

ELEMENTS OF IMPROVED GIP TO BE INCLUDED IN PROGRAM GUIDANCE (MANAGEMENT DIRECTIVE 6.4)

- RES will have overall responsibility for GIP management, including routine GI tracking, as well as periodic reporting to Congress and the Commission.

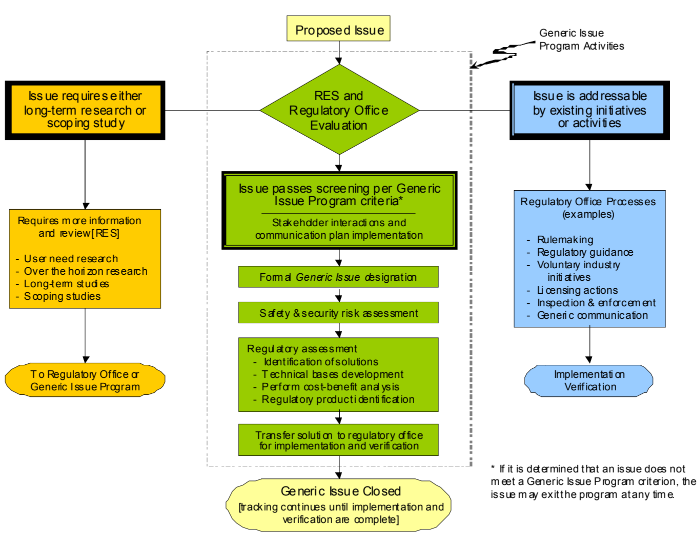

Figure 8.1 Generic Issue Program in Perspective with Other Regulatory Programs and Processes

- Regulatory offices will have well-defined roles, responsibilities, and accountability at all stages of GI assessment and resolution.
- Offices will consistently apply issue screening criteria to identify those issues that are suitable for processing under the GIP. Under the GIP, the staff will not address issues that require extensive studies or investigatory research to determine their risk or safety significance.
- The interface of the GIP with other programs will be clarified so that issues that are suitable for other NRC programs and processes, or industry initiatives, will be appropriately directed to them (e.g., the Differing Professional Opinion Program and the Allegation Program).
- For programmatic efficiency and effectiveness, and to improve timely assessments of GIs, the staff will employ, when feasible, various enhanced risk-informed techniques, which have already been developed as part of other established programs (e.g., the Accident Sequence Precursor Program).
- RES will ensure the necessary interoffice coordination throughout the GI process. After the issue is screened as a formal GI, the staff will consider participation by the key nuclear industry stakeholders, when appropriate, to identify possible solutions (e.g., a regulatory product or an industry initiative).
- The GI process will conclude when the regulatory product is identified. The appropriate regulatory office will proceed, under other established programs and processes, to develop and implement the identified regulatory solution and perform appropriate verification.

EXPECTED RESULTS OF IMPROVED GIP

The staff is planning to complete formal GI assessments within 1–2 years. In some cases, depending on the technical complexity of the individual issue, the GI assessment process may require additional time. Annual reports to the Commission will note such cases.

For More Information

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THE ORGANIZATION FOR ECONOMIC COOPERATION AND DEVELOPMENT HALDEN REACTOR PROJECT

Background

The NRC and its predecessor, the U.S. Atomic Energy Commission, have been participating in the Halden Reactor Project (HRP) since its inception in 1958. During this period, the NRC has used numerous research products from this internationally funded cooperative effort. For example, Halden tests on high-burnup fuel under loss-of-coolant accident conditions supported an NRC research information letter on cladding embrittlement. As another example, Halden's human factors research has supported regulatory guidance in areas such as alarm systems, hybrid control rooms, display navigation, and guidance for the review of proposed staffing configurations in computer-based control rooms.

Approach

The Norwegian Institute for Energy Technology (Institutt for Energiteknikk or IFE) manages the HRP for the Organization for Economic Cooperation and Development/Nuclear Energy Agency. The HRP is based at IFE's facility in Halden, Norway. This facility includes the Halden Boiling Water Reactor (HBWR), which currently operates at 18 to 20 megawatts.

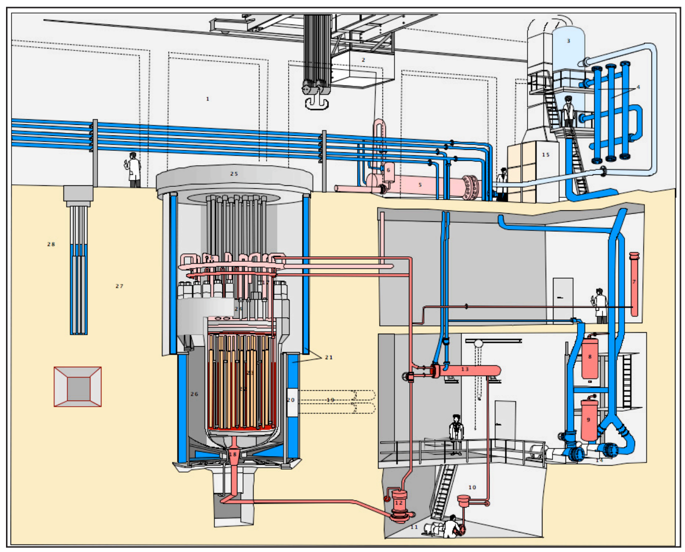
The HBWR is fully dedicated to instrumented in-reactor testing of fuel and reactor materials. It also delivers steam to a nearby paper factory. Since its initial startup, the reactor facility has been progressively updated and is now one of the most versatile test reactors in the world. During this development, more than 300 in-reactor experiments have been performed. The HRP fuels and materials program focuses on the performance of fuel and structural materials under normal or accident conditions using the numerous experimental channels in the core that are capable of handling many test rigs simultaneously.

IFE's Halden facility also includes the IFE Man-Technology-Organization (MTO) Laboratory. The Halden Man-Machine Laboratory (HAMMLAB) is one of the principal experimental facilities in this laboratory. HAMMLAB uses a reconfigurable simulator control room that facilitates research into instrumentation and control, human factors, and human reliability analysis. Currently, HAMMLAB has hardware and software enabling it to simulate the Fessenheim pressurized-water reactor plant in France and the Forsmark-3 boiling-water reactor plant in Sweden. HAMMLAB is the only Western-style light-water reactor reconfigurable simulator that is available to the NRC for human factors research.

Many of the HAMMLAB experiments are performed with the control room configured as a prototype advanced control room with an integrated surveillance and control system. This setup is used to explore the impacts of automation and advanced human-system interfaces on operator performance. HAMMLAB has extensive data collection capabilities and typically uses qualified nuclear power plant operators (who are familiar with the plants being simulated) as test subjects.

The IFE MTO Laboratory also includes a virtual environment center and an integrated operations laboratory. The former is used to perform research involving mixed reality applications (e.g., training), and the latter is used to address issues associated with remote operations. The laboratory, which has a large staff that conducts research on a wide range of technical disciplines, is an acknowledged center of excellence in the nuclear arena.

Figure 8.2 Halden Boiling Water Reactor (HBWR) test reactor



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AGENCY FORWARD-LOOKING AND LONG-TERM RESEARCH

Background

FORWARD-LOOKING RESEARCH

The agency currently identifies, as a matter of routine, long-term, or forward-looking, research activities supporting potential regulatory needs over the longer term (within the next few years). The NRC identifies and pursues these forward-looking research activities during the normal course of planning and budgeting processes.

LONG-TERM RESEARCH

Each year since 2007, the staff has prepared Commission papers on long-term research activities. The papers discuss candidate long-term research topics and estimate funding needs for use in budget preparation. For the purposes of the annual Commission papers, long-term research is defined as research that is not already funded or otherwise being worked on that will provide the fundamental insights and technical information needed to address potential technical issues or identified gaps to support anticipated NRC needs in the future (more than 5 years).

Approach

The NRC performs regulatory research to support the achievement of the goals identified in its Strategic Plan. These goals ensure protection of public health and safety and the environment; ensure the secure use and management of radioactive materials; ensure openness in the NRC's regulatory processes; ensure that NRC actions are effective, efficient, realistic, and timely; and ensure excellence in agency management.

The objectives of forward-looking and long-term research are to identify the research required to support related regulatory decisionmaking, to help determine if research should be conducted by the NRC or by the industry, and to identify collaborative opportunities with domestic and international partners. The identified research could be exploratory, in support of possible new program areas, in support of the development of technical bases for a range of anticipated regulatory decisions, to address emerging technologies that could have future regulatory applications, or to develop plans to implement needed research.

The agency has established the following exploratory long-term research strategies:

1. Ensure that the NRC regulations and regulatory processes have sound technical bases.
2. Prepare the agency for anticipated changes in nuclear technology that could have safety, security, or environmental implications.
3. Develop improved methods by which the agency can carry out its regulatory responsibilities.
4. Develop and maintain an infrastructure of expertise, facilities, analytical capabilities, and data to support regulatory decisionmaking.

The process for determining what projects should be funded under the aegis of the long-term research plan includes soliciting input from the regulatory and regional offices on the exploratory long-term research activities that the agency should consider undertaking. In addition, RES staff reviewed previously suggested long-term exploratory research activities including those not funded in previous budget years for including in the candidate list. Moreover, the process includes establishing a Review Committee composed of seven senior-level system staff from RES and the regulatory offices. The Review Committee reviews, evaluates, and rates activities that resulted from new suggestions and those remaining from previous proposal processes. The Committee's charter specifies five evaluation criteria and their weighting factors to provide a rating, or score, for each activity. The five criteria include leveraging resources, advancing the state-of-the-art, providing an independent tool to NRC, applying to more than one program area, and addressing gaps created by technology advancements.

The Committee forwards the results of the review to the RES Office Director and posts the results on an internal Web site. In this way, the Review Committee's ratings are available to the staff as feedback on the input suggestions. The RES Office Director will consider the rating results as well as other factors external to that review process (e.g., the funding approved by the planning, budgeting, and performance management process for long-term research and factors that may have changed since the scoring process) when determining the projects to fund.

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INTERNATIONAL RESEARCH COOPERATIVE ACTIVITIES AND AGREEMENTS

Background

RES currently maintains more than 70 bilateral or multilateral agreements with over 20 countries and the Organization for Economic Cooperation and Development. These agreements cover a wide range of activities and technical disciplines, including severe accidents, thermal-hydraulic (T/H) code assessment and application, nuclear fuels analysis, seismic safety, fire protection, and human reliability.

RES participation allows broader sharing of data obtained from physical facilities not available in the United States. As a result, NRC tools, data, and safety knowledge are current and founded on state-of-the-art information. This enhances the NRC's ability to make sound, realistic decisions based on worldwide scientific knowledge, as well as promoting the effective and efficient use of agency resources. Data obtained are used to develop new analytical models, updates, and verification and validation of NRC codes; to enhance assessments of plant risk, including decisionmaking, fire, and human performance and reliability; and to develop risk-informed approaches to regulation.

Code Applications and Maintenance Program

The largest international research cooperative activity is the Code Applications and Maintenance Program (CAMP). CAMP includes T/H analysts from more than 20 member nations who meet to (1) share experience with NRC T/H computer codes to identify code errors, perform code assessments, and identify areas for additional experiments, model development, and improvement, (2) maintain and improve user expertise, (3) develop and improve user application guidelines, (4) develop a well-documented T/H code assessment database, and (5) share experience in the use of the codes for resolution of safety and other technical issues (e.g., scalability and uncertainty).

Cooperative Severe Accident Research Program

The Cooperative Severe Accident Research Program (CSARP) facilitates the exchange of data and analysis on experimental and analytical research on severe accidents through meetings hosted by the NRC. Approximately one dozen countries exchange the results from severe accident research through in-kind contributions on phenomenological research and data. As each user organization independently assesses the MELCOR code, the feedback (identification of code deficiencies and improvements) will help the NRC to maintain a state-of-the-art severe accident code.

