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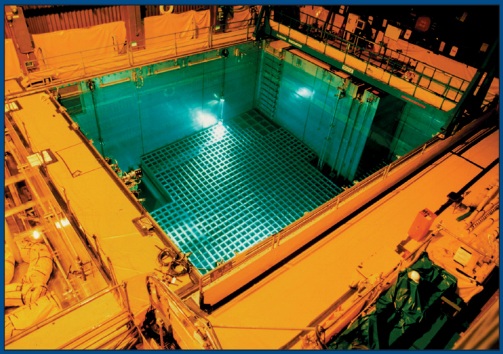


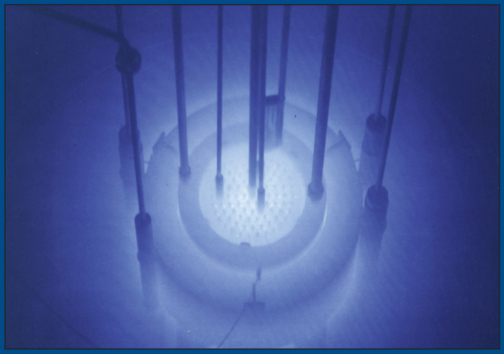

Research Activities
FY 2015–FY 2017

Office of Nuclear Regulatory Research

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Rockville, MD 20852-2738

These standards are available in the library for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from—

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New York, NY 10036-8002
www.ansi.org
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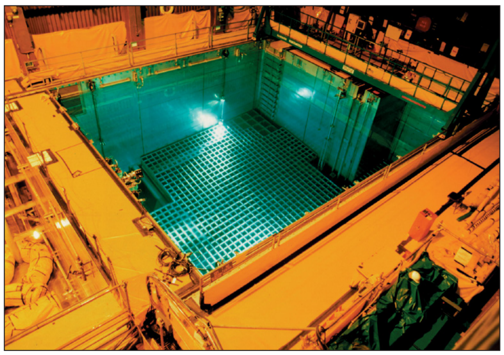
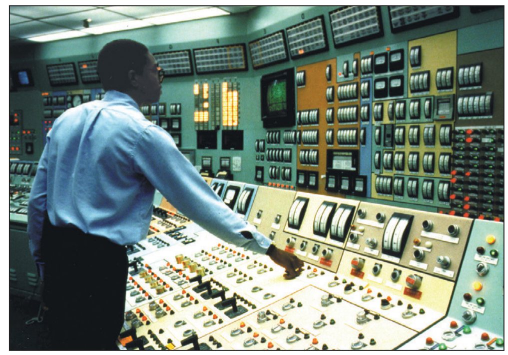
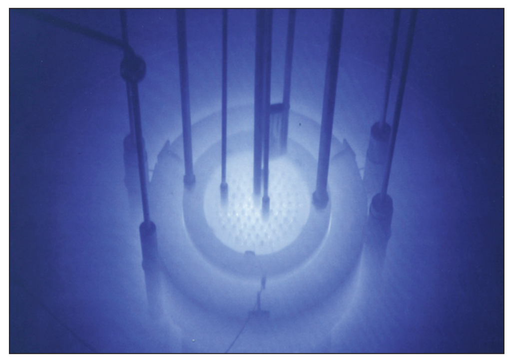
The NUREG series comprises (1) technical and administrative reports and books prepared by the staff (NUREG–XXXX) or agency contractors (NUREG/CR–XXXX), (2) proceedings of conferences (NUREG/CP–XXXX), (3) reports resulting from international agreements (NUREG/IA–XXXX), (4) brochures (NUREG/BR–XXXX), and (5) compilations of legal decisions and orders of the Commission and Atomic and Safety Licensing Boards and of Directors' decisions under Section 2.206 of NRC's regulations (NUREG–0750).

Research Activities
FY 2015–FY 2017







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Abstract

The Office of Nuclear Regulatory Research (RES) supports the regulatory mission of the U.S. Nuclear Regulatory Commission (NRC) by providing technical advice, tools, and information to identify potential safety issues and resolve them as appropriate, make regulatory decisions, and issue regulatory requirements and guidance. This includes conducting confirmatory experiments and analyses, developing technical bases that support the NRC's safety decisions, and preparing the agency for the future by evaluating the safety aspects of new technologies and designs for nuclear reactors, materials, waste, and security.

The NRC faces challenges as the industry matures including potential emergent safety issues, the availability of new technologies, technical issues associated with the deployment of new reactor designs, and knowledge management. The NRC focuses its research primarily on near-term needs related to the oversight of operating reactors as well as to new and advanced reactor designs. RES develops technical tools, analytical models, and experimental data to allow the agency to assess potential safety and regulatory issues. The RES staff uses its own expertise as well as contracts with commercial entities, national laboratories, and universities, or collaborations with international organizations, to develop these tools, models, and data.

This NUREG describes research being conducted by NRC's Office of Nuclear Regulatory Research across a wide variety of disciplines, ranging from fuel behavior under accident conditions to seismology to health physics. This is the fourth issuance of NUREG-1925, revised to capture new research and to update ongoing research projects. This research helps provide the technical bases for regulatory decisions and confirms licensee analyses. RES works closely with the NRC's regulatory offices in the review and analysis of operational events and provides its expertise to support licensing. RES has organized this collection of information sheets by topical areas that summarize projects currently in progress. Each sheet provides the names of the RES technical staff who can be contacted for additional information.

Foreword

A Message from the Director

The Office of Nuclear Regulatory Research (RES) of the U.S. Nuclear Regulatory Commission (NRC) plans, recommends, and implements nuclear regulatory research, confirmatory analyses, standards development, and resolution of potential generic safety issues for nuclear power plants and other facilities and materials regulated by the NRC. The office was established by the U.S. Congress in 1974 to accomplish the NRC mission of protecting the public health and safety, promoting the common defense and security, and protecting the environment. RES partners with other NRC offices, Federal agencies, industry research organizations, international organizations, and universities. This NUREG identifies and describes our key research projects.

Much of our work is shared with the public and stakeholders through NUREG series reports. We issue about 30-40 NUREG reports on a wide variety of topics each year. For example, over the past year, NUREG reports included Technical Basis for Regulatory Guidance on the Alternate Pressurized Thermal Shock Rule (NUREG-2163), Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database (NUREG-2169), Evaluations of NRC Seismic-Structural Regulations and Regulatory Guidance, and Simulation-Evaluation Tools for Applicability to Small Modular Reactors (NUREG/CR-7193), Mechanical Fatigue Testing of High-Burnup Fuel for Transportation Applications (NUREG/CR-7198), and Applying Ultrasonic Testing in Lieu of Radiography for Volumetric Examination of Carbon Steel Piping (NUREG/CR-7204).

Some of the highlighted FY 2015–2017 projects include severe accident analysis (Chapter 4), Level 3 probabilistic risk assessment (Chapter 6), human reliability analysis activities (Chapter 7), seismic and flooding research (Chapter 10), and international and domestic cooperative research (Chapter 15). The NRC also continues to focus on other issues such as dissimilar metal weld cracking inspections and mitigation, cable aging, and other aging-related materials issues, digital instrumentation and control, Fukushima lessons-learned, and new and advanced reactors. These are simply a few of the critical research projects contained in this report that are expected to continue.

We conduct research both in-house and with the use of contractors and interagency agreements. The office's annual budget for contracted work is typically around $50 million (direct contract funds, does not include staff or full cost allocations), which is broken down as follows:

- User needs from NRC's regulatory offices drive over three-fourths of RES activities.
- The Commission drives about 10 percent of RES activities through agency-mandated programs and tasking memoranda.
- A small amount of long-term research supports anticipated NRC regulatory needs on subjects expected to be critical in 5 to 10 years.
- About 3 percent of the office's budget is spent on operations, which includes staff travel and information technology purchases.

Currently, RES has about 220 staff members. This staff continues to reflect diversity in academic degrees, demographics, and technical disciplines. Approximately 30% have PhDs and another 30% have Master's degrees. The wide range of engineering and scientific disciplines includes expertise in thermal-hydraulics, severe accident progression, nuclear materials, human factors and human reliability, health physics, fire protection, seismology, environmental transport, and probabilistic risk assessment.

In summary, we appreciate your interest in and support for nuclear safety and security research. If you have any additional questions or comments on our research projects, please contact the technical staff or the division noted on each specific project summary sheet in this report.

Michael F. Weber
Director, Office of Nuclear Regulatory Research

Research Legislation

The Energy Reorganization Act of 1974, establishes the fundamental role of the Office of Nuclear Regulatory Research to engage in or contract for research to develop recommendations necessary for the performance of NRC's licensing and related regulatory functions. The Joint Explanatory Statement of the Committee of Conference on the Energy Reorganization Act of 1974 states that

> "…the Commission would have an independent capability for developing and analyzing technical information related to reactor safety, safeguards, and environmental protection in support of the licensing and regulatory process."

Section 205 of the Energy Reorganization Act of 1974, as amended, (42 USC 5845) states that the office shall be under the direction of a Director of Nuclear Regulatory Research, who shall be appointed by the Commission, who may report directly to the Commission as provided in section 209, and who shall serve at the pleasure of and be removable by the Commission. The law states that

> "[T]he Director of Nuclear Regulatory Research shall perform such functions as the Commission shall delegate, including:
> (1) Developing recommendations for research deemed necessary for performance by the Commission of its licensing and related regulatory functions.
> (2) Engaging in or contracting for research which the Commission deems necessary for the performance of its licensing and related regulatory functions."

The law also directs the Secretary of Energy (successor to the Administrator, Energy Research and Development Administration) and the head of every other Federal agency to

> (1) "Cooperate with respect to the establishment of priorities for the furnishing of such research services as requested by the Commission for the conduct of its functions;
> (2) Furnish to the Commission on a reimbursable basis, through their own facilities or by contract of other arrangement, such research services as the Commission deems necessary and requests for the performance of its functions; and
> (3) Consult and cooperate with the Commission on research and development matters of mutual interest and provide such information and physical access to its facilities as will assist the Commission in acquiring the expertise necessary to perform its licensing and related regulatory functions."

The law requires that each Federal agency, subject to the provisions of existing law, shall cooperate with the Commission and provide such information and research services, on a reimbursable basis, as it may have or be reasonable able to acquire. In addition, in 1977, the law was amended to direct the Commission to develop a long-term plan for projects for the development of new or improved safety systems for nuclear power plants.

The Office of Nuclear Regulatory Research (RES) is a program office per Title 10 of the Code of Federal Regulations Part 1:

> §1.45 Office of Nuclear Regulatory Research.
>
> The Office of Nuclear Regulatory Research—
> (a) Plans, recommends, and implements programs of nuclear regulatory research, standards development, and resolution of generic safety issues for nuclear power plants and other facilities regulated by the NRC;
> (b) Coordinates research activities within and outside the agency including appointment of staff to committees and conferences; and
> (c) Coordinates NRC participation in international standards-related activities and national volunteer standards efforts, including appointment of staff to committees.

Research Strategy

The Office of Nuclear Regulatory Research (RES) supports the regulatory mission of the U.S. Nuclear Regulatory Commission (NRC), and executes the following strategies to help achieve NRC's safety and security strategic goals:

- Provides expert technical advice, state-of-the-art tools, and information to make safety and security decisions, and issue regulatory requirements and guidance.
- Conducts research activities to independently confirm the safety of licensees' operations and enhance the regulatory framework by addressing changes in technology, science, and policies.
- Conducts independent confirmatory and anticipatory research to resolve potential safety and security issues and confirm the safety and security bases and margins associated with the use of radioactive materials.
- Conducts long-term research to understand any potential safety issues associated with current and emerging technologies.
- Performs independent analyses of operational data and assessments of operating experience that are used to estimate and monitor the risk of accidents at NRC licensed facilities and inform NRC's strategic plan goals.

- Develops and revises regulatory guides in light of knowledge gained from licensing reviews, inspections, operating experience, and research activities.
- Exchanges information, expertise, operating experiences, and research with domestic and international counterparts to increase awareness of, and respond to, emerging technical issues; to participate in the development, evaluation, and implementation of harmonized standards; to seek common approaches to resolving technical issues; to promote best practices; and to leverage resources through shared research programs.
- Incorporates insights gained from research activities, including interactions with international, academic, and other Federal agencies, into the regulatory infrastructure.
- Maintains critical technical expertise to support regulatory functions such as licensing, oversight, rulemaking, policy development, and research.

NRC's licensees are responsible for the safe and secure operation of their facilities and uses of nuclear materials, including performing the research necessary to demonstrate safety and security. Conversely, NRC's research supports independent regulatory decision making by the Commission and NRC's regulatory offices. RES develops its program based on Commission direction and requests from the regulatory offices. RES provides independent, objective evaluations of potential safety and security issues involving the operating fleet of nuclear power plants, other nuclear facilities, and users of nuclear materials; verification studies of new designs and technologies for current and new reactors, other new nuclear facilities, and new uses of nuclear materials; and other assessments deemed necessary to support the Commission's regulatory functions. Much of the office's work is available to the public through NUREG series reports that describe the research and results. Specifically, NUREG-1925 details the topics of research focus for RES. The office typically issues 30-40 NUREG reports on a wide variety of topics each year.

RES also uses cooperative agreements with international and domestic organizations to leverage resources, to acquire data, and to develop and verify numerical procedures and other analytical tools and methodologies to fully understand and characterize the safety and security of nuclear facilities and nuclear materials uses. The development of tools, data, and standards add to the technical basis needed for safety and security determinations and other regulatory decisions. International and domestic cooperative programs have been developed in many research areas to minimize duplication of effort. This enhances the NRC's ability to make sound regulatory and safety decisions based on worldwide scientific knowledge that promotes the effective and efficient use of agency resources.

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Figures

Abbreviations and Acronyms

Numerals

ΔCDP	change in core damage probability
%OLTP	percent of originally licensed thermal power
%RCF	percent of rated core flow
10 CFR	Title 10 of the *Code of Federal Regulations*

A

ABAQUS/SIMULIA	structural analysis codes
AC	alternating current
ACRS	Advisory Committee on Reactor Safeguards
ADAMS	Agencywide Documents Access and Management System (NRC)
AECL	Atomic Energy of Canada Ltd.
AISC	American Institute of Steel Construction
ALARA	as low as reasonably achievable
AMFL	Advanced Multi-Phase Flow Laboratory
AMP	aging management program
AMPX	Advanced Module for Processing Cross-sections
AMUG	Asian MELCOR User Group
ANL	Argonne National Laboratory
ANS	American Nuclear Society
ANSI	American National Standards Institute
ANSYS	engineering simulation software developer
AO	abnormal occurrence
AP	advanced passive
APET	accident progression event tree
APEX	Advanced Power Extraction
APR	Advanced Power Reactor (Korea)
APWR	U.S. Advanced Pressurized-Water Reactor (Mitsubishi)
ASD	aspirating smoke detectors
ASME	American Society of Mechanical Engineers
ASP	accident sequence precursor
ASR	alkali-silica reaction
AST	alternative source term
ASTM	American Society for Testing and Materials
ATD	atmospheric transport, dispersion, and dose
ATLAS	Advanced Thermal-hydraulic Test Loop for Accident Simulation
ATR	Advanced Testing Reactor (INL)
ATWS	anticipated transient without scram
ATWS-I	ATWS (anticipated transient without scram) with instability

B

BADGER	Boron-10 Areal Density Gauge for Evaluating Racks
BAM	German Federal Institute for Materials Research and Testing
BBN	Bayesian Belief Network
BEIR	Biological Effects of Ionizing Radiation
BFBT	BWR full-size fine-mesh bundle test
BFN	Browns Ferry Nuclear Power Plant
BIP	Behavior of Iodine Project (CSNI)
BNL	Brookhaven National Laboratory
BSAF	Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Station
BTP	branch technical position
BWR	boiling-water reactor
B&W	Babcock & Wilcox

C

C	Celsius
CADAK	Cable Aging Data and Knowledge
CAMP	Code Application and Maintenance Program (NRC)
CANDU	Canada Deuterium Uranium reactor
CAPS	CSNI Action Proposal Sheet
CAROLFIRE	Cable Response to Live Fire
CCDP	conditional core damage probability
CCF	common-cause failure
CDET	core damage event tree
CDF	core damage frequency
CE	Combustion Engineering
CEUS	Central and Eastern United States
CFD	computational fluid dynamics
CFR	Code of Federal Regulations
CIRFT	Cyclic Integrated Reversible-bending Fatigue Tester
CISCC	chloride-induced stress corrosion cracking
CHRISTIFIRE	Cable Heat Release, Ignition, and Spread in Tray Installations during Fire

CLB	current licensing basis
CMMI	Capability Maturity Model Integration
CODAP	Component Operational Experience, Degradation and Aging Program
COL	combined license
ComMIT	Community Model Interface for Tsunami
CONOPS	concepts of operations
CONTAIN	Containment Analysis Code
CoP	community of practice
CPRR	containment overpressure protection and release reduction
CR	control room
CRPPH	International Commission on Radiation Protection
CSARP	Cooperative Severe Accident Research Program (NRC)
C-SGTR	consequential steam generator tube rupture
C-S-H	Calcium Silicate Hydrate
CSNI	Committee on the Safety of Nuclear Installations
CSRA	Computational Support for Risk Applications
CT	computed tomography

D

DAKOTA	Design Analysis Kit for Optimization and Terascale Applications
DandD	Decontamination and Decommissioning code
DART	Deep-ocean Assessment and Reporting of Tsunamis
DBA	design-basis accident
DBE	design-basis earthquake
DC	direct current
DCD	design certification documents
ΔCDP	delta (change in) core damage probability
DCF	dose conversion factor
DCPD	direct current potential drop
DCSS	dry cask storage system
DE	dose equivalent
DE	RES Division of Engineering
DEMAC	demountable mechanical gauges
DES	discrete element simulation DESIREE-FIRE Direct Current Electrical Shorting in Response to Exposure Fire
DG	draft (regulatory) guide
DI&C	digital instrumentation and control
DIGREL	reliability assessment of digital systems
DM	dissimilar metal
DOD	U.S. Department of Defense
DOE	U.S. Department of Energy
DRA	RES Division of Risk Analysis
DSA	RES Division of Systems Analysis
DVD	digital versatile disc

E

EAB	exclusion area boundary
ECI	exterior communications interface
ELAP	extended loss of alternating power
ELECTRA-FIRE	Electrical Cable Test Results and Analysis during Fire Exposure
EMUG	European MELCOR User Group
ENDF	Evaluated Nuclear Data File
EPR	U.S. Evolutionary Power Reactor
EPRI	Electric Power Research Institute
EPU	extended power uprate
ESBWR	Economic Simplified Boiling-water Reactor (General Electric)
ESSI	NRC Earthquake Soil-Structure Interaction Simulator
EST	extended storage and transportation
EWR	excavate and weld repair

F

F	Fahrenheit
FAA	U.S. Federal Aviation Administration
FAQ	frequently-asked-questions
FAVOR	Fracture Analysis of Vessels, Oak Ridge
FDA	U.S. Food and Drug Administration
FDS	Fire Dynamics Simulator
FDT	Fire Dynamics Tools
FEA	finite-element analysis
FIRE	Fire Incident Record Exchange
FFD	fitness for duty
FLASH-CAT	Flame Spread over Horizontal Cable Trays
FLECHT	Full Length Emergency Cooling Heat Transfer
FLUENT	computer code used for CFD and FEA
FP	fission product
FR	Federal Register
FRA	fire risk assessment
FPRA	fire probabilistic risk assessment
FRAPCON	Fuel Rod Analysis Program (CONstant (steady state) version)

FRAPTRAN	Fuel Rod Analysis Program (TRANsient version)
FSME	Office of Federal and State Materials and Environmental Management Programs
FT	fracture toughness
ft	feet
FUMAC	Fuel Modeling in Accident Conditions (IAEA)
FUMEX	Fuel Modeling at Extended Burnup (IAEA)
FY	fiscal year

G

Gd2O3	gadolinia
GALE	Gaseous And Liquid Effluent code
GALL	Generic Aging Lessons Learned
GDC	general design criterion
GI	generic issue
GIRP	Generic Issues Review Panel
GMPE	ground motion prediction equations
GMRS	ground motion response spectrum
GUI	graphical user interface
GWd/MTU	gigawatt-days per metric ton of uranium
Gy	gray (unit of radiation dose)

H

HABIT	HABITability code
HAMMLAB	Halden Man-Machine Laboratory
HBU	high burnup
HDPE	high-density polyethylene
HEAF	high energy arcing faults
HELEN-FIRE	Heat Release Rates from Electrical Enclosure Fires
HEP	human error probability
HFE	human factors engineering
HFE	human failure even
hr	hour
HRA	human reliability analysis
HRP	Halden Reactor Project
HRR	heat release rate
HSI	human-system interface
HYMERS	Hydrogen Mitigation Experiments for Reactor Safety
HYSPLIT	HYbrid Single-Particle Lagrangian Integrated Trajectory dispersion and deposition model
Hz	hertz (measure of frequency)

I

IA	International Agreement
I&C	instrumentation and control
IAD	irradiation-assisted degradation
IAE	Institute of Applied Energy
IAEA	International Atomic Energy Agency
IASCC	irradiation-assisted stress-corrosion cracking
ICAP	International Code Assessment and Application Program
ICDE	International Common-cause Data Exchange
ICES	Institute for Nuclear Power Operations Consolidated Events System
ICIC	International Committee on Irradiated Concrete
ICRP	International Commission on Radiological Protection
IDCCS	Integrated Data Collection and Coding System
IDHEAS	Integrated Human Event Analysis System
IEEE	Institute of Electrical and Electronics Engineers
IFBA	Integral Fuel Burnable Absorber
IFE	Institutt for Energiteknikk (Norwegian Institute for Energy Technology)
IFRAM	International Forum for Reactor Aging Management
IGSCC	intergranular stress corrosion cracking
In	inch
INL	Idaho National Laboratory
IPEEE	individual plant examination of external events
IRSN	Institut de Radioprotection et de Surete Nucleaire (French Institute for Radiological Protection and Nuclear Safety)
iSALE	impact Simplified Arbitrary Lagrangian Eulerian
ISB	in-situ bioremediation
ISFSI	independent spent fuel storage installation
ISG	interim staff guidance
ISI	inservice inspection
ISO	International Standards Organization
ISOE	Information System on Occupational Exposure

ISSC-EBP	International Seismic Safety Centre's Extra Budgetary Project (IAEA)
IST	integrated system test
ITP	Industry Trends Program

J

JAEA	Japanese Atomic Energy Agency
JACQUEFIRE	Joint Assessment of Cable Damage and Quantification of Effects from Fire
JCCRER	Joint Coordinating Committee for Radiation Effects Research

K

KATE-FIRE	Kerite Analysis in Thermal Environment of Fire
keff	k-effective reactivity coefficient
KHNP	Korea Hydro and Nuclear Power Co. Ltd.
KM	knowledge management

L

LBB	leak before break
LBLOCA	large-break loss-of-coolant accident
LBNL	Lawrence Berkeley National Laboratory
LCF	latent cancer fatality
LD	leak detection
LER	licensee event report
LERF	large early release frequency
LM	legacy management
LOC	loss of control
LOCA	loss-of-coolant accident
LOCS	Loop Operating Control System
LOH	loss of habitability
LPZ	low population zone
LRGD	license renewal guidance documents
LSDYNA	Livermore Software Technology Corporation for dynamic explicit finite-element analysis
LTO	long term operation (EPRI program)
LTSBO	long-term station blackout
LTRP	Long-Term Research Program
LULCC	land use and land cover change
LWR	light-water reactor
LWRS	Light-Water Reactor Sustainability Research (DOE)

M

MACCS	MELCOR Accident Consequence Code System
MAG	modeling analysis guidelines
MAGIC	fire modeling tool
MARIA-FIRES	Methods for Applying Risk Analysis to Fire Scenarios
MASS	MELCOR Accident Simulation Using SNAP
MATLAB	MATrix LABoratory
MCAP	MELCOR Code Assessment Program
MCCI	Melt Coolability and Concrete Interaction
MCR	main control room
MD	management directive
MELCOR	computer code for analyzing severe accidents in NPPs
MELLLA+	maximum extended load line limit analysis plus
MOST	Method of Splitting Tsunami
MOU	memorandum of understanding
MOX	mixed oxide
MPa	megapascal
MTO	Man-Technology-Organization

N

NAM	neutron-absorbing materials
NAS	U.S. National Academy of Sciences
NASA	National Aeronautics and Space Administration
NCRP	National Council of Radiation Protection and Measurements
NDAA	National Defense Authorization Act of 2005
NDE	nondestructive examination
NDT	nondestructive testing
NEA	Nuclear Energy Agency
NEI	Nuclear Energy Institute
NESCC	Nuclear Energy Standards Coordination Collaborative
NFPA	National Fire Protection Association
NGA	next generation attenuation
NIST	U.S. National Institute of Standards and Technology
NMSS	Office of Nuclear Material Safety and Safeguards
NOAA	National Oceanic and Atmospheric Administration (U.S. Department of Commerce)
NPP	nuclear power plant
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation

NSIR	Office of Nuclear Security and Incident Response
NTTAA	National Technology Transfer and Advancement Act of 1995
NTTF	Near-Term Task Force
NUREG	NRC technical report
NUREG/BR	NRC technical report/brochure
NUREG/CP	NRC technical report/conference proceeding
NUREG/CR	NRC technical report /contractor report
NUREG/IA	NRC technical report /international agreement
NUREG/KM	NRC technical report/knowledge management

O

OECD	Organisation for Economic Co-operation and Development
OLTP	originally licensed thermal power
1-D	one-dimensional
OpE	operating experience
ORNL	Oak Ridge National Laboratory
OSU	Ohio State University

P

PANDA	Passive Non-Destructive Assay of Nuclear Materials
PARCS	Purdue's Advanced Reactor Core Simulator
PAUT	phased array ultrasonic testing
Pb	Lead
PEO	period of extended operation
%OLTP	percent of originally licensed thermal power
%RCF	percent of rated core flow
PFHA	probabilistic flood hazard assessment
Phebus-FP	Phebus-Fission Products
Phebus-ISTP	Phebus-International Source Term Program
PIF	performance influencing factors
PIMAL	phantom with moving arms and legs
PIRT	Phenomena Identification and Ranking Table
PKL	Primarkreislauf-Versuchsanlage (German for primary coolant loop test facility)
PNNL	Pacific Northwest National Laboratory
PRA	probabilistic risk assessment or probabilistic risk analysis
PSA	probabilistic safety assessment
PSHA	probabilistic seismic hazard assessment
PSI	Paul Scherrer Institut
psi	pounds per square inch
PTS	pressurized thermal shock
PUMA	Purdue University Multi-Dimensional Integral Test Assembly
PWR	pressurized-water reactor
PWROG	Pressurized-Water Reactor Owners' Group
PWSCC	primary water stress-corrosion cracking

Q

QA	quality assurance
QC/QV	quality control/quality verification
QHO	quantitative health objective
QUENCH	German fuel experimental program

R

RACHELLE-FIRE	Refining and Characterizing Heat Release Rates from Electrical Enclosures during Fire
RACKLIFE	software calculation package used for mapping of degradation
RADS	Reliability and Availability Data System
RADTRAD	RADionuclide Transport, Removal, And Dose estimation code
RAMP	Radiation Protection Computer Code Analysis and Maintenance Program
RASCAL	Radiological Assessment System for Consequence AnaLysis
RASP	Risk Assessment Standardization Project
RBHT	Rod Bundle Heat Transfer program
RCF	rated core flow
RCS	reactor coolant system
REAcct	Regional Economic Accounting Tool
REAP	Reactor Embrittlement Archive Project
REIRS	Radiation Exposure Information and Reporting System
RELAP	Reactor Excursion and Leak Analysis Program
REMIX	Regional Mixing Model
RES	Office of Nuclear Regulatory Research
RG	regulatory guide
RIA	reactivity initiated accident
RIL	Research Information Letter
RILEM	Committee on ASR of the International Union of Laboratories and Experts in Construction Materials Systems and Experts

RMIEP	Risk Methods Integration and Evaluation Program
Rn	radon
ROP	Reactor Oversight Process
ROSA	Rig of Safety Assessment test facility (Japan)
RPV	reactor pressure vessel
RRT	round robin testing
RSBDBE	repeated and sudden below-design-basis earthquake

S

SACADA	Scenario Authoring, Categorization, and Debriefing Application
SAMG	severe accident mitigation guideline
SAPHIRE	Systems Analysis Programs for Hands-on Integrated Reliability Evaluation
SAREF	Safety Research post Fukushima
SC	steel plate and concrete composite modular construction
SC	Office of Science (DOE)
SCALE	Standardized Computer Analysis for Licensing Evaluation code
SCC	stress-corrosion cracking
SCG	slow crack growth
SCIP	Studsvik Cladding Integrity Project
SDP	Significance Determination Process
SEASET	Separate Effects and Systems Effects Tests
SECY	Office of the Secretary (NRC staff paper for the Commission)
SFP	spent fuel pool
SG	steam generator
SHAC-F	Structured Hazard Assessment Committee Process for Flooding
SLR	subsequent license renewal
SLRA	subsequent license renewal applications
SLRGD	subsequent license renewal guidance documents
SMR	small modular reactor
SNAP	Symbolic Nuclear Analysis Package
SNF	spent nuclear fuel
SNFT	spent nuclear fuel transportation
SNL	Sandia National Laboratories
SOARCA	State-of-the-art Reactor Consequence Analysis
SPAR	Standardized Plant Analysis Risk
SQA	software quality assurance
SRA	senior risk analyst
SRM	staff requirements memorandum (Commission direction to NRC staff)
SRP	Standard Review Plan
SSC	structures, systems, and components
SSHAC	Senior Seismic Hazard Analysis Committee
STAR CCM+	computer code used for CFD
STCP	Source Term Code Package
STDose	Source Term to Dose
STEM	Source Term Evaluation and Mitigation (CSNI experimental program)
STSET	Source Term Separate Effects Test Project
Sv	Sievert
SWS	service water system

T

TDI	Tabbed Document Interface
TBq	terabecquerel (1012 Bq)
TECDOC	IAEA technical document
TEPCO	Tokyo Electric Power Company
TEDE	total effective dose equivalent
TGSCC	transgranular stress corrosion cracking
T/H	thermal-hydraulic
THI	Thermal-Hydraulics Institute
THIEF	Thermally-Induced Electrical Failure TID Technical Information Document
3-D	three-dimensional
TIN	Technical Information Needs report
TMI	Three Mile Island (Nuclear Power Plant)
TRAC	Transient Reactor Analysis Code
TRACE	TRAC/RELAP Advanced Computational Engine
TRITON	Transport Rigor Implemented with Time-dependent Operation for Neutronic depletion code module
2-D	two-dimensional

U

UA	uncertainty analysis
UCF	University of Central Florida
UCLA	University of California at Los Angeles
UMD	University of Maryland
UMTRCA	Uranium Mill Tailings Radiation Control Act
UNSCEAR	United Nations Scientifics Committee on Exposure to Atomic Radiation
UO2	uranium dioxide
U.S.	United States

U.S. APWR	U.S. Advanced Pressurized-Water Reactor (Mitsubishi)
USGS	U.S. Geological Survey
UT	ultrasonic testing

V

V&V	verification and validation
VARSKIN	code used to model and calculate skin dose
VERCORS	French test program
VERDON	French test program
VEWFD	very early warning fire detection
VT	visual testing
VTT	Technical Research Center of Finland

W

WGAMA	Work Group on Analysis and Management of Accidents
WGHOF	Working Group on Human and Organisational Factors
WGRISK	Working Group on Risk
WIAGE	Working Group for Integrity and Aging of Structures and Components
WRS	weld residual stresses

X

xLPR	extremely low probability of rupture

Z

ZrB2	zirconium diboride
ZIRLO	fuel rod cladding material
ZOI	zone of influence

Chapter 1: Agency Programs Support

The objective of the Office of Nuclear Regulatory Research (RES) is to support the regulatory mission of the U.S. Nuclear Regulatory Commission (NRC) by providing technical advice, tools, and information for identifying and resolving potential safety issues; performing the research necessary to support regulatory decisions; and issuing regulatory requirements and guidance. RES's principal product is knowledge; therefore, knowledge management (KM) is an integral part of the RES mission.

RES applies its breadth and depth of technical expertise to coordinating agency wide programs in support of the NRC's regulatory infrastructure. For example, RES coordinates the appointment of RES staff to key committees of various domestic and international standards development organizations to offer technical expertise. RES also coordinates the development and maintenance of about 400 publicly available regulatory guides to present approaches that the NRC staff considers acceptable for nuclear industry use in complying with the agency's regulations.

In addition, RES coordinates the NRC's use of consensus codes and standards. The NRC cooperates with professional organizations that develop consensus standards associated with systems, structures, equipment, or materials that the nuclear industry uses. A standard contains technical requirements, safety requirements, guidelines, characteristics, and recommended practices for performance. Codes are defined as standards or groups of standards that have been incorporated by reference into the regulations of one or more governmental bodies and have the force of law. NRC will endorse consensus codes and standards for use in meeting regulatory requirements when the staff finds them adequate and consistent with NRC regulations.

RES recommends regulatory actions to resolve issues for nuclear power plants and other facilities regulated by the NRC including those issues designated as generic issues (GIs) based on research results and experience. The Generic Issues Program enables the public and NRC staff to raise issues with potential significant generic safety or security implications. The program ensures that those potential safety and security issues are appropriately reviewed and dispositioned through an effective, collaborative, and open process and that any needed actions are taken to ensure continued safety at NRC licensed facilities. The program has identified more than 850 GIs to date, resulting in important safety improvements at NRC licensed facilities and in a variety of regulatory guidance.

RES provides independent analysis of operational data and assessment of operational experience. Data and information reported to the NRC by licensees and collected by RES from the Institute of Nuclear Power Operations (INPO) undergird NRC's Operating Experience Program (OpE) and is used by both RES and NRR to implement a shared responsibility to assess operational safety. NRR focuses its efforts on engineering evaluations of near-term, emerging issues with potential safety implications and trends operational data through its Industry Trends Program. RES uses the industry data to estimate and monitor the probability of potential accidents occurring at nuclear power plants.

The operating experience data is reviewed, evaluated, and coded into databases that form the basis to estimate reliability parameters to maintain and update the Standardized Plant Analysis Risk (SPAR) models that are critical to the staff's risk analysis capability. The SPAR models provide both NRR and RES the ability to independently perform PRA analyses in support of NRR's Reactor Oversight Program (ROP) and RES' Accident Sequence Precursor (ASP) Program. The ASP Program systematically evaluates U.S. nuclear power plant operating experience to identify, document, and rank events and the likelihood they could lead to core damage (precursors) given the estimated probabilities of additional failures. The ASP Program is one of three NRC programs that assess the risk significance of operational events (the other two are the ROP Significance Determination Process and the Incident Investigation Program.)

RES coordinates the abnormal occurrence (AO) process for the agency and authors the AO Report to Congress. Section 208 of the Energy Reorganization Act of 1974 defines an AO as an unscheduled incident or event that the NRC determines to be significant from the standpoint of public health or safety. The NRC reports AOs to Congress annually. The AO process helps to monitor the efficacy of the NRC's regulatory process and to identify any corrective actions that are needed. An accident or event is considered an AO if it involves a predicted reduction in the degree of protection of public health or safety.

As a matter of routine planning, the NRC identifies forward-looking and long-term research activities that support potential future regulatory needs. Both forward-looking and long-term research could support possible new program areas and the development of technical bases for a range of anticipated regulatory decisions. In addition, this research could address emerging technologies that may have future regulatory applications or could be used to develop plans to implement needed research. Long-Term Research Program (LTRP) projects are feasibility or scoping studies that assess if future research on the topic should be pursued, and if so, who should do it and when should it be done.

Regulatory Guide Program

Objective

The NRC issues regulatory guides for licensees, applicants, and the public to use that present approaches the staff considers acceptable in implementing the agency's regulations. The Office of Nuclear Regulatory Research (RES) provides the program management for issuing and updating regulatory guides. Regulatory guides are issued in the following 10 broad divisions:

1. Power Reactors
2. Research and Test Reactors
3. Fuels and Materials Facilities
4. Environmental and Siting
5. Materials and Plant Protection
6. Products
7. Transportation
8. Occupational Health
9. Antitrust and Financial Review
10. General

Research Approach

Program Management

RES is primarily responsible for program management of the regulatory guides including prioritization of updates. RES coordinates with the other program offices to issue revised and new regulatory guides. NRC Management Directive 6.6, "Regulatory Guides," formalizes the regulatory guide development and revision process.

Schedule

To ensure the guides continue to be updated with reasonable frequency, a periodic review cycle has been implemented for each guide. Currently regulatory guides are reviewed every 5 years. Using the results of the review, the program office decides whether a regulatory guide is acceptable for continued use or whether it should be revised or withdrawn. A guide that has been withdrawn can continue to be used if it is part of a facility's licensing basis or for reference.

An online database tracking system is used to track the status of each guide, and the appropriate NRC program office is notified in advance of an upcoming periodic review. A review includes a nominal 65-week schedule that is broken into 21 scheduling activities. This schedule includes a 15-week drafting period for the guide, a review period for internal stakeholders and the NRC's Advisory Committee on Reactor Safeguards, a comment period for the public and other external stakeholders, and a final review period for internal stakeholders including the Office of General Counsel.

Related Activities

RES leads several related activities undertaken in support of agency goals to produce and maintain guidance documents that support transparent, efficient, and effective regulation. These include:

- Assisting NRC regulatory offices in updating related regulatory guidance. Examples of related regulatory guidance include standard review plans, inspection procedures, and technical basis documents.
- Assisting NRC regulatory offices in capturing knowledge from regulatory activities in regulatory guidance documents as a vital part of agency knowledge management efforts.
- Coordinating the agency's use of consensus codes and standards in its regulatory guidance.

Status

As of September 2015, the agency has completed 384 of the 426 regulatory guides in an initial 2006 update program, 37 are in the process of being updated, and an additional 5 have been delayed due to higher priority work. Figure 1.1 depicts the current status of the Regulatory Guide Update Project. The agency continues to update the active guides on an ongoing basis. For example, many of the guides that are shown in Fig. 1.1 as completed have subsequently been reviewed on the current 5-year review schedule and updated as appropriate.