RES Considerations for Proposed International Projects

RES routinely receives proposals from international organizations, foreign regulatory counterparts, and foreign national organizations to participate in cooperative research programs. RES uses seven criteria when considering proposed international projects:

1. The project assists in preparing the NRC for the future.
2. The work contributes to resolving existing or emerging regulatory or safety issues affecting U.S. licensees and applicants.
3. The research reduces known phenomenological uncertainties, enhances the accuracy of NRC computer codes and data, and/or develops state-of-the art safety information.
4. The project contributes to maintaining and/or developing critical skills needed to carry out the NRC's mission.
5. The research supports completion of existing and projected work.
6. The work produces timely results that will support the intended regulatory use.
7. Overall costs (full-time equivalents, travel, and contribution to contractor support) are commensurate with the expected safety benefit.

RES also actively seeks international cooperation in obtaining technical information on safety issues that require test facilities not available domestically and requiring substantial resources to duplicate in the United States. RES will often propose modifications to a project sponsor so that the proposed project can better meet the NRC's needs. In addition, the NRC may propose to sponsor cooperative international participation in the agency's research projects.

The NRC's RES program has long been a leader in enhancing domestic resources with international knowledge, skills, and use of foreign facilities. The staff continues to ensure that the international activities in which the agency participates have direct relevance to the NRC's regulatory program.

For More Information
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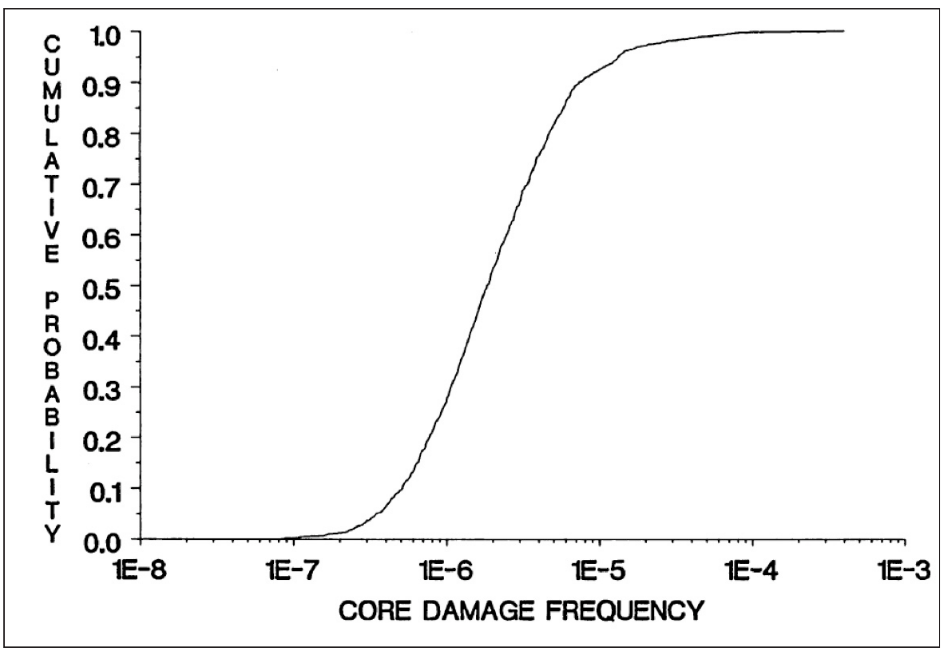
CHAPTER 9: RISK AND CONSEQUENCE ANALYSIS

Accident Sequence Precursor Program

Reactor Operating Experience Data Collection and Analysis

SPAR Model Development Program

Risk Assessment Standardization Project



ACCIDENT SEQUENCE PRECURSOR PROGRAM

Background

The NRC established the Accident Sequence Precursor (ASP) Program in 1979 in response to NUREG/CR-0400, "Risk Assessment Review Group Report," issued September 1978. The ASP Program systematically evaluates U.S. nuclear power plant operating experience to identify, document, and rank the operating events most likely to lead to inadequate core cooling and severe core damage (precursors), given the likelihood of additional failures.

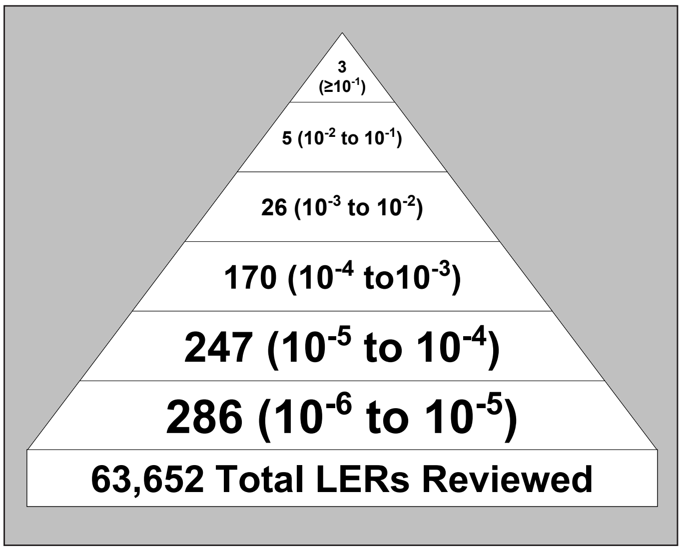


Figure 9.1 Number of precursors within each risk range (FY 1969–2007)

Approach

To identify potential precursors, the NRC staff reviews plant events from licensee event reports (LERs) and inspection reports. The staff then analyzes any identified potential precursors by calculating the probability of an event leading to a core damage state. A plant event can be of one of two types: (1) an occurrence of an initiating event, such as a reactor trip or a loss of offsite power event with any subsequent equipment unavailability or degradation, or (2) a degraded plant condition indicated by unavailability or degradation of equipment without the occurrence of an initiating event.

For the first type, the staff calculates a conditional core damage probability (CCDP). This metric represents a conditional probability that a core damage state is reached, given an occurrence of an initiating event (and any subsequent equipment failure or degradation).

For the second type, the staff calculates an increase in core damage probability (ΔCDP). This metric represents the increase in the probability of reaching a core damage state for the period that a piece of equipment or a combination of equipment is deemed unavailable or degraded from a nominal core damage probability for the same period for which the nominal failure or unavailability probability is assumed for the subject equipment.

The ASP Program considers an event with a CCDP or ΔCDP greater than or equal to 1×10^{-6} to be a precursor. The ASP Program defines a significant precursor as an event with a CCDP or ΔCDP greater than or equal to 1×10^{-3}.

PROGRAM OBJECTIVES

The ASP Program has the following objectives:

- Provide a comprehensive, risk-informed view of nuclear power plant operating experience and a measure for trending nuclear power plant core damage risk.
- Provide a partial check on dominant core damage scenarios predicted by probabilistic risk assessments.
- Provide feedback to regulatory activities.

The NRC also uses the ASP Program to monitor performance against the safety goal established in the agency's Strategic Plan. Specifically, the program provides input to the following performance measures:

- zero events per year identified as a significant precursor of a nuclear reactor accident (i.e., CCDP or ΔCDP greater than or equal to 1×10^{-3})
- no more than one significant adverse trend in industry safety performance (determination principally made from the Industry Trends Program but supported by ASP results)

RECENT RESULTS

- No significant precursors were identified for fiscal year (FY) 2008. The last significant precursor identified was the event at Davis-Besse, which involved multiple degraded conditions (FY 2002).
- A statistically significant decreasing trend was detected for the occurrence rate of all precursors during the FY 2001–2007 period.

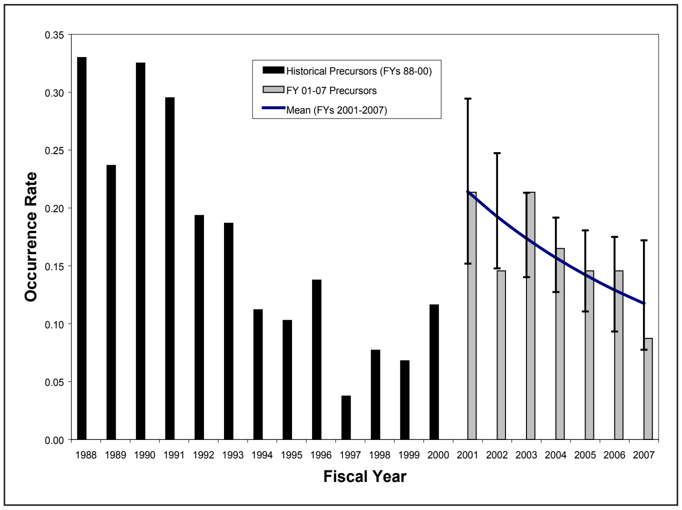


Figure 9.2: Occurrence rate of all precursors. Rates for FY 1988–2000 are shown for historical perspective only.

- Statistically significant decreasing trends were detected in the occurrence rate of precursors involving degraded conditions, precursors with a CCDP or ΔCDP greater than or equal to $1x10^{-4}$, and precursors occurring at pressurized-water reactors.

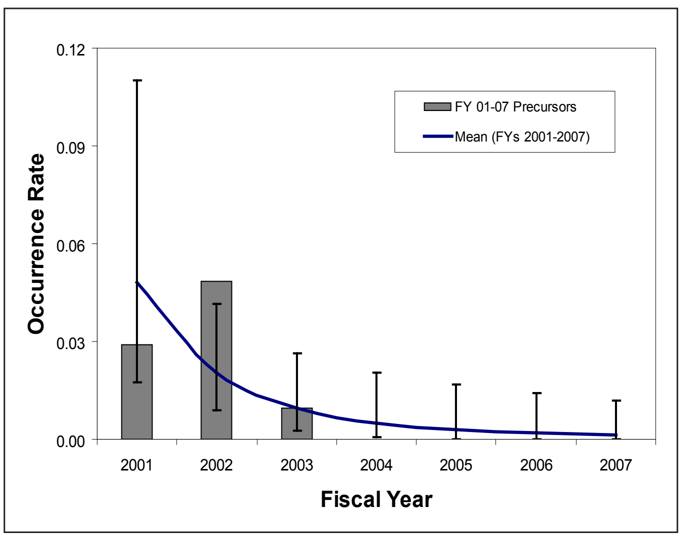


Figure 9.3: Occurrence rate of precursors with CCDP or ΔCDP greater than or equal to $1x10^{-4}$

- No statistically significant trends were detected for precursors involving initiating events, losses of offsite power, and precursors occurring at boiling-water reactors.
- From FY 1998 through FY 2007, five precursors contributed 66 percent of the total risk caused by all precursors.
- During the same 10-year period, 137 precursors contributed the remaining 34 percent of the total risk contribution from all precursors.

Plant	Precursor Description	ΔCDP	Condition Duration
DC Cook, Units 1&2	Long-term degraded conditions involved a number of locations in the plant where the effects of postulated high-energy line break events would damage safety-related components. Event Date: 10/22/1999 ML003768545	$4x10^{-4}$	Original Design Deficiency
Point Beach, Units 1&2	Design deficiency in the air-operated minimum-flow recirculation valves of the auxiliary feedwater (AFW) pumps which could potentially lead to common-mode failure of the AFW pumps. Event Date: 11/21/2001 ML033010140	$7x10^{-4}$	Original Design Deficiency
Davis-Besse	Cracking of control rod drive mechanism nozzles, reactor pressure vessel head degradation, potential clogging of the emergency sump, and potential degradation of the high-pressure injection pumps. ***Event Date: 02/27/2002 ML050260219***	$6x10^{-3}$	~1 year

Figure 9.4 Table of Pant and Precursor Descriptions

ANNUAL SUMMARY OF RESULTS

Updated results from the ASP Program are published in an annual paper to the Commission, usually in October. The most recent paper, SECY-08-0145, "Status of the Accident Sequence Precursor Program and the Standardized Plant Analysis Risk Models," was issued on October 1, 2008.

For More Information

Contact Christopher Hunter, RES/DRA, at 301-251-7575 or Christopher.Hunter@nrc.gov

1. Rates for FY 1988–2000 are shown for historical perspective only.

REACTOR OPERATING EXPERIENCE DATA COLLECTION AND ANALYSIS

Background

The collection and analysis of nuclear power plant operational data are important activities in the NRC's risk-informed regulatory programs. The results of the data collection efforts are primarily used to estimate and monitor the risk of accidents at U.S. commercial nuclear power plants. Data and information reported to the NRC are reviewed, evaluated, and coded into databases which form the basis for estimates of reliability parameters used in probabilistic risk assessment (PRA) models. These models permit the NRC to do the following:

- perform state-of-the-art risk assessments of operating events and conditions
- assess licensee risk-related performance
- conduct special studies of risk-related issues, such as station blackout risk, as part of the Special Reliability Studies Project
- determine trends, develop performance indicators based on operating data, and perform reliability studies for risk-significant systems and equipment

Approach

The NRC maintains a set of PRA models for all operating U.S. commercial nuclear power plants. The staff uses these standardized plant analysis risk (SPAR) models to support risk-informed decisionmaking. For example, the Accident Sequence Precursor (ASP) program uses the SPAR models in analyses to help identify potential precursors, to support the agency's Significance Determination Process (SDP), and to confirm licensee risk analyses submitted in support of license amendment requests.

To maintain current SPAR models, RES collects and analyzes operating data from all nuclear power plants. The data are used to estimate the inputs required for the models. Examples of basic model inputs are initiating event frequencies, component failure probabilities, component failure rates, maintenance unavailabilities, common-cause failure parameters, and human failure probabilities.
The Reactor Operating Experience Data for Risk Applications Project collects data on the operation of nuclear power plants as reported in licensee event reports (LERs), licensees' monthly operating reports, and the Institute of Nuclear Power Operations Equipment Performance and Information Exchange System. The data collected include component and system failures, demands on safety systems, initiating events, fire events, common-cause failures, and system/train unavailabilities. The data are stored in discrete database systems such as the Reliability and Availability Data System (RADS), Common-Cause Failure Database, and ASP Events Database.

Data input into the RADS database are used to verify and validate information used in the Mitigating Systems Performance Index (MSPI) Program. RADS data are used to review the efficiency and effectiveness of the MSPI and to suggest improvements to the index.

The staff can search LERs individually by using the LERSearch program located on the agency's internal Web site.
The Computational Support for Risk Applications Project also uses the data to periodically update PRA parameters, such as initiating event frequencies, component reliabilities, maintenance unavailabilities, and common-cause failure parameters for input into the plant-specific SPAR models. In general, the NRC uses the data collected as described above in support of its established regulatory programs, which help identify potential safety issues, such as the Industry Trends Program (ITP), the ASP Program for evaluation of the risk associated with operating events, and the Reactor Oversight Process.

For example, RES supports the ITP by trending operating experience data and making that information available on the RES internal and public Web sites. Examples of trends that are regularly updated include thresholds for initiating events; system, component, and common-cause failures; and ASP events.
ASP analyses and the SDP use component failure probability estimates and initiating event frequencies to determine the risk significance of inspection findings. The results are then used to decide the allocation and characterization of inspection resources, the initiation of an inspection team, and the need for further analysis by other agency organizations.

The Reactor Operating Experience Results and Databases Web site (http://nrcoe.inel.gov/results/) makes current operating experience information available to the NRC staff and the public. The site also contains results for a variety of previously published studies that include initiating events, system performance, component performance, common-cause failures, fire events, and loss of offsite power.

Figure 9.5 below displays the sources and uses of operating data and analyses in NRC regulatory programs.

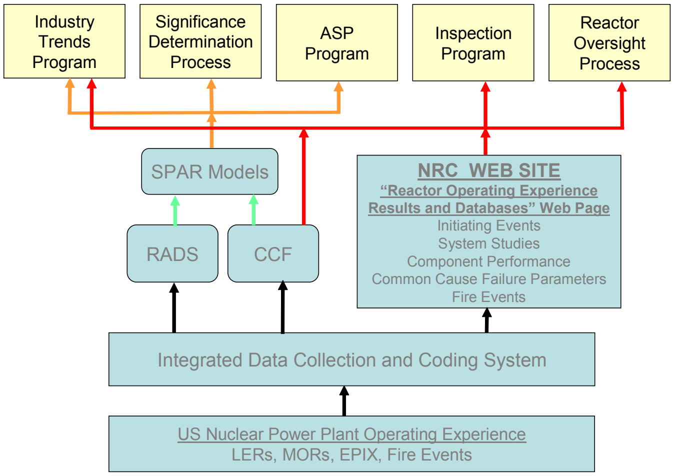


Figure 9.5 Uses of Operational Data and Analyses in NRC Regulatory Programs

Finally, RES also supports the Baseline Risk Index for Initiating Events (BRIIE), a measure used to provide a risk-informed performance indicator for the initiating events "Cornerstone of Safety." This type of information helps the Office of Nuclear Reactor Regulation affirm that operating reactor safety is being maintained and also enhances the NRC's inspections of significant safety systems.

For More Information

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SPAR MODEL DEVELOPMENT PROGRAM

Background

THE RISK

For assessing public safety and developing regula-tions for nuclear reactors and materials, the NRC has traditionally used a deterministic approach that asked "What can go wrong?" and "What are the consequences?" Now, new information for assessing risks also allows the NRC to ask "How likely is it that something will go wrong?" By making the regulatory process "risk informed" (defined as the use of risk insights to focus on the items most important to protecting public health and safety), the NRC can focus its attention on those design and operational issues most important to safety.

In the reactor safety arena, risk-informed activities occur in five broad categories: (1) applicable regulations, (2) licensing process, (3) revised oversight process, (4) regulatory guidance, and (5) risk analysis tools, methods, and data. Activities within these categories include revisions to technical requirements in the regulations; risk-informed technical specifications; a new framework for inspection, assessment, and enforcement actions; guidance on risk-informed inservice inspections; and improved standardized plant analysis risk (SPAR) models.

The SPAR models, in combination with the Systems Analysis Programs for Hands-on Integrated Reliability Evaluation (SAPHIRE) software and Risk Assessment Standardization Project (RASP) Handbook, developed by RES, provide the NRC staff with the probabilistic risk assessment (PRA) tools to support these risk-informed activities.

THE NEEDS

The NRC staff uses the SPAR models, SAPHIRE software, and RASP Handbook in support of risk-informed activities related to the inspection program, incident investigation program, license amendment reviews, performance indicator verification, Accident Sequence Precursor Program, generic safety issues, and special studies. These tools also support and provide rigorous and peer-reviewed evaluations of operating experience, thereby demonstrating the agency's ability to analyze operating experience independently of licensees' risk assessments and enhancing the technical credibility of the agency.

The SPAR models integrate systems analysis, accident scenarios, component failure likelihoods, and human reliability analysis into a coherent model that reflects the design and operation of the plant. The SPAR model gives risk analysts the ability to quantify the expected risk of a nuclear power plant in terms of core damage frequency and the change in that risk given an event or an anomalous condition or a change in the design of the plant. More importantly, the model provides the analyst with the ability to identify and understand the attributes that significantly contribute to the risk and insights on how to manage that risk. The RASP Handbook provides standard methods and guidelines for analyzing internal events using SAPHIRE. The agency is developing additional methods and guidance to support internal fire, internal flooding, external events (e.g., earthquakes), and low-power and shutdown events. In addition to guidelines for resolving technical issues surrounding risk assessments of operating events and conditions, the RASP Handbook will improve the consistency in results when the various NRC programs analyze the same (or a similar) event or condition, improve the coordination among various NRC programs performing risk analyses of licensee performance deficiencies or reactor incidents, reduce the time required to perform risk analyses of operating events and licensee performance issues, and improve internal and external risk communication.

Currently, 77 Level 1 (internal event, at power) SPAR models represent the 104 operating commercial nuclear plants in the United States. The Level 1 SPAR model includes core damage risk resulting from general transients (including anticipated transients without scram), transients induced by loss of a vital alternating current or direct current bus, transients induced by a loss of cooling (service) water, loss-of-coolant accidents, and loss of offsite power. The SPAR models also use a standard set of event trees for each plant design class and standardized input data for initiating event frequencies, equipment performance, and human performance, although these input data may be modified to be more plant- and event-specific, when needed. However, the system fault trees contained in the SPAR models are not as detailed as those contained in licensees' PRA models. In August 2008, RES began the development of a SPAR model for a new reactor design (the AP1000), in support of the Office of New Reactors (NRO). Completion of the AP1000 model is scheduled for November 2009.

To more accurately model plant operation and configuration and to identify the significant differences between the licensees' PRA and SPAR logic, detailed cut-set level reviews have been accomplished for all 77 models. In addition to the Level 1 models, 15 external event models based on the licensees' individual plant examinations of external events (IPEEEs), six low-power/shutdown models, and three extended Level 1 models support large early-release frequency and Level 2II PRA modeling.

A formal quality assurance plan was implemented in September 2006. Limited scope validation and verification are accomplished by comparison to licensee PRA/IPEEE models (as available), or comparison to NRC NUREGs and analyses. Limited-scope peer reviews consist of internal quality assurance review by contractors, NRC Headquarters staff, and the regions' senior reactor analysts (as available). Additionally, detailed peer reviews using the American Society of Mechanical Engineers/American Nuclear Society PRA standards will be completed in early fiscal year 2010 for a representative pressurized-water and boiling-water reactor SPAR model.

The user feedback from staff, peer reviews from licensees, and insights gained from special studies, such as identification of threshold values during Mitigating Systems Performance Index reviews and the "Reevaluation of Station Blackout Risk at Nuclear Power Plants," NUREG/CR-6890, December 2005the loss-of-offsite power and station blackout study), , result in continual improvements to the models. In 2007, the NRC entered into a cooperative effort with the Electric Power Research Institute to improve PRA quality and address several key technical issues common to both the SPAR models and industry models. This cooperative effort has resulted in the joint publication of "Support System Initiating Events: Identification and Quantification Guideline." (EPRI Technical Update, 1016741, December 2008). This report documents current methods to identify and quantify support system initiating events used in PRAs.

Risk-Informed Approach

The NRC has used PRA methods to address complex safety issues and make risk-informed decisions, such as those involved in rules on station blackout, anticipated transients without scram, and pressurized thermal shock; to set the priorities for addressing generic safety issues; and to evaluate responses to generic letters. The NRC has also extended risk-informed decisionmaking approaches to its processes. The risk-informed approach is now part of the Reactor Oversight Process, which includes inspection, enforcement, and assessment. NRO will be using the SPAR models to assist in engineering reviews of the new reactor license applications.

These improvements are intended to better focus inspection resources on the most safety-significant aspects of plant design and operation and to make the process more objective. The SPAR models and SAPHIRE software, in conjunction with RASP methods and guidelines for event analysis, provide analytical tools for use by the NRC staff in regulatory activities undertaken in making risk-informed decisions. NUREG/CR-6952 Volumes 1–7, "Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE)," published September 2008, provides a comprehensive reference for the current SAPHIRE version. A new version of SAPHIRE, Version 8, being developed, will provide improved features and capabilities for the staff's risk-informed activities.