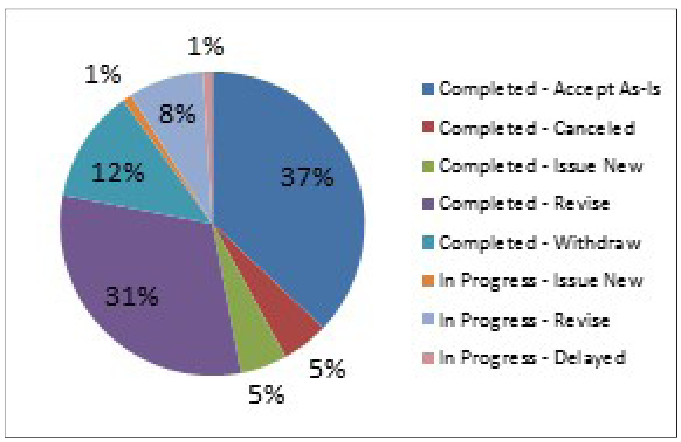

Figure 1.1 NRC Regulatory Guide status.

For More Information
Contact Carol Moyer, RES/DE, at Carol.Moyer@nrc.gov.

Consensus Codes and Standards

Objective

The NRC cooperates with professional organizations that develop consensus standards associated with systems, structures, equipment, or materials that the nuclear industry uses. A standard contains technical requirements, safety requirements, guidelines, characteristics, and recommended practices for performance. The voluntary consensus standards process is based on openness, balance of interests, due process with written records, and consensus—more than a majority but not necessarily unanimity. Codes are defined by the American National Standards Institute as standards or groups of standards that have been incorporated by reference into the regulations of one or more governmental bodies and have the force of law. NRC staff participates in consensus standards writing committees with representatives from industry, academia, and other government agencies. Consensus standards are endorsed by NRC for use in NRC regulations and regulatory guidance, however, only when the NRC staff finds them acceptable for use by licensees and applicants.

For example, the American Society of Mechanical Engineers (ASME) developed the Boiler and Pressure Vessel Code, which is widely acknowledged as an acceptable set of standards used to design, construct, and inspect pressure-retaining components including nuclear vessels, piping, pumps, and valves. Similarly, the National Fire Protection Association (NFPA) has developed a series of consensus standards to define acceptable methods to design, install, inspect, and maintain fire protection systems. The NRC has incorporated into its regulations parts of the ASME Boiler and Pressure Vessel Code and a key NFPA standard, with some limitations, as well as other consensus standards.

The objective of this program is to optimize the NRC's development and use of consensus codes and standards as part of its regulatory framework and in voluntary compliance with Public Law 104-113, the "National Technology Transfer and Advancement Act of 1995" (NTTAA). In addition to issuing regulations that incorporate consensus standards, the NRC staff issues guidance on acceptable methods for complying with its regulations such as regulatory guides. These guidance documents frequently reference consensus standards as acceptable methods for compliance with NRC regulations. A principal reason for using standards is to provide regulatory stability and predictability.

Research Approach

The Office of Nuclear Regulatory Research (RES) coordinates the NRC's use of consensus codes and standards. For example, RES staff provides support for this effort by coordinating standards committee representation, compiling information needed for attendance at standards meetings, collecting and resolving stakeholder comments on draft standards, disseminating documents to other NRC offices for input, and promoting awareness of safety standards. RES implements Management Directive (MD) 6.5, "NRC Participation in the Development and Use of Consensus Standards".

The NRC's use of consensus standards is consistent with the requirements of the NTTAA, as further described in the Office of Management and Budget's Circular A-119, "Federal Agency Participation in the Development and Use of Voluntary Consensus Standards and in Conformity Assessment Activities." Participation of NRC staff in consensus standards development is essential because the codes and standards are an integral part of the agency's regulatory framework. The benefits of this active involvement include cost savings, improved efficiency and transparency, and regulatory requirements of high technical quality. The agency acknowledges the broad range of technical expertise and experience of the individuals who belong to the many consensus standards organizations. Thus, participation in standards development minimizes the time and expenditure of NRC resources that would otherwise be necessary to provide guidance with the technical depth and level of detail of consensus standards.

Nuclear Energy Standards Coordination Collaborative

In 2009, in cooperation with other Federal agencies, the NRC helped establish a new information exchange forum called the Nuclear Energy Standards Coordination Collaborative (NESCC). The NESCC is a cross-stakeholder forum to identify and respond to the needs of the U.S. nuclear industry for updates to codes and standards. The NESCC is a joint effort of the NRC, the U.S. Department of Energy, the American National Standards Institute, standards- developing organizations, and the nuclear industry. Its goals are to identify standards needs, prioritize standards for development or revision, and initiate or support collaboration in writing or updating standards. The group works on a voluntary basis to facilitate and coordinate the timely identification, development, and revision of standards for the design, operation, development, licensing, and deployment of nuclear power plants and other nuclear technologies. Central to the mission of the NESCC is developing a standards database that will provide government agencies, standards developers, nuclear industry users, and the public with clear information about available consensus standards and how the industry can use those standards to meet regulatory requirements.

International Safety Standards

The International Atomic Energy Agency (IAEA) Commission on Safety Standards is a body of senior government officials from member nations that oversees the development of international safety standards. IAEA has four Safety Standards Committees that participate in the development, review, and update of standards and guidance documents related to nuclear safety, radiation protection, waste management safety, and transport safety. A RES manager serves as the U.S. delegate to one of these four committees, the Nuclear Safety Standards Committee. This participation helps harmonize, to the extent practical, NRC standards and guidance with international standards and guidance.

Status

In 2014, about 190 NRC staff members participated in more than 300 standards activities, such as membership on a standards-writing committee. The organizations governing these committees include ASME, NFPA, the Institute of Electrical and Electronics Engineers, the American Concrete Institute, and many others.

Most codes and standards evolve over time through a process that includes development of new standards and revision of existing ones. For example, work is underway with standards developing organizations to update voluntary consensus standards that may be applied to license renewal or new nuclear plant construction including advanced reactor technologies and small modular reactors.

In addition to safety standards and guides issued by the IAEA, the NRC staff is evaluating other international standards, such as documents published by the International Standards Organization and the International Electrotechnical Commission. Where applicable, these documents are referenced for information or guidance. The NRC staff is considering ways to increase its use of international standards within the agency's regulatory framework.

For more information
Contact Carol E. Moyer, RES/DE, at Carol.Moyer@nrc.gov.

Generic Issues Program

Objective

The NRC Generic Issues (GI) Program enables the public and NRC staff to raise issues with potential generic safety or security implications. The purpose of the GI Program is to perform a rigorous evaluation of proposed issues to determine whether additional regulatory requirements are necessary to ensure continued safe operation of the licensed facilities and to disseminate pertinent information addressing generic issues.

Program management includes finding the most appropriate NRC office to work on proposed issues, determining the risk significance of issues, developing consensus on the need for and form of regulatory actions, and managing the communications challenges associated with generic issues.

The GI Program has contributed significantly to the NRC's mission to protect public health and safety. Since 1976, more than 850 generic issues have been resolved and over 40 percent of generic issues that passed the screening stage have resulted in a new regulatory product. These products include rulemaking, Regulatory Guides, NUREG documents, generic communications, Standard Review Plans, staff reports, Commission papers, new policies, and updates to existing regulations.

Research Approach

The Office of Nuclear Regulatory Research authored and coordinates the implementation of Management Directive (MD) 6.4, "Generic Issues Program," that describes the process used to resolve the generic issues. The MD provides the staff with a framework for handling, tracking, and defining the documentation associated with processing generic issues. The GI Program consists of a three-stage process: screening, assessment, and regulatory implementation (refer to Figure 1.2).

In the screening stage, the NRC staff uses seven screening criteria to identify generic issues that the program can effectively evaluate to determine if additional regulatory requirements are necessary. In the assessment stage, the staff explores the technical bases for the issue so that the agency can assess any regulatory actions that may be needed to address the issue. In the regulatory implementation stage, the agency takes regulatory action to address the issue with its licensees.

Status

Recent GI Program changes include (1) program simplification by reducing the number of stages from five to three, (2) increased management involvement and accountability, and (3) new requirements for improved documentation during screening so that it is clear that issues being processed by the program do not involve immediate safety concerns. Generic issues continue to be proposed by NRC staff and the public. Generic issues that satisfy the screening criteria are further developed and evaluated using a regulatory analysis to determine whether a regulatory product is necessary.

Information on the resolution of generic issues is available in NUREG-0933, "Resolution of Generic Safety Issues." NUREG-0933 is now a user friendly, web-based, accessible, and searchable document. Features include full text searches of the full NUREG, filtering by technical area, filtering by facility type, and other enhancements that make NUREG-0933 a better source of information on historical issues. NUREG-0933 is available online at: http://nureg.nrc.gov/sr0933/.

More information on active generic issues is available online at: http://www.nrc.gov/reading-rm/doc-collections/generic-issues/.

For More Information
Contact Stanley Gardocki, RES/DE at Stanley.Gardocki@nrc.gov.

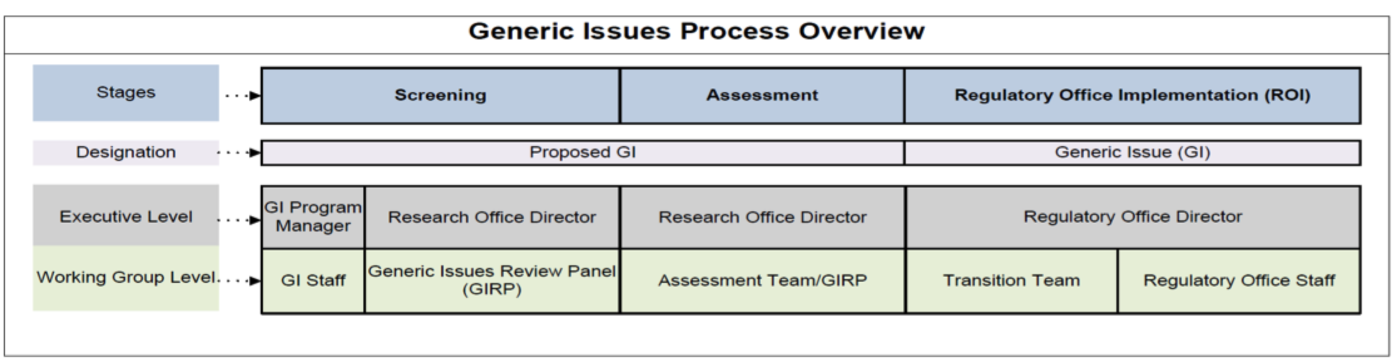

Figure 1.2 Generic Issues Program process overview.

Long-Term Research Program

Objective

The NRC's Office of Nuclear Regulatory Research (RES) conducts long-term research projects as part of the overarching research program which supports regulatory needs. The purpose of the Long-Term Research Program (LTRP) is to help anticipate the agency's future regulatory needs (i.e., within the next 5 to 10 years), to identify any research activities needed to address such needs, to determine if the needed research should be done by industry and/or NRC, and to determine when any needed research should be started. Projects selected for the LTRP are those that are not already the subject of ongoing research activities by the agency. In addition, the scope of the projects in the LTRP is limited to feasibility or scoping studies that typically do not exceed 1 to 2 years. The studies are intended to support possible new program areas, support development of technical bases for anticipated regulatory decisions, address emerging technologies that could have future regulatory applications, or assist in developing plans to implement needed research. These short-term studies may be followed up with future research if required.

Research Approach

Each year, the NRC solicits proposed LTRP projects from the staff. The proposed projects are evaluated by a review committee composed of a subset of the agency's Senior Level Service Technical Advisors. The projects are selected in a timeframe that supports budget formulation and are based on five criteria used in the rating process as follows: (1) leveraging resources, (2) advancing the state of the art, (3) providing an independent tool to the NRC, (4) applying to more than one program area, and (5) addressing gaps created by technology advancements.

The LTRP began funding projects in fiscal year (FY) 2009. Since then, numerous projects have been funded and completed. Examples of these include the following:

Sensors and Monitoring to Assess Grout and Vault Behavior for Performance Assessments. The purpose of this project was to better predict the properties of large grout-based waste isolation structures so that changes in behavior over long periods of time can be anticipated and the long-term performance can be evaluated to obtain better acceptance criteria. The regulatory need is that the NRC has both consultation and monitoring roles for certain U.S. Department of Energy (DOE) waste resulting from reprocessing of spent nuclear fuel as required by the Ronald W. Reagan National Defense Authorization Act (NDAA) of 2005. These "Wastes Incidental to Reprocessing" are contained in cementitious monoliths that are in turn contained within waste vaults and subsurface tanks. Grouted high-level waste tanks and saltstone monoliths rely on performance assessment modeling. However, the critical performance characteristics can only be estimated, and their behavior over long time periods is uncertain. Quantifying the properties of these materials is important because the release of radionuclides is estimated and doses are calculated from them.

This project was started in September 2011 and completed in November 2012. A final report, NUREG/CR-7169, "Sensors and Monitoring to Assess Grout and Vault Behavior for Performance Assessments," was published in June 2014. The report contains an assessment of existing nondestructive testing methods and sensor technologies discussed in the context of collecting information relevant to service life prediction tools. As a result of the information outlined in NUREG/CR-7169, the NRC is contracting with the Center for Nuclear Waste Regulatory Analyses to monitor tests specimens for acoustic emissions (Figure 1.3), which is intended to be representative of DOE wastes.

Figure 1.3 In-site surface air concrete permeability test apparatus.

Safety and Regulatory Issues of the Thorium Cycle. Although almost all of the world's nuclear reactors use uranium as their fuel, the nuclear energy community has increased its level of interest in the use of thorium fuel, both in current light-water reactor core designs and in next-generation reactors. The project looked for potential reactor safety and licensing issues with the use of thorium, especially in design-basis accidents, such as loss-of-coolant accident. This project was done in preparation for a possible license application submittal for the use of thorium fuel.

This project was started in October 2012 and completed in December 2013. NUREG/CR-7176, "Safety and Regulatory Issues of the Thorium Fuel Cycle," was published in February 2014. The report presented important properties of thorium fuel and documented qualitative and quantitative evaluations of both in-reactor and out-of-reactor potential safety issues and requirements specific to a thorium-based fuel cycle for current light-water reactor designs. The report prioritized research areas and identified key knowledge gaps and technical issues that are needed to be addressed in the event a new thorium fuel license application is submitted to NRC. Based on the report, the staff is aware that additional analysis or research will be required to resolve potential impacts on safety requirements and identification of knowledge gaps prior to granting licenses for thorium fuel usage.

Status

Current projects include the following:

- Smart Grid Impacts on Nuclear Power Plants (FY 2012).
- Evaluating Remaining Service Life of Nuclear Power Plant Concrete Structures (FY 2013).
- Quantitative Methods for Assessing Cyber Security Posture (FY 2014).
- Seismic Load Effects on Reactor Materials Degradation (FY 2014).
- Reducing Uncertainty in Dam Risk Analysis (FY 2014).
- Advanced Knowledge Engineering Tools to Support Risk-Informed Decisionmaking (FY 2014).
- Potential Applications for, and Assessment of, Adaptive Automation in Nuclear Plant Processes (FY 2015).
- Strategic Approach for Obtaining Material and Component Aging Information from Decommissioning Nuclear Power Plants (FY 2015).

For More Information
See "The Office of Nuclear Regulatory Research Long-Term Research Program," December 2012, NUREG/BR-0506 at http://www.nrc.gov/reading-rm/doc-collections/nuregs/brochures/br0506/.

Contact Stanley Gardocki, RES/DE at Stanley.Gardocki@nrc.gov.

Report to Congress on Abnormal Occurrences

Objective

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 (Pub. L. No. 104-66) requires the NRC to report AOs to Congress annually.

The NRC initially issued the AO criteria in a policy statement published in the Federal Register on February 24, 1977 (42 FR 10950); several revisions followed in subsequent years. The NRC published its most recent revision to the AO criteria in the Federal Register on October 12, 2006 (71 FR 60198); it took effect on October 1, 2007.

The AO process helps to identify deficiencies in the NRC's regulatory process and ensure that corrective actions are taken to prevent recurrence. An accident or event is 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:

- Moderate exposure to, or release of, radioactive material that the Commission licenses or otherwise regulates.
- Major degradation of essential safety-related equipment.
- Major deficiencies in design, construction, use of, or management controls for facilities or radioactive material that the Commission licenses or otherwise regulates.

Approach

When an incident or event occurs, the NRC uses a generic event assessment process to assess it. This generic event assessment process includes the following actions:

- Internal coordination with NRC offices.
- Systematic review of the cause of the event.
- Follow-up with the reporting licensee.
- Outreach to external stakeholders, as appropriate.
- Communication of lessons learned.

The following are two examples of AO that were reported to Congress in recent years.

Tufts Medical Center

Tufts Medical Center in Boston, Massachusetts, reported a medical event that occurred during a gamma stereotactic radiosurgery unit ("gamma knife, see Figure 1.4) treatment for intense facial pain. The procedure prescribed 75 Gy to the left side of the brain to be delivered from the gamma knife's cobalt-60 source. However, the radiation oncologist erroneously selected the right side of the brain in the treatment planning system, which resulted in the wrong side to be treated.

Figure 1.4 Gamma stereotactic radiosurgery unit (gamma knife) (Source, U.S. NRC TTC-TN).

Caribbean Inspection & NDT Services

Caribbean Inspection & NDT Services reported that a radiographer trainee received an overexposure to his right hand while he was removing a radiography camera guide tube. The trainee stated he noticed the 2.7 TBq iridium-192 source was not fully retracted and protruding from the camera and believed he may have brushed the source with his hand when he removed the tube. About a week later, the trainee had blistering of his fingers, an effect expected with exposure between the range of 20 to 40 Sv.

Status

RES is currently leading an agency effort to revise the current AO criteria to ensure current regulatory framework and technology is included in the determination of events which meet the definition of an AO described in Section 208 of the Energy Reorganization Act of 1974.

For More Information
Contact Luis Benevides, RES/DSA, at Luis.Benevides@nrc.gov.

Operating Experience Program

Objective

The objective of the NRC's Operating Experience (OpE) Program is to collect and analyze nuclear power plant (NPP) operational data to help estimate and monitor the risk of accidents at commercial U.S. NPPs.

Research Approach

The NRC has developed and maintains probabilistic risk assessment (PRA) models for all operating commercial U.S. NPPs. See the discussion in this report on the "SPAR Model Development Program" for additional information on these models. To keep the PRA models current, RES collects and analyzes operating experience data from all U.S. NPPs that are then used to generate up-to-date reliability parameters and event frequencies used in the PRA models. These PRA models support NRC performing state-of-practice risk assessments of operating events and conditions, assessing licensee risk-related performance, conducting special studies of risk-related issues, and determining trends, developing performance indicators based on operating data, and performing reliability studies for risk-significant systems and equipment.

The Reactor Operating Experience Data for Risk Applications Project collects data on the operation of NPPs reported in licensees' monthly operating reports, the Institute for Nuclear Power Operations' Consolidated Events System (ICES), and licensee event reports (LERs). LERs for all plants can be searched using the LERSearch program available on the NRC's public Web site https://nrcoe.inel.gov/secure/lersearch/index.cfm. Operational data collected includes component and system failures, demands on safety systems, initiating events, fire events, common-cause failures, and system and train unavailabilities. The data is stored in the Integrated Data Collection and Coding System (IDCCS) database. The IDCCS database has subsidiary and specialized applications such as the Reliability and Availability Data System, Common-Cause Failure Database, and Accident Sequence Precursor (ASP) Events Database. Information in the RADS database, for example, is used to generate the reliability inputs for the PRA models and to help assess and confirm information reported by licensees as part of the Mitigating Systems Performance Index Program.

The Computational Support for Risk Applications (CSRA) Project uses the operational data collected to periodically update the PRA parameters that constitute generic inputs into NRC PRA models, including for example, component failure probability estimates and initiating event frequencies. This project also supports regulatory programs that help identify potential emerging safety issues, such as the Industry Trends Program (ITP) that monitors operating plants for adverse trends. Examples of trends that are regularly updated and evaluated include automatic scrams while critical, safety systems actuations and failures, forced outages, collective radiation exposure, and reactor coolant system leakage and other activity. CSRA also supports the ASP Program and the Significance Determination Process of the Reactor Oversight Program that use PRAs in the assessment of the risk associated with screened LERs and inspection findings, respectively. The results from these analyses are used as input to the allocation and characterization of inspection resources, the initiation of team inspections, and the need for further analysis by other agency organizations.

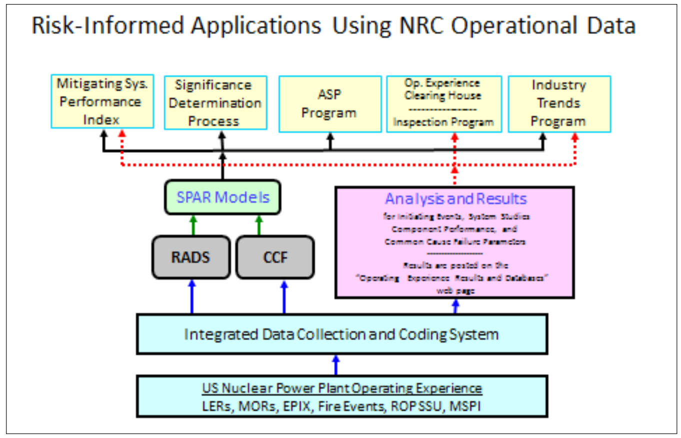

Figure 1.5 Risk-Informed Applications Using NRC Operational Data.

Finally, RES OpE data efforts support the Baseline Risk Index for Initiating Events, a series of trending measures used to provide a risk-informed performance indicator for key initiating events including general transients, losses of condenser heat sinks, losses of main feedwater, losses of offsite power, and steam generator tube ruptures. This type of information helps the Office of Nuclear Reactor Regulation affirm that operating reactor safety is being maintained and enhances the NRC's inspections of risk-significant safety systems.

Status

The collection and analysis of nuclear plant operational data is an ongoing Office of Nuclear Regulatory Research activity whose performance is directed by NRC Management Directive 8.7, "Reactor Operating Experience Program." The operational data analyses performed by CSRA are updated annually on the Reactor Operational Experience Results and Databases Web site (http://nrcoe.inel.gov/results/) that contains current OpE information and is available to the NRC staff and the public. The site also contains results from a variety of previously published studies that include, for example, reports on initiating events, system performance, component performance, common-cause failures, fire events, and loss-of-offsite-power.

For More Information
Contact John C. Lane, RES/DRA, John.Lane@nrc.gov.

Accident Sequence Precursor Program

Objective

The ASP Program has the following objectives:

- Provide a comprehensive, risk-informed view of nuclear power plant operating experience and a measure for trending core-damage risk.
- Provide a partial validation of the current state of practice in risk assessment.
- Provide feedback to regulatory activities, such as the Operating Experience and Industry Trends Program.

ASP Program results are also used as an input to the agency's Abnormal Occurrence Report and to monitor performance against the safety measures in the agency's Congressional Budget Justification. Specifically, the ASP Program provides input into the Performance Indicator for Performance Goal 4 under Safety Objective 1 that provides insights into the effectiveness of NRC's regulatory programs at preventing and mitigating accidents, ensuring radiation safety, and protecting the environment. The Performance Indicator is that there shall be less than or equal to 3 malfunctions, deficiencies, events, or conditions at commercial nuclear power plants (operating or under construction) that meet or exceed Adbnormal Occurrence (AO) criteriaII.A-II.D. Significant Precursors, which are defined as having a conditional core damage probability (CCDP or a change in core damage probability greater than or equal to 1 x 10^{-3}, are examples of events that are of high safety significance that are reported to Congress as AOs (criterion II.D.).

Research Approach

To identify potential precursors, the staff reviews operational events, including the impact of external events (e.g., fires, floods, and seismic events) from licensee event reports and inspection reports. The staff then analyzes any identified potential precursors by calculating the probability of an event leading to a core damage state.

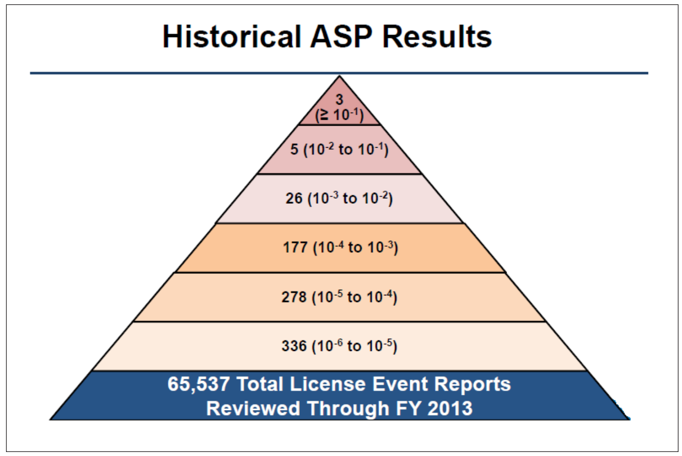

Figure 1.6 Historical ASP Results

An operational event can be one of two types: (1) an occurrence of an initiating event, such as a reactor trip or a loss of offsite power, with or without any subsequent equipment unavailability or degradation or (2) a degraded plant condition characterized by the unavailability or degradation of equipment without the occurrence of an initiating event. For the first type of event, the staff calculates a CCDP. This metric represents a conditional probability that a core damage state is reached given the occurrence of an initiating event (and any subsequent equipment failure or degradation). For the second type of event, the staff calculates a ΔCDP. This metric represents the change in core damage probability for a time period during which component(s) are deemed unavailable or degraded.

The ASP Program considers an event with a CCDP or ΔCDP greater than or equal to 1×10-6 to be a precursor. However, to focus ASP analyses on the more safety significant events, the ASP Program excludes (i.e. screens out as precursors) initiating events whose results would be similar to or less significant than a reactor trip coincident with the loss of balance-of-plant systems (e.g., feedwater and condenser heat sink) with no degradation of safety-related equipment.

Status

Updated results from the ASP Program are published in an annual paper to the Commission. The latest paper, SECY-14-0107, "Status of the Accident Sequence Precursor Program and the Standardized Plant Analysis Risk Models," was issued on October 6, 2014.

For More Information
Contact Keith Tetter, RES/DRA, at Keith.Tetter@nrc.gov.

Knowledge Management in the Office of Nuclear Regulatory Research

Objective

The mission of the Office of Nuclear Regulatory Research (RES) is to support the regulatory mission of the U.S. Nuclear Regulatory Commission (NRC) by providing technical advice, technical tools, and information for identifying and resolving potential safety issues, performing the research necessary to support regulatory decisions, and issuing regulatory requirements and guidance. RES's principal product is knowledge; therefore, knowledge management (KM) is an integral part of the RES mission. The KM program's objective is to capture, preserve, and transfer key knowledge among employees and stakeholders. This body of knowledge can be used when making regulatory and policy decisions and ensures that issues are viewed and analyzed within a historical context.

Research Approach

RES KM activities fall into several categories as follows:

Agency-Level KM Steering Committee and KM Staff Leads
The NRC has a KM Steering Committee in which senior-level managers consider new KM ideas and discuss future plans. The meetings cultivate an awareness of the value of KM initiatives agency wide and support staff with KM-oriented projects and goals. The Steering Committee also provides an opportunity for senior level managers to participate in the agency's various KM initiatives, such as Marketing and Standardization.

RES is a member of the committee and sends a representative to the meetings, which occur several times per year. The office presents KM ideas and concepts for discussion and also participates in agency KM initiatives. In addition, the KM Staff Leads meet several times a year and provide assistance to the KM Steering Committee and their individual office staff. The office also supports initiatives such as:

- Seminars
- Communities of Practice (CoPs)
- Regulatory Guides
- NUREG/KM series reports

RES Seminars

For several years, RES has sponsored monthly seminars on technical topics of broad agency interest. RES also sponsors special in-depth technical symposia on topics such as the Three Mile Island (TMI) accident, Fukushima, and Davis Besse Reactor Head Degradation. These events include staff presentations and also may feature special guests who have unique knowledge of the topic. For example, for the TMI seminar in 2009, speakers included Governor Richard Thornburgh of Pennsylvania (see Figure 1.7) and Ed Frederick, who was an operator on shift at the time of the accident at TMI in 1979. Some of these seminars are also captured in other KM activities such as NUREG/KM, which are discussed in further detail below. For example, NUREG/KM-0001 "Three Mile Island Accident of 1979 Knowledge Management Digest" has records of major TMI seminars conducted by NRC.

Figure 1.7 Governor Dick Thornburgh (PA) at a RES seminar on the 1979 accident at Three Mile Island.

Communities of Practice

A key aspect of the RES KM Program is the development of virtual CoPs in which RES staff members can share and collect information in their area of interest. RES now has several CoPs on topics such as human factors; high temperature gas reactors; liquid metal cooled reactors; fire protection; health effects; and structural, geotechnical, and seismic engineering.

Capturing Knowledge in NRC Regulatory Guides

The NRC Regulatory Guides are managed by RES. The Regulatory Guide series provides guidance to licensees and applicants on implementing specific parts of the NRC's regulations, techniques used by the NRC staff in evaluating specific problems or postulated accidents, and data needed by the staff in its review of applications for permits or licenses. The guides also serve as a vital repository for agency regulatory knowledge. The RES staff proactively seeks to capture knowledge from all NRC staff by routinely reviewing and updating these guides. The NRC staff develops this regulatory knowledge while addressing emerging technical and regulatory issues. Updating the guides keeps this institutional knowledge in a permanent and long-lasting record. The Regulatory Guide process is also a transparent and provides the opportunity for all stakeholders to contribute to capturing the knowledge behind agency regulatory decisions.

Publications - NUREGs

Official NRC reports or brochures on regulatory decisions, results of research, results of incident investigations, and other technical and administrative information are called NUREGs. RES is the agency leader for publishing KM focused NUREGs that compile historic information, video, and references. These publication series focuses exclusively on collecting and interpreting historical information on technical topics for the benefit of future generations of NRC professionals. A list of NUREG/KMs is available at http://www.nrc.gov/reading-rm/doc-collections/nuregs/knowledge/.

RES is currently developing two NUREG/KMs on Chernobyl and hydrogen and is revising NUREG/KM-0001 to a three-volume report.

Status

Knowledge Management at the NRC is an ongoing activity and RES will continue efforts to capture, preserve, and transfer knowledge among employees and stakeholders.

For More Information
Contact Felix Gonzalez, RES/DRA, at Felix.Gonzalez@nrc.gov.

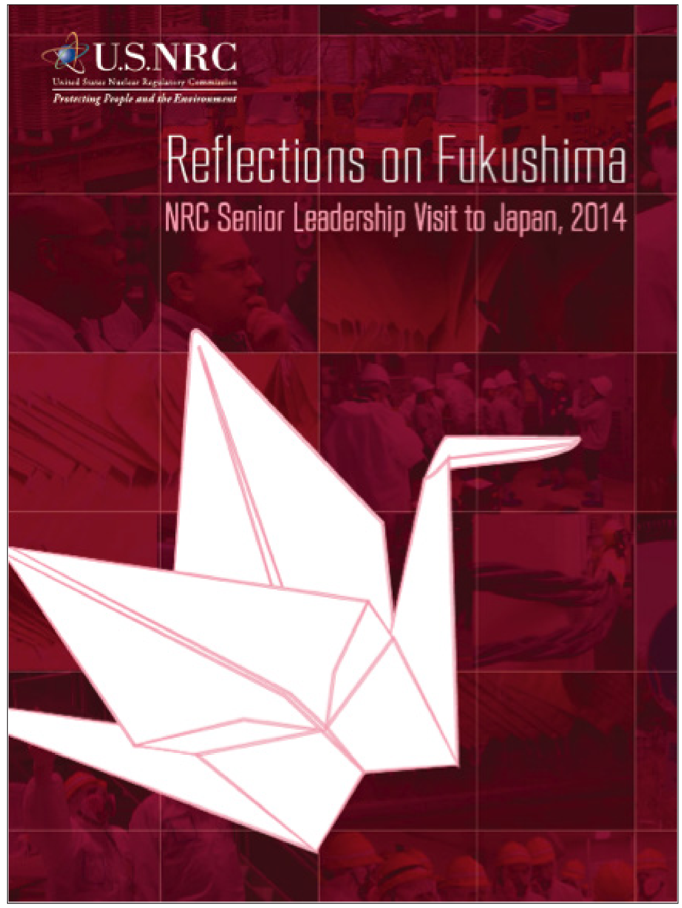

Figure 1.8 NUREG/KM-0008, "Reflections on Fukushima: NRC Senior Leadership Visit to Japan, 2014".

Chapter 2: Thermal-Hydraulic Research

The Office of Nuclear Regulatory Research (RES) provides the tools and methods that the U.S. Nuclear Regulatory Commission (NRC) program offices use to review licensee submittals and to evaluate and resolve potential safety issues. For thermal-hydraulic analyses, the NRC uses the Transient Reactor Analysis Code/Reactor Excursion and Leak Analysis Program (TRAC/RELAP) Advanced Computational Engine (TRACE) code to perform the following:

- Confirmatory calculation reviews of licensee submissions, such as those for extended power uprates and license renewals.
- Exploratory calculations to establish the technical bases for rule changes such as the proposed revisions to the emergency core cooling system rule in Title 10 of the Code of Federal Regulations (10 CFR) 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."
- Exploratory calculations for the resolution of generic issues such as Generic Issue 191, "Assessment of Debris Accumulation on PWR [pressurized-water reactor] Sump Performance."
- Confirmatory calculations in support of design certification and combined operating license reviews for new reactors. The modeling of various small modular reactor (SMR) designs has been undertaken to assess the applicability of NRC codes in anticipation of confirmatory analyses.

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.

The NRC maintains several experimental research programs that directly support reactor safety code development. These experimental programs investigate thermal-hydraulic phenomena and provide data and analysis used to improve the predictive capability of the codes. The TRACE code is currently assessed against a matrix of more than 500 cases. However, when a new phenomenon or design is identified that falls outside of the assessment base, new experimental programs must be developed to collect relevant data to support further TRACE development. The data collected in these programs are used to develop TRACE models as well as the validation of those models as assessment cases that are added to the already substantial assessment matrix. Figure 2.1 shows an example of a simplified reactor system nodalization for TRACE.

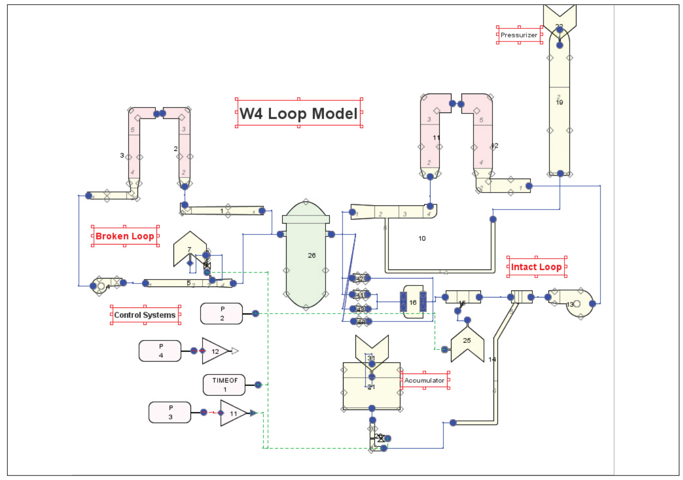

Figure 2.1 Simplified plant model nodalization.

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 multidimensional details can enhance the understanding of certain phenomena and thus play a role in reducing the uncertainty in the technical bases for licensing decisions. RES has developed a state-of-the-art CFD capability that supports multiple offices within the agency. 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 more than 200 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 thermal-hydraulic programs and maintains a high level of expertise in the field. NRC conducts the Code Application and Maintenance Program (CAMP) to assess and improve its thermal-hydraulic transient computer codes. About 30 nations signed bilateral cooperative agreements with the United States providing contributions in the form of model development, code assessment, and information generated from applying the codes to operating nuclear power plants. This state-of-the-art capability provides a robust infrastructure for both confirmatory and exploratory thermal-hydraulic computations.

TRAC/RELAP Advanced Computational Engine (TRACE) Thermal– Hydraulics Code

Objective

The TRAC/RELAP Advanced Computational Engine (TRACE) Version 5.0 code is a single code developed by the NRC that has improved ease of use, speed, robustness, flexibility, maintainability, and upgradability compared to past codes and code versions. NRC reactor systems engineers use TRACE to analyze operational and safety transients such as small and large break loss-of-coolant accidents (LOCA) in pressurized-water reactors (PWRs) and boiling-water reactors (BWRs) as well as the interactions between the related neutronics and thermal-hydraulic systems. The thermal-hydraulic and neutronics capabilities of TRACE V5.0 enable the NRC to make independent evaluations of transients for existing and new reactor designs. The NRC uses these capabilities to perform sensitivity assessments of system hardware and phenomena using different analytical or modeling approaches.

Research Approach

The TRACE 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 the Purdue Advanced Reactor Core Simulator (PARCS). The code is capable of performing any type of reactor analysis previously performed by each of the predecessor codes and has component models and mesh connectivity that allow a full reactor and containment system to be easily modeled.

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). ECI is a request-driven interface that allows TRACE to communicate with any code that implements the ECI without actually having to make any modifications to TRACE. ECI has allowed TRACE to be easily coupled to codes such as Symbolic Nuclear Analysis Package (SNAP), Containment Analysis Code (CONTAIN), Regional Mixing Model (REMIX), and Matrix Laboratory (MATLAB). The interface should allow TRACE to be coupled to computational fluid dynamics (CFD) or other special purpose codes in the future. TRACE uses a modern code architecture that is portable, easy to maintain, and easy to extend with new models to address future potential safety issues (Figure 2.2 depicts a graphical representation of TRACE). TRACE runs successfully on multiple operating systems including Windows, Linux, and Mac OSX.