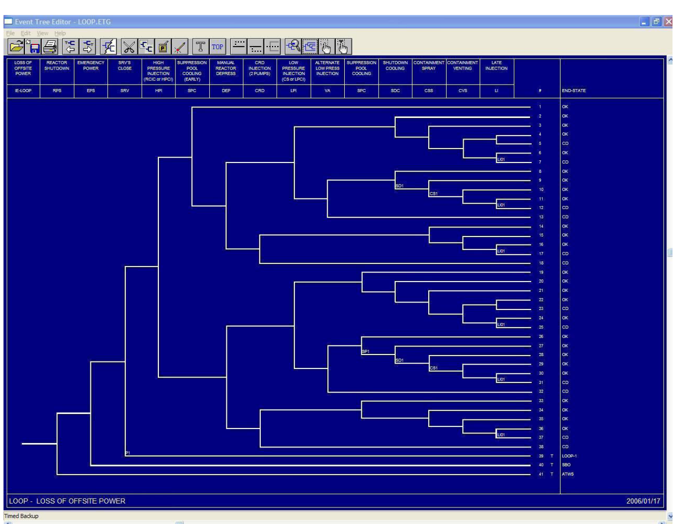

Figure 9.6 Example Loss of Offsite Power SPAR Model Event Tree

SPAR Model Applications

SPAR models, SAPHIRE software, and the RASP Handbook are used to support the following activities:

Inspection Program (Significance Determination Process, Phase 3): Determine the risk significance of inspection findings or of events to decide the allocation and characterization of inspection resources, the initiation of an inspection team, or the need for further analysis or action by other agency organizations.

Management Directive 8.3, "NRC Incident Investigation Program": Estimate the risk significance of events and conditions at operating plants so that the agency can analyze and evaluate the implications of plant operating experience in order to compare the operating experience with the results of the licensees' risk analyses, identify risk conditions that need additional regulatory attention, identify risk-insignificant conditions that need less regulatory attention, and evaluate the impact of regulatory or licensee programs on risk.

Accident Sequence Precursor Program: Systematically screen and analyze operating experience data to identify those events or conditions that are precursors to severe accident sequences.

Generic Safety Issues: Provide the capability for resolution of generic safety issues, both for screening (or prioritization) and conducting more rigorous analysis to determine if licensees should be required to make a change to their plant or to assess if the agency should modify or eliminate an existing regulatory requirement.

License Application and License Amendment Reviews: Enable the staff to make risk-informed decisions on plant-specific changes to the licensing basis as proposed by licensees and provide risk perspectives in support of the agency's reviews of licensees' submittals.

Special Studies (e.g., NUREG/CR-6890, Volumes 1 and 2, "Reevaluation of Station Blackout Risk at Nuclear Power Plants," issued December 2005): Perform various studies in support of regulatory decisions as requested by the Commission, the Office of Nuclear Reactor Regulation, and other NRC offices.

For More Information
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RISK ASSESSMENT STANDARDIZATION PROJECT

Background

In the NRC's Reactor Oversight Process, the NRC staff performs risk assessments of inspection findings and reactor incidents to determine their significance for appropriate regulatory response. Currently, several NRC groups are performing these risk assessments for Accident Sequence Precursor (ASP) and Significance Determination Process (SDP) Phase 3 analyses and Incident Investigation Program assessments under Management Directive (MD) 8.3. Because of the different objectives of each NRC program, the NRC staff initiated the Risk Assessment Standardization Project (RASP) to establish standard procedures, improve the methods, and enhance risk models that are used in risk assessment in various risk-informed regulatory applications.

Approach

PROJECT OBJECTIVES

The primary objective of RASP is to provide standard methods and tools for risk analysis of inspection findings or reactor incidents for the ASP Program, Phase 3 analysis of the SDP, and Incident Investigation Program, while recognizing differences in purpose among the programs. By using these standard methods and tools, NRC analysts from various Headquarters and regional offices will achieve more consistent results when performing risk assessments of operational events and licensee performance issues.

RASP ACTIVITIES

The major activities of RASP are the following:

- developing standard procedures and methods for the analysis of internal events, internal fire and flooding events, external events, and shutdown events
- providing enhanced quality, integrated NRC Standardized Plant Analysis Risk (SPAR) models for internal and external events, including shutdown events
- enhancing the SAPHIRE/GEM code for SPAR model analyses
- providing readily available technical support to SDP analysts

The NRC staff in the RES Division of Risk Analysis is performing these RASP activities as part of a multiyear project expected to result in the revision and development of procedures to consolidate and streamline risk analysis. Staff from the Office of Nuclear Reactor Regulation (Division of Risk Assessment and Division of Inspection and Regional Support) and regional senior reactor analysts provide detailed peer review of RASP-related products, as well as feedback for future enhancements.

Specific details of the proposed work on each RASP activity are discussed below.

Development of Risk Assessment of Operational Events Handbook (RASP Handbook)

The NRC staff has issued a RASP Handbook for risk assessment of "internal events" and "external events" at U.S. commercial nuclear power plants. This handbook, entitled "Risk Assessment of Operational Events Handbook for SDP Phase 3, ASP, and MD 8.3," is in the form of a practical "how to" guide to the methods, best practices, examples, tips, and precautions for using SPAR models to evaluate the risk of inspection findings and reactor incidents. The handbook represents best practices based on feedback and experience from the analyses of over 600 precursors in the ASP Program (since 1969) and numerous SDP Phase 3 analyses (since 2000).

The handbook consists of three volumes, designed to address internal events analysis (Volume 1), external events analysis (Volume 2), and SPAR model reviews (Volume 3). A fourth volume is being developed for shutdown events analysis. The scope of each of these volumes is described below.

Development of Standard Guidance for Internal Events Analysis. Volume 1 of the RASP Handbook, "Internal Events," provides guidance on generic methods and processes to estimate the risk significance of initiating events (e.g., reactor trip, loss of offsite power) and degraded conditions (e.g., a failed high-pressure injection pump, failed emergency power system) that may have occurred at a nuclear power plant. Specifically, this volume provides guidance on the following analysis methods: exposure time determination and modeling, failure determination and modeling, mission time modeling, test and maintenance outage modeling, recovery modeling of failed equipment, and multiunit considerations modeling.

In addition, Volume 1 contains an appendix that provides guidance on the process to perform risk analysis of operational events. The appendix, "Roadmap—Risk Analysis of Operational Events," provides an overview of the risk analysis process and detailed steps on how to perform a risk analysis of an operational event.

Future revisions of Volume 1 of the handbook will include additional method guides, such as common-cause failure analysis in event assessment, human reliability analysis in event assessment, parameter estimation and update, convolution of failure to run parameters, uncertainty analysis in event assessment, and a simplified expert elicitation.

Development of Standard Guidance for Evaluating Internal Fires and Flooding Events, External Events, and Shutdown Events. Volume 2 of the RASP Handbook, "External Events," provides methods and guidance for the risk analysis of initiating events and conditions associated with external events. External events include internal fire, internal flooding, seismic events, and other external events such as external flooding, external fire, high winds, tornado, hurricane, and other extreme weather-related events. This volume is intended to complement Volume 1 for internal events. The guidance for risk analysis of external events provides a systematic process to initiate and complete a preliminary analysis, including examples and worksheets for the required steps of the analysis method. Specifically, this volume provides guidance on the following analysis methods: internal fire modeling and fire risk quantification, internal flood modeling and risk quantification, seismic event modeling and seismic risk quantification, and other external event modeling and risk quantification.

A future volume of the handbook is being developed for risk analysis of shutdown events. The guidance development will be closely coordinated with the development of SPAR models for shutdown events.

Development of Standard Guidance for Reviews of SPAR Model Modifications. Volume 3 of the RASP Handbook, "SPAR Model Reviews," provides analysts and SPAR model developers with additional guidance to ensure that the SPAR models used in the risk analysis of operational events represent the as-built, as-operated plant to the extent needed to support the analyses. This volume provides checklists that can be used following modifications to the SPAR models for performing risk analysis of operational events. These checklists are based on NUREG/CR-3485, "PRA Review Manual," September 1985; Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities"; and experiences and lessons learned from SDP and ASP analyses.

Enhancements to SPAR Models, and SAPHIRE/GEM Interface for SPAR Model Analyses

This task involves enhancing SPAR models and the SAPHIRE/GEM interface to ensure that quality risk assessment tools are readily available to NRC staff performing risk assessments. The expected enhancements will include improvements in the fidelity of SPAR models for risk analysis of internal events, external events, and shutdown events. Efforts to enhance the Revision 3 SPAR models for internal events involve comparing the SPAR models against a licensee's probabilistic risk assessments (PRAs); updating station blackout and loss-of-offsite power models; updating parameter estimates for failure probabilities and initiating event frequencies; and reevaluating success criteria based on thermal-hydraulic analyses of key accident sequences known to be significantly different than those used in a licensee's PRAs. Additional description of SPAR model enhancement and development activities appears in the information sheet "SPAR Model Development Program."

Now under development, Version 8 of SAPHIRE/GEM expands upon the current version of the code software (Version 7) and provides new features and capabilities. Version 8 has been designed to improve user-friendliness and analyses with large, complex models and to support NRC user need requirements for SPAR model development and risk-informed programs.

Technical Support for SDP Analysts

This activity involves providing technical support to SDP analysts on the efficient use of the various RASP products such as guidance for standard risk assessment methods, enhanced SPAR models, new software tools, and the Web-based toolbox. The expected technical support will include the maintenance of RASP products and their quality, as-requested enhancements to risk assessment methods and SPAR models, and peer reviews of SDP Phase 3 analyses. Peer reviews of SDP Phase 3 analyses will focus on unique and complex cases to ensure consistency and scrutability of analysis results.

For more information, please see S.M. Wong, et al., "Risk Assessment Standardization Project (RASP) Handbook for Risk Assessment of Operational Events," ANS PSA 2008 Topical Meeting, Knoxville, TN, September 7–11, 2008.

For More Information
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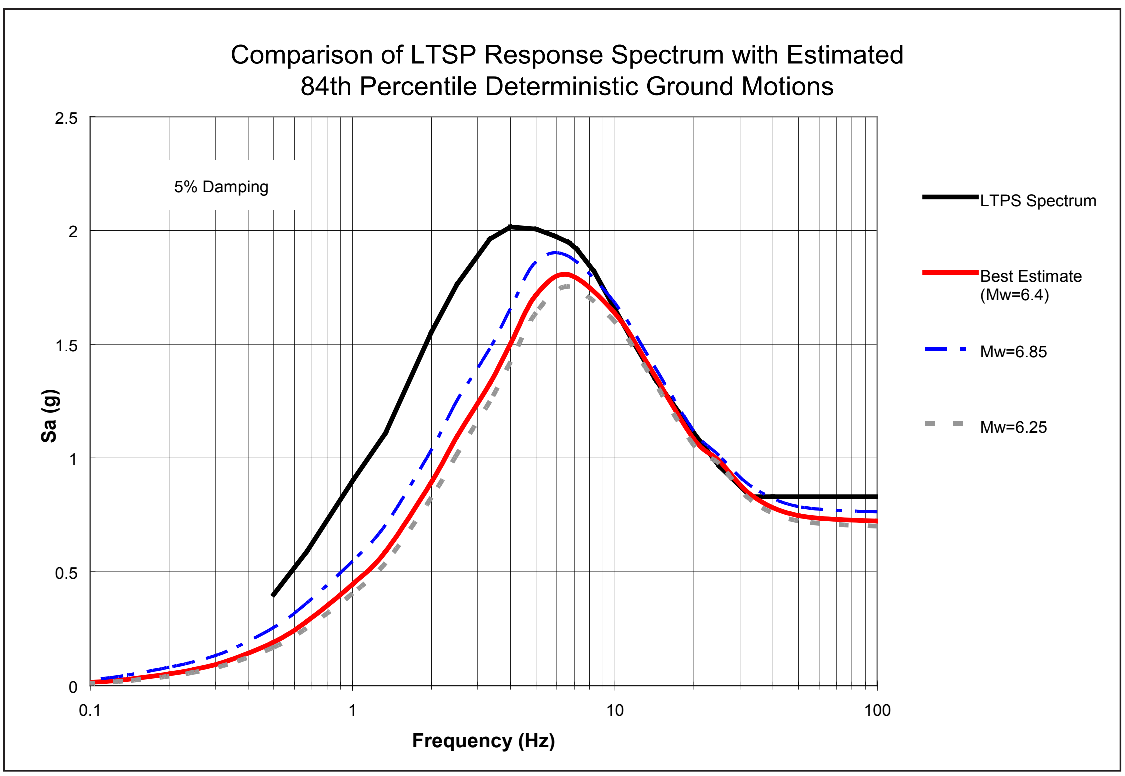
Package Performance Study

Seismic and Tsunami Research Program



Comparison of long-term seismic program (LTSP) response factor for Diablo Canyon with estimated 84% deterministic ground motions from the newly discovered shoreline fault. Average results of 84th percentile ground motions with varying maximum magnitudes.

PACKAGE PERFORMANCE STUDY

Background

The NRC staff has conducted (or sponsored) a series of studies assessing the risks associated with transporting spent nuclear fuel. The latest study, known as the Package Performance Study (PPS), proposes full-scale testing and analyses of spent nuclear fuel transportation (SNFT) casks.

The purpose of the PPS is to enhance public confidence in the ability of SNFT casks to withstand the effects of severe accidents during transportation without endangering public health and safety by exposure to radiation.

Approach

The staff plans to conduct the PPS in three steps. Step 1 will reaffirm staff and industry practices, and enhance public confidence in the use of computer modeling and scale-model tests as a basis for certification of the SNFT casks. To achieve this, the study will use detailed structural simulation and analysis of full-scale and scale models of two SNFT casks and compare their results with the drop tests previously performed on these casks by the German Federal Institute for Materials Research and Testing (BAM).

In Step 2, the study will include computer simulations of a realistically severe train accident scenario (demonstration test) for the two SNFT casks previously subjected to regulatory drop tests at the BAM facility. The study will use the results of these computer simulations, the results of the analyses of drop tests, and data from those drop tests to compare the response of SNFT casks during a demonstration test and regulatory drop tests. Step 2 also will include computer simulations of drop tests and demonstration test scenarios on two additional SNFT casks likely to be used to transfer spent nuclear fuel to a repository.

Step 3 will demonstrate visually the performance of an NRC-certified SNFT cask in a realistically severe accident and demonstrate the ability to predict such performance by analysis using computer-based simulations. A demonstration test will use one of the SNFT casks likely to be used to transport spent nuclear fuel to a repository and selected for the computer simulations in Step 2.

DEMONSTRATION TEST

The plan for the demonstration test calls for a fully assembled rail SNFT cask containing surrogate fuel assemblies tied to and supported on a carrier railcar. This plan calls for a train to impact the carrier railcar at a 90-degree angle on a simulated rail crossing. The impacting train consists of a locomotive with several freight railcars. That locomotive is similar to the one that various railroads use for hauling freight cars. The selected train impact speed is 60 miles per hour, which represents a realistic and conservative scenario. The transportation cask is to have a current NRC certificate of compliance to transport commercial spent nuclear fuel, in accordance with Title 10 of the Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

Following the crash test, the demonstration test planning calls for subjecting the transportation cask to a fire test involving a fully engulfing, optically dense hydrocarbon fire for the duration of one-half hour.

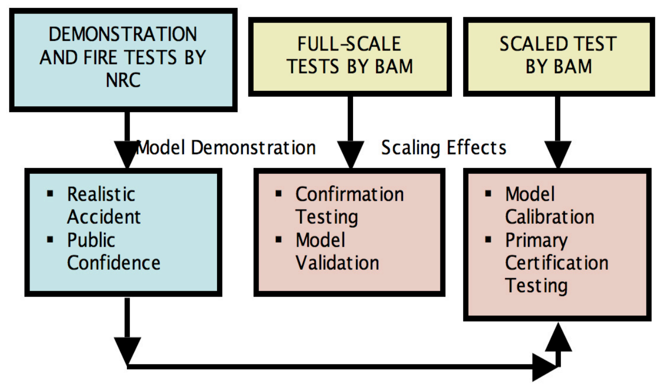


Figure 10.1 Activities of the Package Performance Study

SCHEDULE

The PPS schedule depends on the Commission's approval of resource allocation, the availability of design and test data from BAM, and the U.S. Department of Energy's schedule for ordering an SNFT cask that is likely to be used to transport spent nuclear fuel to a repository. On the basis of current information, the staff expects to complete the activities of Step 1 in the fourth quarter of fiscal year 2011.

For More Information

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SEISMIC AND TSUNAMI RESEARCH PROGRAM

Background

Over the last decades, significant advances have been made both in the ability to assess seismic hazard and in earthquake-resistant engineering. In addition to specific technical advances, both areas have moved toward a more integrated probabilistic and performance-based design methodology. Seismic research at the NRC is focused on bringing the latest technical advances to the regulatory process and on exploring topics unique to nuclear facilities. The current seismic research program focuses on three key areas: seismic and tsunami hazard assessments, earthquake-resistant design, and development of the technical basis for implementation of performance-based design in NRC guidance. Three key research projects in these areas are briefly described below.

Approach

TSUNAMI HAZARD ASSESSMENT FOR THE U.S. ATLANTIC AND GULF COASTS

In the past, significant research has been focused on the tsunami hazard of the U.S. Pacific coast. While the tsunami hazard for the U.S. east and gulf coasts (USEGC) is known to be lower than in the Pacific, the actual hazard is not well understood. This is because an important tsunamigenic source in the USEGC, submarine landslides, has not been well characterized. To address this uncertainty, a two-phase project is being undertaken. In Phase 1, the NRC is working with the U.S. Geological Survey (USGS) to develop a database of all the seismic and landslide tsunamigenic sources that may impact the USEGC. This database will be used both for reviews of individual plant applications and as input for Phase 2, which will focus on tsunami generation and propagation modeling (being done by the National Oceanic and Atmospheric Administration, USGS, and Texas A&M University) to better understand the possible impacts that the identified sources could have on the coasts. This research will greatly improve the NRC's understanding of the tsunami hazard at existing and proposed sites near these two coasts.

NEXT GENERATION ATTENUATION RELATIONSHIP DEVELOPMENT FOR THE CENTRAL AND EASTERN UNITED STATES

The prediction of ground motions at a site for an earthquake with a specific magnitude and distance has always constituted a significant source of uncertainty in seismic hazard results. This research program will develop new ground motion prediction equations for the Central and Eastern United States by following up on the successful multi-investigator project known as the Next Generation Attenuation (NGA) Relationship project that focused on the Western United States. That project, coordinated by the Pacific Earthquake Engineering Research Center, produced a set of consensus ground motion attenuation relationships that are now viewed as the state of the practice. The NGA-East project also represents a multi-investigator approach that will focus on four areas: earthquake records database development, site response characterization, numerical modeling of earthquake wave propagation, and development of ground motion prediction equations. Because the results of this project will have broad application, the NRC, the Electric Power Research Institute (EPRI), the U.S. Department of Energy (DOE), and USGS are jointly sponsoring the work The NGA-East project, in conjunction with the ongoing Central and Eastern United States Seismic Source Characterization Project for Nuclear Facilities, a joint undertaking with DOE and EPRI, will provide a set of state-of-the-art seismic hazard assessment tools for new nuclear facilities.

PRACTICAL PROCEDURES FOR IMPLEMENTING THE SENIOR SEISMIC HAZARD ANALYSIS COMMITTEE GUIDELINES AND UPDATING EXISTING PROBABILISTIC SEISMIC HAZARD ANALYSES

In an effort to standardize probabilistic seismic hazard analyses (PSHAs), the NRC sponsored the development of NUREG/CR-6372, "Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts," issued April 1997. While the guidelines developed in this report by the Senior Seismic Hazard Analysis Committee (SSHAC) provide a framework for performing PSHAs of different levels of complexity, they do not provide specific details on how to implement PSHAs. In the years since the report's publication, domestic and international projects have acquired practical experience in conducting PSHA in accordance with the SSHAC guidelines. The objective of this project is to capture this experience and knowledge in a NUREG-series report that will complement the SSHAC guidelines by providing practical guidelines for implementing the SSHAC framework, by capturing lessons learned during SSHAC Level 3 and Level 4 projects, and by providing practical guidelines for updating SSHAC-based PSHAs when new information becomes available.

ADDITIONAL SEISMIC RESEARCH PROJECTS

- cooperative research program with USGS on seismic source characterization
- seismic analysis of advanced reactor designs
- random vibration theory-based site response

For More Information

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CHAPTER 11: SEVERE ACCIDENT RESEARCH

Melt Coolability and Concrete Interaction Follow-on Program

State-of-the-Art Reactor Consequence Analysis

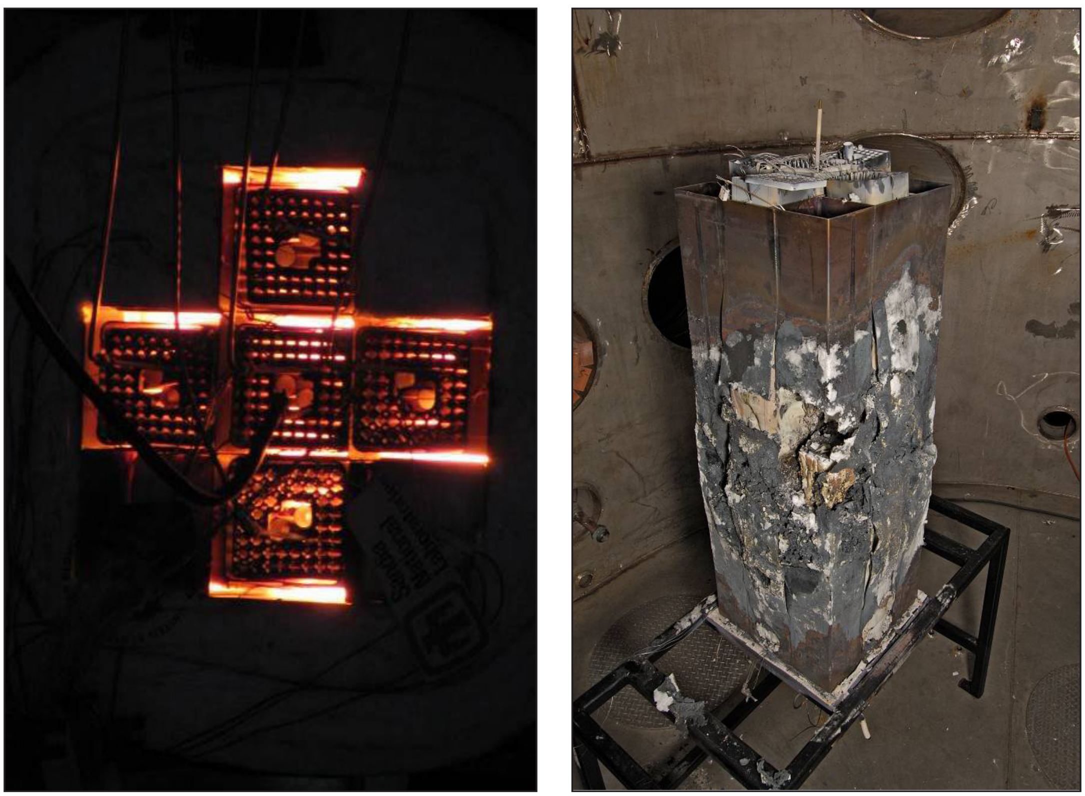
MELCOR Accident Consequence Code System
and Its New Graphical User Interface, WinMACCS

Containment Analyses

Severe Accidents and the MELCOR Code

Phebus-Fission Products and Phebus-International
Source Term Program

Zirconium Fire Research



BWR assembly during and after Zirconium fire propagation

MELT COOLABILITY AND CONCRETE INTERACTION FOLLOW-ON PROGRAM

Background

The goal of the Melt Coolability and Concrete Interaction Follow-on (MCCI-2) research program is to conduct reactor material experiments and associated analysis to achieve the following two technical objectives: (1) resolve the ex-vessel debris coolability issue through a program that focuses on providing both confirmatory evidence and test data for coolability mechanisms identified in earlier integral tests and (2) address remaining uncertainties related to long-term, two-dimensional core-concrete interactions (CCIs) under both wet and dry cavity conditions. Achievement of these objectives will demonstrate the efficacy of severe accident management guidelines for existing plants and provide the technical basis for better containment designs.