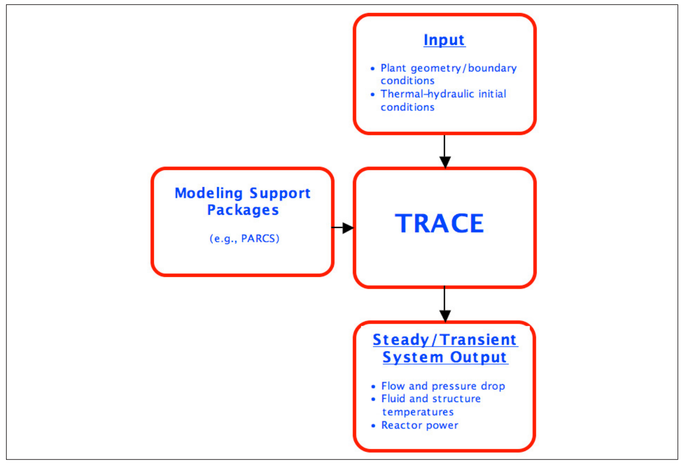

Figure 2.2 TRACE, an advanced, best-estimate reactor system code used to model the thermal-hydraulic performance of nuclear power plants.

Code quality is the goal of a stringent development process. 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. The current assessment test matrix for TRACE contains more than 500 cases. The TRACE assessment test matrix contains a comprehensive set of 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. The assessment matrix includes experimental data obtained through NRC-funded experiments and international collaboration.

Status

TRACE code development and assessment is an ongoing process. Recently, the NRC addressed modeling issues identified during (1) an independent peer review completed in 2008, (2) the development of input models used to support the licensing of new and operating reactors, and (3) code assessment activities leading up to the release of Version 5.0. These efforts ultimately led to the release of TRACE V5.0 Patch 4 in April 2014.

Improvements underway for future versions of TRACE are focused on enhancing capabilities related to the simulation of advanced reactor designs such as the U.S. Advanced Pressurized-Water Reactor, the U.S. Evolutionary Power Reactor, and the Advanced Passive 1000 Megawatt as well as small modular reactors. Significant efforts also are directed towards fixing bugs, addressing peer review findings, and improving code robustness and run time performance. The TRACE development team recently released V5.0 Patch 4 to address some of the issues identified to date, and additional patch releases are planned. TRACE will provide a robust and extensible platform for safety analyses well into the future.

For More Information
Contact Chris Hoxie, RES/DSA, at Chris.Hoxie@nrc.gov.

Symbolic Nuclear Analysis Package (SNAP) Computer Code Applications

Objective

The Symbolic Nuclear Analysis Package (SNAP) is a single, standardized graphical user interface (GUI) that is used with many NRC analytical codes. Currently, SNAP has interfaces for the Reactor Excursion and Leak Analysis Program (RELAP5), TRAC/RELAP Advanced Computational Engine (TRACE), SCALE, Containment Analysis Code (CONTAIN), MELCOR, Radionuclide Transport, Removal, and Dose code (RADTRAD), and FRAPCON3. The input models for most codes are text based, requiring the user to write an input file (or deck) in a text editor and then run the analysis program. Each computer code uses different input formats and variable names. This adds to the burden on the analysts, who usually use more than one modeling program to perform a review. To lessen this model development burden, the NRC decided that it would be cost effective and efficient for the analysts to develop a common GUI for its codes. SNAP removes the need for analysts to use the text-based entry methods and to transfer or replicate data among several different programs. Because the core look and feel of SNAP is the same, it is less likely that an error will be made due to differences in input formats.

Research Approach

SNAP provides a powerful, flexible, and easy-to-use GUI both to prepare analytical models (Figure 2.3) and to interpret results (Figure 2.4). SNAP contributes to efficient model development by instituting component subsystems that are a convenient way to improve the logical layout of a model by allowing the components that make up physical systems (such as steam generators) to be grouped together. Therefore, a library of common steam generator types can be stored and reused in future model development (and similarly for other nuclear plant components).

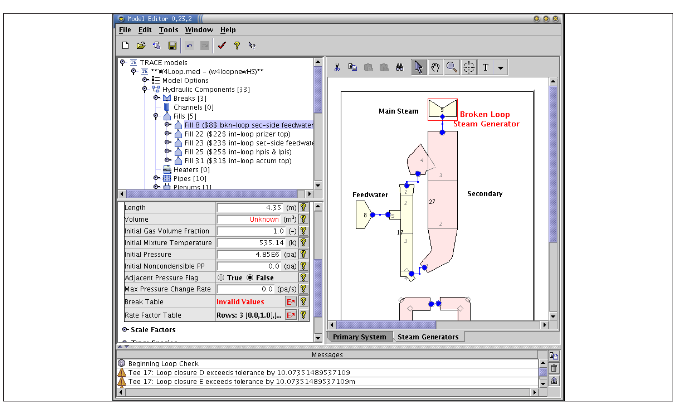

Figure 2.3 Creating input models using SNAP.

In addition, modeling guidelines for code users have been integrated into SNAP, thus enabling the analyst to quickly build models that follow best practices for input model creation. SNAP verification tools and automated model checking tools correct input model errors before the input model is ever deployed, thus saving the analyst time and effort.

SNAP's interactive and post-processing capabilities are predominately realized within its animation displays. Within such a display, the results of a calculation may be animated and visualized in a variety of ways. Animation models (or "masks") are composed of "views" containing a number of visual elements (e.g., time-dependent plots of axial reactor core power, or coolant temperatures.) Thus, an animation display retrieves data from a Calculation Server and represents it visually in some fashion. This data can be from actively running calculations or completed calculations.

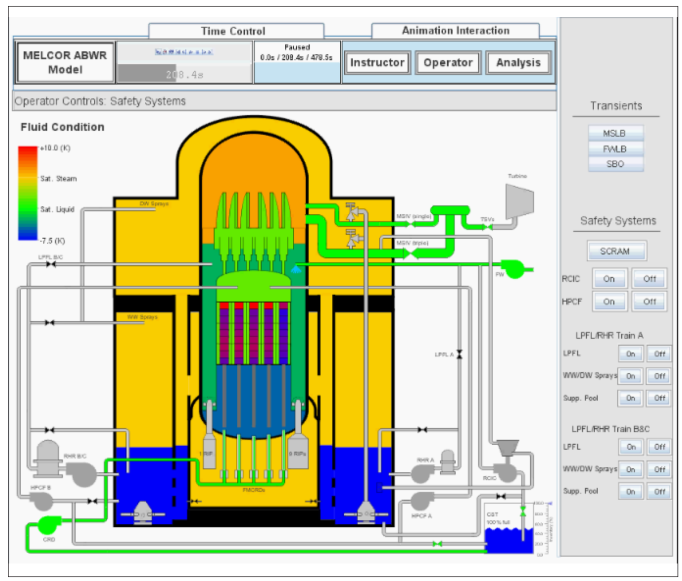

Figure 2.4 Animating analysis results using SNAP.

SNAP provides the analyst with a convenient framework and set of tools to allow the user to select certain parameters for "ranging" and study. These can be model parameters (e.g., parameters from TRACE's interfacial drag models) or input parameters (such as flow rates and temperatures).

Typically, the analyst would direct SNAP to vary the parameter over a certain range and have the values distributed according to a probability distribution function. With this series of inputs and a properly "instrumented" code that has been modified to expose to the analyst access to the parameters of interest, SNAP can direct the execution of multiple instances of the code (e.g., TRACE). SNAP then collects the code results and packages them up for automated submission to Sandia National Laboratories' Design Analysis Kit for Optimization and Terascale Applications (DAKOTA) code for a statistical analysis. The result is a DAKOTA report that contains the results of the uncertainty quantification.

Status

Improvements to the SNAP-RADTRAD user interface were introduced based on user feedback and stakeholder comments. A major portion of these improvements included a new source term editor. The new source term editor features built-in source terms referenced in the commonly used Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design-basis Accidents at Nuclear Power Reactors," for offsite dose estimation as well as access to the International Commission on Radiological Protection (ICRP) Publication 38, 838 nuclide database.

New nonmodel-based uncertainty quantification inputs were added as well as improvements to the uncertainty quantification user interface and generated reports.

A SNAP-SCALE plug in was developed that provides a user interface for the current SCALE 6.1 code. Specifically, this new SNAP-SCALE interface currently only supports the TRITON depletion sequence in SCALE. The user interface previously used for the TRITON sequence was re-engineered and implemented in SNAP to further consolidate user interface functionality for the analytical codes that the NRC uses.

For More Information
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Thermal-Hydraulic Simulations of Operating Reactors, New Reactors, and Small Modular Reactors

Objectives

The U.S. Nuclear Regulatory Commission (NRC) uses the TRACE code to confirm industry calculations submitted to the NRC to meet the requirements of Title 10 of the Code of Federal Regulations (10 CFR) 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors." RES analysts evaluate transient and accident scenarios and assess potential generic safety issues and licensing issues for both operating plants and new reactor designs. These can be analysis of design-basis loss-of-coolant accidents, anticipated operational occurrences, anticipated transient without scram, and other transients. TRACE is able to simulate the multifaceted evolution of these events, capturing all of the major system operations and thermal-hydraulic processes that unfold.

TRACE calculations support design certification and combined operating license reviews for new reactors—the Advanced Passive 1000 Megawatt, U.S. Advanced Pressurized-Water Reactor, the U.S. Evolutionary Power Reactor, the Economic Simplified Boiling-Water Reactor, and the Advanced Boiling-Water Reactor. In addition, the modeling of small modular reactor (SMR) designs has been undertaken to assess the applicability of NRC codes in anticipation of confirmatory analyses.

Research Approach

TRACE plant input decks are developed for specific simulations (Figure 2.5). Depending on the simulations performed, the size and complexity of plant input decks can range from single-system components to the entire nuclear steam supply system.

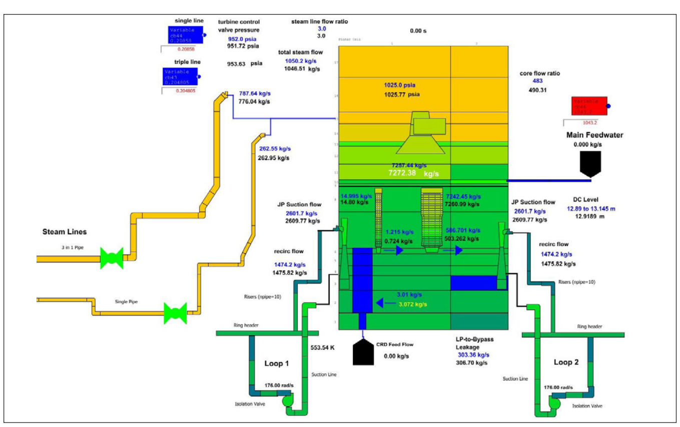


Figure 2.5 Steady-state conditions in a boiling-water reactor.

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 or cells (Figure 2.6) 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. Once the arrangement of the plant deck is complete and each component is set with initial values for normal operating pressures, temperatures, and flow 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.

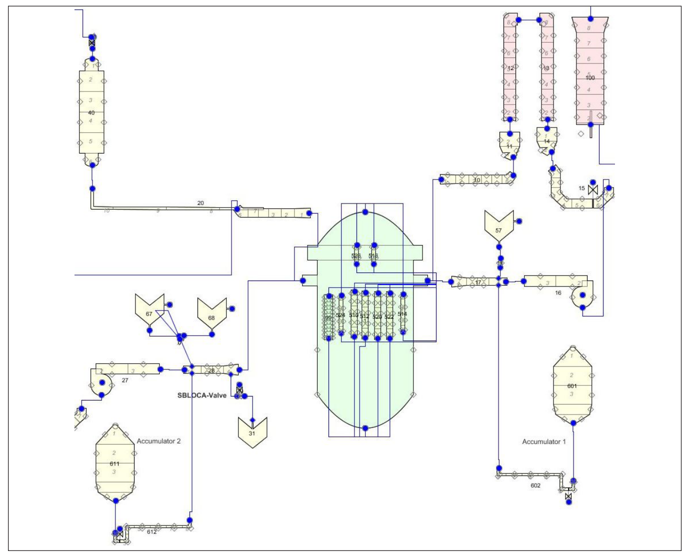


Figure 2.6 Key primary coolant T/H components, including reactor vessel, pumps, and steam generator, for a two-loop pressurized-water reactor.

The NRC updated plant input decks developed for other system codes and converted them into the TRACE format to support the licensing reviews of extended power uprate applications. It uses these models to assess the effects of increased power on system behavior and safety margins. As part of the NRC's pre-design certification application reviews for SMRs (mPower by Babcock & Wilcox (B&W), NuScale, and Westinghouse), efforts have been initiated to identify and examine key technical and policy issues potentially shared by various SMR designs.

Status

RES is developing a library of TRACE input decks for simulating currently operating PWRs and boiling-water reactors (BWRs). Building a comprehensive library of plant input decks will enhance the ability of the NRC staff to efficiently perform confirmatory analyses to support regulatory decisions.

For More Information
Contact Scott Elkins, RES/DSA, at Scott.Elkins@nrc.gov.

Simulation of Anticipated Transients Without SCRAM with Core Instability for Maximum Extended Load Line Limit Analysis Plus for Boiling-Water Reactors

Objectives

The industry has proposed the maximum extended load line limit analysis plus (MELLLA+) domain for boiling-water reactors (BWRs) that have extended power uprates (EPUs). The MELLLA+ domain would allow operation at high reactor thermal power (up to 120 percent of originally licensed thermal power [%OLTP]) at reduced reactor core flow (as low as 80 percent of rated core flow [%RCF]). The high power-to-flow operating point (120 %OLTP / 80 %RCF) introduces new concerns related to the consequences of anticipated transient without SCRAM (ATWS) events initiated from this point. In particular, the plant will evolve to a condition of high power-to-flow ratio during an ATWS in which large amplitude power oscillations are expected to occur. Figure 2.7 illustrates the transient evolution of postulated ATWS events for a plant operating at the low flow corner of the MELLLA+ upper boundary.

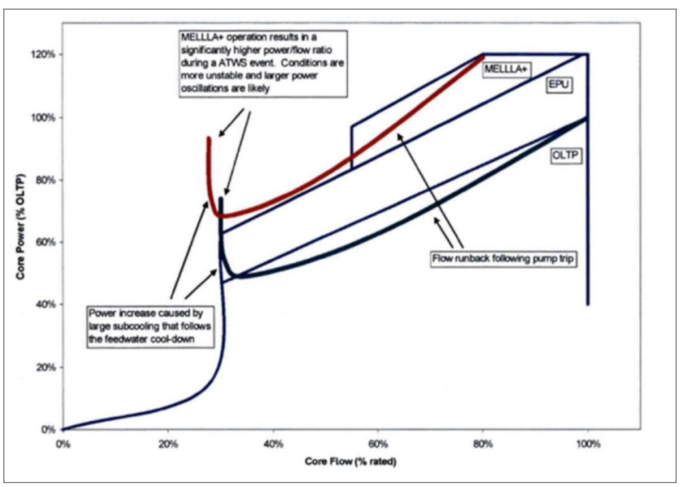


Figure 2.7 Operating state evolutions during ATWS for different operating domains.

Research Approach

Studies of ATWS with instability (ATWS-I) events using TRACE/PARCS were performed to better understand the dynamic coupling during ATWS-I and the safety implications associated with the MELLLA+ operating domain. Simulation of ATWS-I required several codes and a defined methodology for the use of these codes and interfaces. RES developed a methodology for generating large core models in TRACE comprising many channels to represent the thermal-hydraulic and fuel thermal-mechanical response of the core. The model uses FRAPCON calculations to generate dynamic gap conductance properties for the fuel.

Once the core model was generated and incorporated into the TRACE model, calculations were performed using TRACE and PARCS in a coupled manner. Figure 2.8 illustrates the process for performing these coupled calculations. One key feature of the TRACE/PARCS method is the use of flux harmonic calculations to excite in-phase and out-of-phase core oscillations.

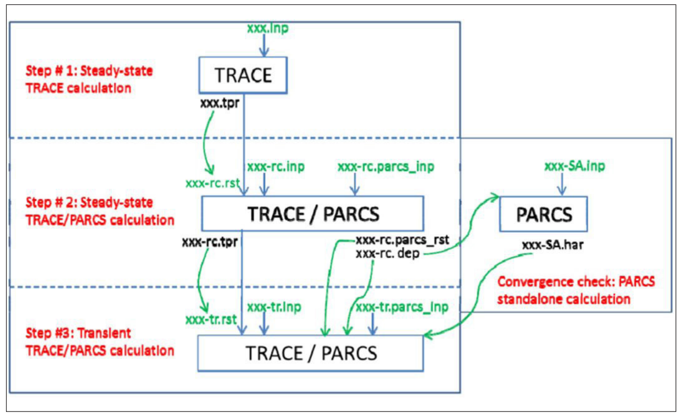


Figure 2.8 TRACE/PARCS coupled methodology.

Status

Visualization tools have been developed to analyze the evolution of the power oscillations during ATWS-I and to study the oscillation contour. Figure 2.9 illustrates the result of an ATWS-I simulation. TRACE/PARCS predicts the onset of large amplitude power oscillations and the evolution of an out-of-phase oscillation contour in this example. Results from this analysis are still under investigation.

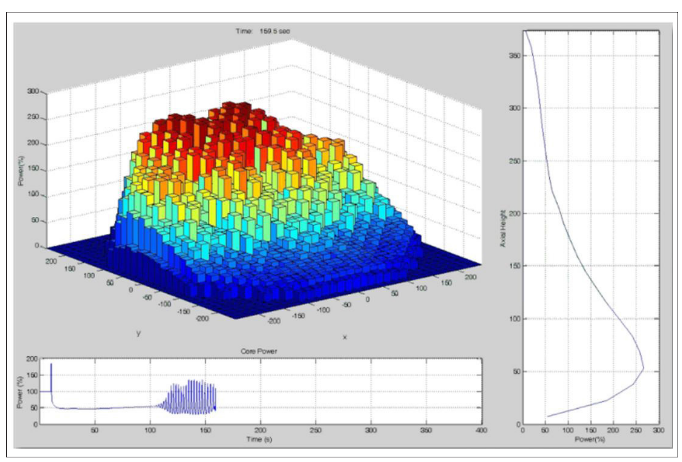


Figure 2.9 Power oscillation visualization during simulated ATWS-I.

For More Information
Contact Scott Elkins, RES/DSA, at Scott.Elkins@nrc.gov.

Computational Fluid Dynamics in Regulatory Applications

Objectives

Computational fluid dynamics (CFD) provides detailed three-dimensional fluid flow information not available from system code thermal- hydraulic simulations. These multidimensional details can enhance the understanding of certain phenomena and thus play a role in reducing the uncertainty in the technical bases for licensing decisions.

Research Approach

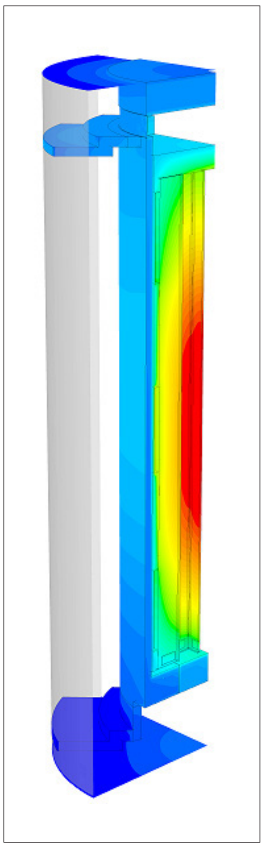
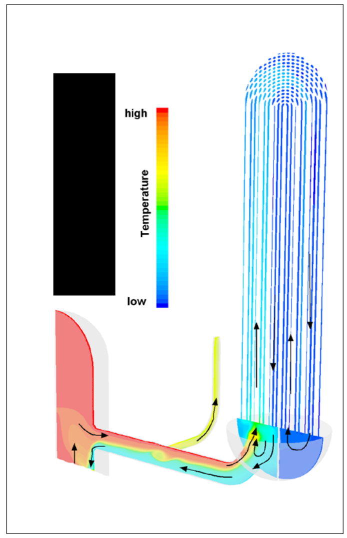


Figure 2.10 Temperature contours of a ventilated dry cask.

The U.S. Nuclear Regulatory Commission's Office of Nuclear Regulatory Research (RES) has developed a state-of-the-art CFD capability that supports multiple regulatory reviews within the agency. RES uses the commercial CFD codes from ANSYS (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 more than 200 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.

Status

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 further informing the agency's technical bases for licensing decisions (see Figure 2.10 above).



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 (see Figure 2.11).

Figure 2.11 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.

RES used CFD to confirm the distribution of injected boron in the ESBWR. In the design certification of the U.S. APWR, CFD was used to investigate the performance of an advanced accumulator (see Figure 2.12). The phenomena of interest are cavitation and nitrogen ingress, which exceed typical system code capabilities. CFD results also were used to examine possible scale effects.

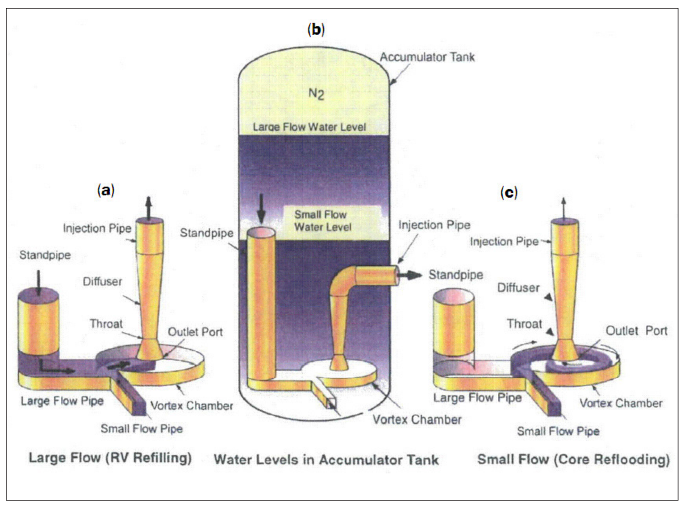


Figure 2.12 The advanced accumulator (b) is a water storage tank with a flow damper in it that switches the flow rate of cooling water injected into a reactor vessel from a large (a) to small (c) flow rate.

For More Information

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Code Application and Maintenance Program (CAMP)

Purpose

The Code Application and Maintenance Program (CAMP) evolved from the earlier International Code Assessment Program (ICAP). CAMP facilitates cooperation and sharing among the 30 participant countries in the areas of thermal-hydraulic (T/H) research and analysis. In addition to exchanging technical information and data, CAMP members contribute funds to help maintain and improve the TRAC/RELAPS Advanced Computational Engine (TRACE), Purdue's Advanced Reactor Core Simulator (PARCS), and Symbolic Nuclear Analysis Package (SNAP) codes. The research conducted and in-kind information exchanged under this program enables the NRC to leverage agency resources while expanding the agency's knowledge and database. It also provides independent verification and validation of the accuracy of the TRACE code through the expansion of the international user community and their subsequent applications and feedback.

Research Approach

The CAMP program provides members with the TRACE, PARCS, and SNAP codes, and the Reactor Excursion and Leak Analysis Program (RELAP5). The TRACE code is the NRC's primary T/H reactor system analysis code. PARCS is a multidimensional reactor kinetics code coupled to TRACE. SNAP is a graphical user interface to the codes that provides preprocessing, runtime control, and postprocessing capabilities. RELAP5 is a legacy NRC T/H computer code, and no further development is being done; however, bugs are patched when found. These codes are used to analyze accidents and transients in operating reactors, support the resolution of generic issues, evaluate emergency procedures and accident management strategies, confirm licensees' analyses, test the fidelity of NRC simulators, provide training exercises for NRC staff, and support the certification of advanced reactor designs.

During CAMP meetings held two times per year, members have an opportunity to present their technical findings. More specifically, the members (1) share their experience using NRC T/H computer codes to identify errors and to perform 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).

The CAMP program has provided more than 100 NUREG/IAs that have contributed to the development, assessment, and application of the NRC T/H analysis codes. The NUREG/IAs are listed on NRC's public Web site at http://www.nrc.gov/reading-rm/doc-collections/nuregs/agreement/. Technical areas span the entire range of accident and transient analysis. These include 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; and condensation with noncondensibles. The reports document the contributions made to assessment, plant analysis, and physical model development.

Status

In several recent cases, contributions to the CAMP program provided important code improvements and efficiencies for NRC's regulatory programs. For example, Slovenia developed a subcooled boiling model that may be used in TRACE. Taiwan used TRACE to simulate the station blackout (SBO) event that occurred at the Maanshan pressurized-water reactor (PWR) and also performed additional TRACE calculations to study potential SBO mitigation strategies. Korean modeling of the advanced accumulator in the proposed AP1400 reactor design has helped guide NRC efforts to model the advanced accumulator of the U.S. advanced pressurized-water reactor (APWR), which has similar design features.

Another recent Korean in-kind contribution was its sharing of a multi-energy group solver for NRC's PARCS code. This addition to PARCS removes the present limitation of two neutron energy groups and allows PARCS to more accurately model situations in which a multigroup approach is desirable (e.g., mixed oxide [MOX] fueled cores).

Several CAMP members have built large, detailed TRACE models to facilitate their in-kind technical contributions. For example, CAMP members have shown good results in TRACE assessments of the Rig of Safety Assessment (ROSA) and Primärkreislauf - Versuchsanlage (PKL, primary coolant loop test facility) integral test facilities, in separate effects condensation tests, and in the boiling-water reactor full-size, fine-mesh bundle test single-channel steady-state and transient tests. Members also have demonstrated coupling TRACE to computational fluid dynamics (CFD) codes.

For More Information
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Thermal-Hydraulic Cooperative Programs

Objectives

The NRC conducts and participates in domestic and international thermal-hydraulic (T/H) experimental programs to improve TRACE code predictive capability. Data from these experimental programs are used for code assessment and validation and to develop correlations used in the code. 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 boiling-water reactors and pressurized-water reactors. The set of experimental data against which TRACE has been validated is more comprehensive than any other NRC T/H code in terms of scope, quantity, and quality.

Research Approach

Three primary domestic experimental research programs as well as several international programs have played a fundamental role in providing necessary T/H data for improving TRACE code predictive capability.

- Thermal-Hydraulics Institute (THI): The THI is a consortium of universities that has been performing separate effects experiments for the NRC since 1997. Several unique test facilities are used to perform a wide variety of T/H experiments. The emphasis of these tests has been interfacial area transport in pipes, annuli, and rod bundles. In addition, work has been conducted to investigate post critical heat flux heat transfer.
- Rod Bundle Heat Transfer (RBHT) Program: The RBHT program involves separate effect experiments using a full-length rod bundle designed to simulate a light-water reactor rod bundle. The facility is capable of high temperatures and is heavily instrumented. In addition, the RBHT facility has the capability for advanced droplet visualization techniques. The tests focus on steam cooling and reflood T/H, including the influence of spacer grids and the behavior of droplets because these items are important in determining key regulatory figures of merit, such as peak clad temperature.
- Advanced Multi-Phase Flow Laboratory (AMFL): The AMFL performs two-phase flow experiments in a highly instrumented flow loop facility that is used to design and perform scaled experiments as well as to pursue theoretical and computational treatment of multiphase flows. Researchers have used the AMFL to enhance the database for Interfacial Area Transport Models. The experimental data are acquired by state-of-the-art two-phase flow instrumentation including the four sensor conductivity probe, high-speed camera, and laser Doppler anemometer. The obtained data will be used for developing the two-group interfacial area transport model that has been implemented in test versions of the TRACE code. This new interfacial area transport model will improve TRACE code capabilities in predicting two-phase flow characteristics and heat-transfer phenomena. The use of this new model will effectively avoid the shortcomings of the traditional experimental correlations that are based on flow regimes and regime transition criteria.
- International Experimental Programs: In addition to data from NRC-funded experiments, the assessment matrix includes experimental data obtained through international collaboration. Among these are experiments at the loop for the study of T/H systems (BETHSY), Rig of Safety Assessment (ROSA), and passive decay heat removal and depressurization test (PANDA) facilities. NRC also participates in a series of experimental programs fostered by the Organisation for Economic Co-operation and Development (OECD) (e.g., the Primärkreislauf - Versuchsanlage [PKL] primary coolant loop test facility) to investigate safety-related issues relevant to current and new reactor designs.

Status

The THI program delivered experimental data on void fraction, pressure drop, and interfacial area transport. Among other things, these data have been used to develop assessment cases for several geometric configurations and in the development and validation of interfacial area transport models for a future version of TRACE. Likewise, the RBHT and AMFL programs have provided valuable data that is being applied to two-phase flow, spacer grid, and droplet behavior models.

To demonstrate the applicability of TRACE to the EPR, code predictions were assessed against data acquired from separate and integral test facilities, such as Advanced Power Extraction (APEX), Full Length Emergency Cooling Heat Transfer Separate Effects and Systems Effects Tests (FLECHT SEASET), Rig of Safety Assessment (ROSA). Integral test data from the Purdue University Multi-Dimensional Integral Test Assembly (PUMA) and Passive Non Destructive Assay of Nuclear Materials (PANDA) facilities were used to assess the code for the prediction of void distributions and two-phase natural circulation for the ESBWR design. Integrated System Test (IST) facilities are being used to assess TRACE for applicability for use in system analysis of small modular reactor designs.

For More Information

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Chapter 3: Fuel and Core Research

The Office of Nuclear Regulatory Research (RES) oversees and executes a wide range of experimental and analytical research programs in the areas of nuclear fuel and reactor core behavior. These research programs are summarized below and detailed in this chapter.

The NRC develops and maintains the neutronics code SCALE. SCALE is a nuclear analysis code system to perform independent reactor and criticality safety analyses for existing and new nuclear reactor designs, spent fuel pools, and spent fuel storage and transportation casks. The broader term nuclear analysis describes the use of analytical tools and experimental data to predict and understand interactions between nuclear radiation and matter within various nuclear systems.

Also in the area of neutronics and criticality, RES recently completed the implementation of full burnup credit (i.e., actinides and fission products isotopes) for pressurized-water reactor spent nuclear fuel. RES is currently developing the technical basis to support an agency-wide, integrated approach to further expand the application of burnup credit in spent nuclear fuel storage and transportation systems to boiling-water reactor spent nuclear fuel.

The NRC is engaged in various research activities related to the performance of high-burnup light-water reactor (LWR) fuel. Many of these activities are related to maintaining the ability to predict all important aspects of high-burnup LWR fuel performance via NRC's steady-state and transient fuel performance codes. These research activities include the development of methods to assess the potential for fuel dispersal during loss-of-coolant accidents (LOCA) and to evaluate the potential consequences of fuel dispersal under LOCA conditions.

The research activities also include measurement of mechanical properties of high-burnup fuel rods. For example, tests have been performed at Oak Ridge National Laboratory to determine the fatigue characteristics of high-burnup spent nuclear fuel and how much the fuel participates with the cladding to increase the bending stiffness and strength of the fuel rod. These measurements will be used in analysis to evaluate safety of the transportation of high-burnup spent nuclear fuel under normal transport conditions and hypothetical accident conditions.

The NRC maintains computer codes for the analysis of both steady-state and transient conditions. The agency uses these codes to evaluate experimental data and to audit licensees' safety analyses. As new fuel designs and materials are introduced and higher burnups are sought (beyond 62 gigawatt day per ton), the materials' properties and models in the codes must be revised. In-reactor tests are often used to obtain data for these model revisions. The ability to perform quantitative analyses of fuel rod behavior is an essential part of the NRC's assessment of safety in reactor operations and spent fuel transportation and storage.

The NRC interacts in various ways with the Department of Energy (DOE) on fuels related research programs such as the Used Fuel Disposition Campaign and the Advanced Fuel Campaign. The staff's interactions with DOE on these programs are typically oriented to maintain awareness of research developments.

The NRC engages in multiple international cooperative research programs related to nuclear fuel. These programs include the Halden Reactor Project in Norway, where about 10-12 test rigs are under irradiation at any one time and a similar number are either undergoing post irradiation examination or in preparation for starting irradiation. The NRC relies on fuel property data from Halden to validate its steady-state and transient fuel performance codes, including steady-state gas release and thermo-mechanical behavior and fuel behavior under demanding operation conditions and accident scenarios.

The NRC participates in the Studsvik Cladding Integrity Project (SCIP III) in Sweden that is focused on issues related to high-burnup fuel under LOCA conditions, in particular on fuel fragmentation, relocation, and dispersal. The NRC also is working actively with partners at the Nuclear Regulation Authority in Japan and the Institut de Radioprotection et de Sûreté Nucléaire in France on LOCA issues.

NRC also participates in the Organisation for Economic Co-operation and Development/Nuclear Energy Agency Working Party on Nuclear Criticality Safety and the Committee on the Safety of Nuclear Installations Working Group on Fuel Safety. NRC participates in various international benchmark exercises to compare our neutronics codes against experimental data, the development of criticality methodologies, and the development of technical basis for the application of burnup credit.

Nuclear Analysis and the SCALE Code

Objective

An objective of the NRC's Office of Nuclear Regulatory Research (RES) is to perform independent neutronics and criticality analyses for existing and new nuclear reactor designs, spent fuel pools, and spent fuel storage and transportation casks.

Research Approach

Nuclear analysis combines 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 encompasses the analyses of (1) fission reactor neutronics, both steady-state and dynamic; (2) nuclide generation and depletion as applied to predicting 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 fluence leading to material damage, 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).

Nuclear analysis efforts support the staff's ongoing and anticipated nuclear safety evaluation activities for the licensing and oversight of (1) existing reactors, front-end fuel cycle activities, and spent fuel storage, transport, and disposal systems; and (2) 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 Advanced Module for Processing Cross Sections (AMPX) code for processing fundamental nuclear data in the Evaluated Nuclear Data File (ENDF) into code-usable libraries of continuous energy or fine-group nuclear cross-sections and related nuclear data, (2) the SCALE 6.2 modular code system, and (3) the Purdue Advanced Reactor Core Simulator (PARCS) core neutronics simulator code. When appropriate, RES integrates planned nuclear analysis activities into larger NRC research plans for the respective applications.

An example of the need for additional data for current and near-term activities is in the area of boiling-water reactor (BWR) burnup credit for the criticality safety analysis of spent fuel casks. Operating and new reactors need experimental data to validate codes and to reduce uncertainties. Such validation currently relies on limited data or code-to-code comparisons. The NRC has validated nuclear codes for partial mixed-oxide fueling in pressurized-water reactors (PWR) and for PWR burnup credit application in spent fuel casks.

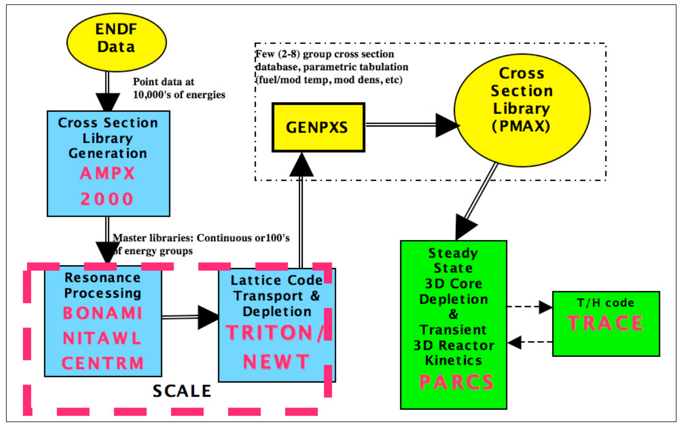

Figure 3.1 NRC nuclear analysis codes for reactor physics.

Status

The NRC is currently modifying and extending codes to accommodate different fuel and core designs and operating features of existing and new reactors. A new SCALE automated calculation sequence called Polaris is being developed to allow quicker lattice cross-section generation execution times and engineering evaluations. In addition, the NRC is updating the radiation shielding codes for application to high-capacity spent fuel cask systems and advanced reactor systems. The NRC also is validating its codes against plant operating and test data for use in steady-state and transient analyses of existing PWR and BWR cores and for new reactors such as the small modular reactors and Economic Simplified BWR.

For More Information

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High-Burnup Light-Water Reactor Fuel

Objective

Current research on high-burnup (HBU) light-water reactor fuel is focused in the following general areas:

- Development of methods to assess the potential for fuel dispersal during loss-of-coolant accidents (LOCA) and evaluate the potential consequences of fuel dispersal under LOCA conditions.
- Fuel rod properties for transportation and storage analysis.
- Fuel rod computer codes used to audit licensees' evaluation models that demonstrate compliance with criteria and to analyze test data.

Research Approach

The research to develop methods to address the potential for fuel dispersal during LOCAs and on fuel rod computer codes in general will be discussed on the next page that covers Fuel Rod Thermal and Mechanical Modeling and Analyses.

The research on fuel rod properties for transportation and storage analysis is conducted at Oak Ridge National Laboratory (ORNL). In this program, the flexural rigidity and fatigue life of HBU fuel were investigated using an innovative system, the Cyclic Integrated Reversible-bending Fatigue Tester (CIRFT), shown in Figure 3.2.

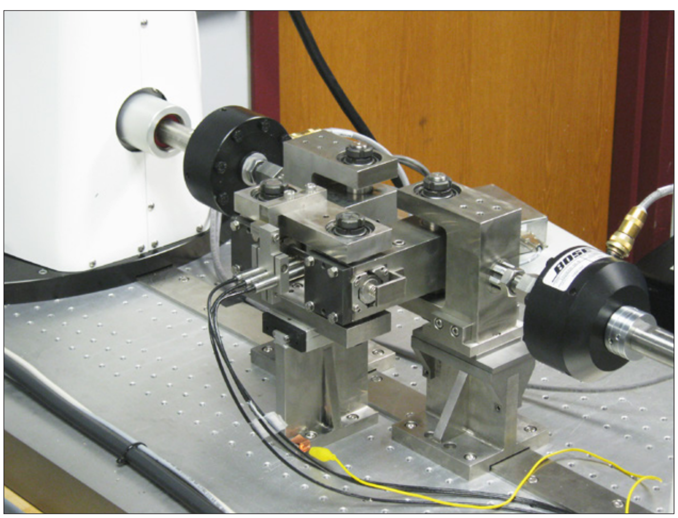

Figure 3.2 The CIRFT device at ORNL. A sister device is installed in a hot-cell to allow for testing of irradiated materials.

Status

The NRC recently published the results of the CIRFT testing program in NUREG/CR-7198. Two highlights of the research results are the measurement of bending moment as a function of curvature in static tests (Figure 3.3) and the maxima of absolute strain extreme as a function of number of cycles (Figure 3.4).

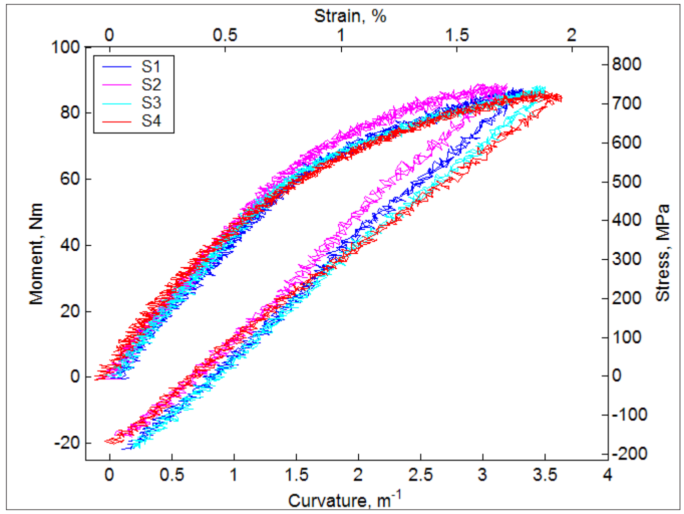

Figure 3.3 Moment-curvature measurements in static tests showing loading and unloading response. The corresponding stress/strain is displayed on right/top axes, respectively.

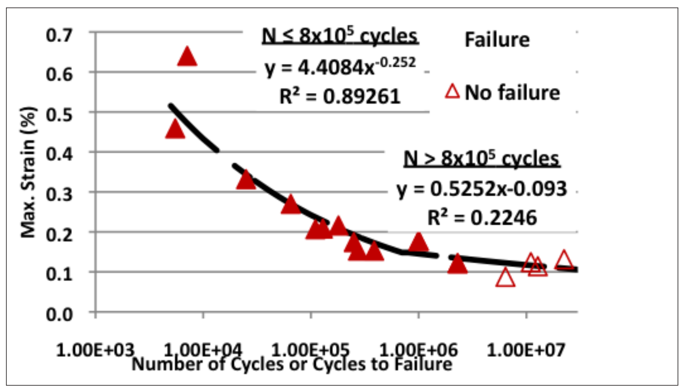

Figure 3.4 Maxima of absolute strain as a function of number of cycles to failure with curve-fitting extended to include the no-failure data points.

The results reported in NUREG/CR-7198 represent a significant advancement in the understanding of fuel rod properties as it is one of the few sources of data that allows for the evaluation of the high-burnup fuel rod as a system, including the presence of intact fuel inside the cladding and any fuel/cladding bonding effects. The properties measured in this testing program will be used in the evaluation of spent nuclear fuel (SNF) integrity under normal conditions of transport when combined with details of an SNF cask design and expected transportation loading conditions.

For More Information

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Fuel Rod Thermal and Mechanical Modeling and Analyses

Objective

To comply with safety regulations, licensees must demonstrate the acceptable thermal and mechanical performance of nuclear fuel during steady-state operation and anticipated transients and accidents.

The NRC maintains the FRAPCON and FRAPTRAN computer codes to reliably predict fuel rod thermal and mechanical behavior under steady-state and transient conditions, respectively. 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 steady-state and transient conditions. The NRC fuel behavior codes must be able to model current fuel designs deployed in the United States.

Research Approach

Early versions of FRAPCON and FRAPTRAN date back to the 1970s, and both codes have evolved to incorporate new modeling capabilities and new fuel and cladding materials to follow industry trends.

Currently, the NRC fuel behavior codes model uranium dioxide (UO2) pellets as well as mixed-oxide pellets (MOX), gadolinia (Gd2O3) doped pellets, and zirconium diboride (ZrB2) coated pellets (Integral Fuel Burnable Absorber–IFBA fuel). Moreover, new pressurized-water reactor (PWR) cladding alloy models were added to the code as these new alloys were introduced in the U.S. fleet of reactors. Examples include AREVA M5™ and Westinghouse ZIRLO™. Finally, the codes have been validated up to the current licensed U.S. burnup limit of 62 GWd/MTU peak rod average. The latest code versions FRAPCON-3.5 and FRAPTRAN-1.5 as well as their extensive validation are documented in NUREG/CR-7022 and NUREG/CR-7023. The predictive bias and sensitivity in the fuel performance codes are documented in NUREG/CR-7001.

Status

Ongoing development efforts aim to better integrate FRAPCON and FRAPTRAN within the NRC's suite of safety codes including TRACE, SNAP, and DAKOTA for sensitivity analyses for both FRAPCON and FRAPTRAN. In parallel, a source code modernization effort is underway, and the codes are gradually being adapted for the modeling of spent fuel behavior. The code benchmarking database also is continuously being expanded, and the material and failure models are constantly being adjusted to incorporate the latest available data.

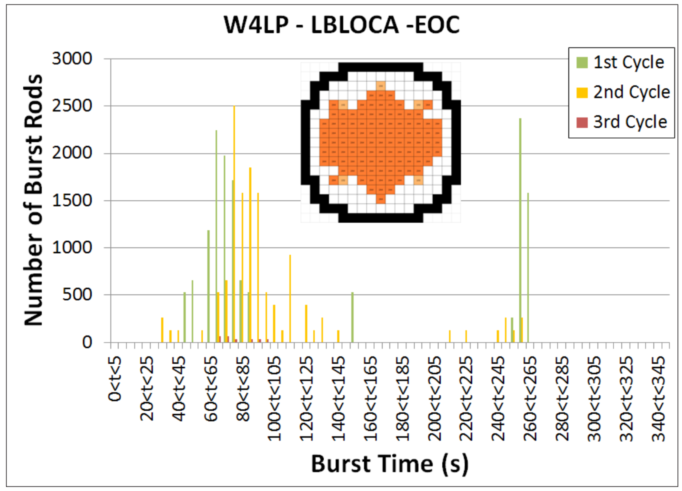

Figure 3.5 Number of ruptured rods versus time and location of ruptured rods for a large-break, loss-of-coolant accident (LBLOCA) at the end of cycle.

NRC regularly participates in code benchmarking exercises, such as the OECD RIA benchmarks phases 1 and 2, the SCIP-2 power ramp benchmarks, and the IAEA FUMEX-3 and FUMAC projects. Best-practice modeling methods and model improvements are continuously derived from these exercises. Since 2012, FRAPCON and FRAPTRAN are being used with TRACE boundary conditions to produce best-estimate core-wide LBLOCA fuel rod behavior predictions as shown in Figure 3.5. In addition, a recent modification of FRAPCON was used to predict the stress in spent fuel cladding for a period of 300 years of dry storage while taking into account gas generation and release as well as fuel pellet swelling during storage.

For More Information
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Spent Nuclear Fuel Burnup Credit

Objective

The purpose of this research is to develop a technical basis to support the allowance of full (fission product and actinides) burnup credit for spent fuel. Whereas the focus in on primarily on transportation and storage casks, it is intended for this research to ensure an integrated approach to criticality analysis among the various NRC offices and would therefore be applicable to spent fuel pool storage as well.

Research Approach

Background:

Spent nuclear fuel (SNF) refers to uranium-bearing fuel elements that have been used at commercial nuclear reactors and are no longer producing enough energy to sustain full-power reactor operation. The fission process stops 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. Moreover, the SNF storage or shipping system needs to ensure sub-criticality, thereby preventing criticality accidents.

Burnup Credit Methodology:

The approach relies on a two-step methodology:

1. Evaluation of available measured isotopic composition data to support isotopic validation. Under this activity, two-dimensional (SCALE/TRITON) depletion calculations are performed for comparison to the available measured data with the goals of developing a basis for isotopic validation, determining a representative bias and bias uncertainty for the SCALE/TRITON code, and determining the range of applicability associated with the bias and bias uncertainty. Much of the existing and recently available measured data has not been previously modeled, thus considerable effort is required in this activity to first model and then evaluate these data.
2. Evaluation of available critical experimental data to support criticality validation for spent boiling-water reactor (BWR) fuel. Under this activity, the sensitivity/uncertainty tools (TSUNAMI) in SCALE are used to evaluate relevant critical experiments and to identify those that are applicable for validation of spent fuel pool racks and dry cask storage and transportation designs. The evaluation considers experiments from the International Criticality Safety Benchmark Experiment Project Handbook as well as other proprietary and nonproprietary experiments, with the goals of developing a basis for criticality validation, determining a representative bias and bias uncertainty for the SCALE/KENO code, and determining the range of applicability associated with the bias and bias uncertainty.

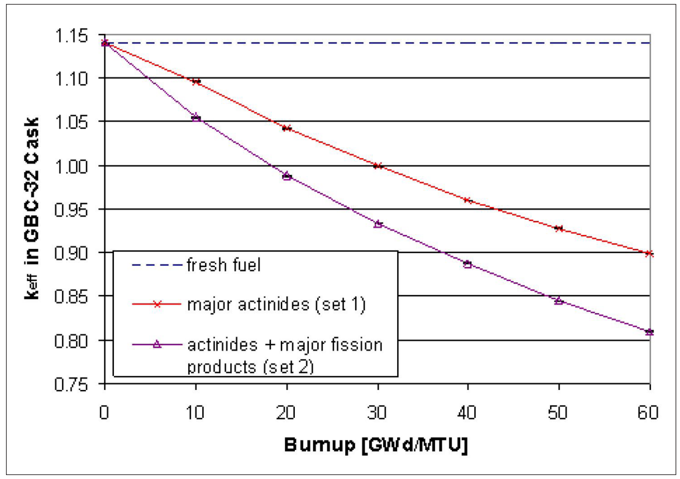

Figure 3.6 Comparison of typical reactivity decrements associated with burnup credit allowance.

Status

The NRC recently completed the work associated with pressurized-water reactor burnup credit. This work supported the released of Revision 3 of Interim Staff Guidance 8 (ISG-8) of the Office of Nuclear Material Safety and Safeguards (NMSS). The research has now shifted to the implementation of burnup credit for BWR spent nuclear fuel. A BWR burnup credit sensitivity study (NUREG/CR-7157 and NUREG/CR-7158) and the peak reactivity burnup credit technical basis (NUREG/CR-7194) have been recently completed. The focus is currently on the treatment of reactor coolant moderator density profiles, control blades usage, and axial burnup distributions. This will be followed by the validation of isotopic composition and criticality analysis calculations that will yield the technical basis for BWR burnup credit allowance.

For More Information
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Fuel Cooperative Research

Objective

The NRC engages in multiple international cooperative research programs related to nuclear fuel. These programs provide an opportunity for the agency to leverage resources to conduct complex research programs in collaboration with international counterparts. In the area of nuclear fuel research, these programs include the Halden Reactor Project (HRP) in Norway and the Studsvik Cladding Integrity Project (SCIP III) in Sweden. Both the HRP and SCIP III programs include participants from Europe, Japan, the United States, Russia, and Korea. The participants generally represent four categories—those who supply and manufacture the fuel, the power companies themselves, regulators, and laboratories. The NRC also is working actively with partners at the Nuclear Regulation Authority in Japan and the Institut de Radioprotection et de Sûreté Nucléaire in France on loss-of-coolant accident (LOCA) issues.

The NRC also interacts in various ways with the Department of Energy (DOE) on fuels related research programs such as the Used Fuel Disposition Campaign and the Advanced Fuel Campaign. The staff's interactions with DOE on these programs are typically oriented toward maintaining awareness of research developments.

Research Approach

The Halden boiling-water reactor, which currently operates at 18 to 20 megawatts, is fully dedicated to instrumented in-reactor testing of fuel and reactor materials. About 10-12 test rigs are under irradiation at any one time, and a similar number are either undergoing post irradiation examination or in preparation for starting irradiation. Test rigs are specially designed to obtain measurements of:

- Fuel thermal conductivity degradation and recovery as a function of burnup and temperature
- Fuel creep
- Cladding response to rod overpressure
- Fuel and cladding properties important in LOCA evaluation, including fuel dispersal
- Cladding creep
- Cladding corrosion

The NRC relies on fuel property data from Halden to validate its steady-state and transient fuel performance codes, including steady-state gas release and thermo-mechanical behavior, and fuel behavior under demanding operation conditions and accident scenarios.

The SCIP III project is focused on issues related to high-burnup fuel under LOCA conditions, in particular on fuel fragmentation, relocation, and dispersal. A large portion of the testing conducted within SCIP III uses an integral LOCA test device first built for the NRC's LOCA program, which ran a number of integral LOCA tests from 2010-2012 (see Figure 3.7). The SCIP III program will allow for greater understanding of the phenomena of fuel fragmentation, relocation, and dispersal through separate effects tests. The NRC relies on the tests performed through SCIP III to develop models and analysis methods to complete predictions of fuel dispersal under postulated LOCA conditions.

Status

The NRC remains actively engaged in both the HRP and SCIP III programs through periodic program review meetings. These meetings provide the staff with an opportunity to express emerging agency needs, collaborate with international counterparts regarding the analysis of research results, and maintain awareness of state-of-the art research in the area of nuclear fuel.

For More Information
Contact Michelle Bales, RES/DSA, at Michelle.Bales@nrc.gov.

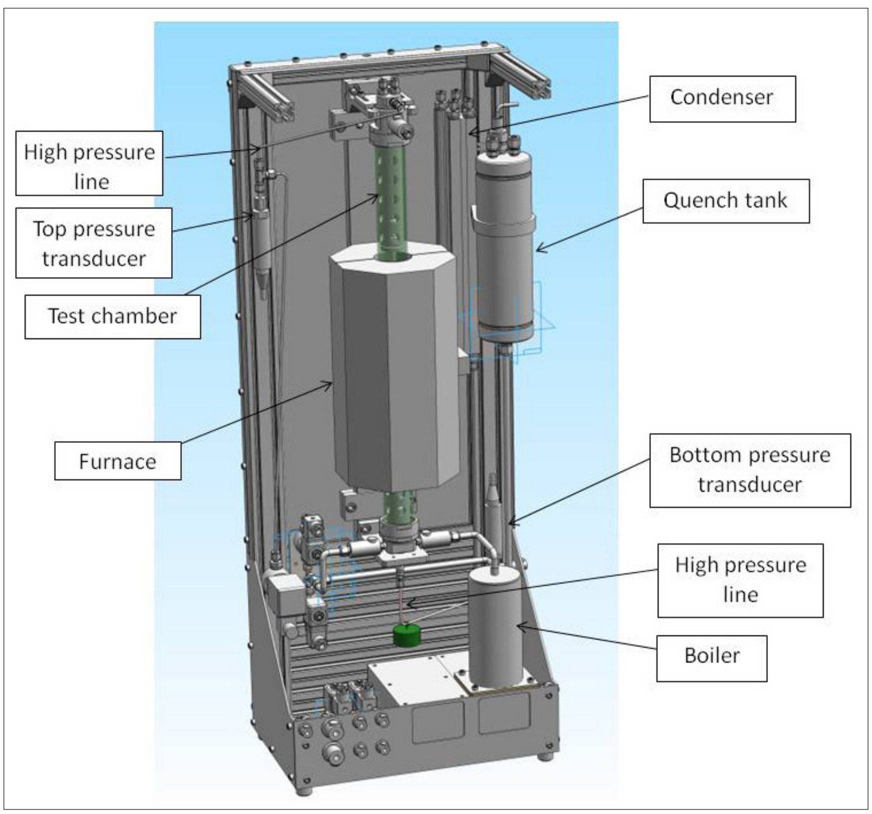

Figure 3.7 Integral LOCA test device. Test segments up to 12 inches long can be tested in this device.

Chapter 4: Severe Accidents and Accident Consequences Research

The Office of Nuclear Regulatory Research (RES) provides the tools and methods that the U.S. Nuclear Regulatory Commission (NRC) uses to evaluate and resolve potential safety issues and to perform risk-informed decisionmaking. The risk to the public from nuclear power generation arises if an accident progresses to the point at which fuel degradation occurs and radioactive materials are released into the environment. The use of postulated releases of radioactive materials is an integral part of defining the NRC's regulatory policy and practices. Title 10 of the Code of Federal Regulations (10 CFR) Part 100, "Reactor Site Criteria," requires licensees to postulate the occurrence of an accidental fission product release resulting from "substantial meltdown" of the core and to evaluate the potential radiological consequences of such a release under the assumption that the containment remains intact but leaks at its maximum allowable rate.

The NRC continues to maintain and develop its expertise in understanding severe accident phenomena and has developed computer codes for the analysis of severe accident progression, which provides quantitative predictive capabilities for simulating nuclear power plant response to severe accidents. The role of such expertise and analytical capability is wide-ranging in the regulatory environment, which includes a transition to a more risk-informed regulatory framework and the study of nuclear power plant vulnerabilities.

The NRC uses the MELCOR code for the analysis of postulated severe accident progression. The MELCOR code represents the current state-of-the-art in severe accident analysis, which has developed through the conduct of NRC and international research since the accident at Three Mile Island in 1979. MELCOR is a fully integrated computer code that is capable of modeling the progression of severe accidents in light-water reactors. MELCOR has been integrated into the NRC-developed Symbolic Nuclear Analysis Package (SNAP) graphical user interface that provides a user-friendly system for accident analysis. Using plant-specific or generic design data, MELCOR generates a source term representing the release of fission products from core degradation into the containment and out to the environment.

The NRC developed the MELCOR Accident Consequence Code System (MACCS) to evaluate offsite consequences from a hypothetical release of radioactive material into the atmosphere. The MACCS code models atmospheric transport and deposition, emergency response and protective actions, exposure pathways, health effects, and economic costs using the source term generated by MELCOR. MACCS is used to evaluate the consequences of severe radiological releases for 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, to assist in emergency planning, and to provide input to cost/benefit analyses.

MELCOR and MACCS are used for targeted regulatory research applications including, for example: (1) technical support for the NRC's full-scope site Level 3 probabilistic risk assessment; (2) State-of-the-Art Reactor Consequence Analyses; (3) the Spent Fuel Pool Study; (4) analysis of new and advanced reactors for design certification review, including small modular reactors; and (5) analysis of the event at Fukushima and support of Japan Lessons Learned and Near-Term Task Force recommendations to more effectively meet the NRC's mission to protect the health and safety of the public.

Future needs include developing insights into severe accident behavior of advanced reactor designs and extending the expertise gained on current reactor designs to unique phenomenological challenges present during severe accidents. Experimentation and the development of new or revised models are necessary to better the understanding of severe accident progression. One possible extension of MELCOR capabilities relates to analyses of localizations of highly contaminated coolant within containment. This could lead to the analysis of offsite impacts of the release of contaminated water to the environment based on plant-specific accident progressions. Cooperative international experimental programs such as the Phébus-Fission Product experiments, CSNI-STEM, and CSNI_BIP3 were performed to better understand containment iodine behavior. The Melt Coolability and Concrete Interaction research program investigates ex-vessel debris coolability mechanisms and provides insights and data for code upgrades.

In addition, the NRC coordinates the Cooperative Severe Accident Research Program (CSARP) that includes more than 20 member nations that focus on the analysis of severe accidents using the MELCOR and MACCS codes. CSARP includes MELCOR and MACCS user group meetings where participants share experience with the NRC codes, identify code errors, perform code assessments, and identify areas for code improvements, experiments, and model development.

Severe Accidents and the MELCOR Code

Objective

The objective of the research is to maintain the NRC staff's expertise on severe accident phenomenological behavior and to develop a computer code for analysis of nuclear power plants' response to severe accidents. The MELCOR code represents the current state-of-the-art in severe accident analysis and containment thermal-hydraulics.

Research Approach

The MELCOR code is a fully integrated, engineering-level computer code designed to model the progression of postulated accidents in light-water reactors and in non-reactor 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 flowpaths, heat and mass transfer to structures, gas combustion, and aerosol and vapor physics); (2) reactor-specific phenomena (i.e., decay heat generation, core degradation and relocation, ex-vessel phenomena, and engineering safety systems); and (3) support functions (i.e., thermodynamics, equations of state, material properties, data-handling utilities, and equation solvers). These packages model the major systems of a nuclear power plant and their associated interactions. A code modernization effort initiated in early 2000 resulted in conversion of the source code from Fortran 77 (MELCOR 1.8.6) to Fortran 95 (MELCOR 2.1). MELCOR 2.1 is currently the main computational tool for accident analysis, and the early versions of the code are no longer maintained. MELCOR can run under both Windows and Linux environments and has extensive capabilities for sensitivity and parametric analysis. SNAP is used for pre/post processing, visualization, and accident simulation. Code development meets the following criteria:

- Prediction of phenomena is in qualitative agreement with current understanding of physics and uncertainties are in quantitative agreement with experiments.
- Focus is on mechanistic models where feasible with adequate flexibility for parametric models.
- Code is portable, robust, and relatively fast running, and the code maintenance follows established Software Quality Assurance (SQA) standards.
- Availability of detailed code documentation (including user guide, model reference, and assessment).
- Software quality assurance (SQA) is an integral part of the MELCOR development process. The SQA program is adapted from two internationally recognized standards—CMMI and ISO 9001—that provide elements of traceability, repeatability, visibility, accountability, roles and responsibilities, and objective evaluation. These standards encompass the SQA program outlined by the NRC in NUREG/BR-0167.

Status

MELCOR has been under continuous development by the NRC and Sandia National Laboratories. Current activities include development and implementation of new and improved models to predict the severe accident behavior of various reactor and spent fuel pool designs and to reduce modeling uncertainties. Examples of recent model additions include a new ex-vessel core debris cooling model to better represent water ingression and melt eruption, and more mechanistic treatment of the engineered safety features such as fan coolers and heat exchangers. The MELCOR development team is in the process of improving the code by implementing several models including an external core catcher, more mechanistic treatment of core debris spreading and aerosol resuspension as well as a multiple fuel rod types for spent fuel pool heat transfer analysis. Plans are underway to revise the code to improve stability and efficiency of explicit coupling and time integration. The improvements in the code numeric involve casting all implicit equations in residual form and enable use of modern solver libraries. Code maintenance and user support will continue as more users are becoming involved in the code.

For More Information
Contact Hossein Esmaili, RES/DSA, at Hossein.Esmaili@nrc.gov.

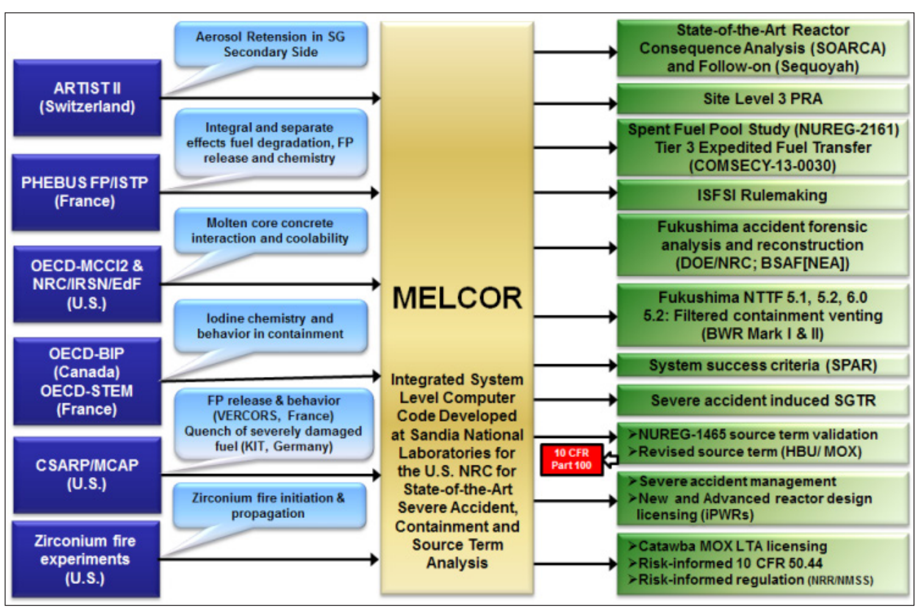

Figure 4.1 Severe accident experimental programs and MELCOR regulatory applications.

MELCOR Accident Consequence Code System (MACCS)

Objectives

The U.S. Nuclear Regulatory Commission (NRC) developed MELCOR Accident Consequence Code System (MACCS) to evaluate offsite consequences from a hypothetical release of radioactive material into the atmosphere. MACCS/WinMACCS is used to evaluate severe accident consequences as part of the environmental reports and environmental impact statements for early site permits. These analyses support plant-specific evaluation of severe accident mitigation alternatives required as part of the environmental assessment for license renewal to assist in emergency planning and to provide input to cost/benefit analysis.

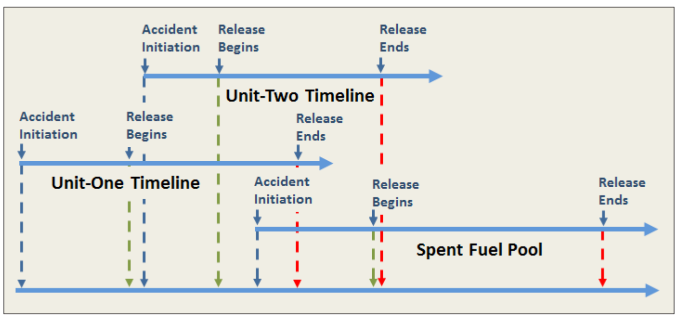

Figure 4.2 Example source release timeline for multiple releases at a single site with multiple units.

Research Approach

The MACCS code was developed to evaluate the impacts of severe accidents at nuclear power plants on the surrounding public. It is an integrated engineering level code designed to model severe accident consequences from a source term resulting from an accident progression scenario. The principal phenomena considered are atmospheric transport and deposition under time-variant meteorology, short- and long-term mitigative actions and exposure pathways, deterministic and stochastic health effects, and economic costs. MACCS Version 3.10 incorporates the following improvements:

- More cohorts for evacuation (up to 20).
- More compass directions (up to 64) and plume segments (up to 500).
- A long-range lateral plume spread model and an improved Briggs plume rise model.
- A plume meander based on Regulatory Guide 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants.”

In addition, new capabilities to the code include the ability to model releases from two or more separate inventories (i.e., modeling a site that contains multiple units and the ability to model keyhole evacuations, which can account for wind shifts and fluctuations in the release path). The SECPOP databases used by MACCS also were updated to include the most recent census data (2010) and economic information. Version 3.10 of the code has been released with the graphical user interface WinMACCS. The three most important modeling features implemented in WinMACCS are (1) the ability to easily evaluate the impact of parameter uncertainty, (2) the ability to manipulate input parameters for network evacuation modeling, and (3) the ability to model alternative dose-response relationships for latent cancer fatality evaluations.

Status

Work is ongoing to update the MACCS code based on current technology. For uncertainty analyses, capabilities are being implemented to sample dose conversion factor values and to distribute numerous MACCS runs into a computer network cluster; this effort will include post processing of the results.

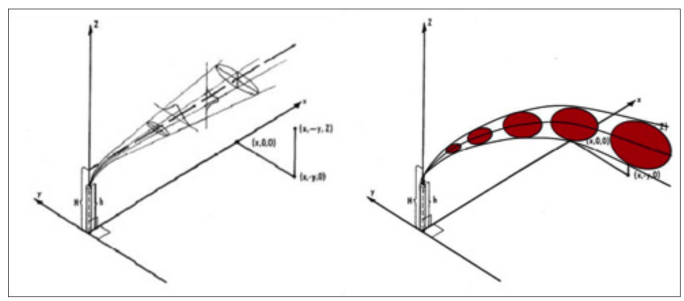

Figure 4.3 MACCS includes the Gaussian plume segment model (left) for atmospheric transport, but efforts are underway to introduce the Gaussian puff model (right) and Lagrangian particle tracking.

The new economic model under development is based on the existing Regional Economic Accounting Tool (REAcct) that Sandia National Laboratory developed for the U.S. Department of Homeland Security and will allow the ability to estimate economic impacts using input-output modeling techniques. An external peer review for the new economic model is underway.

Alternate atmospheric transport models also will be introduced to MACCS by integration of the HYSPLIT code to allow the use of the Gaussian puff model and Lagrangian particle tracking in addition to the Gaussian plume segment model already available. These models handle higher dimensional wind fields, are used when steady-state assumptions along a straight line, as assumed by the Gaussian plume segment model, are not appropriate.

For More Information
Contact Patricia Santiago, RES/DSA, at Patricia.Santiago@nrc.gov.

State-of-the-Art Reactor Consequence Analyses

Objectives

The NRC initiated the State-of-the-Art Reactor Consequence Analyses (SOARCA) project to develop best-estimates of the offsite radiological health consequences for potential severe reactor accidents. SOARCA aimed to assess the benefits of the mitigation measures required by Title 10 of the Code of Federal Regulations (10 CFR) 50.54(hh) that were put in place after the terrorist attacks of September 11, 2001, for responding to fires and explosions in other accident scenarios. Additional studies were performed to determine the benefits of flex equipment installed at nuclear power plants (NPPs) following the events that took place at the Fukushima-Daiichi NPP.

Research Approach

SOARCA modeled selected severe accident scenarios in a representative pressurized-water reactor with a large dry containment (Surry) and boiling-water reactor with Mark I containment (Peach Bottom). Selected scenarios were run twice, first assuming the event proceeds without mitigation measures required by 10 CFR 50.54(hh) followed by a case where mitigation strategies were successful. This method provided an indication of the benefit of the mitigation strategy.

In addition, the NRC conducted an uncertainty analysis (UA) for the SOARCA study. The goals of this UA are (1) to develop insights into the overall sensitivity of SOARCA results to uncertainty in inputs, (2) to identify the most influential input parameters for releases and consequences, (3) and to demonstrate a UA methodology that could be used in future source term, consequence, and site Level 3 probabilistic risk assessment studies. The uncertainty analysis involved perturbing numerous uncertain model parameters based on a monte carlo sampling of parameter probability distributions. A number of experts in this area were consulted in order to determine the most important uncertain parameters. Approximately 900 calculations were performed and statistical regressions were utilized to quantify uncertainty and to determine which parameters had the greatest influence on the results.

A model of a nuclear power plant with an ice condenser containment (Sequoyah Nuclear Power Station) is under development to apply the lessons learned from the earlier SOARCA analyses. Accident scenarios are chosen to challenge this style of containment, which is smaller than the large dry type. Also, the effects of flex equipment, which are required in response to the events that took place at Fukushima-Daiichi, are modeled to characterize their benefits.

Status

The first part of the SOARCA project is documented in a series of NUREG reports; NUREG-1935 and NUREG/CR-7110 Volumes 1 and 2, Revision 1. Key results of the analysis include:

- Operators can prevent core melting or can delay or reduce radioactive releases to the environment when successful in using available onsite equipment.
- Modeled accidents progress more slowly and release smaller amounts of radioactivity than calculated in previous studies.
- Longer term cancer fatality risks for the accident scenarios analyzed are millions of times lower than the general U.S. cancer fatality risk from all causes.

An uncertainty analysis has been performed on for the Peach Bottom SOARCA analysis. The results of this uncertainty analysis corroborated the SOARCA project conclusions in regards to delayed radionuclide releases as compared to earlier studies. The uncertainty analysis indicated that parameters describing the behavior of the safety relief valve and dry deposition velocity of contaminants are the most important uncertain model inputs for the chosen scenario. The results of the Peach Bottom Uncertainty Analysis are documented in NUREG/CR-7155 which is to be published. An Uncertainty Analysis is also underway for the Surry SOARCA analysis.

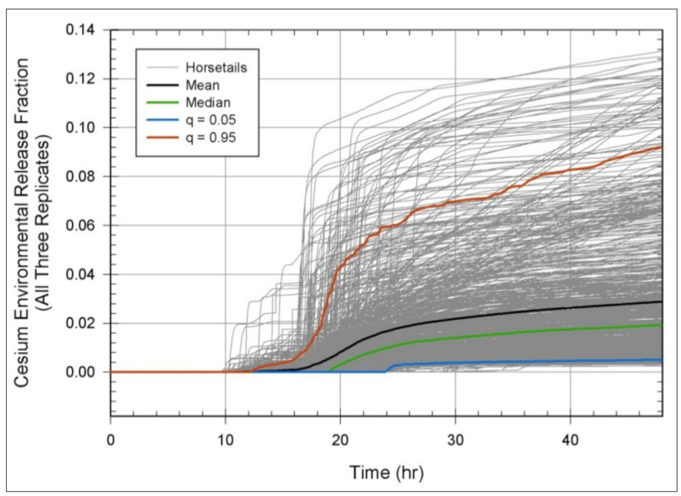

Figure 4.4 Horsetail plots demonstrate the variability of calculational results due to parameter perturbations. The mean, 5 percent, and 95 percent percentiles characterize the uncertainty.

The MELCOR model of the Sequoyah Nuclear Power Station is being updated to the current state of the plant and analyses are underway to quantify accident progression results.

For More Information
Contact Patricia Santiago, RES/DSA, at Patricia.Santiago@nrc.gov.

MELCOR Accident Simulation Using SNAP (MASS)

Objective

The simulation models should provide the users with the capability to define accident sequences, alter the system availabilities, and provide a visual progression of the accident using MELCOR for the prediction of the accident outcome and the SNAP animation capabilities.

Research Approach

The design concept requires minimal user training in both MELCOR and SNAP. The objective is to provide users with an easy-to-use tool to analyze accident scenarios. The end user controls the type of accident (e.g., size and location of a loss-of-coolant accident) and the availability of plant safety systems and any operator actions. For containment design-basis analysis, the mass and energy and fission product sources into the containment can be provided as an external source. The end user can then view the results and perform sensitivity calculations.

One of the advantages of the visualization is to provide an overview of the accident progression in terms of interpretation of results, input model checking, and user training.

Because of the desire to make MELCOR more user friendly through the SNAP graphical user interface, an additional program was added to the SNAP suite—the SNAP-KIOSK. The SNAP-KIOSK allows the normal SNAP model editing features to be disabled while still allowing users to interact with the models and to control the simulation. A socket interface and new MELCOR control functions also were developed as part of the project for MELCOR and SNAP to more effectively communicate. In addition, several MELCOR-specific SNAP modules (e.g., dynamic core degradation and hydrogen flammability diagrams) were developed.

Status

The accident simulation models for new reactor designs, including the U.S. Evolutionary Power Reactor, Advanced Boiling-Water Reactor, U.S. Advanced Pressurized-Water Reactor, Advanced Passive 1000 Megawatt, and Economic Simplified Boiling-Water Reactor have been completed. The models run in severe accident and design-basis accident modes (containment peak pressure and source term) and provide a convenient display system for the user to define an accident sequence by introducing system malfunctions (e.g., loss-of-coolant accident) and controls (e.g., emergency core cooling system) to mitigate the consequences of the accident. In addition, the user can visually see the progression of an accident (e.g., core heatup and degradation) as the calculation is progressing. Similar masks also have been developed for the existing reactors (a pressurized-water reactor and a boiling-water reactor) for user training and accident analysis. Future work will focus on developing models for other reactor designs.

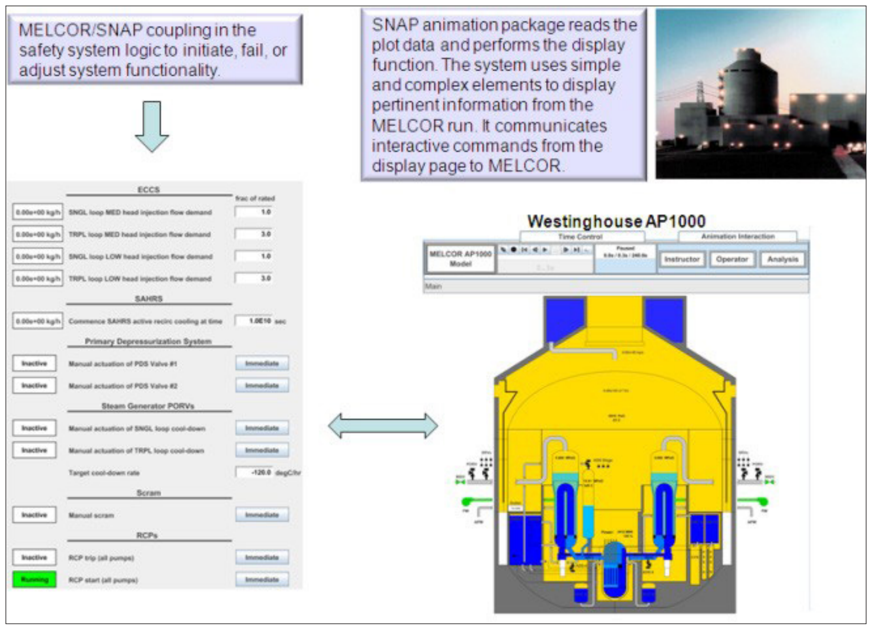

Figure 4.5 MASS user interface for AP1000.

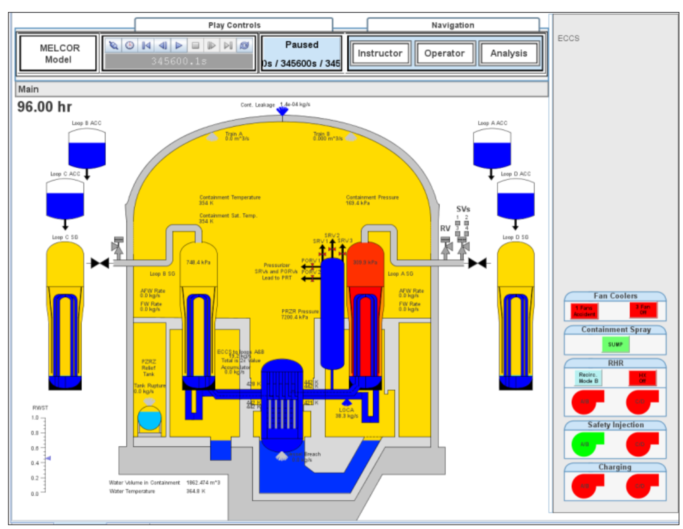

Figure 4.6 Accident progression for a PWR.