Figure 11.1: SSWICS test stand

Approach

The risk to the public from nuclear power generation arises if an accident progresses to the point where fuel degradation occurs. In the most extreme postulated event sequences, molten fuel could hypothetically fail the reactor vessel, leading to melt discharge into the containment. The NRC has computer codes that simulate the progression of severe accidents. The agency uses these codes to evaluate the consequences of beyond-design-basis accidents, and thus they are an important tool in the transition to a more risk-informed regulatory framework.

The improved understanding of phenomenological behavior under both design- and beyond-design-basis accident sequences has direct implications for the analytical methods. The improved models for debris coolability and molten CCI gained from the MCCI program will reduce uncertainties when applied to risk assessments of the current fleet and new plant designs.

In terms of the ex-vessel debris coolability, two types of separate effects tests were conducted to provide data on key cooling mechanisms. In the original MCCI program, the melt eruption test focused on providing data on the melt entrainment coefficient under well-controlled experimental conditions. The entrainment rate data provided coefficient estimates that can be used in models for evaluating the effect of melt ejection on mitigation of accident sequences. The small-scale water ingression and crust strength (SSWICS) tests provide data on the ability of water to ingress into core material, thereby augmenting the otherwise conduction-limited heat transfer process. These tests showed that the dry-out limit is a strong function of melt composition, but weakly dependent on system pressure. Crust strength data obtained as part of this work verified the concept of sustained melt/crust contact as the result of crust instability in the typical cavity span of most power plants.

With regard to CCI, the approach was to conduct integral effect tests that replicate as closely as possible the conditions at plant scale, thereby providing data that can be used to verify and validate the codes directly. To augment the amount of information gathered from these tests, the experiments were flooded from above after a predefined concrete ablation depth was reached to provide debris coolability data under conditions involving late-phase flooding. The input power levels for the tests were selected so that the heat fluxes from the melt to concrete surfaces and the upper atmosphere were initially in the range of the heat flux expected early in the accident sequence. The results of these tests indicate that the directional power split is a strong function of concrete characteristics: the split is approximately unity for limestone/common sand concrete, whereas the split is significantly larger than unity for siliceous concrete.

In terms of the applicability to plant conditions, the tests provided information that contribute to the database for reducing modeling uncertainties related to two-dimensional, molten CCI under both wet and dry cavity conditions. Data from these and other test series thus form the technical basis for developing and validating models of the various cavity erosion and debris cooling mechanisms. These models can then be deployed in integral codes that are able to link the interrelated phenomenological effects, thereby forming the technical basis for extrapolating the results to plant conditions. Furthermore, current experiments are designed to address special mitigation

Figure 11.2: Posttest debris from CCI tests

features that can enhance coolability in new reactor designs. For example, an integral test will be conducted to investigate the effect of cooling the molten corium from the bottom through a system of pipes, which is expected to expedite stabilization.

As part of the project, analytical models were upgraded to include the experimental findings related to debris coolability and to scope out an approximate debris coolability envelope for the two concrete types that were evaluated as part of the program. The results for limestone/common sand concrete indicate that melt stabilization may be achievable in under 1 meter of axial ablation as long as the cavity is flooded before the melt concrete content exceeds 15 weight percent for initial melt depths ranging up to 40 centimeters. For siliceous concrete, stabilization may not be achieved in under 1 meter of ablation unless the initial melt depth is fairly shallow (i.e., less than 20 centimeters), and the cavity is flooded before the melt concrete content exceeds 10 weight percent.

As a whole, the results of the CCI tests have indicated trends in the ablation front progression that cannot be fully explained on the basis of the current understanding of the phenomenology involved with this type of physical process. These trends are currently under investigation, and data acquired in the MCCI-2 program will allow analysts to extrapolate experimental results to plant scale with higher confidence.

For More Information

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STATE-OF-THE-ART REACTOR CONSEQUENCE ANALYSIS

Background

The NRC is conducting a project to estimate the possible public health and safety consequences in the unlikely event of a commercial nuclear power plant accident releasing radioactive material into the environment. The agency has used accident assessment tools since their creation in the 1970s to help focus attention on the reactor design and operational features that are most important to safety. The State-of-the-Art Reactor Consequence Analysis (SOARCA) takes maximum advantage of hundreds of millions of dollars of national and international reactor safety research and reflects improved plant design, operation, and accident management implemented over the past 25 years. Using computer models and simulation tools, the NRC plans to develop a set of realistic consequence estimates of very unlikely accidents at an initial set of no more than eight U.S. reactor sites representative of reactor and containment designs used in the United States. This kind of research into accident phenomena, such as core damage and containment performance, has provided the basis for industry procedures to mitigate such accidents.

Approach

SOARCA'S PLANT-SPECIFIC BASIS

Accident progression and consequences will be developed for each of the reactor and containment designs in use in the United States: General Electric boiling-water reactors (BWRs) with Mark I, Mark II, and Mark III containments; Westinghouse pressurized-water reactors (PWRs) with ice condenser, subatmospheric, and large dry containments; Combustion Engineering PWRs; and Babcock &Wilcox PWRs. A combination of NRC and Sandia National Laboratory staff are performing the work.

SOARCA PROCESS AND SCHEDULE

This study uses state-of-the-art information and calculation tools to develop best estimates of radioactive material released into the environment based on the reactor/containment classes. The study assesses those releases to determine best estimates of offsite radiological consequences, including uncertainties in those results.

Some areas considered in these new assessments include (1) design-specific reactor accident sequence progression; (2) design-specific containment failure timing, location, and size; (3) site-specific emergency planning assumptions including evacuation and sheltering; (4) credit for operator actions; and (5) site-specific meteorological conditions and updated population data.

The project uses standardized plant analysis risk models or other available probabilistic risk analyses to determine the sequences and initiating events (internal and external) that should be considered for inclusion in the study. Scenario selection is based on an estimated release frequency of 10-6 per reactor-year (one in a million). Insights gained from NRC research programs on containment performance and severe accident phenomena are also being incorporated. A computer code that models accident progression (MELCOR) is being used to estimate the radioactive material released into the environment for each scenario. Finally, a computer code that models offsite consequences (MACCS2) is being used to generate site-specific consequence estimates that account for site-specific weather conditions, population distribution, and emergency planning assumptions.

SOARCA Status

Of the initial scope of no more than eight plants, the staff has been able to secure three volunteers. Of those three plants, the analyses of a BWR and a PWR plant have been completed, and an external peer-review of these results is underway. The staff plans to initiate an uncertainty study in late 2009 and expects the public release of the results from these two plants by the middle of 2010. Preliminary results shown in Figure 11.3 demonstrate that current predictions are dramatically different than those of previous studies.

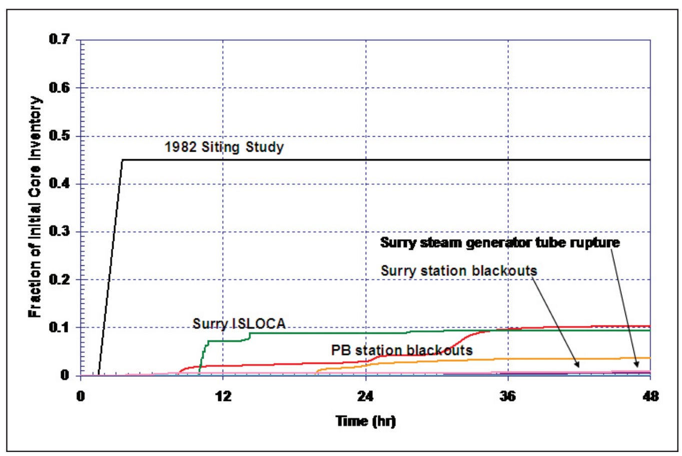

Figure 11.3: Iodine Release for Unmitigated Cases

For More Information

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MELCOR ACCIDENT CONSEQUENCE CODE SYSTEM AND ITS NEW GRAPHICAL USER INTERFACE, WINMACCS

MACCS2

The NRC uses the MELCOR Accident Consequence Code System (MACCS) to estimate the offsite consequences from radioactive material released into the atmosphere. The MACCS code, first released in 1987, was developed to remedy limitations in CRAC, a code developed in the 1970s for the WASH-1400 study (the Reactor Safety Study: A probabilistic risk assessment (PRA) of hypothetical nuclear power plant accidents). The MACCS code has evolved through the years into a more complex and realistic set of models for offsite consequences.

MACCS2 and Its Graphical User Interface WinMACCS

Recently, a new Version 2.4 of MACCS2 has been released, along with the graphical user interface WinMACCS Version 3.4.

Meteorological sampling capabilities have been maintained from the earliest version of the code (CRAC). Now, uncertainty in source term and in many other parameters, including parameters related to emergency response, can be easily input through WinMACCS. The two most important improvements implemented in MACCS2/WinMACCS are the ability to easily evaluate the impact of parameter uncertainty and the ability to model alternative dose-response relationships for latent cancer fatality evaluation (e.g., the Health Physics Society type of threshold for latent cancer).

OTHER IMPROVEMENTS IN MACCS V.2.4

- more cohorts for evacuation (20)
- potassium iodine ingestion model
- more compass directions (up to 64)
- more plume segments (up to 200)
- more aerosol bins and chemical groups (20)
- multiple meteorological data intervals (15, 30, or 60 minutes)
- diurnal mixing-height model
- long-range, lateral plume spread model
- improved Briggs plume rise model
- plume meander based on Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants"
- dynamic memory allocation

FEATURES OF WINMACCS V.3.4

- cyclic handling of MELCOR source terms
- graphical manipulation of MACCS2 network evacuation parameters (e.g., direction and speed)
- editing of grand mean and arbitrary quantile levels for uncertainty calculations
- option to remove food pathway

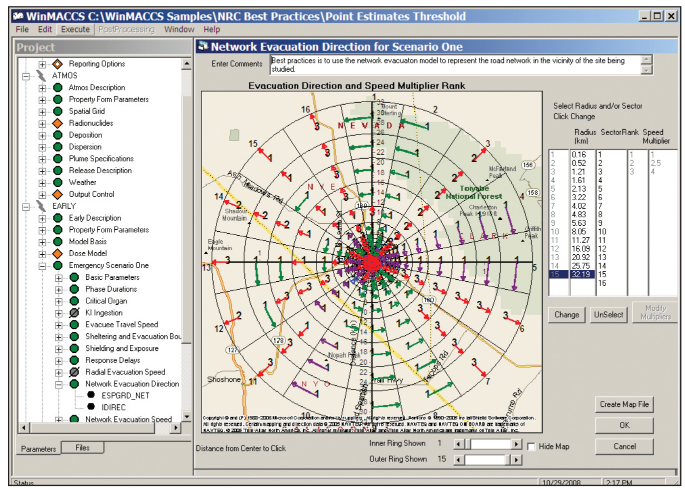

Figure 11.4: Graphical view of WinMACCS (network evacuation model is shown)

MACCS2/WinMACCS Uses

Offsite consequence evaluations are used to evaluate the consequences of severe radiological accidents as part of the environmental reports and environmental impact statements for early site permits, to support plant-specific evaluation of severe accident mitigation alternatives required as part of the environmental assessment for license renewal, for emergency planning, and for cost/benefit analyses.