For More Information

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Severe Accident Progression in Advanced Nuclear Reactors

Objectives

The NRC has received, or is expecting to receive, Design Certification Documents (DCD) for advanced reactor designs in the near future. These new designs incorporate safety features that do not exist in the present fleet of nuclear reactors. Although these designs incorporate features that minimize the possibility of core damage events, design basis accidents and beyond design basis accidents must be analyzed for design certification.

The NRC has recently received the DCD for the Advanced Power Reactor 1400 (APR-1400). The APR-1400, designed by Korea Hydro and Nuclear Power Co. Ltd. (KHNP), is a two-loop pressurized-water reactor (PWR). Confirmatory analyses are to be performed by the NRC to verify the statements of fact set forth in the DCD in regards to design and beyond design basis accidents.

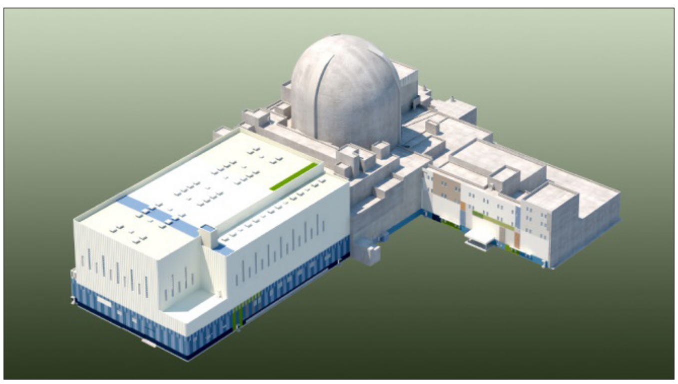

Figure 4.7 APR-1400 Reactor Building.

Small Modular Reactors (SMRs) are advanced reactor concept designs that utilize the proven technologies of traditional large PWRs and incorporate enhanced passive safety features. These designs integrate the steam generator into the reactor pressure vessel, eliminating the possibility of traditional large break loss-of-coolant accident (LOCA) events because the entire primary coolant loop is contained within the pressure vessel. Although the frequency of core damage events in SMRs is expected to be significantly lower than a traditional PWR plant, severe accidents cannot be totally eliminated from consideration and must be analyzed.

Research Approach

The NRC intends to use the NRC-sponsored MELCOR computer code for confirmatory analyses of design basis and beyond design basis severe accidents. A MELCOR model of the APR-1400 was developed, and the results were compared to the licensee's when possible.

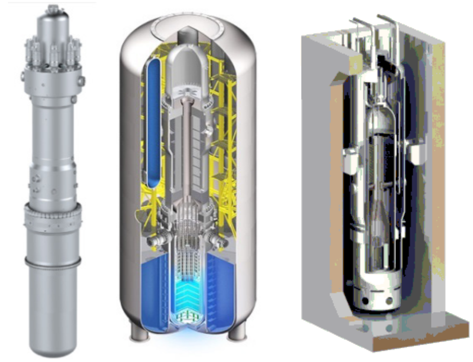

Figure 4.8 Babcock & Wilcox mPower (left), Westinghouse W-SMR (center), and NuScale (right) are three SMR designs that will be modeled in MELCOR to analyze severe accident progression.

The unique design of SMRs, as compared to conventional PWRs, may introduce phenomenological challenges during severe accidents that may require experimentation or the development of new or revised models into MELCOR. The objective of this research is to identify thermal-hydraulic, melt progression, and fission product release and transport phenomena that are relevant to modeling of severe accidents in SMRs and to provide an assessment of the applicability of the MELCOR computer code to those analyses.

Status

Phenomena Identification and Ranking Tables (PIRTs) were developed to identify important phenomena and processes that need to be considered for the analysis of containment system design basis and beyond design basis accidents in the APR-1400. This analysis is used to determine the applicability of MELCOR to perform confirmatory analyses. A MELCOR model of the APR-1400 was developed, and preliminary simulations of steady-state, design basis, and beyond design basis accidents were performed to assess the performance of the MELCOR model. The events under consideration successfully completed, and the model has been updated to reflect the applicant's most recent submittal. Further pre-confirmatory calculations will be performed to ensure the model's capabilities.

MELCOR models have been developed for the mPower, NuScale, and Westinghouse SMRs and demonstrate that the code can be readily applied to these type of reactors. No modifications of the MELCOR code were required to model the performance. Workarounds were required to model some of the new design features, such as the cooling of core debris produced by the unlikely event of a severe accident. Design basis and beyond design basis events were simulated and successfully completed. Further updates would be required for confirmatory analysis when the applicants submit design certification documentation.

For More Information
Contact Patricia Santiago, RES/DSA, at Patricia.Santiago@nrc.gov.

Source Term Analysis

Objective

The objective of this research is to extend the source term described in NUREG-1465 ("Accident Source Terms for Light-Water Nuclear Power Plants," issued February 1995) for both light-water reactors with conventional reactor fuel burnup to high burnups (55 to 75 gigawatt days per ton) and to mixed-oxide (MOX) fuel made with weapons-grade plutonium dioxide.

Approach

The use of postulated accidental releases of radioactive materials is an integral part of defining the U.S. Nuclear Regulatory Commission's (NRC's) regulatory policy and practices. The regulations at Title 10 of the Code of Federal Regulations (10 CFR) Part 100, "Reactor Site Criteria," require licensees to postulate, for licensing purposes, the occurrence of an accidental fission product release resulting from "substantial meltdown" of the core into the containment. The regulations also require licensees to evaluate the potential radiological consequences of such a release under the assumption that the containment remains intact but leaks at its maximum allowable leak rate.

Radioactive material escaping from the containment is often referred to as the "radiological release to the environment." The radiological release is obtained from the containment leak rate and knowledge of the airborne radioactive inventory in the containment atmosphere. The radioactive inventory within containment is referred to as the "in-containment accident source term."

Regulatory source term ("release to the environment" and "in containment") provides a prescription of fission product release magnitude and timings that represent a broad range of accident scenarios. The release of radioactive material to the environment during a hypothetical reactor accident is an input to models of radionuclide dispersal and accident consequences. It drives measures taken for emergency preparedness and accident response. It is a crucial element of Level III probabilistic risk assessments and is an important consideration in the cost-benefit analyses of safety improvements that go beyond regulatory requirements to provide adequate protection of public health and safety. The "in-containment" source term is used in the analysis of a defense-in-depth measure to assess the adequacy of reactor containments and engineered safety systems. This source term also figures into the environmental qualification of equipment within the containment that must function following a design-basis accident.

Previously, operating power reactors in the United States were designed and licensed based on the source term described in Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactors," issued by the U.S. Atomic Energy Commission in 1962. Since then, sufficient new data and calculation tools (source term codes package - STCP) were developed to define a new source term that is more realistic in modeling of fission products release from fuel and transport to the containment (Figure 4.9). NUREG 1465 (known as the alternative source term-AST) delineated a new source term for regulatory analysis. Since the development of AST, additional fission products release tests for high-burnup pressurized-water reactor (PWR) and boiling-water reactor (BWR) UO2 fuel and PWR mixed-oxide (MOX) fuel had been performed. The state-of-the-art integrated system level analysis MELCOR code (replacing STCP) was developed and validated against experiments involving high-burnup and MOX fuels (VERDON and VERCORS).

Status

MELCOR analysis has been completed to synthesize a high-burnup source term for both light-water reactors with conventional reactor fuel burnup to high burnups (55 to 75 gigawatt days per ton) and to MOX fuel made with weapons-grade plutonium dioxide. Review of the results had been performed, and the comments and additional analysis have been completed. NRC is in the process of completing the documentation.

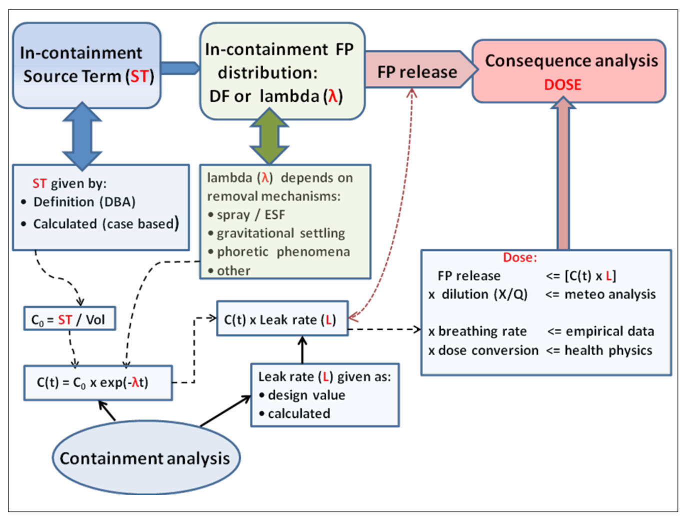

Figure 4.9 Use of source term and relation to other factors in dose calculations.

For More Information
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Severe Accident Waste Water and Consequences

Objectives

This project will provide information and analysis on potential causes, prevention, mitigation, and safety significance of the accumulation or loss of control of large volumes of highly contaminated water generated during severe accidents.

Research Approach

This project consists of several tasks that are discussed below.

Containment Failure Mechanisms.

Earlier studies have provided a good understanding of the performance of containments such as those used in U.S. nuclear power plants under severe (beyond design-basis) accident conditions. It is clear from those studies that margins between ultimate pressure capacity and design pressure capacity of containment structures are about a factor of 2 to 5. This task will include an examination of the evidence from the Fukushima accident that relates to the release of contaminated water from containment. It will review and evaluate historical analyses and evaluations regarding failure modes for boiling-water reactor Mark I containments. A review also will be made of operating experience, in-service inspections, and inspection records related to license renewal for incidents of corrosion and leaks that may indicate vulnerabilities that have implications for potential release locations or processes.

Models of Severe Accidents: In-plant Consequences of the Aqueous Pathway.

The loss of control, accumulation, and radionuclide content of highly contaminated water is typically not considered in severe reactor accident simulations. Staff will determine the best approach for modeling: failure of containment leading to loss of water, the aqueous source term, and fate of highly contaminated water (determining frequency, accident progression sequences, estimating aqueous source terms, and estimating in-plant consequences). The product will be a report that includes brief descriptions of the available codes, their current capabilities, and the pros and cons of incorporating the modeling of containment failure, aqueous source terms, and the fate of highly contaminated water in MELCOR, including how that could be accomplished.

Options for Preventing and Managing Highly Contaminated Water Releases on Site.

If findings from the tasks above conclude that safety significant issues are present, RES staff in collaboration with NRR will make recommendations pertaining to managing highly contaminated waste water during and after severe accidents.

Analysis of Highly Contaminated Water Releases Offsite.

In this task, staff will assess potential offsite consequences of the loss of control of highly contaminated water in a severe accident in which the contaminated water flows to a body of water. This task will provide analysis from models and accident experience at Fukushima on potential offsite waterborne impacts from severe reactor accidents. Simplified scenarios will be developed by applying an aqueous source term based on aqueous phase releases from Fukushima in models of a variety of postulated bodies of water (river, lake, etc.). The model will be used to estimate the spatial and temporal distribution of radionuclides, to calculate dose distributions, and to evaluate the resultant public health and safety implications. The capacity of readily available transport/dose codes to model offsite aqueous releases will be assessed.

Status

The analysis of offsite releases is nearing completion, and a peer review will be conducted. The other portions of this project are just being started.

For More Information
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Containment Iodine Behavior Research

Objective

The objective of this research is to develop mechanistic models of the phenomena that govern the containment iodine behavior observed in the Phébus-FP experiments to scale this observed behavior to operating power reactors.

Approach

The integral Phébus-Fission Product (FP) experiments provided an opportunity to test code predictions of containment iodine behavior. Previously conducted pure water benchtop experiments suggested that preventing pressurized-water reactor (PWR) sump water from becoming acidic is necessary and sufficient to prevent significant gaseous iodine from evolving in a reactor containment following an accident involving core damage. However, the observations of the Phébus-FP experiments, the complexity of which more closely matches prototypic severe accident behavior, show that this may not necessarily be the case for power reactors.

Iodine is one of the major contributors to dose in analyses of postulated reactor accidents and, therefore, merits more attention than less dose-important elements do. Because iodine's dose contribution results from gaseous and particulate fission products contained in gas leaking from the reactor and containment, reducing the amount of airborne fission products reduces the contribution to dose. To minimize the iodine dose, PWR sumps are buffered to keep the sump water alkaline, thus preventing the iodine that reaches the sump from converting to volatile forms that can then be released to the containment atmosphere.

The results of the Phébus-FP tests indicate that controlling the sump pH may not significantly affect the development of a gaseous iodine concentration in the reactor containment in the immediate aftermath of an accident involving core degradation. Two aspects of the Phébus-FP experiments that influenced this iodine behavior were the presence of condensing surfaces and the presence of additional materials in the sump. The buffer in the sump does not affect the liquid films that develop on surfaces; therefore, these films do not remain alkaline. Consequently, the buffer in the sump does not prevent the iodine in these films from converting to volatile forms that may subsequently be released to the containment atmosphere.

The Office of Nuclear Regulatory Research (RES) is using the following approach to resolve the iodine issue:

- Test hypotheses against experiments.
- Develop models and validate models with further experiments.
- Simulate the Phébus-FP experiment.
- Simulate power plants.
- Evaluate sensitivities and uncertainty.
- Conduct peer review models and analyses.
- Make recommendations related to gaseous iodine behavior.

The approach for developing models to scale the iodine behavior of the Phébus-FP experiments has been to systematically test various working hypotheses that describe the persistent gaseous iodine behavior. For a steady-state concentration of gaseous iodine to exist, sources of gaseous iodine must balance the sinks of gaseous iodine. The experimental work and modeling is directed towards identifying and characterizing the sources and sinks of gaseous iodine. Based on observations of the Phébus-FP experiments, the results of additional separate-effects experiments, and analyses, the source of the persistent gaseous iodine in the Phébus-FP experiments is believed to be the containment surfaces upon which iodine deposited. Figure 4.10 shows a schematic of the hypothesized mechanism for this source. The general mechanism can be described as follows:

- Particulate and gaseous iodine is released to the containment from the reactor coolant system.
- Particles deposit and gases adsorb on surfaces in the containment.
- Particles decompose and gases absorb into paint.
- Irradiation releases iodine vapors.
- Vapors react in the atmosphere to form iodine oxide particles.
- The particles and vapors can redeposit on paint, thus continuing the cycle.

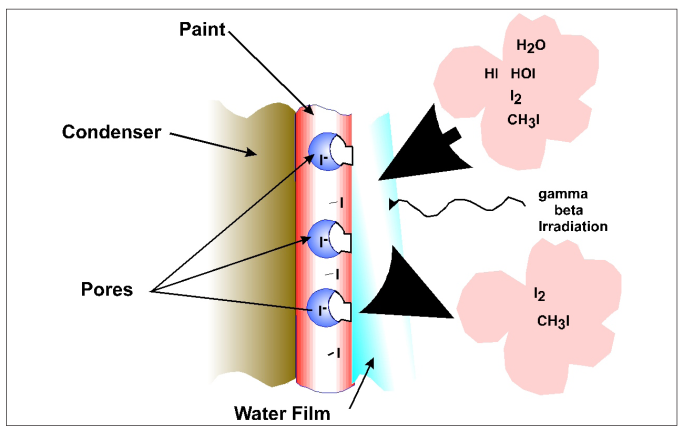

Figure 4.10 Hypothesized mechanism for gaseous iodine source in the Phébus-FP tests.

To obtain data to test hypotheses for gaseous behavior and to validate the developed models, RES is participating in the following international separate effects research programs:

- Behavior of Iodine Project (BIP).
- Source Term Evaluation and Mitigation (STEM).

The Organisation for Economic Co-Operation and Development (OECD) organized both programs. The Atomic Energy of Canada, Ltd (AECL) is conducting BIP and Institut de Radioprotection et de Surete Nucleaire (IRSN) is conducting STEM.

Figure 4.11 BIP irradiation vessel with sample coupons.

Status

The BIP2 and STEM projects have recently finished.

The BIP and STEM experiments provide information useful for the containment chemistry modeling. BIP2 identified chemical compounds within paint that contribute to iodine adsorption and release upon irradiation. Only polyamides (nylon) absorbed iodine in a similar manner to paint suggesting that the nylon content is responsible for much of the iodine adsorption on paint. It was speculated that the greater adsorption rate on paint results from the increased porosity in paint which can provides easier access to iodine.

A few competition experiments were conducted near the end of BIP2 by exposing the paint to chlorine gas prior to iodine loading. Chlorine could potentially react at the same location in paint as iodine thereby limiting iodine adsorption. No appreciable change in the iodine adsorption rate or the organic iodine generation rate was observed although this could potentially change for prototypic conditions.

Paint in the experimental programs was either fresh, heated in an oven for a specified time at temperature to remove solvents, or stored in a desk for several years. Paint treated in these ways does not necessarily correspond to those used in NPPs.

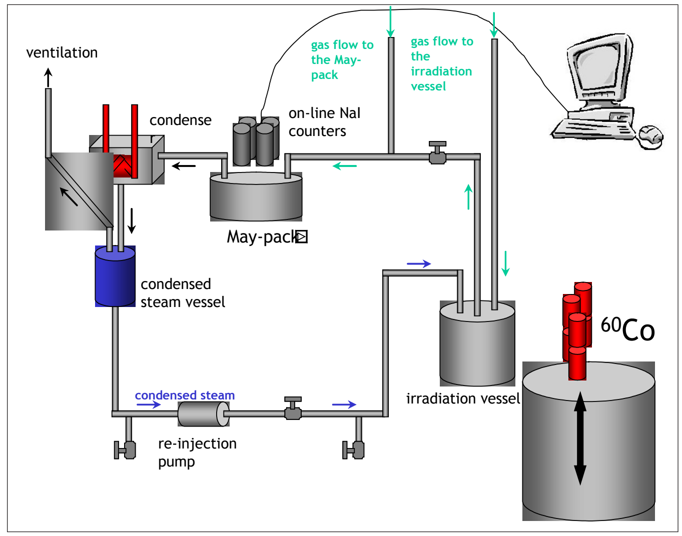

Figure 4.12 The EPICUR experimental setup (used in the OECE STEM project).

OECD is planning to conduct additional experimental study on prototypic paint used in NPPs under the OECD BIP3 and STEM2 combined research.

For More Information
Contact Michael Salay, RES/DSA, at Michael.Salay@nrc.gov.

Cooperative Severe Accident Research Program (CSARP)

Objective

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. Cooperative Severe Accident Research Program (CSARP) technical review meetings provide a forum to exchange technical information on severe accident research to gain insight into regulatory and potential safety issues and to improve modeling capabilities.

Research Approach

CSARP is an international program on severe accident phenomenological research and code development activities organized by the NRC since 1988. Through CSARP, NRC facilitates the exchange of severe accident research among NRC and participating countries. Participating countries provide contributions (phenomenological research, data, codes, and code assessment) to the NRC, thereby lessening the resources required by the NRC to maintain a core competency and staff expertise in severe accident analysis. Currently, 27 foreign countries are participating in CSARP. For NRC, the current thrust is on the development, assessment, and application of MELCOR. Through CSARP, NRC has access to a large body of international severe accident research.

Status

The NRC hosts a CSARP technical review meeting once a year (in September) to exchange progress in severe accident research and to report code development and assessment status. Topics that are discussed at the meeting include recent advances in severe accident research programs such as:

- Latest information and analysis from the Fukushima accident and status of code modeling.
- Molten Core Concrete Interaction (MCCI) Program, Organization for Economic Co-operation and Development (OECD) and Argonne National Laboratory. This project consists of separate-effects experiments to further address the ex-vessel debris coolability issue.
- Status of fuel-coolant interaction experiments and modeling.
- Overview of the severe accident research at various international organizations.
- Investigation of zirconium fire in spent fuel assemblies under a postulated complete loss-of-coolant accident.
- Behavior of Iodine Project (BIP), Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI). This project involves experimental investigations of iodine behavior in containment during conditions following a severe accident for computer code model development and validation.
- Phébus-Fission Products (Phébus-FP), VERCORS (a French test program), and follow-on program (Phébus-Source Term Separate Effects Test Project [STSET]), French Institute for Radiological Protection and Nuclear Safety (IRSN). This collaboration investigates fission product releases and degradation of uranium dioxide fuel, including burnup greater than 40 gigawatt days per metric ton. It also investigates mixed-oxide fuel under severe accident conditions and the effects of air ingress on core degradation and fission product release. The results are used to validate the NUREG-1465 source term and MELCOR code.
- The QUENCH experimental program at Karlsruhe Institute of Technology to investigate overheated fuel.

The MELCOR Code Assessment Program (MCAP) is an annual technical review meeting that focuses on the MELCOR code development and assessment and provides a forum for the presentation and discussion of the user experience, in particular (1) assessment using integral and separate-effect tests, (2) model development efforts, and (3) code application for plant safety studies, including probabilistic risk analysis. MCAP follows the CSARP meeting so that code users can also benefit from the latest severe accident research. Currently, two other MELCOR-related technical meetings are held (i.e., the European MELCOR User Group [EMUG] and the Asian MELCOR User Group [AMUG] meetings). These meetings provide an opportunity for more code users to interact with the code development team.

The first EMUG meeting was held in December 2008 in Switzerland. This group was founded to facilitate collective discussion and exchange of experience between European MELCOR users and the U.S. NRC and Sandia National Laboratories and to support the training of new MELCOR users. The host organization is among the European MELCOR community and from a country that is a member of CSARP. The 7th EMUG was held in Belgium in March 2015. The first AMUG meeting was organized in South Korea in October 2014 in consultation with other Asian CSARP members (Japan, China, and Taiwan) to introduce the latest version of MELCOR. The next meeting is tentatively is scheduled to be held in Japan in 2015.

For More Information
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Severe Accident Cooperative Research

Objective

The objective of domestic and international cooperative severe accident phenomenological research is to develop an improved understanding of those phenomena that are important to reactor safety and where knowledge gaps exist, and to reduce residual uncertainties through a combination of experimental and analytical research activities.

Research Approach

The research approach consists of (1) identification of knowledge gaps by taking a stock of the current knowledge base, (2) development of a comprehensive experimental program plan to address the gaps, (3) systematic implementation of the plan to generate needed information, and (4) development of analytical tools to extrapolate experimental data for reactor safety applications.

Over the last three decades or so, this approach proved useful in addressing and resolving a number of severe accident issues either deterministically or from a risk perspective. However, two severe accident issues in which residual uncertainties remained somewhat large are (1) ex-vessel melt coolability and core-concrete interaction (MCCI) and (2) ex-vessel steam explosion. After the Fukushima Daiichi accidents in March 2011, the MCCI issue received further attention from the international research community. Two other issues—hydrogen management and spent fuel pool—also received renewed attention. Other phenomenological issues that received attention in light of Fukushima include in-vessel melt progression behavior and fission product behavior in the containment.

The cost of experimental research investigating severe accident phenomena—in particular, experiments involving prototypic core material at large scale—has become prohibitively expensive for any single organization to carry out. Therefore, in recent years, an increasing effort has been made to participate in international cooperative research programs using one-of-a-kind facilities in member countries.

The MCCI experimental facility at the Argonne National Laboratory is one such facility where prototypic MCCI experiments at large scales were carried out in the last two decades under the Organisation for Economic Co-operation and Development (OECD)-MCCI program.

The TROI facility at the Korea Atomic Energy Research Institute and the KROTOS facility at the Commissariat l'Energie Atomique aux Alternatives are two other facilities where steam explosion experiments with prototypic materials were carried out in the past under the OECD-SERENA program.

The hydrogen stratification issue had been looked at under the OECD-THAI program in the past at the THAI facility in Germany. Retention of fission products in steam generator tubes was investigated in the international ARTIST program at the PSI facility in Switzerland. Limited experimental work on spent fuel pool fire risk was carried out under an OECD program at Sandia National Laboratories. Finally, the PHEBUS experimental program at the IRSN PHEBUS facility in France has been investigating fission products release and transport behavior in containment.

Concurrently, analytical work is performed to supplement the experimental activities, and such work involves analysis of experimental data and development of phenomenological models. Much of this work is done under the auspices of the OECD Working Group on Analysis and Management of Accidents (WGAMA).

Status

The OECD-MCCI experiments produced a database of information on various coolability mechanisms, and this information is being used to develop improved coolability models for incorporation into severe accident analysis codes. The new information also reduced residual uncertainties. The Fukushima event, however, pointed to the need for additional data that are representative of more prototypic plant design and operating conditions.

Likewise, the OECD-SERENA program produced a database of information on prototypic melt steam explosion potential, and this information is being used to develop new models and to improve existing models. Again, the Fukushima event brought to the forefront the potential for stratified steam explosion contingent on certain severe accident management actions.

In other areas (e.g., hydrogen risk management, fission products scrubbing, etc.), there is a renewed interest to perform additional research to strengthen the technical bases for regulatory actions. The recent OECD initiative SAREF (Safety Research post Fukushima) is expected to provide insights into future needs and opportunities in severe accidents.

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Chapter 5: Radiation and Environmental Protection Research

Overview

The U.S. Nuclear Regulatory Commission's (NRC's) Radiation Protection Research Program and the Environmental Transport Research Program are two agency-wide resources that provide technical support in the areas of radiation and environmental protection. Both programs are conducted by the NRC's Office of Nuclear Regulatory Research (RES). The Radiation Protection Research Program provides technical support in 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 Environmental Transport Research Program provides the technical bases, including data and analytical tools, to provide more realistic analyses of releases to environmental systems.

Radiation Protection Research

The mission of the NRC's Radiation Protection Program is to assist the NRC in its goals of regulatory licensing, policy making, and increasing public confidence. RES conducts research to support the NRC's evaluation and implementation of improvements to licensing, regulations, nuclear regulatory policy updates and changes; and oversees studies toward publishing guidelines and publications for public consumption. RES is also responsible for providing and maintaining computer codes for reactor licensing, decommissioning, and radiation safety/dose calculations.

RES is responsible for the following activities:

Development of technical basis for radiation protection regulations, licensing, rulemaking, and regulatory guides; health effects and dosimetry research; computer codes and databases development; participation in and monitoring of radiation research activities by National and International scientific and standard setting organizations; exposure and abnormal occurrence reports; and the RAMP (Radiation Protection Computer Code Analysis and Maintenance) program for developing, maintaining, and distributing the NRC's radiation protection, dose assessment, and emergency response computer codes.

Radiation protection research supports the following NRC program areas: Operating Reactors, New Reactors, Materials, Low-Level Waste/ Decommissioning, Health Studies and Nuclear Security.

Environmental Transport Research

The mission of the NRC's Environmental Transport Research Program is to provide improved technical bases and analytical tools for reviewing site characterization, monitoring, modeling, and remediation programs submitted by current and prospective licensees with regard to the release of radioactive materials to the environment from licensed nuclear facilities. Regulatory guidance is needed on environmental assessments and performance monitoring associated with nuclear reactors, fuel cycle and waste disposal facilities, and the decommissioning of nuclear facilities. Current projects within this program are addressing the long-term behavior of engineered barriers (specifically rates of release of sequestered radon), bio-remediation of uranium contamination, and source terms from aqueous reprocessing facilities.

NRC Standards for Protection Against Ionizing Radiation and ALARA for Radioactive Material in Light-Water Reactor Effluents

Objective

Technical information is being developed for possibly updating the NRC's radiation protection regulatory framework. A key component of this regulatory initiative is the development of new dose coefficients for occupational and public exposure to radionuclides that are based on International Commission on Radiological Protection (ICRP) Publication 103 recommendations. The results of this work directly supports the NRC staff in developing a technical regulatory basis for agencywide rulemakings on 10 CFR Part 20 and 10 CFR 50 Appendix I.

Research Approach

The NRC provides the fundamental radiological protection criteria for licensees to use in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, "Standards for Protection against Radiation." The last major revision to 10 CFR Part 20 was completed in 1991. It was primarily based on the 1977 recommendations contained in ICRP Publication 26, "Recommendations of the International Commission on Radiological Protection." Since 1991, the NRC has made minor revisions to 10 CFR Part 20, such as a reduced public dose limit that incorporates the recommendations of ICRP Publication 60, "1990 Recommendations of the International Commission on Radiological Protection," issued in 1991. The Agreement States' requirements for their licensees are essentially identical to 10 CFR Part 20.

In other NRC regulations, such as Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," some radiation dose criteria are based primarily on ICRP Publications 1 and 2 (the 1958 and 1959 "Recommendations of the International Commission on Radiological Protection"). Also, NRC fuel cycle licensees have received authorization, on a case-by-case basis, to use the newer ICRP methodology (ICRP Publication 66, "Human Respiratory Tract Model for Radiological Protection," issued January 1995 and beyond) in their licensed activities.

Updated technical information based on ICRP Publication 103 could be used to replace the three different sets of ICRP recommendations that are in use today by various licensees. The NRC staff works with Oak Ridge National Laboratory on the development of new dose coefficients for occupational and public exposure to radionuclides. Close coordination with other Federal agencies and participation in domestic and international working groups are beneficial for assessing potential technical and policy issues associated with implementation of new dose coefficients.

In support of this project, fundamental radiation dosimetry research is conducted to improve the capability to model radiation interactions and behavior within humans by employing advanced computational methods and state-of-the-art biokinetic models (Figure 5.1).

Status

The NRC staff is providing technical support with the U.S. Environmental Protection Agency for developing new dose coefficients for occupational and public exposures based on ICRP Publication 103. This joint interagency effort also supports the preparation of revised Federal Guidance reports on radiation protection and the development of analytical tools that are needed for possible revision of 10 CFR Part 20 and 10 CFR Part 50.

Figure 5.1 Biokinetic model.

For More Information
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Research on Patient Release, Post-Radioisotope Therapy

Objective

Each year, thousands of patients in the United States undergo diagnostic and therapeutic procedures using radioactive isotopes, and the medical discharge of these patients are governed by the Title 10 of *the Code of Federal Regulations* Part 35.75 (10 CFR 35.75), “Release of individuals containing unsealed byproduct material or implants containing byproduct material.” The Commission on May 9, 2011, directed the staff to evaluate the potential gaps in the available data regarding the doses actually being received by members of the public resulting from the release of patients treated with medical isotopes as well as the methodology to collect such data. In addition, the staff was asked to provide its recommendations to the Commission, in a notional vote paper, on whether data gaps exist and whether and how such data could be collected and used. In the direction, the Commission stated, “We should continually satisfy ourselves that we are aware of doses that result from use of radioactive material. The current 10 CFR 35.75 for patients treated with radioactive material set appropriate dose limits and appears to properly balance public health and safety with individual necessities of medical care.”

In response to the Commission’s mandate, the staff identified a gap in the empirical data and provided the Commission with information and options on gathering the missing data, and sensitive to patient confidentiality, determining how much data could be collected. The objective of the study is to provide information to evaluate existing regulatory guidance and its application as it pertains to members of the general public.

Research Approach

The study was designed to take a three-phase research approach. The three-phases will be Phase I - Pilot survey-tool (questionnaire) study, Phase II - Full survey-tool study, and Phase III - Detailed interviews and health-physics calculations based on Phase II. In Phase I (pilot study), the scope of the issue will be accessed by using a survey tool distributed to nine private medical institutions to Federal partners such as the U.S. Army, U.S. Navy, U.S. Air Force, Department of Veterans Affairs, National Institutes of Health, and the Bureau of Federal Prisons. Using the pilot-study data, Phase II will expand the survey tool to all U.S. medical facilities and will include focused interviews to access the state of the common practice and its interrelationship with patient activities post treatment. As part of Phase II, the study will be reaching out to advocate groups, agreement state regulators, and other interested parties to isolate and identify current best practices and their resulting impact to a member of the general public. Phase III would bring together Phase I and Phase II data to develop and refine members-of-the-general-public exposure scenarios from patients released from study facilities. The final outcome of this study is to provide the Commission data to inform regulatory revisions.

Figure 5.2 I-131 Radiation treatment of the thyroid.

Status

The contract for the study was awarded in late 2014 to Sanford Cohen and Associates Incorporated (SC&A Inc.). The contractor will perform all three phases of the study. Phase I was initiated in February 2015 with an expected completion by the summer 2015. Phase II is anticipated to start in summer to late autumn 2015. The study is expected to be complete by 2017.

For More Information
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Effectiveness of Surface Covers for Controlling Fluxes of Water and Radon at Disposal Facilities for Uranium Mill Tailings

Objective

Engineered covers are designed, constructed, and maintained to minimize infiltration of water into the subsurface to preclude contaminant leaching, mobilization, and migration of buried hazardous and/or radioactive waste to the accessible environment. The objective of this study is to determine the effects of soil structure evolution on the hydraulic conductivity and gaseous diffusivity of radon (Rn) barriers.

The study will determine how structural development varies with depth and thickness of the Rn barrier, and how structure influences transmission of radon and seepage carrying ground water contaminants. Ultimately, NRC needs to understand the behavior of cover materials over time, how changes impact releases of radon and influx of water, and if these changes impact regulatory decisions.

Research Approach

In December 2011, NRC issued a peer-reviewed report, NUREG/CR-7028, "Engineered Covers for Waste Containment: Changes in Engineering Properties and Implications for Long-Term Performance Assessment[1]," for use in assessing performance of engineered covers and systems for waste containment. An important conclusion of NUREG/CR-7028 was that compacted soil materials used in engineered covers at the sites studied did not retain "as built" properties over periods of regulatory interest. Changes in low permeability cover soils can be rapid and within several years can result in an increase to the saturated hydraulic conductivity by three to four orders of magnitude. It is anticipated that radon emission would behave in a similar manner, increasing substantially above "as-built" measurements. The durability/sustainability of Uranium Mill Tailings Radiation Control Act (UMTRCA) covers will be evaluated with respect to hydraulic flux and radon emissions by measuring a range of cover conditions that may include Rn barrier age, soil structure, soil moisture content, hydraulic conductivity, lead-210 profiles, plant rooting depths, vegetation types, and vegetation maturity.

Four UMTRCA radon barriers under surveillance by Department of Energy/LM (Legacy Management) will be selected for evaluation. Sites will be selected that have Rn barriers varying in age, depth, and thickness that are in locations representing a range of vegetation and climates. At each location, three conventional size and one large size flux chambers will be deployed. After the flux chamber tests are complete and the chambers are removed, large-scale (450 mm diameter) undisturbed block samples will be collected from the Rn barrier. Data from the hydraulic conductivity and other parameters will be used to construct profiles of hydraulic properties of the Rn barrier as a function of depth. These profiles will be compared to profiles anticipated during design and measured during construction of the Rn barrier. Samples of cored material will be measured for Lead-210 sorbed on the Rn barrier material. Profiles of sorbed Pb-210 vs. depth will be used as an indicator of Rn flux, as Pb-210 is a relatively long-lived (22-year half-life) decay product of Rn. The data collected will be used to assess radon fluxes and percolation rates for each of the UMTRCA covers relative to the predictions made during design. Percolation rates will be predicted using the program WinUNSAT-H using measured hydraulic properties and existing and historic meteorological records for each site. Long-term radon fluxes will be estimated using source concentrations assumed during design. These predictions will be compared to the radon fluxes and water saturation profiles measured in the field allowing evaluation of changes in water and radon transport as a function of cover properties.

Status

This project began in 2015 and is anticipated to be completed in 2018 with the analysis of cover materials from four sites.

For More Information
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1 http://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr7028/

In-Situ Bioremediation of Uranium in Ground Water

Objective

As a result of uranium in-situ recovery (ISR) (solution mining of uranium) and the processing of uranium for nuclear fuel, elevated concentrations of uranium and other elements exist in some ground-water systems. In many cases, the traditional remediation methods have not reduced aqueous uranium concentrations to acceptable levels. As a result, a new approach, in-situ bioremediation (ISB), has been proposed using native bacterial populations to alter geochemical conditions. The objective of this project is to provide experimental and modeling information for staff to use in evaluating ISB.

Research Approach

During ISB, nutrients are injected through wells into the contaminated aquifer and used by bacteria to increase their growth and reproduction. In the process, the oxygen is used in the subsurface to generate reducing conditions. As a result, iron and uranium are chemically reduced, and uranium is precipitated from solution. Although uranium is removed from solution, it is precipitated as a mineral and left in place.

Two approaches were used for both shallow contamination sites (uranium ore processing facilities) and the ISR sites (deep ore bodies)—laboratory-scale experimental work and advanced modeling.

For the experimental program, sediments from shallow and ISR sites were placed in columns, and reducing conditions were established by biostimulation. The behavior of uranium and other elements were followed in both the aqueous and solid phase during reduced conditions and then as oxygen-containing water was introduced into the columns. Solid-phase analysis included determination of the oxidation state of uranium and iron and their microscale distributions under the reduced and oxidized conditions of the columns.

The modeling work was conducted by Pacific Northwest National Laboratory and evaluated short and long-term chemical processes of bioremediation and it focused on processes controlling changes of uranium mobility during and after bioremediation. The approach used coupled models of biological, geochemical, and transport processes to determine how the chemistry in these systems changed and what the effects were on parameters that can be monitored in the field.

This modeling was based on the experiments conducted by the U.S. Geological Survey. Results of these experiments, the first done on material from an ISR site, showed that biological processes leading to uranium precipitation at an ISR site appear to be quite different from those at shallow sites and required significant alteration of the modeling approach. Of these, the most important were reduction in the growth rate compared to shallow sites and the dependence of uranium bioreduction rate on biomass, which increases with continuous acetate addition to the ground water. Detailed results are given in NUREG/CR-7167, "Assessing the Potential for Biorestoration of Uranium In Situ Recovery Sites."

From this work, RES staff have recommended against using bioremediation at a shallow site because the availability of oxygen would readily redissolve and mobilize uranium. However, bioremediation of deep ISR sites where oxygen is limited may be an option.