New Work

Work is ongoing to update the MACCS2 code based on current technology. The new work will develop and implement a more detailed and up-to-date economic model and an approach for treating complex wind patterns. Other modifications will allow additional flexibility in specifying population groups (i.e., at a specific location in a defined grid area and with a finer resolution) as a function of distance from the release location.

For uncertainty analyses, capabilities are being implemented to sample dose conversion factor values and distribute numerous MACCS2 runs into a computer network cluster; this effort will include postprocessing of the results.

The current schedule envisions a new version of MACCS2/WinMACCS by December 2011.

For More Information

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CONTAINMENT ANALYSES

Background

The containment encloses the reactor system and is the final barrier against the release of radioactive fission products in the event of a breach of either the primary or secondary coolant system. Evaluations entail a variety of postulated design-basis and beyond-design-basis (including core melt) events, involving accident progression and radiological source term calculations. Computer codes, such as CONTAIN and MELCOR, are used in licensing reviews (including new reactor designs), in addressing regulatory safety issues (e.g., generic safety issues, risk-informing regulations), and in addressing changes in containment safety margins. These computer codes serve as a repository of accumulated knowledge in the area of containment and severe accident research and will be improved as new information is collected and disseminated.

Approach

CONTAIN and MELCOR are state-of-the-art lumped parameter codes which offer a greater robustness in analyzing a broader array of reactor containment designs.

The CONTAIN code is used to perform containment response analysis in light-water reactor plants under postulated design-basis accident (DBA) and beyond-DBA events in order to predict thermal-hydraulic conditions (i.e., pressure and temperature) inside containment. This code can also be used to perform best-estimate containment thermal-hydraulic analyses; to confirm applicant and licensee analyses; to review industry models and correlations; and to benchmark other RES codes incorporating containment-related models.

The MELCOR code is used to perform integrated analysis (encompassing the reactor coolant system and the containment building) in light-water reactor plants under postulated beyond-DBA events (including core melt) in order to predict the accident progression from thermal-hydraulic conditions inside the reactor coolant system to fission product release and transport to the environs. The code is also used to analyze selected DBA containment applications. Specifically, because of the integrated nature of this code, MELCOR has been used successfully to perform design-basis containment analysis for the Economic Simplified Boiling-Water Reactor (ESBWR). Moreover, MELCOR is being used in analyses of the Evolutionary Power Reactor (EPR) and the U.S.-Advanced Pressurized Water Reactor (U.S.-APWR) containment design basis.

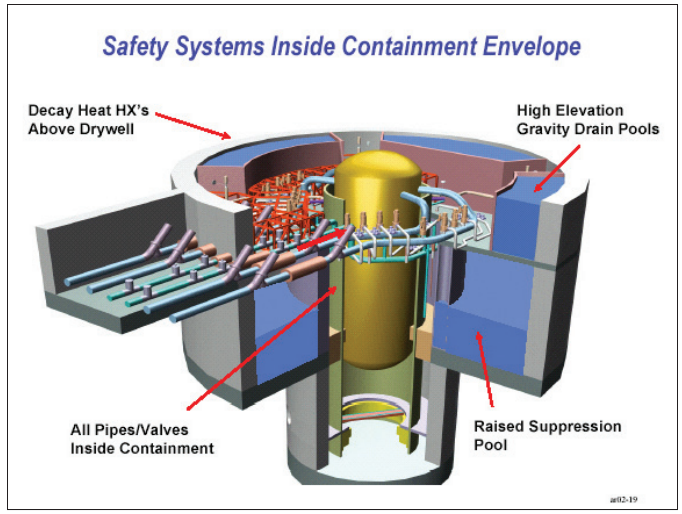

Figure 11.5: ESBWR containment model

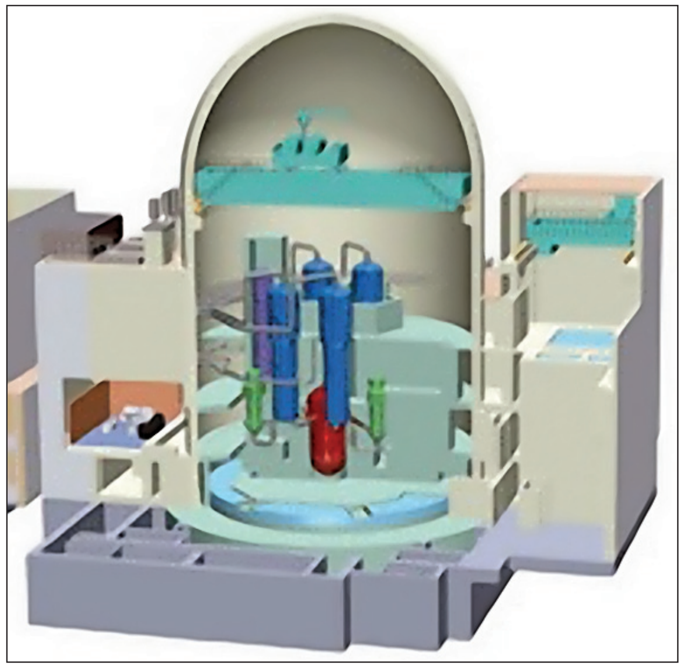

Figure 11.6: U.S.-APWR containment model

For More Information
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SEVERE ACCIDENTS AND THE MELCOR CODE

Background

The risk to the public from nuclear power generation arises if an accident progresses to the point where fuel degradation occurs, and large quantities of radioactive materials are released into the environment. The NRC has invested heavily in the investigation of severe reactor accidents and has developed computer codes for the analysis of severe accident phenomena and progression. Expertise on severe accident phenomenological behavior and a quantitative predictive capability for simulating the response of nuclear power systems to severe accidents are essential to the NRC's mission. The role of such expertise and analytical capability is potentially wide ranging in the regulatory environment, which includes the transition to a more risk-informed regulatory framework and to the study of vulnerabilities of nuclear power plants. MELCOR represents the current state of the art in severe accident analysis, which has developed through NRC and international research performed since the accident at Three Mile Island in 1979.

Approach

The MELCOR code is a fully integrated, engineering-level computer code whose primary purpose is to model the progression of postulated accidents in light-water reactors as well as in nonreactor systems (e.g., spent fuel pool and dry cask). MELCOR is a modular code consisting of three general types of packages: (1) basic physical phenomena (i.e., hydrodynamics—control volume and flow paths, heat and mass transfer to structures, gas combustion, aerosol and vapor physics), (2) reactor-specific phenomena (i.e., decay heat generation, core degradation, ex-vessel phenomena, sprays and engineering safety systems), and (3) support functions (thermodynamics, equations of state, material properties, data-handling utilities, equation solvers). These packages model the major systems of a nuclear power plant and their associated interactions. MELCOR 1.8.6 (Fortran 77) was released in September 2005; the code modernization effort resulted in the release of MELCOR 2.0 (Fortran 95) in September 2006. The latest version (MELCOR 2.1) was released in September 2008. Future activities will include development and implementation of new and improved models to predict the severe accident behavior of advanced non-light-water reactor designs.

Severe accident competency will be needed to evaluate new generic severe accident issues and to address risk-informed regulatory initiatives and operating reactor issues associated with plant changes, as in the case of steam generator tube integrity. Licensees will continue to pursue plant modifications that require assessment of incremental risk impacts that will necessitate analysis of phenomena related to severe accidents.

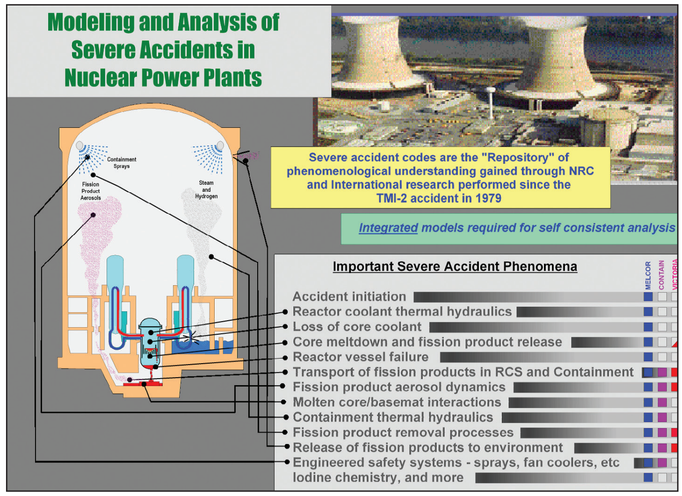

Figure 11.7: MELCOR modeling capabilities

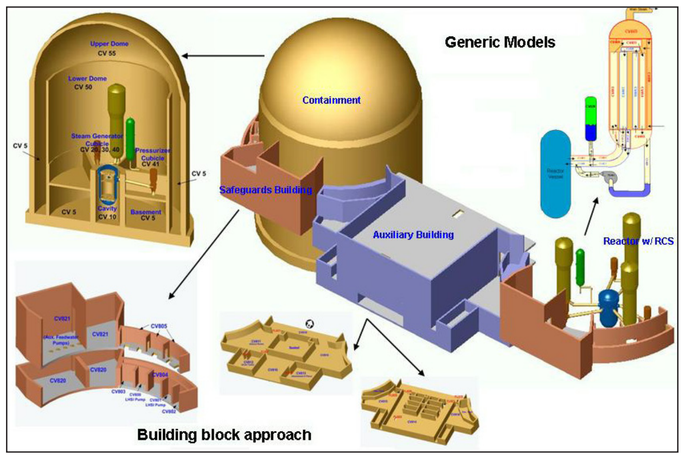

Figure 11.8: MELCOR plant modeling approach

APPLICATIONS

The improved understanding of phenomenological behavior and modeling in severe accidents and their implementation in MELCOR has had a direct impact on the analytical methods and criteria adopted for design-basis accidents (e.g., source term research and the revised source term). The development of best-estimate severe accident models in the future is expected to improve the licensing evaluation models. The development of best-estimate models reveals, quantitatively, margins in existing models.

Activities associated with the development, assessment, and applications of MELCOR include the following:

- safety analysis and risk decisionmaking
  - revision of the NRC's alternative source term (NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," issued February 1995) for high-burnup fuel and mixed-oxide (MOX) fuel

- new reactor certification (AP-1000, Economic Simplified Boiling-Water Reactor [ESBWR], Evolutionary Power Reactor [EPR], Advanced Pressurized-Water Reactor [APWR])

- experimental analyses and code validation activities
- nuclear power plant beyond-design-basis accidents
- aerosol transport and deposition in steam generators during bypass accidents
- risk of steam generator tube rupture induced by a severe accident
- effects of air ingress on fission product release
- vulnerabilities of spent fuel pool to accidents
- state-of-the-art consequence analysis

National laboratories, universities (e.g., Texas A&M), and international organizations (e.g., Paul Scherrer Institute in Switzerland and the Institut de Radioprotection et de Sûreté Nucléaire [IRSN] in France) are involved in the MELCOR code development effort.

INTERNATIONAL COLLABORATIONS

The following are examples of international collaborations that resulted in MELCOR improvements:

- NRC Cooperative Severe Accident Research Program (CSARP).
- MELCOR Code Assessment Program (MCAP).
- Phébus-Fission Products (Phébus-FP), VERCORS, and follow-on program (Phébus-Source Term Separate Effects Test Project [STSET])—IRSN: Fission product releases and degradation of UO2 fuel (including burnup greater than 40 gigawatt day per metric ton) and MOX fuel under severe accident conditions, and the effects of air ingress on core degradation and fission product release. Results are used to validate the NUREG-1465 source term and MELCOR code.
- German QUENCH experiment program, investigating overheated fuel.
- ARTIST—Paul Scherrer Institute (Switzerland): To investigate experimentally the potential mitigation of radioactive material releases through the secondary side of a steam generator. Results from this research would allow the NRC to decide whether improved source term bypass models are needed.
- Molten Core Concrete Interaction Program— Organization for Economic Cooperation and Development and Argonne National Laboratory (U.S.): Separate effects experiments to further address the ex-vessel debris coolability issue. The results will be used to develop coolability models for incorporation into severe accident codes.
- Behavior of Iodine Project (BIP)—Nuclear Energy Agency, Committee on the Safety of Nuclear Installations (France): Experimental investigations of behavior of iodine in containment during post-severe-accident conditions for computer code model development and validation. BIP addresses the uncertainties related to iodine behavior (especially with respect to iodine interactions with paints). With complementary testing at Atomic Energy of Canada Limited and at IRSN, the state of the art on modeling of iodine behavior in the containment can be advanced and quantified. Adequate modeling of iodine behavior is crucial in determining the need for pH control in containment sump. The proposed research will complement the ongoing IRSN of France Phébus-FP and follow-on program Phébus-STSET.

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PHÉBUS-FISSION PRODUCTS AND PHÉBUS-INTERNATIONAL SOURCE TERM PROGRAM

Background

In the unlikely event of a commercial nuclear power plant accident, the NRC has developed computer codes for the analysis of severe accident phenomena and progression. The NRC maintains its analytical tool to evaluate severe accident risk in the transition to a more risk-informed regulatory framework and for use in the study of vulnerabilities of nuclear power plants.

Future needs include development of insights into the severe accident behavior of advanced reactor designs and extending the expertise acquired on current reactor designs to address future design-specific considerations.

The improved understanding of phenomenological behavior and modeling in severe accidents has had direct implications for the analytical methods and criteria adopted for design-basis accidents (e.g., source term research and the revised source term described in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," issued February 1995). The development of improved severe-accident, best-estimate models in the future will also be likely to influence the improvement of licensing evaluation models since the development of best-estimate modeling reveals, quantitatively, margins in existing modeling.

Approach

The purpose of the Phébus-Fission Products (Phébus-FP) program is to conduct integral tests to study the processes governing the transport, retention, and chemistry of fission products under severe accident conditions in light-water reactors. The aim of the follow-on program, the Phébus-International Source Term Program (Phébus-ISTP), is to conduct separate-effects experiments in various experimental facilities to resolve findings from Phébus-FP and continue the investigation done in Phébus-FP (e.g., research into air ingress and fission product chemistry, fission product release from high burnup fuel and mixed-oxide fuel, iodine chemistry, and control rod oxidation and degradation).

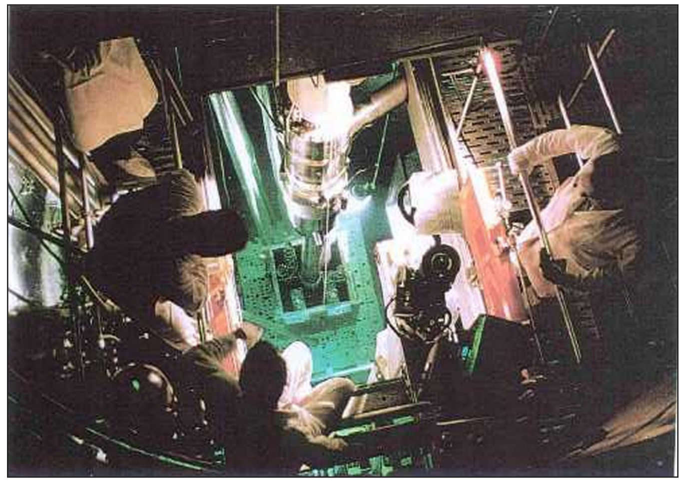

Figure 11.9: Phébus reactor and the test loop (top view)

The key features of the Phébus-FP Program include the following:

- The program uses a loop-type test reactor with a low-enrichment driver core of 20 to 40 megawatt power, using fuel rod elements.
- Core cooling and moderation are achieved by demineralized light water.
- Light water and graphite are used as reflectors.
- Tests (four out of five) primarily involve a cluster of 20 fuel rods (about 10 kilograms), 1-meter long, located in the central hole of the driver core of the PHÉBUS-FP reactor. One test (FPT-4) consists of a rubble bed instead of fuel rods.
- The facility is instrumented to measure fission product release, deposition in the primary circuit, and release to the containment.
- The facility includes a representative primary circuit, including a steam generator tube, containment, and a sump.

Applications

The Phébus-FP integral experimental data support the assessment and development of new MELCOR models (e.g., iodine chemistry, iodine behavior in containment, and fuel degradation). The improved MELCOR is used for safety analysis and risk decisionmaking. The data were also used to confirm many of the important features of the NRC revised/alternative source term as specified in NUREG-1465, such as the finding that iodine release is predominantly in aerosol form, with allowance for small fractions (5 percent) in gaseous form.

The results of the Phébus-FP tests indicate that controlling the sump pH may not significantly impact the development of a gaseous iodine concentration in the reactor containment in the immediate aftermath of an accident involving core degradation. This finding may have implications for the sump screen blockage issue (Generic Safety Issue 191, "Assessment of Debris Accumulation on PWR Sump Performance"). Moreover, on the basis of experiments reported in NUREG/CR-6917, "Experimental Measurements of Pressure Drop Across Sump Screen Debris Beds in Support of Generic Safety Issue 191," issued February 2007, interactions between the chemicals used to control sump pH and some insulation materials dispersed to the sump can exacerbate sump screen blockage.

The Phébus-ISTP is expected to provide prototypical experimental data on air ingress, fission product chemistry, and fission product release from high-burnup fuel and mixed-oxide fuel for MELCOR code assessment and development. The data will enable the NRC to address the issue of ruthenium (RuO4) behavior under accident conditions in an air environment. If the ruthenium released is significant, it will impact the evaluation of early and latent health effects under Title 10 of the Code of Federal Regulations (10 CFR) Part 100, "Reactor Site Criteria." In addition, assessments will be made of the separate-effects results on NUREG-1465 (the NRC revised/alternative source term). NUREG-1465 is used for design-basis accident analysis in operating plants and in new reactor design certification reviews (10 CFR Part 100)

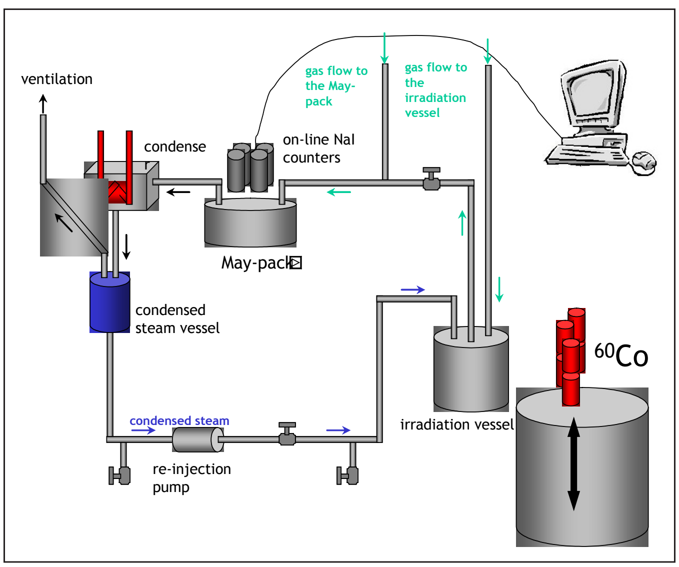

Figure 11.10: EPICUR experimental setup (one of the experiments under the Phébus-STSET program)

For More Information

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ZIRCONIUM FIRE RESEARCH

Background

In 2001, the NRC staff performed an evaluation of the potential accident risk in a spent fuel pool (SFP) at decommissioning plants in the United States. NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," described a modeling approach for a typical decommissioning plant with design assumptions and industry commitments, the thermal-hydraulic analyses performed to evaluate spent fuel stored in the SFP at decommissioning plants, the risk assessment of SFP accidents, the consequence calculations, and the implications for decommissioning regulatory requirements. Some of the assumptions in the accident progression in NUREG-1738 were known to be necessarily conservative, especially the estimation of the fuel damage. The NRC continued SFP accident research by applying best-estimate computer codes to predict the severe accident progression following various postulated accident initiators. The best-estimate computer code studies identified various modeling and phenomenological uncertainties that prompted a need for experimental confirmation. The present experimental program was undertaken to address thermal-hydraulic issues associated with complete loss-of-coolant accidents in pressurized-water reactor (PWR) SFPs. The NRC also plans to expand the study to include accidents in the SFPs of operating power plants.

Objective

The objective of this project is to provide basic thermal-hydraulic data associated with an SFP complete loss-of-coolant accident. The accident conditions of interest for the SFP were simulated in a full-scale prototypic fashion (electrically heated, prototypic assemblies in a prototypic SFP rack) so that the experimental results closely represent actual fuel assembly responses. A major impetus for this work is to facilitate code validation (primarily MELCOR) and reduce modeling uncertainties within the code.

Testing Approach

The study will be conducted in two phases. Phase 1 will focus on axial heating and burn propagation. A single full-length test assembly will be constructed with zirconium-alloy clad heater rods. As demonstrated in the previous study for boiling-water reactors (BWRs), the thermal mass of the compacted magnesium-oxide (MgO) powder used to make the electric heater is an excellent match to spent fuel. The assembly will be characterized in two different-sized storage cells and conclude with an ignition test to determine where in the assembly ignition first occurs and the nature of the burn along the axis of the assembly. The insulated boundary conditions will experimentally represent a "hot neighbor" situation, which is an important bounding scenario.

Phase 2 will address radial heating and burn propagation and will include effects of fuel rod ballooning. Five full-length assemblies will be constructed in which the center assembly will be of the same heated design as used in Phase 1. The four peripheral assemblies will be unheated but highly prototypic, incorporating prototypic fuel tubes and end plugs. These boundary conditions experimentally represent a "cold neighbor" situation, which complements the bounding scenario covered by Phase 1. The peripheral fuel rods will be filled with high density MgO ceramic, sized to precisely match the thermal mass of spent fuel. Studies using this test assembly will conclude with a fire test in which the center assembly is heated to ignition, which eventually propagates radially to the peripheral assemblies. All of the fuel rods in two of the four peripheral assemblies will be pressurized with helium so that these fuel rods will balloon when the zirconium-alloy cladding reaches a high enough temperature. The two peripheral assemblies without pressurized rods will serve as a control for evaluating the effect of ballooning.

ANALYSIS SUPPORT

As in the previous BWR study, all stages of testing will use MELCOR modeling results. Pretest MELCOR modeling results will be used to guide the experimental test assembly design and instrumentation. MELCOR modeling results will also be used to choose experimental operating parameters such as the applied assembly power. At each step in the testing, improvements will be made to the MELCOR model such that confidence in the modeling validity will continually increase.

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11. ABSTRACT *(200 words or less)*

The Office of Nuclear Regulatory Research (RES) develops technical tools, analytical models, and experimental data with which the agency assesses safety and regulatory issues for operating reactors as well as for new and advanced reactor designs. RES staff develops these tools, models, and data through contracts with commercial entities, national laboratories, and universities, or in collaboration with international organizations.

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