Status

This work is now concluded. Final reports for the ISR work (NUREG/CR-7167) and for the shallow aquifer work (NUREG/CR-7178, "Uranium Sequestration During Biostimulated Reduction and In Response to the Return of Oxic Conditions In Shallow Aquifers") have been published and are available on the NRC Web site. Future work may include modeling bioremediation experiments being conducted at an ISR site.

For More Information
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Analysis of Cancer Risk in Populations Near Nuclear Facilities

Objective

The objective of this research is to provide an up-to-date, technically credible resource for NRC staff to use to communicate with our stakeholders about recurrent cancer risk concerns from living near NRC-licensed facilities. NRC-licensed facilities sometimes release very small radiation doses during normal operations. Facility operators must follow NRC regulations by closely monitoring and controlling these releases to meet very strict radiation dose limits. The plants also must publicly report them to the agency. Some people are concerned these releases could affect the health of communities around nuclear facilities. The NRC staff has used a 1990 study conducted by the National Institutes of Health/National Cancer Institute, "Cancer in Populations Living Near Nuclear Facilities," as a valuable risk communication tool for addressing stakeholder concerns. This study is now over 25 years old and ready for an update.

Research Approach

The NRC and National Academy of Sciences (NAS) agreed on a two-phase approach. The NAS Phase 1 committee completed their report in May 2012 and recommended two approaches for assessing cancer risks. The committee also recommended a pilot study of seven nuclear facilities to assess whether the approaches could work on a larger scale—this is the Phase 2 pilot study. The Phase 1 committee identified many technical challenges for the pilot study including:

- The need for large groups of people to detect very small changes in risk.
- Uneven availability and quality of cancer data for areas smaller than a county.
- Difficulty in reliably capturing information on population movement, risk factors, and other variables that could make interpreting the results difficult.
- The pilot study will determine if these technical challenges can be overcome. The study will also develop procedures and data collection methods while estimating the necessary time and resources.

The Phase 1 committee specifically recommended the pilot study have two parts: (1) a population study of cancer diagnosis and mortality rates for multiple cancer types and all age groups down to the census-tract level, and (2) a "case control" study of childhood cancers in children born within a fixed distance of a nuclear facility.

NRC-regulated facilities record information on their releases and report it once a year to the NRC. The committee recommended using this data and examining populations within about 30 miles (50 kilometers) of nuclear facilities to cover a range of potential radiation exposures. The committee also recommended adapting existing computer models (or developing a new model) to estimate radiation doses to individual organs from airborne and liquid radioactive releases.

The NAS committee recommended these facilities for the pilot study:

- Dresden Nuclear Power Station, Illinois.
- Millstone Power Station, Connecticut.
- Oyster Creek Nuclear Generating Station, New Jersey.
- Haddam Neck, Connecticut (decommissioned).
- Big Rock Point Nuclear Power Plant, Michigan (decommissioned).
- San Onofre Nuclear Generating Station, California (permanently shut down).
- Nuclear Fuel Services, Tennessee.

These facilities were selected because they started operation at different times and represent both currently operating and decommissioned nuclear facilities. Moreover, these facilities have some variation in surrounding population sizes, the quality and maturation of the State's cancer registry, and level of complexity for the registry's research approval processes and research support.

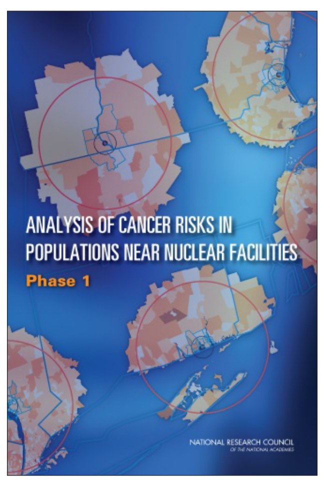

Figure 5.3 NAS Phase 1 Report.

Status

The NRC is ceasing work on the study because of the significant amount of time and resources needed to perform the study and the agency's current budget constraints. For details on the decision to end the study see SECY-15-0104 at http://pbadupws.nrc.gov/docs/ML1514/ML15141A404.pdf.

For More Information

Contact Terry Brock, RES/DSA, at Terry.Brock@nrc.gov.

The One Million Worker and Atomic Veteran Study

Objective

The objective of the One Million Worker and Atomic Veteran Study is to determine the cancer risk of radiation workers who received occupational doses (low dose rates) over a career of exposure compared to the cancer risk known at high doses and high dose rates from past studies (e.g., the Japanese atomic bomb survivors).

Research Approach

The NRC has entered into an interagency agreement with the U.S. Department of Energy (DOE) Office of Science (SC) Low Dose Radiation Research Program to study the health effects of more than 1 million radiation workers and atomic veterans. Supporting DOE and this multi-agency effort will provide valuable new information for future radiation protection standards-setting bodies and any resultant occupational radiation dose standards. The significance of the proposed research is considerable because it applies directly to existing concerns about standards for chronic radiation exposure. Much knowledge has been gained from the study of atomic bomb survivors, but exposure was acute and among a Japanese population living in a war-torn country.

Scientific and medical committees continue to grapple with how best to estimate risks associated with the gradual exposures received from environmental, medical, and occupational radiation. Recent studies, though limited, have suggested that chronic exposures may be more hazardous than currently accepted. Governmental agencies must deal with the complex issues of compensating prior workers, veterans, and citizens who may have been potentially harmed by past exposures. Protection committees deliberate over how best to estimate and apply a "dose and dose rate effectiveness factor" to scale the risks from the A-bomb survivor data for relevant and current circumstances. Evaluation of risk among persons with intakes of radioactive substances assumes greater importance as society debates the expansion of nuclear energy and deals with nuclear waste and threats of terrorist attacks with nuclear devices.

The significant increase in population medical exposures to CT scans and other nuclear imaging technologies has raised concerns about future health consequences. The methodology will follow the state-of-the-art approach recently used in studying cancer and other diseases among Rocketdyne radiation workers ("Updated mortality analysis of radiation workers at Rocketdyne (Atomics International), 1948-2008," Boice et al., August 2011 http://www.ncbi.nlm.nih.gov/pubmed/21381866).

Figure 5.4 Radiation worker taking measurements.

Status

Research on the NRC early nuclear power plant and industrial radiographer worker is well under way. The cohorts are now established, and we expect to see cancer risk results for these workers in late 2015.

For More Information
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Radiation Exposure Information and Reporting System (REIRS)

Objective

The Radiation Exposure Information and Reporting System (REIRS) project collects and analyzes the occupational radiation exposure records that NRC licensees submit under Title 10 of the Code of Federal Regulations (10 CFR) 20.2206, “Reports of Individual Monitoring.”

The objective of the REIRS database is to provide NRC staff with occupational exposure data for evaluating trends in licensee performance in radiation protection and for research and epidemiological studies. The exposure reports in this database can provide facts about routine occupational exposures to radiation and radioactive material that can occur in connection with certain NRC-licensed activities.

Approach

To maintain compliance with 10 CFR 20.2206, NRC licensees must submit their occupational radiation exposure data to the NRC by April 30 of each year. Licensees can submit this data electronically or on paper using either NRC Form 5, “Occupational Dose Record for a Monitoring Period,” or a Form 5 equivalent.

Each year, about 200,000 radiation exposure reports are submitted by five 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.

The NRC does not receive radiation exposure reports from the remaining two licensee categories—low-level waste disposal facilities and geologic repository for high-level waste—because these facilities are either not under NRC jurisdiction or not currently in operation.

The radiation exposure reports that NRC licensees submit are used to meet the following NRC regulatory goals:

- Evaluate the effectiveness of licensee’s as low as is reasonably achievable (ALARA) programs at commercial nuclear power plants (see Figure 5.5).
- Evaluate the radiological risk associated with certain categories of NRC-licensed activities.
- Compare occupational radiation risks with potential public risks.
- Establish priorities for the use of NRC health physics resources such as research and development of standards and regulatory guidance.
- Answer congressional and public inquiries.
- Provide radiation exposure history reports to current and former occupational radiation workers who were exposed to radiation or radioactive materials at NRC-licensed or regulated facilities.
- Conduct occupational epidemiological studies.

Status

The analysis of REIRS data is published annually in NUREG-0713, “Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities.” The annual NUREG-0713 reports are available on the NRC’s public Web site at http://www.nrc.gov or the REIRS Web page at www.reirs.com.

For More Information
Contact Luis Benevides, RES/RES/DSA, at Luis.Benevides@nrc.gov.

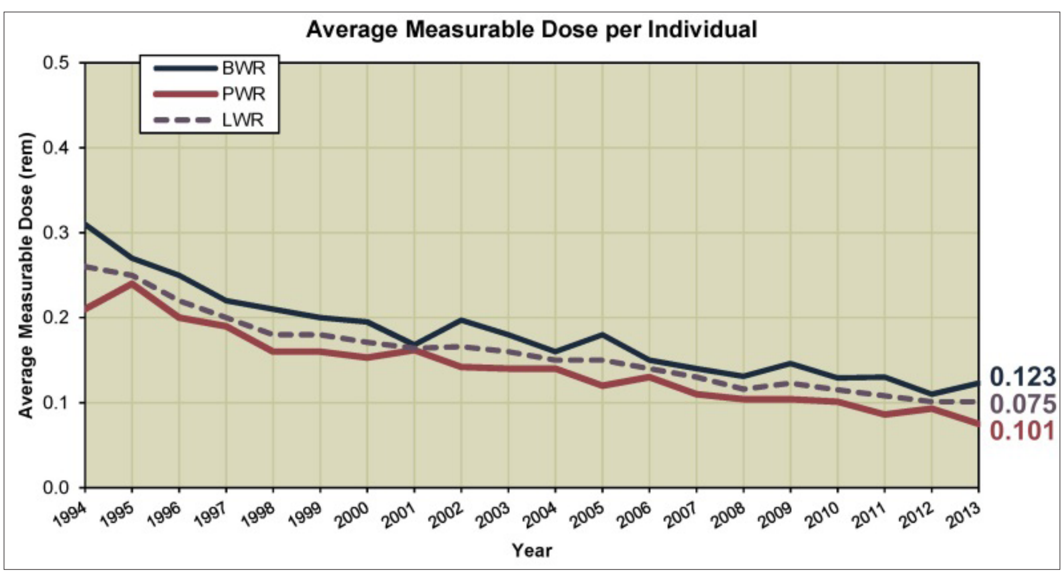

Figure 5.5 Annual Occupational Radiation Dose for PWR/BWR/LWR Reactors.

Radiological Assessment System for Consequence AnaLysis (RASCAL) Code

Objective

The Radiological Assessment System for Consequence AnaLysis (RASCAL) code is a tool used by the Protective Measures Team in the NRC's Operations Center for making independent dose and consequence projections during radiological incidents and emergencies. The NRC developed RASCAL over 25 years ago to provide a tool for the rapid assessment of an incident or accident at an NRC-licensed facility and to aid decisionmaking such as whether the public should evacuate or shelter in place. RASCAL evaluates atmospheric releases from nuclear power plants, spent fuel storage pools and casks, fuel cycle facilities, and radioactive material handling facilities (see Figure 5.6).

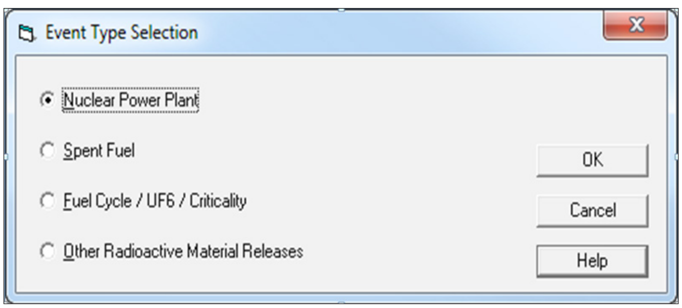

Figure 5.6 RASCAL v4.3.1 Source Term Event Type Selection Screen.

Research Approach

RASCAL has been continually upgraded and improved upon to include updated source term models, atmospheric transport models, nuclear power plant site-specific data, and updated computer calculation methods. RASCAL version 4.3, which was issued in September 2013, incorporated the NRC Near Term Task Force's lessons learned on the Fukushima Daiichi nuclear power plant accident in Japan. Some of these included updates to the atmospheric transport, dispersion, and dose calculation (ATD) model to increase the RASCAL 4.3 domain from a 50-mile radius to a 100-mile radius; changes to the Source Term to Dose (STDose) models to include the option for long-term station blackout (LTSBO); and 96-hour duration for the accident and the addition of Source Term Merge/Export option that allows users to assess the consequences from a multi-reactor event (see Figure 5.7).

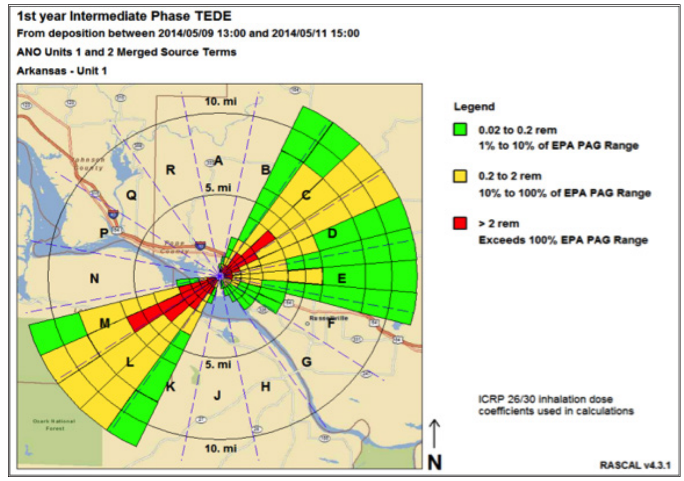

Figure 5.7 Example of RASCAL v4.3.1 Merged Source Term TEDE Plume.

Status

In December 2014, the Office of Nuclear Regulatory Research (RES) released RASCAL version 4.3.1 (update), to resolve coding issues to RASCAL version 4.3 that were identified by RASCAL users (see Figure 5.8). Some of these issues include updates to the Source Term to Dose (STDose) module for reactor events and the spent fuel pool. Specifically, the STDose module for reactor events includes updates to the LTSBO State-of-the-Art Consequences Analyses (SOARCA) option, revised containment leak rate (pressure/hole) models, and updates to the LOCA (NUREG-1465) calculations. The spent fuel pool graphic user interface was changed allowing the user greater clarity of the RASCAL models used for these calculations and to aid the user with the selection of RASCAL options during a spent fuel pool event. In addition, this update to RASCAL provides for improved source-term import, export and merge options, resolution of issues related to the ATD models and meteorological data handling, updates to the RASCAL facility database and site data files, and RASCAL software installation and other coding fixes.

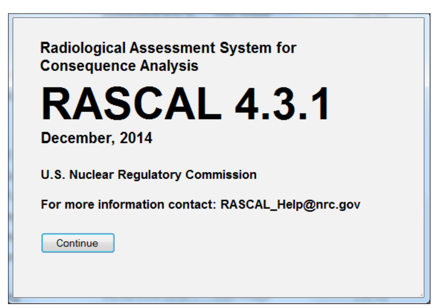

Figure 5.8 RASCAL v4.3.1 Welcome Screen.

RASCAL version 4.3.1 is one of the radiation protection codes available through RAMP.

For More Information
Contact John Tomon, RES/DSA, at John.Tomon@nrc.gov.

RADionuclide Transport, Removal, And Dose Estimation (RADTRAD) Code

Objective

The potential radiological consequences of nuclear power reactor accidents depend in part on the amount, form, and species of the radioactive material released during the postulated accident. The Radionuclide Transport, Removal, And Dose Estimation Code (RADTRAD) models doses at the exclusion area boundary, the low-population zone, and the control room (CR) from a release of radionuclides during a design basis accident (DBA). RADTRAD is a licensing analysis tool used to show compliance with nuclear plant siting and CR dose limits for various loss-of-coolant accidents (LOCAs) and non-LOCA accidents. As radioactive material is transported through the containment, the user can account for sprays and natural deposition that may reduce the quantity of radioactive material. Material can flow between buildings, from buildings to the environment, or into control rooms through high efficiency particulate air filters, piping, or other connectors. Decay and ingrowth of daughters can be calculated over time as the material is transported.

Research Approach

To improve RADTRAD's maintainability, remove platform and compiler dependencies, and add new features, RADTRAD was re-implemented in the JAVA language. This JAVA-based version of RADTRAD was named Version 4.5. In addition, the Microsoft Visual Basic GUI was replaced with the Symbolic Nuclear Analysis Package (SNAP) GUI (see Figure 5.9).

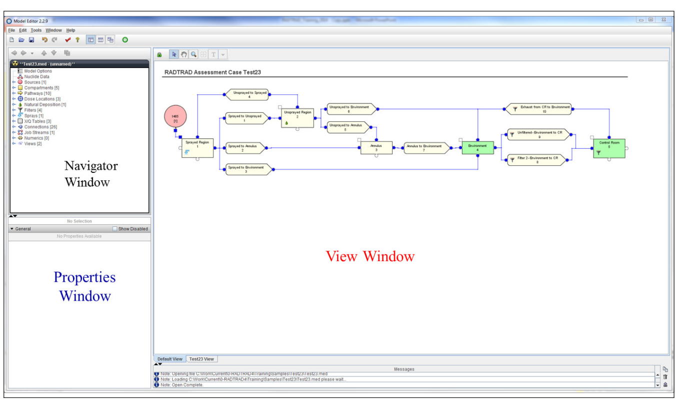

Figure 5.9 Creating RADTRAD input model using SNAP GUI.

SNAP uses a plugin-based architecture that "wraps" all of the interfaces to an analytical code in a special file called a "SNAP plug-in." Placing RADTRAD in the SNAP framework allows for the use of SNAP features, including the Model Editor for developing plant models, and provides tools for user input checking and monitoring calculations.

Status

In 2015, the Office of Nuclear Regulatory Research released RADTRAD version 4.5 and the SNAP-RADTRAD plug-in version 4.9.4. Verification testing of RADTRAD version 4.5 and the SNAP-RADTRAD plug-in version 4.9.4 was performed independently using the MATHCAD engineering calculation software package. This version of the RADTRAD code and SNAP-RADTRAD plug-in included new features such as:

- Option for an adaptive time step algorithm.
- Option for a default time step with error calculation.
- Addition of the reactor coolant system (RCS) activity calculator to the code (see Figure 5.10).
- Ability to add Technical Specification equilibrium activity values for dose equivalent (DE) I-131 and Xe-133, and pre-incident or coincident iodine spiking (see Figure 5.10).
- Ability for the user to model alternative source term non-LOCA DBAs described in Regulatory Guide 1.183 (RG 1.183).
- Updated dose conversion factors (DCFs) to Federal Guidance Reports 11 and 12 (DCFPAK2).
- Updated a larger radionuclide database from ICRP-38 (838 nuclides).

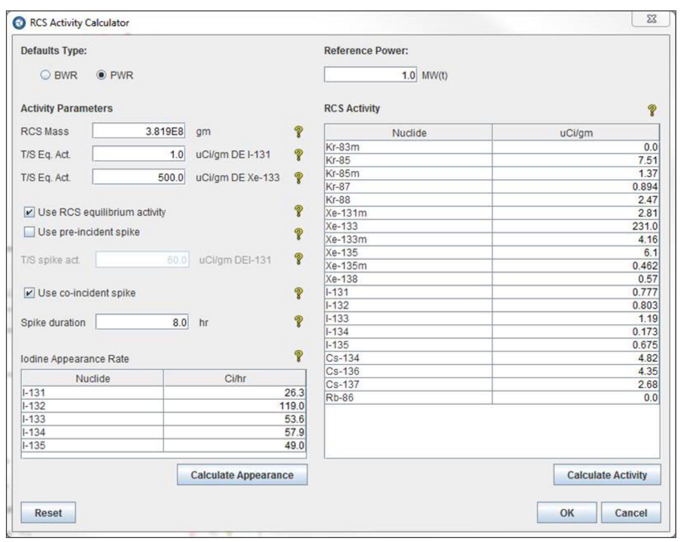

Figure 5.10 SNAP/RADTRAD RCS Activity Calculator.

RADTRAD version 4.5 and the SNAP-RADTRAD plug-in version 4.9.4 are one of the radiation protection codes available through RAMP.

For More Information
Contact John Tomon, RES/DSA, at John.Tomon@nrc.gov.

Radiation Protection Computer Code Analysis and Maintenance Program (RAMP)

Objective

The Radiation Protection Computer Code Analysis and Maintenance Program (RAMP) (see Figure 5.11) is a program for developing, maintaining, and distributing the NRC's radiation protection, dose assessment, and emergency response computer codes. The codes in RAMP include RASCAL, SNAP/RADTRAD, VARSKIN, GALE, DandD, HABIT, PIMAL, and Radiological Toolbox.

Figure 5.11 RAMP Logo.

Goals of RAMP:

- Ensure codes are appropriately updated.
- Ensure codes reflect computer programming language updates.
- Ensure updates are in accord with International Regulations and Guidance Documents.
- Ensure codes are updated based on lessons learned from events such as Fukushima.
- Ensure costs are shared among users of the codes.
- Provide a centralized management structure for reporting, prioritizing, and resolving code issues.

Benefits of RAMP:

- Access to the most current versions of the code.
- Code maintenance, development, benchmarking, and uncertainty studies.
- A cooperative forum to resolve code errors and inefficiencies.
- Technical basis documents and user guidelines for applying the codes, and periodic meetings to share experiences, discuss code development.
- Periodic training on the codes.

Research Approach

The NRC conducts regulatory research in partnership with international nuclear safety agencies and organizations. As such, the NRC and international entities carry out cooperative research projects to achieve mutual research needs with greater efficiency by sharing experiences and costs for code development and maintenance. RAMP is an integral part of this research because it supports regulatory decisions on radiation protection, dose assessment, and emergency response computer codes used by these agencies and institutes. In addition, RAMP has one domestic and one international meeting per year to exchange information and discuss state-of-the-art models, emerging technologies, and various other radiation protection issues.

Status

Radiological Assessment System for Consequence AnaLysis (RASCAL) Code

The RASCAL code is a tool used by the Protective Measures Team in the NRC's Operations Center for making independent dose and consequence projections during radiological incidents and emergencies. RASCAL evaluates atmospheric releases from nuclear power plants, spent fuel storage pools and casks, fuel cycle facilities, and radioactive material-handling facilities. These data from RASCAL represent an important part of the total information used by the local authorities during an accident.

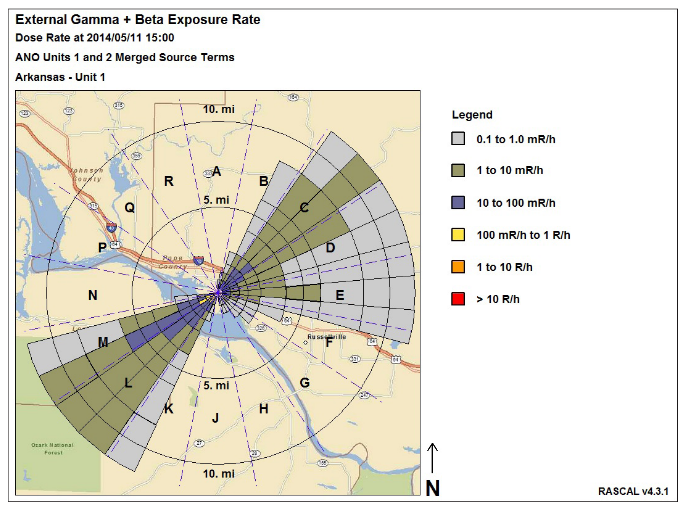

Figure 5.12 RASCAL v4.3.1 Output Screen.

RADionuclide Transport, Removal, And Dose Estimation (RADTRAD) Code

The RADionuclide, Transport, Removal, and Dose Estimation (RADTRAD) code is a licensing analysis code used to show compliance with nuclear plant siting criteria for the site boundary radiation doses at the Exclusion Area Boundary (EAB) and the Low Population Zone (LPZ) and to assess the occupational radiation doses in the control room (CR) and/or Emergency Offsite Facility for various loss-of-coolant accidents (LOCA) and non-LOCA design basis accidents (DBAs). RADTRAD uses a combination of tables and numerical models of source term reduction phenomena to determine the time-dependent dose at the CR, EAB, and LPZ for given DBA scenarios.

Figure 5.13 RADTRAD Logo.

Gaseous and Liquid Effluent (GALE) Code

The GALE code estimates the Gaseous And Liquid Effluent (GALE) from commercial light-water nuclear power plants. This Fortran-based code can provide estimates for gaseous and liquid effluent from either boiling or pressurized light-water reactors for pre-licensing reviews. GALE is maintained at the Pacific Northwest National Laboratory under contract for the NRC's Office of Nuclear Regulatory Research. The calculations are based on data generated from operating reactors, field tests, laboratory tests, and plant-specific design considerations incorporated to reduce the quantity of radioactive materials that may be released to the environment. These data are based on (1) standardized coolant activities derived from American Nuclear Society (ANS) 18.1 Working Group recommendations, (2) release and transport mechanisms that result in the appearance of radioactive material in liquid and gaseous waste streams, (3) plant-specific design features used to reduce the quantities of radioactive materials ultimately released to the environs, and (4) information received on the operation of nuclear power plants.

Currently, GALE is undergoing a modernization effort that will result in GALE version 12 (v12) that will incorporate a new graphical user interface (GUI). The GUI will facilitate the user in developing input data, executing the GALE v12 suite of codes, and operating in a modern windows environment. As part of this effort, the associated documentation will be also updated. This includes completing the necessary code documentation, the verification-validation package, a detailed quality-control and assurance program, and associated revisions to NUREG-0016 and NUREG-0017. Reaching into the future, programmers will be incorporating the capability to perform GALE calculations for small modular reactors. The GALE code has now been incorporated into the RAMP program. The status of the effort is that RES staff is evaluating the publications and revisions being made to GALE. Now as part of the RAMP program, GALE User Group will enable end-users to provide user interface feedback and identify program improvement issues. GALE v12 is expected to be released in the summer 2015.

Figure 5.14 GALE Logo.

HABITability (HABIT) Code

The HABITability (HABIT) is a package of computer codes designed to assist in the evaluation of light-water reactor control room habitability in the event of accidental spills of toxic chemicals or the accidental release of radionuclides, including noble gas. It consists of a number of program modules and produces files containing tabular output that can be printed, viewed, or imported into spreadsheet programs for further applications. All technical bases can be found in NUREG/CR-6210, "Computer Code for Evaluation of Control Room Habitability (HABIT)," June 1996. The recent re-hosting of HABIT v1.2 code is compatible fully to Windows 7 (64 bit) operating environment and supports 508 accessibility and compliance.

The new user interface for HABIT v1.2 uses a Tabbed Document Interface (TDI) that allows all the major functions to be contained within tabs in a single window. Each instance of a module (EXTRAN, CHEM, FPFP_2, TACT5, and CONHAB) is contained within its own tab document. The single window containing the TDI is the Design Package window (see Figure 5.15). This window coordinates the display and execution of the individual modules as well as performing functions that are design-centric (e.g., saving all the files that are part of the design package). The window layout contains a main menu bar on the top that has standard drop-down menus that allow the user to access options and output settings for the overall design and for individual modules. The application has been designed for optimal display at a resolution of 1024x768 pixels but is adaptable to support other resolutions. In addition to the improved computational stability and consistency, many improvements have been made on the end-user interface and "Help" manual. For example, a HABIT v1.2 Quick User Guide has also been integrated into the code, and interactive pop-up "HELP" screens are available throughout the program, when activated, to guide the proper execution of the calculations.

Continual user feedback is being currently incorporated into future versions of the code.

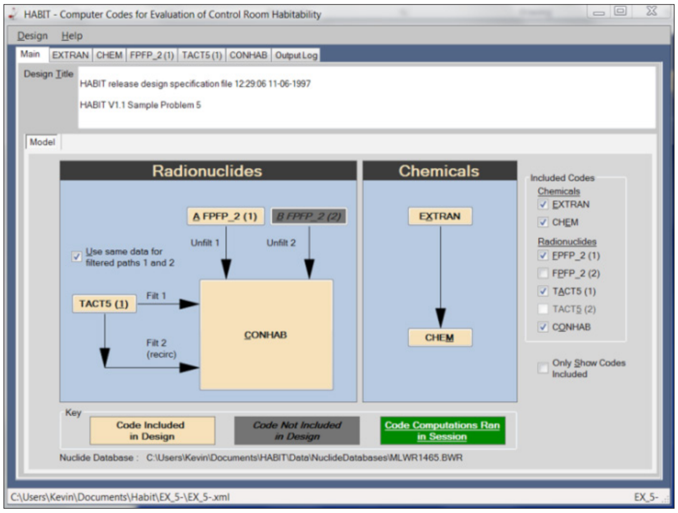

Figure 5.15 HABIT v1.2 Main Screen.

Decontamination and Decommissioning (DandD) Code

The Decontamination and Decommissioning (DandD) software package developed by the NRC provides a user-friendly analytical tool to assess compliance with the dose criteria of 10 CFR Part 20, Subpart E. Specifically, DandD embodies the NRC's guidance on screening dose assessments to allow licensees to convert residual radioactivity from residual in soils and on building surfaces at their site to annual dose in a way consistent with NRC regulations and guidance.

DandD provides the option of specifying only the level of contamination at a site and running the code with default screening parameters to determine a conservative dose estimate or entering site-specific information, modifying scenario pathways, or providing site-specific parameter distributions to determine a site-specific dose. The screening methodology in DandD employs reasonable conservative scenarios, fate and transport models, and default parameter values and parameter distributions (based on NUREG/CR–5512, Volumes 1 and 3) to allow the NRC to quantitatively estimate the risk of terminating a license given only information about the level of contamination. In addition, DandD includes a sensitivity analysis module to identify parameters in the screening analysis that have the greatest impact on the results of the dose assessment. With DandD's screening methodology and the sensitivity analysis, licensees are able to make informed decisions regarding allocation of resources needed to collect additional site-specific information.

VARSKIN Code

The VARSKIN computer code is used by the NRC, Agreement States, licensees, vendors, and international partners to calculate radiation dose to the skin resulting from exposure to radiation emitted from hot particles or other contamination on or near the skin. These dose assessments are required by 10 CFR 20.1201(c) in which the assigned shallow dose equivalent is to the part of the body receiving the highest exposure over a contiguous 10 cm^2 of skin at a tissue depth of 0.007 centimeters (7 mg/cm^2).

PIMAL Code

Phantom with Moving Arms and Legs (PIMAL) is a graphical user interface with pre-processor and post-processor capabilities that assists users in developing Monte Carlo N-Particle Transport Code input decks and running the codes. It allows users to easily generate quantitative figures of merit regarding positioning arms and legs in different geometries. PIMAL software is considered an efficient and accurate tool for performing dosimetry calculations for radiation workers and exposed members of the public.

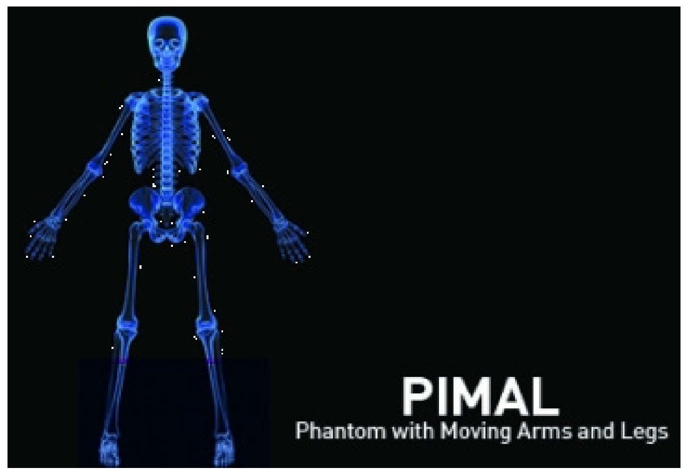

Figure 5.16 PIMAL Logo.

Radiological Toolbox

Radiological Toolbox provides ready access to data of interest in radiation safety and protection of workers and members of the public. The data include dose coefficients for intakes of radionuclides, external exposure to radioactive materials distributed in environments, and exposures to photon and neutron radiation fields described in ICRP Publication 74. Other supportive data include interaction constants and coefficients for alpha, beta (i.e., electron), gamma (i.e., photon or x-ray), and neutron radiations; nuclear transformation data in ICRP Publications 38 and 107; biological, radiological, and physiological data; and information on various related topics.

For More Information
Contact Stephanie Bush-Goddard, RES/DSA, at Stephanie.Bush-Goddard@nrc.gov.

Radiation Protection Cooperative Research

Objective

The NRC monitors the latest scientific information on radiation cancer risks to ensure our regulatory programs continue to adequately protect the public health and safety. Toward that end, NRC staff participate in and monitor the activities and research efforts of scientific and standard setting organizations—such as the National Academy of Sciences (NAS), the United Nations Scientific Committee on Exposure to Atomic Radiation (UNSCEAR), the International Commission on Radiological Protection (ICRP), the U.S. National Council on Radiation Protection and Measurements (NCRP), the International Atomic Energy Agency, and the joint U.S.-Russian Health Studies Program.

Research Approach and Status

Ongoing scientific work continues to increase our understanding of the health effects and risks associated with radiation exposure. For example, in the United States, the NAS published the report entitled, "Health Effects of Exposure to Low Levels of Ionizing Radiation," which the Biological Effects of Ionizing Radiation (BEIR) VII Committee prepared as an update to the 1990 BEIR V report entitled, "Health Effects of Exposure to Low Levels of Ionizing Radiation." As such, the BEIR VII report constitutes the updated scientific basis for radiation safety standards in the United States.

One of the benefits of the Radiation Protection 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 Radiation Protection Program staff collaborates with national and international experts in health physics at national laboratories, universities, and other organizations.

International Commission on Radiological Protection

The NRC participates in the ICRP, an independent registered charity established to advance the science of radiological protection for the public benefit, in particular by providing recommendations and guidance on all aspects of protection against ionizing radiation. The NRC uses ICRP recommendations, in part, to form the technical bases for the agency's radiation protection program and regulations. NRC staff participates in ICRP committees and collaborates with stakeholders to ensure consistency in the application of radiation protection standards and dosimetry modeling. Toward that end, the NRC played a pivotal role along with other Federal partners in establishing the biennial ICRP Symposium that brings together the world's experts in radiation protection.

For More Information
Contact Terry Brock, RES/DSA, at Terry.Brock@nrc.gov.

U.S. National Council on Radiation Protection and Measurements

The NCRP was chartered by the U.S. Congress in 1964 and seeks to formulate and disseminate information, guidance, and recommendations on radiation protection and measurements that represent the consensus of leading scientific thinking. The Council seeks out areas in which the development and publication of NCRP materials can make an important contribution to the public interest. The NRC is currently supporting three specific NCRP projects with staff expertise, NRC-collected data, and financial resources: (1) the U.S. One Million Worker and Atomic Veterans Study; (2) Guidance on Radiation Dose Limits for the Lens of the Eye; and (3) Radiation Protection Guidance for the United States.

For More Information
Contact Terry Brock, RES/DSA, at Terry.Brock@nrc.gov.

Russian Health Studies Program

NRC staff participates with the U.S. Department of Energy (DOE) staff on the Russian Health Studies Program. This program encompasses a portfolio of cooperative health research and radiation studies with the Russian Federation Joint Coordinating Committee for Radiation Effects Research (JCRRER). The NRC is a member of the U.S. delegation to JCRRER and involves staff participation in the Executive Committee. The program evaluates long-term health effects on workers and populations living near the Russian nuclear weapons production site at Mayak. The effort is expected to answer critical questions on the health impacts associated with long-term, low-dose-rate radiation exposures, and other mutually beneficial radiation health effects programs in our respective agencies.

For More Information
Contact Terry Brock, RES/DSA, at Terry.Brock@nrc.gov.

Committee on Radiation Protection and Public Health

The Committee on Radiation Protection and Public Health (CRPPH) sponsored by the Organization for Economic Co-operation and Development (OECD)/Nuclear Energy Agency is a valuable resource for its member countries including the United States represented by the NRC. The committee is made up of regulators and radiation protection experts with the broad mission of providing timely identification of new and emerging issues, analyzing their possible implications, and recommending or taking action to address these issues to further enhance radiation protection regulation and implementation. The NRC supports the CRPPH on emerging issues, policy and regulation development in member countries, and disseminating good practices.

For More Information
Contact Rebecca Tadesse, RES/DSA, at Rebecca.Tadesse@nrc.gov.

Information System on Occupational Exposure

Another important collaboration is the NRC involvement with the Information System on Occupational Exposure (ISOE). The ISOE was created in 1992 and is jointly sponsored by the OECD/Nuclear Energy Agency and the International Atomic Energy Agency. The focus is to provide an international forum for radiation protection professionals from nuclear power utilities and national regulatory authorities to share best practices in dose reduction information, operational experience to improve the radiological protection at nuclear power plants. Other national and international outreach include the Interagency Steering Committee on Radiation Standards, the International Commission on Radiation Units and Measurements, and the French Institute for Radiological Protection and Nuclear Safety.

For More Information
Contact Luis Benevides, RES/DSA, at Luis.Benevides@nrc.gov.

Chapter 6: Risk Analysis Research

For assessing public safety and developing regulations for nuclear reactors and materials, the U.S. Nuclear Regulatory Commission (NRC) traditionally used a deterministic approach that asked, "What can go wrong?" and "What are the consequences?" Now, the development of risk-assessment methods and tools allows the NRC to also ask, "How likely is it that something will go wrong?" According to the traditional definition, risk is the product of the likelihood and consequences of an adverse event. Probabilistic risk assessment (PRA) is a systematic analysis tool consisting of specific technical elements that provide both qualitative insights and a quantitative assessment of risk. In this way, PRAs allow the identification, prioritization, and mitigation of significant contributors to risk to improve nuclear power plant safety.

Modern PRAs also have incorporated uncertainty analyses to address a fourth question: "How confident are we in our answers to these three questions?" The NRC staff has developed guidance to address the types of uncertainties reflected in PRAs, and it has documented these in NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking."

In 1995, the NRC issued a policy statement on the use of PRA encouraging its use in all regulatory matters. That policy statement directs that "the use of PRA technology should be increased to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach." Since the NRC issued its PRA policy statement, the agency has added a number of risk-informed activities to the NRC regulatory structure (i.e., regulation and guidance, licensing and certification, oversight, and operational experience). The NRC also has developed technical documents to provide guidance on the use of PRA information to support these activities.

The NRC recognizes that PRA has evolved to the point where it can be used as a tool in regulatory decisionmaking. These risk tools also allow the NRC to consider multiple hazards and combinations of equipment and human failures that go beyond what is traditionally considered. By making the regulatory process risk-informed (using risk insights to focus on those items most important to protecting public health and safety), the NRC can focus its attention on the design and operational issues most important to safety. Consequently, confidence in the information derived from a PRA is an important issue. The accuracy of the technical content must be sufficient to justify the specific results and insights that are used to support the decision under consideration.

In the reactor safety arena, risk-informed activities occur in five broad categories: (1) rulemaking, (2) licensing process, (3) reactor oversight process, (4) regulatory guidance, and (5) development of 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 in-service inspections; and improved Standardized Plant Analysis Risk (SPAR) models.

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. Although different NRC programs have different objectives, they use the same risk tools—the Systems Analysis Programs for Hands-on Integrated Reliability Evaluation (SAPHIRE) code and Standardized Plant Analysis Risk (SPAR) models for performing risk assessments. Therefore, 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.

Factor	Scoping Options for Operating Nuclear Power Plants
Radiological hazards	Reactor core Spent fuel Other Radioactive Sources
Population exposed to hazards	Onsite population Offsite population
Plant operating states	At-Power Low Power/Shutdown
Initiating event hazards	Traditional internal events (transients, loss-of-coolant accidents) Internal floods Internal fires
	Seismic events (earthquakes) High winds Other external hazards
Level of risk characterization	Level 1 PRA: Core damage frequency Level 2 PRA: e.g., Large early release Level 3 PRA: Early fatality risk Latent cancer fatality risk

Figure 6.1 Factors affecting the scope of PRAs for operating nuclear power plants.

SAPHIRE provides the functions required for performing a PRA. Users can supply basic event data, create and solve fault trees and event trees, perform uncertainty analyses, and generate reports. The SPAR models are used to support a number of risk-informed initiatives. The fidelity and realism of these models are ensured through a number of processes including cross-comparison with industry models, review and use by a wide range of technical experts, and confirmatory analysis.

Full-Scope Site Level 3 Probabilistic Risk Assessment Project

Objectives

The full-scope site Level 3 PRA project includes the following objectives:

- Develop a Level 3 PRA, generally based on current state-of-practice methods, tools, and data that (1) reflects technical advances since completion of the NUREG-1150 studies and (2) addresses scope considerations that were not previously considered.
- Extract new risk insights to enhance regulatory decisionmaking and to help focus limited agency resources on issues most directly related to the agency's mission to protect public health and safety.
- Enhance PRA staff capability and expertise and improve documentation practices to make PRA information more accessible, retrievable, and understandable.
- Obtain insight into the technical feasibility and cost of developing new Level 3 PRAs.

Research Approach

The scope of the Level 3 PRA project includes the major radiological sources onsite (i.e., both reactor units, both spent fuel pools, and dry cask storage), considered both individually and in terms of integrated site risk; all modes of reactor operation; and all internal and external hazards (excluding malevolent acts). Consistent with the objectives of this project, the Level 3 PRA study is generally based on current state-of-practice methods, tools, and data and is only pursuing new research in a few limited cases (e.g., multi-unit risk).

Based on a set of site selection criteria and with the support of the utility, Southern Nuclear Operating Company's Vogtle Electric Generating Plant, Units 1 and 2 was selected as the volunteer site for the Level 3 PRA study. The Level 3 PRA project team is leveraging the existing and available information on Vogtle and its licensee PRA in addition to related research efforts (e.g., SOARCA) to enhance efficiency in performing the study.

The Level 3 PRA project team is using the following NRC tools for performing the Level 3 PRA study:

- Systems Analysis Programs for Hands-on Integrated Reliability Evaluation (SAPHIRE), Version 8, which is the NRC's standard software application for performing PRAs.
- MELCOR Severe Accident Analysis Code, for modeling the progression of postulated accidents in both light-water reactors and in non-reactor systems.
- MELCOR Accident Consequence Code System, Version 2 (MACCS), for evaluating the public health effects and economic costs of mitigation actions for severe accidents at diverse reactor and non-reactor facilities.

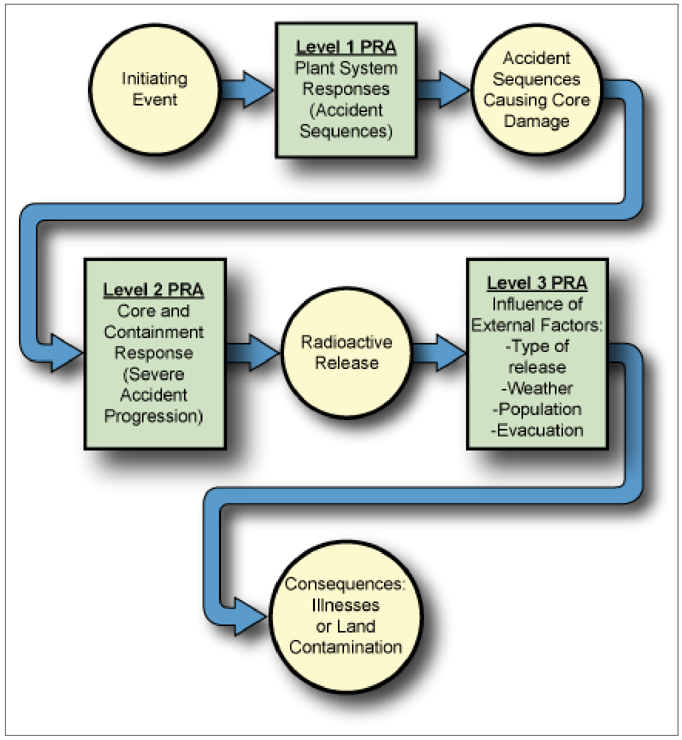

Figure 6.2 Level 3 PRA Analysis.

Status

Initial reactor, at-power, PRA models for internal events and internal floods (Level 1 and Level 2), high winds (Level 1), and "other hazards" have been completed and subjected to a PWR Owners Group (PWROG)-led peer review based on the ASME/ANS PRA standards. Initial reactor, at-power, PRA models for internal events and internal floods (Level 3); internal fires (Level 1); and seismic events (Level 1) are expected to be completed in 2015. In addition, an initial reactor, low power and shutdown PRA model for internal events and floods (Level 1) as well as a combined Level 1 and Level 2 PRA for dry cask storage are expected to be completed in or early 2016.

For More Information
Contact Alan Kuritzky, RES/DRA at Alan.Kuritzky@nrc.gov.

Probabilistic Risk Assessment Technical Acceptability and Standards

Objective

The objective of this activity is to define probabilistic risk assessment (PRA) technical acceptability so that there is confidence in the results of a PRA being sufficient for risk-informed regulatory decisionmaking and that the technical acceptability is commensurate with the activity (or decision) under consideration.

Research Approach

PRA technical acceptability is defined in Regulatory Guide (RG) 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, dated March 2009. A major goal of RG 1.200 is to eliminate the need for an in-depth review of licensee's base PRA allowing NRC reviewers to focus on key assumptions and areas identified during the peer review as a concern and relevant to the application. Consequently, RG 1.200 is meant to provide for a more focused and consistent review process. For PRA technical acceptability, RG 1.200 defines the scope of a base PRA to include Level 1, 2, and 3 analyses; at-power, low-power, and shutdown operating conditions; and internal and external hazards to support operating and new light-water reactors (LWRs). It also defines a set of technical elements and associated attributes that need to be addressed in a technically acceptable base PRA. Moreover, it provides guidance to ensure that a PRA model represents the plant at a component level of detail, incorporates plant-specific experiences, and reflects a realistic analysis of plant responses. Further, it includes a process to develop, maintain, and upgrade a PRA to ensure that the model represents the as-built, as-operated (or as-designed) plant.

RG 1.200 allows the use of consensus PRA standards and peer review methods endorsed by the NRC to demonstrate the technical acceptability of a base PRA. It provides guidance for an acceptable peer review process and peer reviewer qualifications, and it endorses the American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA standard and the Nuclear Energy Institute (NEI) peer review guidance documents with certain objections. A PRA used in an application needs to address the staff objections in RG 1.200, where applicable, if the PRA standard is to be considered met.

For PRA technical acceptability in support of regulatory applications, RG 1.200 recognizes that the needed PRA scope is commensurate with the specific risk-informed application under consideration. It also acknowledges that some applications may only use a portion of the base PRA, whereas other applications may require use of the complete model. In addition, it demonstrates an approach for technical acceptability of a PRA, independent of application. Inherent in this is the concept that a PRA need not only have the scope and level of detail necessary to support the application for which it is being used, but it also needs to be technically acceptable.

Status

ASME/ANS RA-Sa-2009 was published to support operating LWRs covering Level 1 large early-release frequency (LERF) PRAs for at- power conditions addressing both internal and external hazards. A new edition is expected to be published in 2016 that will address issues with internal events, internal flood, internal fires, and seismic events. Work is ongoing to extend ASME/ANS RA-Sa-2009 to low-power shutdown conditions and to support new LWRs. In addition, PRA standards for Levels 2 and 3 are under development.

NEI published NEI-00-02, "Probabilistic Risk Assessment Peer Review Process Guidance;" NEI-05-04, "Process for Performing Follow-on PRA Peer Reviews Using the ASME PRA Standard;" and NEI-07-12, "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines" that include a peer review process for a Level 1 LERF PRA for internal events and internal floods, PRA updates and upgrades, and fire PRA, respectively. NEI revised NEI-07-12 in June 2010 and published NEI-12-13, "External Hazards PRA Peer Review Process Guidelines," in August 2012.

Revision 2 to RG 1.200 endorses ASME/ANS RA-Sa-2009 and the NEI peer review guidance documents except for the revised NEI-07-12 and the new NEI-12-13. A draft Revision 3 to RG 1.200 is expected to be published in early 2016 to provide draft staff positions on the trial use standards for PRAs on Level 2, low power and shutdown, and advance LWRs; and the revised NEI-07-12 and the new NEI-12-13. Insights from the trial use of these standards will be incorporated into Revision 3 of RG 1.200. Once the trial use period has ended and ASME/ANS publish the standards for use, Revision 3 of RG 1.200 will be finalized providing the NRC's endorsement and conditions for use of the next edition of ASME/ANS RA-Sa-2009.

For More Information
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Treatment of PRA Uncertainties in Risk-Informed Decisionmaking

Objective

The objective of this activity is to provide guidance on how to treat uncertainties associated with probabilistic risk assessments (PRAs) used by a licensee or applicant to support a risk-informed application to the NRC. The guidance is intended for use by both the NRC staff and its licensees or applicants. Specifically, guidance is provided that addresses identifying and characterizing the uncertainties associated with PRA, performing uncertainty analyses to understand the impact of the uncertainties on the results of the PRA, and factoring the results of the uncertainty analyses into decisionmaking.

Research Approach

NRC guidance on the treatment of uncertainties is provided in NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking." NUREG-1855 describes an approach the NRC finds acceptable for how licensees or applicants address PRA uncertainties in the context of risk-informed licensing actions and how the impact of those uncertainties is evaluated by the NRC. In parallel with NRC efforts, the Electric Power Research Institute (EPRI) developed guidance on the treatment of uncertainties (EPRI 1016737, "Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments;" and EPRI 1026511, "Practical Guidance on the Use of PRA in Risk-Informed Applications with a Focus on the Treatment of Uncertainty"). The NRC and the EPRI guidance have been developed to complement each other and are intended to be used as such when assessing the treatment of uncertainties in PRAs used in risk-informed decisionmaking.

Factors addressed as the guidance was developed included the need to identify the different types of uncertainties, the treatment of uncertainty to be performed by the licensee or applicant, and how the staff accounts for the treatment of uncertainty in its decisionmaking. Generally, the two main types of uncertainty are aleatory and epistemic. Aleatory uncertainty (random or stochastic uncertainty) is based on the random nature of events or phenomena and cannot be reduced by increasing the knowledge of the systems being modeled. Epistemic uncertainty (state-of-knowledge uncertainty) is the uncertainty related to the lack of knowledge about or confidence in the system or model.

The guidance for the treatment of uncertainties has seven major stages. The first stage (Stage A) covers assessing the risk-informed activity and associated risk analysis to determine if the treatment of uncertainties should be based on the approach provided in NUREG-1855. This guidance generally involves understanding the type of application and the type of risk analysis and results needed to support the application. Stages B through F provide guidance for licensees on understanding the risk-informed application and determining the scope of the PRA needed to support the application, evaluating the completeness uncertainties and determining if bounding analyses are acceptable for the missing scope items, evaluating the parameter uncertainties, evaluating model uncertainties to determine their impact on the applicable acceptance guidelines, and developing strategies to address key uncertainties in the application. The last stage (Stage G) provides guidance for the NRC staff for evaluating the PRA for technical adequacy, determining if the uncertainties were adequately addressed, determining if the risk element of the risk-informed decisionmaking (in light of the uncertainties) is adequately achieved in the context of the application, and evaluating the licensee strategy for addressing the key model uncertainties resulting in exceeding the acceptance guideline (e.g., risk metrics).

Status

Revision 1 to NUREG-1855 is scheduled to be published by summer 2015.

For More Information
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SPAR Model Development Program

Objective

Standardized Plant Analysis Risk (SPAR) models are plant-specific NRC-developed probabilistic risk assessments (PRAs) that use standardized modeling conventions and data. This standardization allows agency risk analysts to efficiently use SPAR models for diverse plant designs in support of a variety of regulatory activities. The regulatory uses of the SPAR models include:

- Significance Determination Process (SDP) used to perform a risk-informed prioritization of inspection issues.
- Management Directive 8.3, "NRC Incident Investigation Program," assessments to determine an appropriate agency response to operational events.
- The Accident Sequence Precursor Program used to assess the risk significance of operational events and conditions.
- Generic Issues screening and prioritization.
- Special system and component studies.

Research Approach

The SPAR models allow agency risk analysts to perform independent evaluations of regulatory issues without reliance on licensee-developed models and analyses. 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 models give risk analysts the capability to quantify the expected risk of a nuclear power plant in terms of core damage frequency. More importantly, the models provide 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.

Currently, 75 SPAR models representing 99 operating commercial nuclear plants in the United States are used for analysis of the core damage risk from internal events at operating power. The SPAR models include core damage risk resulting from general transients, 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. Some of the SPAR models contain additional scope such as other hazard categories (e.g., fire, seismic, high winds); plant operating states (e.g., shutdown); or severe accident modeling (Level 2).

Status

The staff continues to develop new SPAR model capabilities and provide technical support for SPAR model users and risk-informed programs. The staff maintains and implements a quality assurance (QA) plan for the SPAR models to ensure that the models appropriately represent the as-built, as-operated nuclear plants to support the assessment of operational events within the staff's risk-informed regulatory activities. The SPAR QA Plan provides mechanisms for model benchmarking and reviews, validation and verification, and configuration control of the SPAR models. In addition, about half of the SPAR models are updated to reflect significant plant modifications or other plant or modeling changes in a typical year.

The staff also developed design-specific internal events SPAR models for new reactor designs such as the AP1000, the General Electric Advanced Boiling Water Reactor (ABWR), the Toshiba ABWR, the U.S. Advanced Pressurized-Water Reactor, and the U.S Evolutionary Power Reactor.

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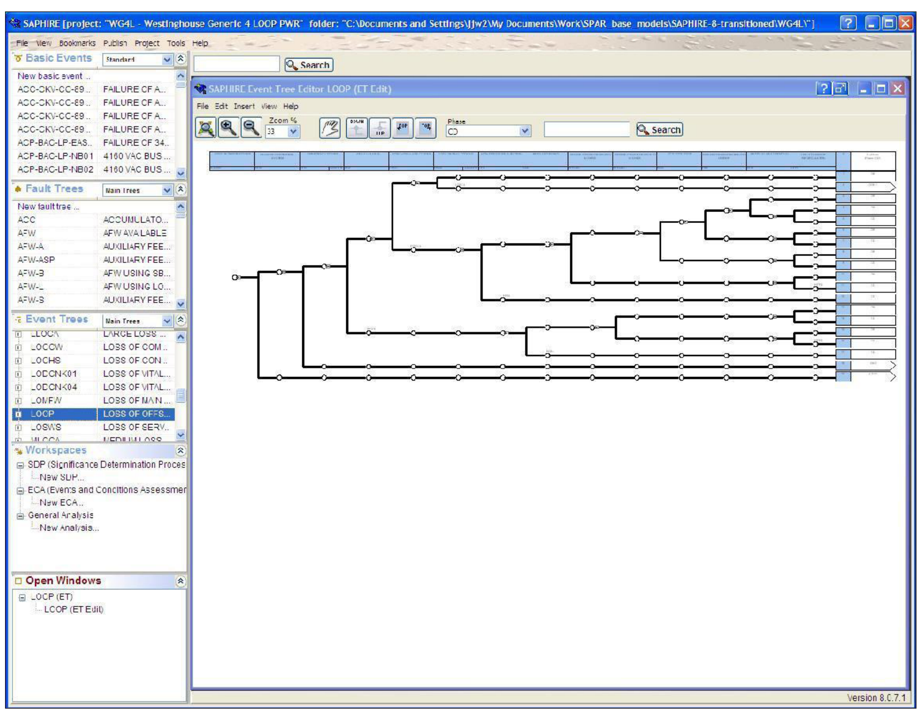

Figure 6.3 Example of loss-of-off-site-power SPAR model event tree display with SAPHIRE.

SAPHIRE PRA Software Development Program

Objective

The Systems Analysis Programs for Hands-on Integrated Reliability Evaluation (SAPHIRE) computer code is an NRC-developed probabilistic risks assessment (PRA) analysis tool that supports risk-informed regulatory activities. Although capable of modeling any technological risk in an event tree/fault tree framework, SAPHIRE is primarily used to model a nuclear power plant's response to events that could result in core damage, quantify the associated core damage frequencies, and identify important contributors to core damage (Level 1 PRA). It also can be used to evaluate containment failure and to characterize release of radioactive materials for severe accident conditions (Level 2 PRA). The objective of the SAPHIRE software development program is to provide a tool that performs risk calculations accurately and efficiently and reports the results in a clear and concise manner to support risk-informed decisionmaking.

Research Approach

SAPHIRE contains graphical editors for creating, viewing, and modifying fault trees and event trees. The graphical editors in SAPHIRE are used for creating the logical representations of accident scenarios that can occur at a nuclear power plant. One unique aspect of SAPHIRE, in comparison to other available PRA software, is the availability of features and tools to support event and condition assessments. SAPHIRE uses analysis modules called "workspaces." These workspaces assist the user with performing the analysis steps needed to assess the change in risk associated with the occurrence of an initiating event and/or degraded conditions. The workspaces assist the staff in producing accurate, consistent, and repeatable analyses to support NRC programs such as the Accident Sequence Precursor (ASP) program and the Significance Determination Process (SDP).

SAPHIRE uses the event tree and fault tree models, along with accident sequence linking rules and post processing rules, to generate unique combinations of individual failures that can cause core damage (for Level 1 PRA). These unique failure combinations are called minimal cut sets. SAPHIRE quantifies the frequencies and probabilities associated with the minimal cut sets to estimate a plant's total core damage frequency. SAPHIRE includes many useful features to support the quantification of PRA models and identification of significant contributors to risk. SAPHIRE calculates traditional PRA importance measures such as Fussell-Vesely, risk increase ratio or interval, risk reduction ratio or interval, and Birnbaum. SAPHIRE can be used to perform uncertainty analysis. Both Monte Carlo and Latin Hypercube sampling methods are available, and uncertainty analysis can be performed on importance measures.

Status

The Office of Nuclear Regulatory Research (RES) supports the ongoing maintenance and development of the SAPHIRE software. Areas of continuing development include improving the capabilities for reporting and documenting risk insights and results, exploring alternate quantification techniques for areas in which the typical approximations are challenged, and enhancing the ability to integrate different PRA model types (e.g., fire PRA, Level 2 PRA). In addition, work is currently underway to develop a Web-based version of SAPHIRE. This Web-based or "cloud" version will allow users to perform analyses on a remote server taking advantage of high-performance computing resources. The SAPHIRE developers have created a software quality assurance program to ensure that SAPHIRE continues to meet its requirements as new features and changes are implemented.

For More Information

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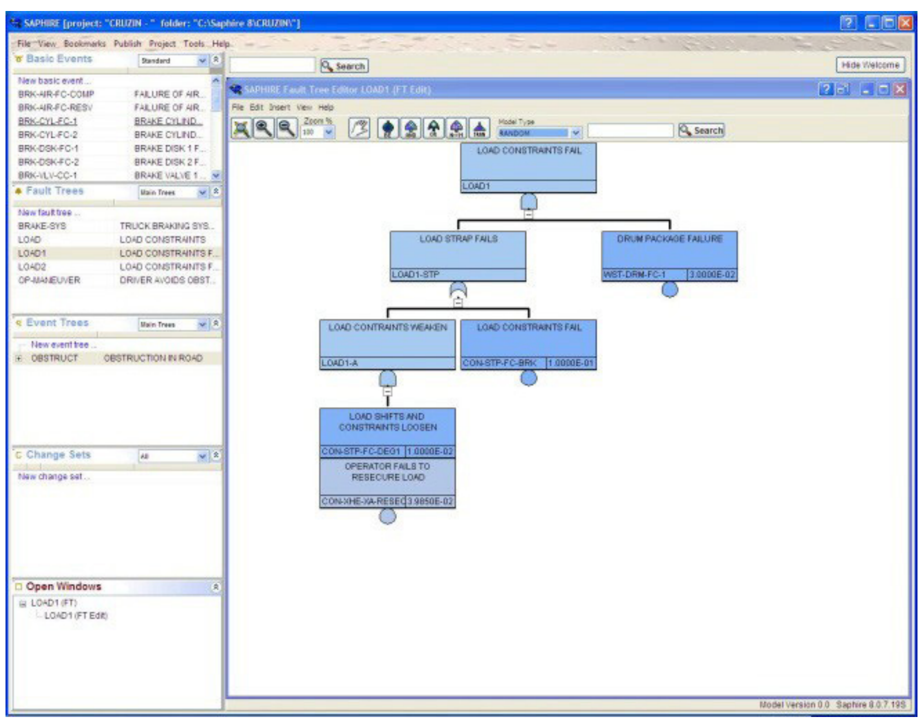

Figure 6.4 A graphical representation of a simple fault tree.

Thermal-Hydraulic Level 1 Probabilistic Risk Assessment (PRA) Success Criteria Activities

Objectives

The objectives of this project are:

- To perform thermal-hydraulic analyses that can update or confirm specific underlying assumptions in the agency's PRA (SPAR) models.
- To enhance in-house expertise and knowledge transfer for the purpose of improving the Office of Nuclear Regulatory Research's ability to consult to the program offices and regions on PRA modeling issues.
- To promote collaboration between thermal-hydraulic and PRA analysts.

Research Approach

Using a mixture of in-house and contractor capabilities, specific modeling aspects are identified, scoped, and analyzed. These analyses are then used as the technical basis for making changes (as needed) to the PRA models themselves. Examples of the type of issues that have been investigated to date include the following:

- Feed and bleed decay heat removal—the minimum number of pressurizer-power-operated relief valves and high-head pumps needed for small loss-of-coolant accidents, loss of a direct current bus, etc.
- Spontaneous steam generator tube rupture—time available for operators to mitigate the accident before core damage.
- Station blackout—time available to recover power.

Analysis for the Surry and Peach Bottom stations can be found in NUREG-1953, "Confirmatory Thermal-Hydraulic Analysis to Support Specific Success Criteria in the Standardized Plant Analysis Risk Models – Surry and Peach Bottom," September 2011.

In addition, a closely related study investigated modeling assumptions that affect success criteria findings: NUREG/CR-7177, "Compendium of Analyses to Investigate Select Level 1 Probabilistic Risk Assessment End-State Definition and Success Criteria Modeling Issues," May 2014.

Status

As of mid-2015, ongoing activities include:

- Analysis for the Byron station including small- and medium-break loss-of-coolant accidents, loss of a direct current bus, steam generator tube rupture, and loss of decay heat removal during shutdown operations – final NUREG to be issued in 2015.
- Analysis for the Vogtle station (Units 1 and 2), for a mix of issues of importance to the Vogtle Level 3 PRA project (SECY-11-0089), - documented in project documents to be issued at the completion of the Level 3 PRA project

For More Information
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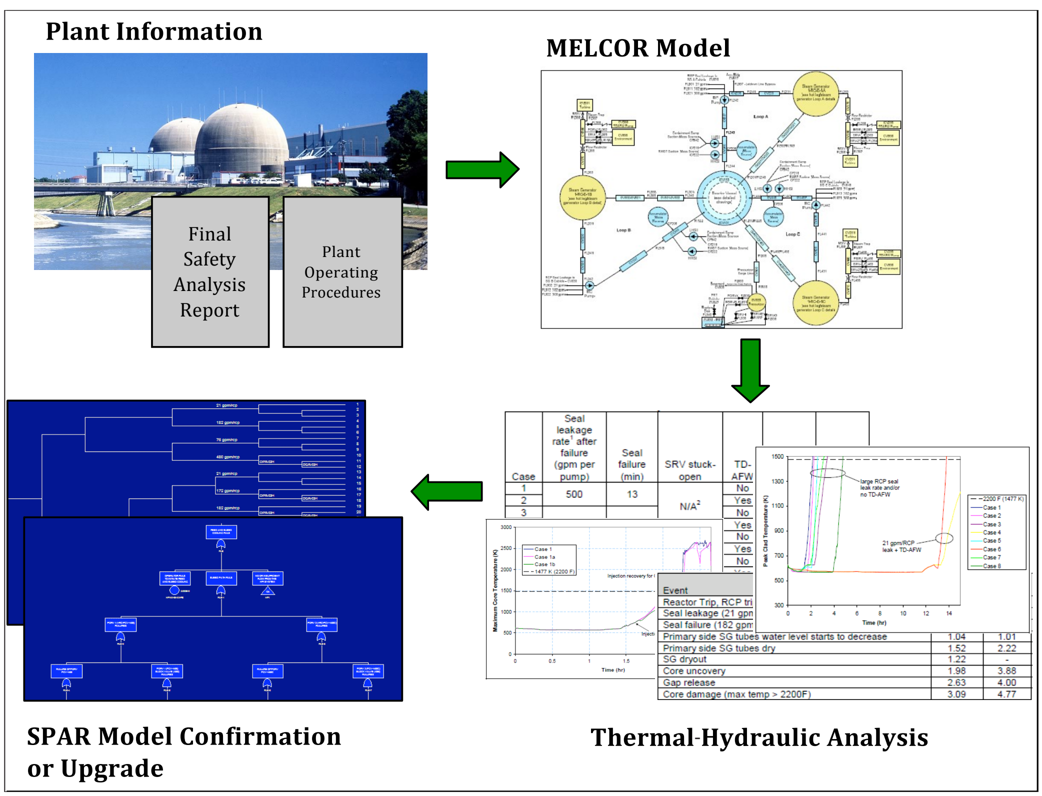

Figure 6.5 Level 1 Success Criteria Analysis.

Consequential Steam Generator Tube Rupture Program

Objective

Consequential steam generator tube ruptures (SGTRs) are potentially risk-significant events because thermally-induced steam generator tube failures caused by hot gases from a damaged reactor core can result in a containment bypass event and a large release of fission products to the environment. The main accident scenarios of interest are those that lead to core damage with high reactor pressure, dry-steam generator, and low-steam generator pressure (high-dry-low) conditions. A typical example of such an accident scenario is a station blackout with loss of auxiliary feedwater. The objective of this program is to develop a simplified methodology for the quantitative assessment C-SGTR probability and large early-release frequency (LERF) for pressurized-water reactors (PWRs).

Research Approach

Over the last two decades, the NRC and the nuclear industry have investigated the safety implications and risk associated with C-SGTR events (i.e., events in which steam generator [SG] tubes leak or fail as a consequence of the high differential pressures and/or elevated temperatures during accident sequences. Accidents involving SG tube ruptures have shown to be contributors to plant risk in various probabilistic risk assessments (PRAs) due to their potential for causing a release of fission products outside containment (containment bypass sequences).

It has been previously understood that the geometry of the steam generator reactor coolant inlet plenum region and the hot leg influences the temperature of the gases reaching the steam generator tubes during closed-loop-seal natural circulation conditions. Hotter gases reaching the steam generator tube reduce the time before tube failure, which increases the likelihood of containment bypass. To address C-SGTR risk, simplified PRA methods are being developed and applied to two representative PWR plants—a Westinghouse and a Combustion Engineering design. The study uses the latest available thermal-hydraulic analysis for the representative plants, updated SG tube flaw statistics pertinent to current in-service SGs, and enhanced calculation tools. A C-SGTR calculator containing the latest available model for estimating the failure probability/timings of SG tubes and other reactor coolant system RCS components (i.e., hot leg) and surge line) has been developed to improve the efficiency of the analysis.

Although the methodologies developed by this project are intended to provide a straightforward the screening approach, this method was developed in a manner that can establish the framework to perform a more comprehensive PRA that can address the C-SGTR at a level of detail suitable for other needs. Extension of these methodologies could support the risk-informed decisionmaking process and also be used to update the PRA Standards and PRA Procedure Guide.

Status

A draft report is being prepared to document the research results from this study. It is expected that the report will be issued for public review and comment in late calendar year 2015 and finalized in 2016.

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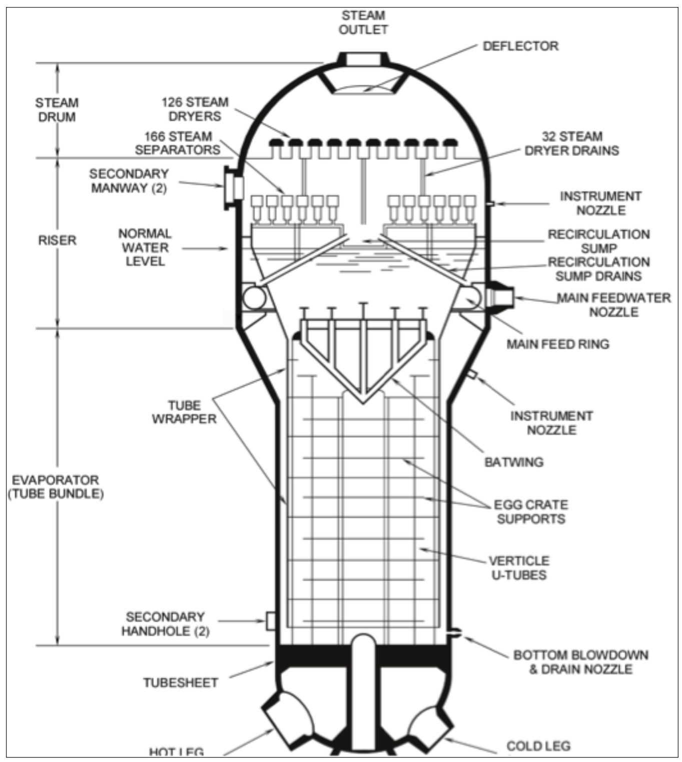

Figure 6.6 Combustion Engineering Steam Generator.

Risk Analysis Cooperative Research

Objective

The NRC's Office of Nuclear Regulatory Research (RES) develops and maintains state-of-the-art risk assessment methods, tools, data, and technical information to support the agency's safety mission and increasing use of risk-informed regulatory decisionmaking. To ensure risk analyses are performed using high-quality tools and data by the most knowledgeable researchers in a cost-effective manner, RES has developed numerous cooperative partnerships. These cooperative partnerships include other government agencies, universities, industry organizations, international regulators, and technical support organizations. By leveraging cooperative probabilistic risk assessment (PRA) research, the NRC is better able to ensure our research programs take advantage of state-of-the-art research results and ensure efficient use of our resources by avoiding overlapping research programs.

Research Approach

Domestically, RES actively participates with consensus standards organizations such as the American Society of Mechanical Engineers and the American Nuclear Society to promote the use of consistent guidelines for the building of PRA models for nuclear power plants. The agency staff participates in PRA standard working and writing groups in addition to providing grants to allow recognized experts to support the consensus standard development process. The agency often endorses these standards in regulatory guidance documents such as Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," to support risk-informed regulatory licensing decisions.

RES also maintains Memoranda of Understanding (MOU) with the Electric Power Research Institute (EPRI) and the National Aeronautics and Space Administration (NASA). The NRC-EPRI MOU recognizes that while research programs conducted by the NRC and EPRI may have differing objectives, results and data from these programs may have common value to both organizations. Therefore, the NRC-EPRI MOU has the overarching objective of avoiding unnecessary duplication of effort through sharing of information related to research programs of mutual interest whenever such cooperation and cost sharing is appropriate (e.g., when such cooperation does not represent a conflict of interest or compromise the NRC's role as an independent regulator). Areas of cooperation with EPRI have included developing PRA modeling approaches for support system initiating events and offsite power and development of guidance in addressing uncertainties. The NASA-NRC MOU supports the development of advanced risk analysis techniques and tools to support risk-informed decisionmaking. Areas of collaboration with NASA have included sharing of information related to digital system PRA modeling, human performance, fire risk, and staff training.

RES periodically provides research grants to universities to support state-of-the-art PRA method development. Recently, grants have been provided to the University of California, Los Angeles (UCLA); the Ohio State University (OSU); and the University of Maryland (UMD). The grant to UCLA is supporting the methodological and software enhancements of dynamic PRA platforms for event assessment applications. The goal of the UCLA work is to develop a more practical and realistic modeling tool for a number of regulatory applications, primarily event assessments and precursor studies. Recent work at OSU involved the development of an automated reliability prediction system software assessment tool. Cooperative work at UMD involves a study of the implications of multi-unit accidents in the context of NRC's Quantitative Health Objectives. The proposal's objective is to work toward addressing formal approaches to site-based risk assessments for multi-unit sites.

In the international arena, RES participates on the Organization for Economic Cooperation and Development (OECD) Nuclear Energy Agency (NEA) Committee for the Safety of Nuclear Installations (CSNI) Working Group on Risk Assessment (WGRISK). The main objective of WGRISK is to advance the understanding of PRA and to enhance PRA utilization for improving the safety of nuclear installations. WGRISK, through its support of risk-related issues, fosters continual improvement in the application of risk assessment methods by NEA member countries. WGRISK has been active in a number of critical PRA activities including human reliability, digital system reliability, low power and shutdown risk, external hazard and fire risk assessment, and use of operating experience data for PRA.

Status

By engaging in productive cooperative research partnerships, RES is able to take advantage of state-of-the-art domestic and international research results while efficiently targeting specific research needs. This supports the objective of developing and maintaining state-of-the-art methods, tools, data, and technical information in support the agency's safety mission.

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Chapter 7: Human Reliability Research

Consistent with the NRC's policy statement on the use of probabilistic risk assessment (PRA) and staff requirements memoranda (SRMs) for achieving an appropriate PRA quality for NRC risk-informed regulatory decisionmaking, the NRC has established a phased approach to PRA quality. This phased approach includes an action plan for stabilizing the PRA quality expectations and requirements to address PRA technical issues. Human reliability analysis (HRA) is an important PRA element. HRA is a structured approach used to identify potential human failure events and to systematically estimate the probability of those errors using data, models, or expert judgment.

The Commission identified the need for HRA data and models in the SRMs M061020, dated November 8, 2006, and M090204B, dated February 18, 2009. In SRM M061020, the Commission directed the Advisory Committee on Reactor Safeguards (ACRS) to work with the staff and external 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. In SRM M090204B, the Commission directed the staff to work with industry and international partners to test the performance of U.S. nuclear power plant operating crews and to keep the Commission informed of the status of its HRA data and benchmarking projects.

In response to the Commission's direction, the Office of Nuclear Regulatory Research (RES) developed the human performance data collection method and tool (i.e., Scenario Authoring, Categorization, and Debriefing Application), evaluated different human reliability methods, and participated in international and domestic HRA empirical studies to benchmark HRA models.

Outside the nuclear power plant arena, RES addresses needs from other NRC program offices related to HRA. For example, in 2011, the former Office of Federal and State Materials and Environmental Management Programs (now Office of Nuclear Material Safety and Safeguards) provided RES with a user need to develop HRA-informed materials for understanding and addressing potential human errors for medical application of byproduct materials.

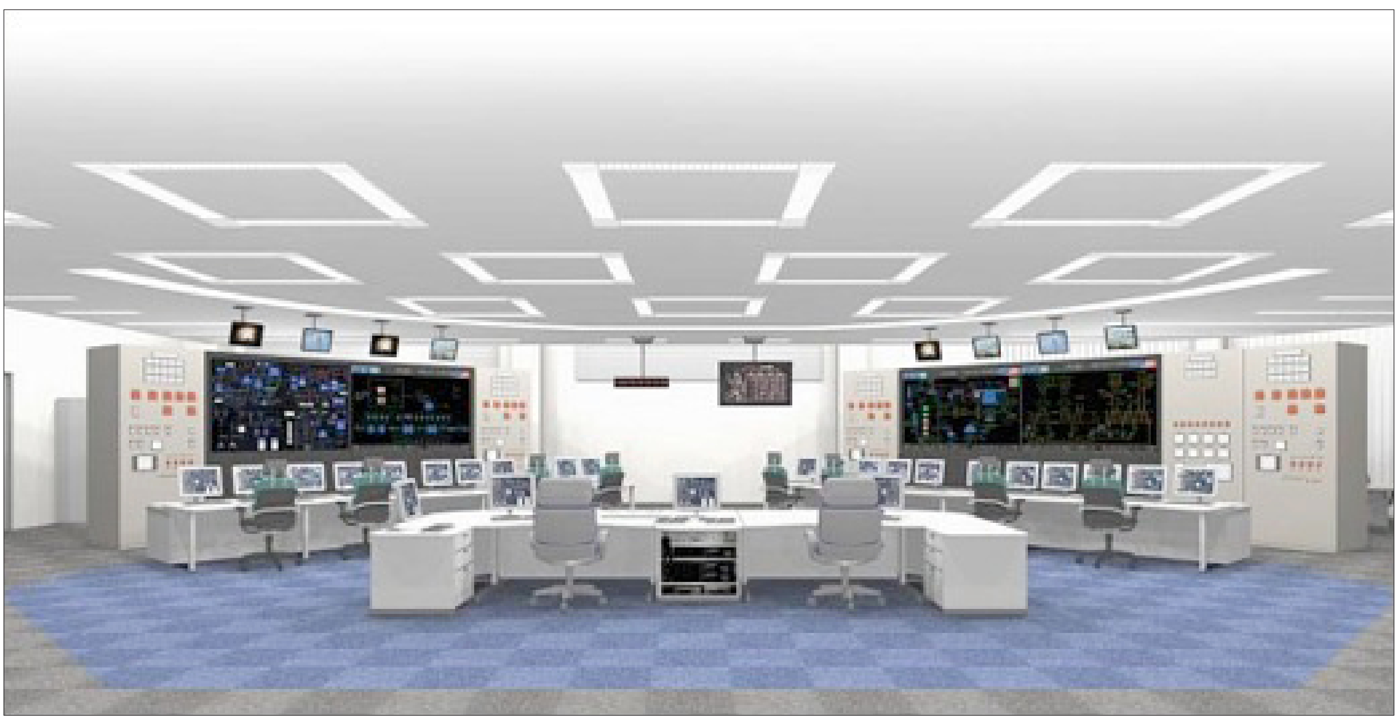

Figure 7.1 One conceptualization of an advanced control room design.

Human Reliability Analysis Data Repository

Objective

The NRC's Office of Nuclear Regulatory Research (RES) developed the human performance data collection method and tool (i.e., Scenario Authoring, Categorization, and Debriefing Application [SACADA]) with emphasis on collecting the licensed operator simulator training data to inform the human error probability (HEP) estimations in human reliability analysis (HRA)/probabilistic risk assessment (PRA). The objective is to collect a large quantity of licensed operator simulator exercise data to provide statistical indications on human reliability of performing various tasks inside the main control room.

Research Approach

The staff's approach is to use the similarity-matching concept to identify the empirical data that can be used to inform the HEPs of the human failure events of interest. The similarity matching is based on the situational profile in challenging nuclear power plant operators in detecting the cues of plant malfunctions, understanding the situations, making correct decisions, and executing correct actions with the additional consideration of team communication and supervision. This human-centered approach differs from the traditional task-centered or component-centered approaches (e.g., turn a switch) and allows combining data of different tasks with similar situational profiles to inform HEP estimates. This approach is expected to significantly increase the data usability.

A successful data collection program should include high data reliability and a long-term data collection to collect a large number of data for statistical indications. To achieve the objective of high data reliability, the SACADA data are entered by the plant staff (operator trainers and reactor operators) when the information is still fresh in the individuals' memories. The key SACADA human performance data can be divided into two types.

The first type of data is the situational or performance challenge profile, which is entered by the scenario designers (i.e., operator trainers). Each human task identified in the simulation scenario has its own situational profile that is represented by a set of performance-influencing factors whose states can be objectively identified. Therefore, the scenario designers could enter the data with high reliability.

The second type of data is the operators' performance results. The subset of this type of data includes the operators' performance in meeting the expectations and, if there are performance deficiencies, then the information related to the performance deficiency is collected. This type of data is entered by the plant operating crew during post simulation debriefing to ensure data reliability. For both types of data, the master set of factors are provided by SACADA for the operator trainers and operators to identify the most appropriate factors and factor statuses to characterize the situational profile and operator performance results.

To achieve the objective of long-term data collection, the strategy is to emphasize mutual benefits to the data providers and the NRC. The data providers are the plants' training department and the operations department, whose main interest is to improve human performance. The SACADA tool allows for the plants to replace their current tool in collecting operators' simulator performance information. Using SACADA to replace the plants' existing tools has not shown to increase the plant's training workload. In fact, the SACADA tool has streamlined data entry that, in turn, has reduced data entry effort for other plant training applications. The intent of using these features is to increase the likelihood that plants will collaborate by using SACADA in their operator simulator training. During routine operations, all data is entered by plant staff as part of their normal practices. The NRC only audits the data for data quality purposes. This strategy reduces uncertainty and the NRC's workload in maintaining a long-term data collection program.

Status

A U.S. nuclear power station has used the SACADA tool in its operator simulator training since 2012. The collected data are accessible to NRC under a bilateral agreement. The SACADA tool is also used by the Halden Reactor Project (HRP) to collect the data of operator simulator experiments. A few international research institutes have signed agreements with NRC to test the SACADA tool. The staff continues to outreach to domestic and international nuclear power companies for SACADA collaboration. The SACADA database has collected a sufficient amount of operator training and experiment data for demonstrating how the data would be used to inform HRA and human performance. An HRA data workshop was held in April 2015 to discuss how the SACADA data could be used to inform HRA and human performance and to share user experiences.

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Human Reliability Analysis Methods

Objective

The objective is to address the issue identified by the NRC in Staff Requirements Memorandum (SRM) M061020 regarding the use of different human reliability analysis (HRA) methods contributing to the variability of probabilistic risk assessment (PRA)/HRA results.

Research Approach

The research includes three parts: (1) develop a cognitive basis framework for HRA; (2) develop a stand-alone HRA method that reduces analyst-to-analyst variability for internal, at-power scenarios (referred to as "Integrated Human Event Analysis System" [IDHEAS]); and (3) develop a comprehensive HRA method that can be used for general HRA applications including external events, shutdown, and level-3 PRA. The cognitive framework serves as the basis for the HRA methods.

The cognitive framework, while developed as the technical basis for IDHEAS, is a stand-alone product. The staff conducted a literature review to document the understanding of the cognitive aspects of nuclear power plant (NPP) crew behavior in response to plant upsets based on research results and findings in cognitive psychology, human factors, and organizational behavior. A cognitive framework was developed to organize the results of cognitive research related to human performance in NPPs and to identify relevant performance influencing factors (PIFs) leading to crew failure. The framework presents the links between the PIFs, cognitive mechanisms, proximate causes of failure, and ultimately to macrocognitive functions. The development of the linkages is an important accomplishment of this work to bring the understanding of human performance underlying the HRA up-to-date. The outcome of the literature review and the cognitive basis framework for HRA were documented in NUREG-2114. The content of the report also serves as the technical basis for human factors and human- performance-related research and regulatory activities.

NRC staff collaborated with the Electric Power Research Institute (EPRI) under a memorandum of understanding between RES and EPRI on PRA to develop a stand-alone HRA method that reduces analyst-to-analyst variability for internal, at-power scenarios. The method, IDHEAS, integrates the strengths of existing methods and addresses the key sources contributing to analyst-to-analyst variability. The project team addressed the four key sources of variability by incorporating the following features in IDHEAS:

- Integrating qualitative analysis guidance in existing HRA methods and developing additional guidance for task analysis.
- Incorporating the cognitive framework of the mechanisms underlying human errors as the human performance model for HRA.
- Developing the IDHEAS human error quantification model based on the cognitive framework and experts' understanding of operator actions in internal, at-power scenarios; IDHEAS explicitly describes the assessment of PIFs and their effects on different types of human failures.
- Verifying the quantification model and estimating the base HEPs through an expert panel that consists of human factors/cognitive engineers, PRA/HRA analysts, and operational personal from U.S. NPPs; the estimation of base HEPs also used human error data from operational experience and human factors studies.

The NRC staff has also been developing an extensive version of the IDHEAS method to be used for general HRA applications. The extended method is based on the cognitive basis framework and models human errors in four basic cognitive functions: (1) detecting information, (2) understanding and assessing plant status, (3) making decisions and planning actions, and (4) executing planned actions. The method models a broad set of factors that may lead to human errors under various operating conditions from internal to external events, from at-power to shutdown, and the full span of Level-3 PRA. As the result, IDHEAS as an integrated HRA method can provide the following information for risk-informed decisionmaking:

- Operational narrative of imperfect, unexpected, and nontypical conditions that challenge human performance.
- Identification of human actions that may lead to undesired or unsafe plant status.
- Potential ways that crews may fail required actions.
- Performance-influencing factors that impact crew performance.
- Likelihood of personnel performing the actions.

Status

The cognitive framework report, NUREG-2114, is in the publication process and will be published in 2015. The staff is currently engaged in the improvement of IDHEAS for internal, at-power events and working on testing the method. The staff expects to complete the work by December 2015. The staff also has completed the initial development of the extensive IDHEAS method for general applications and will pilot the method in 2015.

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Using a Simulator to Improve Nuclear Power Plant Control Room Human Reliability Analysis

Objective

The objective of this study is to evaluate a specific set of human reliability analysis (HRA) methods used in regulatory applications by comparing HRA predictions to crew performance in simulator experiments performed at a U.S. nuclear power plant (NPP). The results will be used to determine the potential limitations of data collected in non-U.S. simulators when used to evaluate U.S. applications and improve the insights developed from the international HRA empirical study. This study is responsive to SRM M090204B.

Research Approach

The NRC established a memorandum of understanding with a U.S. NPP utility that volunteered to participate in this study and offered simulator facilities, operator crews, and expertise to support the design and execution of the experimental runs. This utility also has used the SACADA tool since 2012 (see Human Reliability Analysis Data Repository sheet). The study consists of the following four steps:

1. Experimental Design and Performance of Simulations

The experimental design is focused on collecting information on the predictive power and consistency of HRA methods. This effort involves analysis of crew performance in simulated NPP initiating events modeled in PRAs. It stipulates the collection of information to be used by HRA analysts to evaluate the HFEs involved in the scenarios and to estimate the HEPs.

The study provided the following information to HRA analysts for analyses: (1) the plant status before the initiating event, (2) the initiating event, and (3) the associated plant design capabilities and operational characteristics to deal with the event. The actual experiment consists of running of the accident scenarios and collecting and documenting observations about plant behavior and crew performance by experts (typically plant trainers and PRA/HRA experts). In addition to live observations, crew performance observations are collected through videotapes and debriefings of both the crews and the plant experts who observed the performance of the crews during the experiments.

The experimenters evaluate crew performance by analyzing the information collected during the experiment according to predefined protocols and performance metrics. This part of the study is supported by the staff of the Halden Reactor Project where the non-U.S. simulator data was collected.

2. Information Collection and Evaluation of HEPs by HRA teams

Each HRA method is applied by two or three HRA teams. The HRA teams interview plant personnel, observe operating crews in the simulator responding to simulated initiating events other than the study simulations, and collect relevant plant information. Based on the information collected, the teams use their selected HRA methods to perform predictive analysis and to estimate HEPs for the HFEs involved in the simulated scenarios, document the results, and submit them for review and evaluation.

One goal of the study is to understand the types of information considered by HRA teams in performing HRA analysis using a given method. Documenting this information provides insights about differences and commonalities among HRA methods; in particular, it helps staff to develop an understanding of how methods (or analysts) are using the collected information and of how the different ways of using information affect consistency among methods or analysts.

3. Evaluation of the HRA Submittals

An independent group of experts reviews the submitted analyses and compares them to the observed simulator data. These experts perform method-to-method and HRA team-to-team comparisons to determine if and how method differences and analyst differences influence the HRA results. Their analysis includes both qualitative and quantitative comparisons. Qualitative comparisons examine the extent to which HRA analysts were able to identify key drivers (such as misdiagnosis of equipment failures or lack of adequate procedural guidance for performing the required actions) that could influence the crew's capability to accomplish the required actions. Quantitative comparisons involve (1) the ranking of the estimated HEPs, (2) the ranking of the human actions in terms of the level of difficulty that crews appear to have experienced during the simulation, and (3) comparison of the resulting rankings in (1) and (2).

Status

A NUREG report will (1) document the results for each method tested, including the performance characteristics of each method and potential implications for regulatory applications and (2) assess the consistency of the methods and identify how practitioners can achieve better consistency in HRA. RES expects this report to be published by December 2015.

For More Information
Contact Y. James Chang, RES/DRA, at James.Chang@nrc.gov.

Potential Human Errors for Medical Applications of Byproduct Materials

Objective

The objective is to address a 2011 user need provided by the Office of Federal and State Materials and Environmental Management Programs (FSME) (now the Office of Nuclear Material Safety and Safeguards [NMSS]) to the Office of Nuclear Regulatory Research (RES). The user need requested RES to (1) develop a report of understanding human error in radiation therapy, (2) publish human reliability analysis (HRA)-informed training materials, and (3) demonstrate how to use the HRA-informed job aid through illustrative examples. NMSS provided the user need because human error has been identified as an important contributor to significant events across multiple technologies and industries including medical application of byproduct materials.

Research Approach

This work builds on an earlier user need, provided by NMSS to RES in 2003, to develop HRA capability specific to materials and waste applications. This earlier work was conducted in two-phases:

- 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.
- 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 in human performance in medical applications.

In this earlier work, 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- see Figure 7.2), were presented to FSME (now NMSS) staff and delivered to NRC in December 2008.

In all three cases, the products delivered to FSME (now NMSS) in December 2008 are the starting point for new development. However, new information and background material will be added as needed and appropriate for the first two products. The illustrative examples of how to use the job aid will be developed with NMSS staff input and guidance.

Status

RES is currently working on a draft of the NUREG on understanding human error in radiation therapy. RES plans to have this NUREG ready for publication in 2015.

For More Information
Contact Susan E. Cooper, RES/DRA, at Susan.Cooper@nrc.gov.

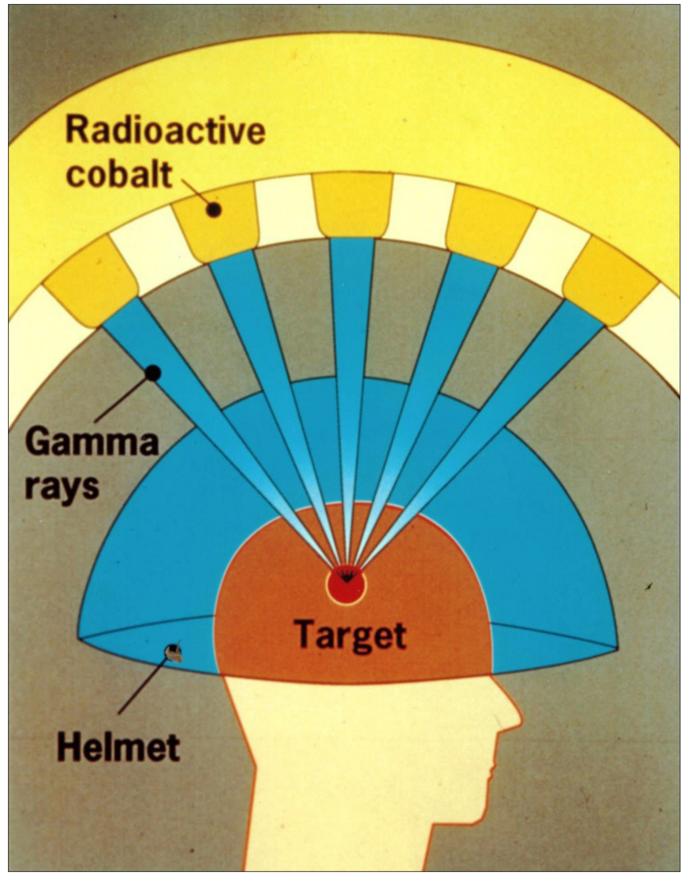

Figure 7.2 Graphic of Gamma Knife.

Human Reliability Cooperative Research

Objective

As part of its efforts to improve human reliability analysis (HRA) performed as part of probabilistic risk assessments (PRAs), the NRC's Office of Nuclear Regulatory Research (RES) participated in and supported the International HRA Empirical Study to benchmark HRA models. The objective of the International HRA Empirical Study was to develop an empirically based understanding of the performance, strengths, and weaknesses of the various HRA methods used to model human response to accident sequences in PRAs.

Research Approach

The International HRA Empirical Study was a multinational, multi-team effort supported by the Organization for Economic Co-Operation and Development Halden Reactor Project (HRP). The HRP provided facilities (i.e., the HAlden huMan-Machine LABoratory [HAMMLAB]), crews, and expertise to collect and analyze simulator crew performance data. HRA analyst teams from multiple organizations used their preferred HRA methods to analyze and predict the performance of those crews to certain initiating events modeled in nuclear power plant PRAs. The results of the predictions were compared to actual operating reactor control room crew performance in the simulator.

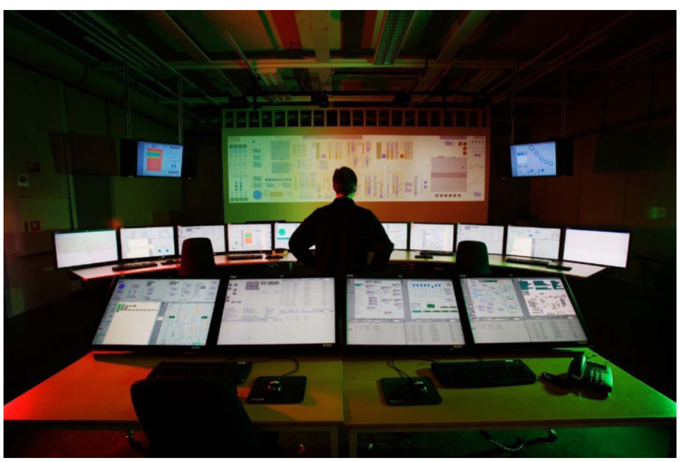

Figure 7.3 HAMMLAB Control Room Simulator at Halden.

The study was structured in three phases. The results of these phases were documented in the following reports:

- NUREG/IA-0216, Volume 1, "International HRA Empirical Study – Phase 1 Report, Description of Overall Approach and Pilot Phase Results from Comparing HRA Methods to Simulator Performance Data," November 2009.
- NUREG/IA-0216, Volume 2, "International HRA Empirical Study – Phase 2 Report, Results from Comparing HRA Method Predictions to Simulator Data from [Steam Generator Tube Rupture] Scenarios," August 2011.
- NUREG/IA-0216, Volume 3, "International HRA Empirical Study – Phase 3 Report, Results from Comparing HRA Methods Predictions to HAMMLAB Simulator Data on [Loss of Feedwater] Scenarios," December 2014.

Status

The overall conclusions and lessons learned from the International HRA Empirical Study have been documented in NUREG-2127, "The International HRA Empirical Study: Lessons Learned from Comparing HRA Methods Predictions to HAMMLAB Simulator Data," August 2014.

For More Information
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Chapter 8: Human Factors Research

Humans are integral to the safe operation of a nuclear power plant (NPP). In the late 1970s, the NRC began to focus on both protecting and ensuring adequate training of plant staff to perform their assigned tasks. The NRC studied factors affecting performance such as the effects of shift work on health and whether control room simulators would improve training. The Office of Nuclear Regulatory Research (RES) continues to look at these factors affecting human performance.

Currently, the nuclear power community is modernizing control room designs; building new plants, which have automated computer-based control rooms; and designing advanced reactors to support power generation for decades to come. The new generation of plants will differ from the existing fleet in several important ways including the reactor technology, the design of the instrumentation and control (I&C) systems, and the types of human-system interfaces (HSI).

The introduction of new NPPs will bring about a host of changes, including new technology and tools to support plant personnel and adjustments to plant staffing configurations. Moreover, the old analog control panels (Figure 8.1) will be replaced by computer-based human-system interfaces that will be used for process and component control. These new digital workstations change the analog spatially dedicated and continuously visible instrument and control (I&C) design to one that no longer has all the I&C elements necessary to support operator interaction immediately available and visible at all times. This change from parallel to serial information display and component control increases the opacity of the interface, further restricting the HSI with regard to the efficiency of navigation and timely access to the required information and to the means of control. If the new technology is being used to replace tasks that were previously done by the operators, as is often the case with automation, the operators now are presented with a different job that includes supervising the automation. However, if implemented well, HSI can be enhanced by digital I&C through organizing the information presented to operators in ways that are more useful with better context.

Taken together, these technological advances will lead to concepts of operation and maintenance that are different from those found in currently operating NPPs. The potential benefits, as explained above, of the new technologies should result in more efficient operations and maintenance. However, if the technologies are poorly designed and implemented, they will potentially reduce human reliability, increase errors, and negatively impact human performance—resulting in a detrimental effect on safety. For these reasons, it is important that the potential impact of these developments is evaluated and understood by prospective operators and regulators responsible for determining the acceptability of new designs to support human performance and maintain plant safety.

In addition to the work related to human performance in control rooms, RES supports activities related to fitness-for-duty (FFD) programs and safety culture. The NRC requires certain licensees to have an FFD program to provide reasonable assurance that licensee personnel (1) are trustworthy; (2) will perform their tasks in a reliable manner; (3) are not under the influence of any substance, legal or illegal, that may impair their ability to perform their duties; and (4) are not mentally or physically impaired from any cause that can adversely affect their ability to safely and competently perform their duties. The Safety Culture Policy Statement (76 Federal Register (FR) 34773, June 14, 2011) provides the Commission's expectation that individuals and organizations establish and maintain a positive safety culture commensurate with the safety and security significance of their activities and the nature and complexity of their organizations and functions.

Lastly, RES participates and supports the Working Group on Human and Organisational Factors (WGHOF) of the Nuclear Energy Agency. The main mission of the WGHOF is to improve the understanding and treatment of human and organizational factors within the nuclear industry to support the continued safety performance of nuclear installations and to improve the effectiveness of regulatory practices in member countries.

Figure 8.1 Human-System Interface in the Control Room.

Human Performance for New and Advanced Control Room Designs

Objective

To address the concerns related to new and advanced control room (CR) designs, the NRC sponsored a study to identify and prioritize human performance research that will be needed to support technical basis development and the corollary review of licensees' implementation of new technology in new and advanced nuclear power plants (NPPs).

Research Approach

Current industry trends and developments were evaluated in the areas of reactor technology, instrumentation and controls (I&C) technology, human-system interface (HSI) integration technology, and human factors engineering (HFE) methods and tools. These four broad research areas were then organized into seven HFE topic areas:

1. Role of personnel and automation.
2. Staffing and training.
3. Normal operations management.
4. Disturbance and emergency management.
5. Maintenance and change management.
6. Plant design and construction.
7. 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 areas, which resulted in 64 issues distributed among four categories with 20 research issues placed into the top priority category. NUREG/CR-6947, "Human Factors Considerations with Respect to Emerging Technology in Nuclear Power Plants," dated October 2008, documents the results of the study. The findings from this study are being used to develop a long-term research plan addressing human performance within these technology areas for the purpose of establishing a technical basis from which regulatory review guidance can be generated. The three projects that are underway are provided below.

Advances in Human Factors Engineering Methods and Tools

The outcome of this project to date has been the development of detailed review guidance for applying human performance models to the evaluation of NPP designs.

Roles of Automation and Complexity in Control Rooms

The present study will further the state of the art by examining the impact of automation on CR design, specifically the impact of automation on (1) operator performance during normal, abnormal, and emergency operations; (2) the reliability of operator's use of automation systems including existing methods for assessing impacts; and (3) operator performance when the automation fails or is in a degraded state.

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." NUREG-0800 references NUREG-0711, Revision 2, "Human Factors Engineering Program Review Model," and NUREG-0700, Revision 2, "Human-System Interface Design Review Guidelines," as the acceptance criteria. NUREG-0711, Rev. 2 and NUREG-0700, Rev. 2 were published in 2004 and 2002, respectively. The guidance is benefitting from further updates to keep pace with the modern I&C systems and advanced levels of automation that will be found in the next generation of NPP control rooms. Keeping the guidance up-to-date reduces the uncertainty, both for vendors and plant owners who worry about the acceptability of such systems to the regulators, as well as for the regulators who would have up-to-date technical bases on which to judge the acceptability of the new highly integrated control rooms employing state-of-the-art digital system designs.

Status

In addition to the technical report previously discussed under the Methods and Tools research area that presents guidance for applying human performance models to the design and evaluation of NPPs, two additional reports are in the process of being developed. The first report is on integrated system validation, and the second one is on cognitive task analysis. Both reports should be issued by the end of 2015. Under the Automation and Complexity research area, two technical reports are currently available—one on human-system interfaces for automatic systems and the other on the effects of degraded digital I&C control systems on operator performance. Under our HFE Regulatory Guidance Update program, NUREG-0711 was revised and published in 2012. Due to its size, NUREG-0700 is being updated in two phases. The phase 1 update is nearly complete and should be issued by the end of 2015, with the phase 2 update to follow.

For More Information
Contact Stephen Fleger, RES/DRA, at Stephen.Fleger@nrc.gov.

Human Performance Test Facility Research

Objective

The objective of this work is to conduct research assessing the impact of new designs on human performance with a larger and lower cost research subject pool as a supplement to the research being performed at the Halden Reactor Project.

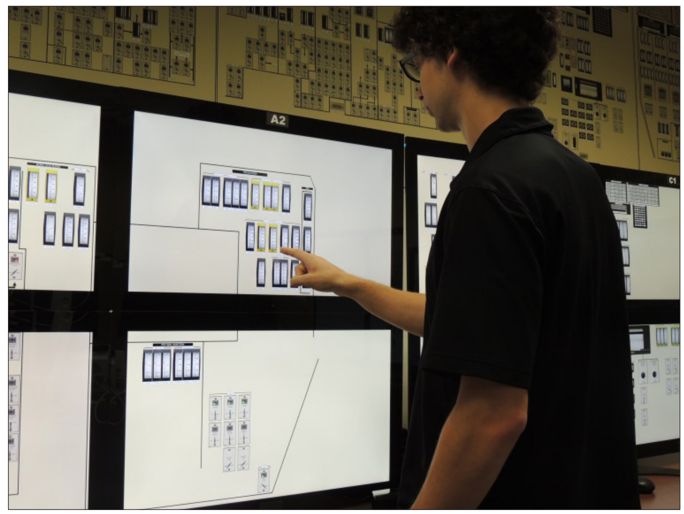
Research Approach

To meet this objective, the NRC's Office of Nuclear Regulatory Research (RES) recently procured two copies of a desktop computer-based nuclear control room simulator to conduct this research—one copy is housed at NRC headquarters and the other is at the University of Central Florida (UCF) under contract with the NRC.

The simulators have the following characteristics:

- Generic pressurized-water reactor.
- Westinghouse, 3-Loop.
- RETACT thermal-hydraulics code.
- Reprogrammable analog panel, soft controls, digital interfaces.
- Supporting documents (e.g., procedures, tech specs).

The NRC and UCF are working together to design and conduct human-in-the-loop experiments. This research is expected to produce nuclear-specific human performance data that aid in the evaluation of prioritized issues identified in NUREG/CR-6947, "Human Factors Considerations with Respect to Emerging Technologies in Nuclear Power Plants." These issues include the impact that new designs, technologies, and concepts of operations have on human performance.



Figures 8.2 and 8.3 NRC simulation facility at the University of Central Florida.

Status

The information gained will be incorporated in updates to the NRC staff's human factors review guidance NUREG-0700, "Human-System Interface Design Review Guidelines;" NUREG-0711, "Human Factors Engineering Program Review Model;" and in updates to the NRC's Human Reliability Analysis method development initiatives.

For More Information
Contact Amy D'Agostino, RES/DRA, at Amy.DAgostino@nrc.gov, or Niav Hughes, RES/DRA, at Niav.Hughes@nrc.gov.

Fitness for Duty

Objective

The objective of this work is to support the NRC regulatory offices in the development of the technical basis for rulemaking and implementation of the Fitness for Duty rule, Title 10 of the Code of Federal Regulations (10 CFR) Part 26, “Fitness for Duty Programs.”

Research Approach

The NRC’s Office of Nuclear Regulatory Research (RES) participates in and provides technical support to several working groups engaged in Fitness for Duty (FFD) rulemakings and program implementation. Two main initiatives related to Part 26 are described below.

Fatigue

RES is assisting other office to address several petitions for rulemaking related to the fatigue management provision of Part 26 and has worked on developing guidance for implementing the fatigue management requirements. Specific to research, RES has been looking at new methods to manage fatigue in the workplace and technologies assessing fatigue as well as other possible types of impairment.

Drug and Alcohol Testing

RES continues to evaluate the latest advancements in the area of drug and alcohol testing. The latest topics of interest have included the use of alternate specimens such as breath and saliva for testing. This is following the recent policy adoptions of these new testing methods in the private sector and by the Department of Health and Human Services. In addition to rulemaking support, RES has been assisting in the development of regulatory guidance that describes the methods that the NRC staff considers acceptable for complying with the drug testing provisions in Part 26.

Status

The results from the drug and alcohol initiatives will be published as a NUREG/CR in the ongoing series of technical basis reports the NRC has periodically published since the FFD rule was first implemented in the early 1990s. RES will continue to support rulemaking activities on fatigue and update guidance documents for implementing the fatigue management requirements.

For More Information

Contact Valerie Barnes, RES/DRA, at Valerie.Barnes@nrc.gov or DaBin Ki, RES/DRA, at DaBin.Ki@nrc.gov.

Figure 8.4 Operators in a NPP control room.

Safety Culture

Objective

The objective of this work is to provide technical expertise related to human and organizational performance to support the NRC's safety culture activities.

Research Approach

The culture of an organization affects the performance of the people in it. 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. The NRC has long recognized the importance of maintaining a positive safety culture in nuclear operations, most recently emphasized in the Commission's Safety Culture Policy Statement (76 FR 34773; June 14, 2011).

RES supports various safety culture activities including conducting research to understand the underlying relationship between safety culture and safety performance, reviewing methods for assessing safety culture, and developing educational materials to increase awareness and understanding of the importance of a positive safety culture.

RES staff also participates in the Safety Culture Advisory Committee led by the Office of Enforcement, which coordinates safety culture activities across the agency.

Status

Research on the relationship between safety culture and safety performance is documented in a technical report titled, "Independent Evaluation of INPO's Nuclear Safety Culture Survey and Construct Validation Study." This technical report can be found in the NRC's Agencywide Documents Access and Management System using the accession number ML12172A093.

Updates on safety culture activities and new educational materials can be accessed from the NRC's safety culture Web site at http://www.nrc.gov/about-nrc/safety-culture.html.

For More Information
Contact Stephanie Morrow, RES/DRA, at Stephanie.Morrow@nrc.gov.

Figure 8.5 Plant Maintenance Crew.

Figure 8.6 NRC Staff at Plant Control Room Simulator.

Figure 8.7 Nuclear materials scientist.

Human Factors Cooperative Research

Objective

As part of the NRC's ongoing participation with international partners, NRC participates and supports the Working Group on Human and Organisational Factors (WGHOF). NRC staff members ensure that activities remain aligned with NRC research goals and priorities and address the Organisation for Economic Co-operation and Development (OECD) Committee on the Safety of Nuclear Installations (CSNI) strategic priorities, potential safety issues, and topics.

Research Approach

The WGHOF is a working group under the OECD that focuses on human and organizational factors affecting safety at nuclear facilities. This group consists of representatives from over 20 countries and international organizations. The group also works on specific initiatives that are of interest to the members such as Human Intervention and Performance under Extreme Conditions, Establishing Desirable Attributes of Current Human Reliability Assessment Techniques, Human Performance Improvement Programs, Integrated System Validation, and Safety and Organizational Culture Influences on the Japanese Accident.

Status

The WGHOF meets two times a year, and the NRC supports and helps guide the cooperative research opportunities identified in these meetings.

For More Information
Contact Sean Peters, RES/DRA, at Sean.Peters@nrc.gov.

Chapter 9: Fire Safety Research

The results of the Individual Plant Examination of External Events (IPEEE) program conducted in the 1990s juxtaposed with actual fire events demonstrate that fire can be a significant contributor to nuclear power plant (NPP) risk. In particular, these studies show that failures of fire protection defense-indepth features can lead to risk-significant conditions. Fire protection programs in U.S. NPPs are based on the concept of defense-in-depth to achieve the required degree of fire safety. The three elements for fire protection are (1) prevent the fire from starting; (2) rapidly detect, control, and promptly extinguish those fires that do occur; (3) protect structures, systems, and components important to safety so that a fire not promptly extinguished by the fire suppression activities will not prevent the safe shutdown of the plant.

To address these lessons-learned about NPP fire risk, the Office of Nuclear Regulatory Research (RES) plans, develops, and manages the safety- and risk-related Fire Research Program for the NRC. Through this state-of-the-art program, RES supports other NRC offices by developing and validating fire analysis methodologies, tools, and supporting data. These include fire probabilistic risk assessment, fire human reliability analysis, and mathematical fire modeling to provide a structured, integrated approach to evaluate the impact of failure in the fire protection defense-in-depth strategy on safety. The staff then uses the results of its research activities as the basis for recommending improvements in NRC programs and/or processes to risk-inform regulations and achieve the desired outcomes of enhanced safety, efficiency, and effectiveness.

Figure 9.2 High Energy Arc Fault Testing of Electrical Components.

For example, in 2004, the NRC amended its fire protection requirement to allow existing reactor licensees to voluntarily adopt the risk-informed, performance-based requirements in Title 10 of the Code of Federal Regulations (10 CFR) 50.48(c). This rule endorses National Fire Protection Association Standard 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," as an alternative to the existing prescriptive fire protection requirements. RES staff is actively involved in developing the state-of-the-art methods, tools, data, and technical information required to implement these new requirements.

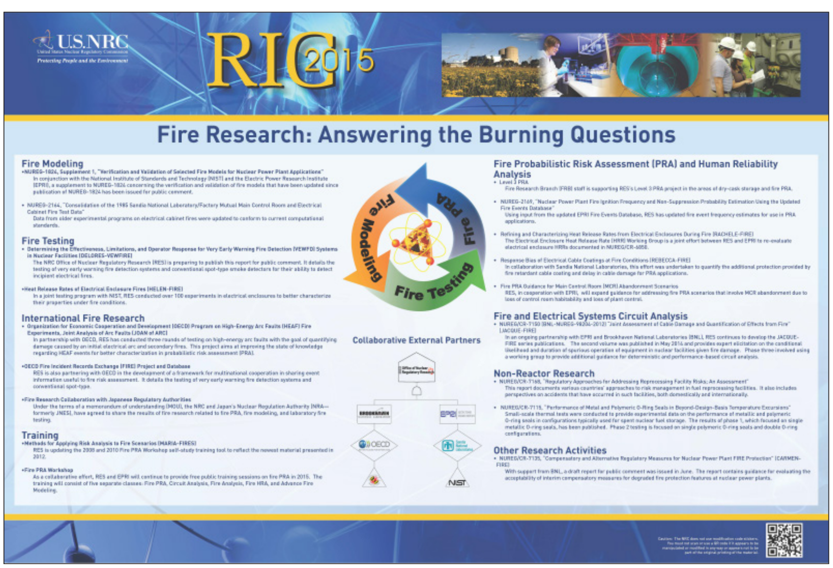


Figure 9.1 Fire Research Regulatory Information Conference (RIC) Poster.

The RES staff performs a variety of activities to establish a solid foundation for the agency's fire safety research and to support other NRC offices. Moreover, the RES staff supports the NRC's knowledge management initiative by training other NRC staff and by identifying and documenting relevant information.

In addition, RES staff works with both national and international fire research entities to assess and improve the agency's fire research program and to maintain a high level of expertise in the field. This work and cooperation provide a robust infrastructure for NPP fire research. The largest area of international cooperation in fire research is that on High Energy Arcing Fires with the Organisation for Economic Cooperation and Development (OECD) and the Nuclear Energy Agency (NEA).

Fire Probabilistic Risk Assessment Methodology for Nuclear Power Facilities

Objective

The primary objective of this research is to advance the state-of-the-art in fire probabilistic risk assessment (PRA) methods, tools, and data for use in regulatory decisionmaking.

Research Approach

In 2001, the Electric Power Research Institute (EPRI) and the NRC's Office of Nuclear Regulatory Research (RES), working under a memorandum of understand (MOU) on fire risk research, embarked on a cooperative project to improve the state-of-the-art in fire risk assessment to support this new riskinformed environment in fire protection. This project produced a consensus fire PRA document (NUREG/CR-6850 (EPRI TR-1011989), "EPRI/ NRC-RES Fire PRA Methodology for Nuclear Power Facilities," issued September 2005) that addresses nuclear power plant (NPP) fire risk for at- power operations. Plants making the transition to the rule, 10 CFR 50.48(c), rely on NUREG/CR-6850 (EPRI TR-1011989) to develop their fire PRAs whereas the NRC uses it to support reviews. The NRC, working with EPRI, has produced interim solutions to fire PRA issues raised by plants and EPRI related to NUREG/CR- 6850 (EPRI TR-1011989) in the NFPA Standard 805 frequently asked questions (FAQ) program and issued it as Supplement 1 to NUREG/CR-6850 in September 2010.

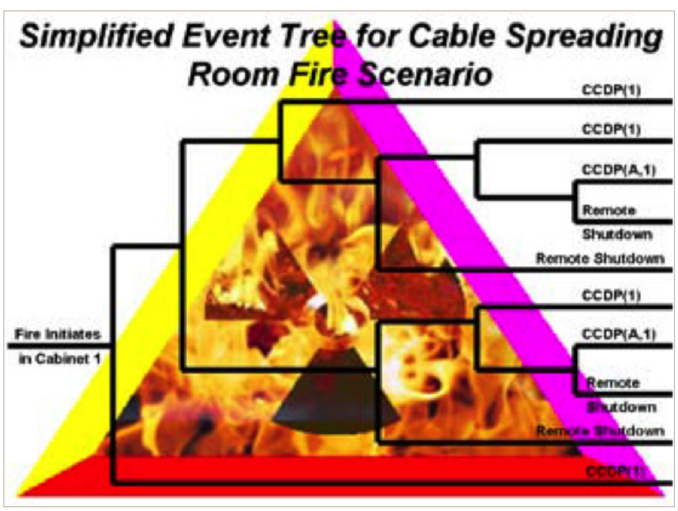


Figure 9.3 Simplified fire PRA event tree representing different sets of fire damage and plant response.

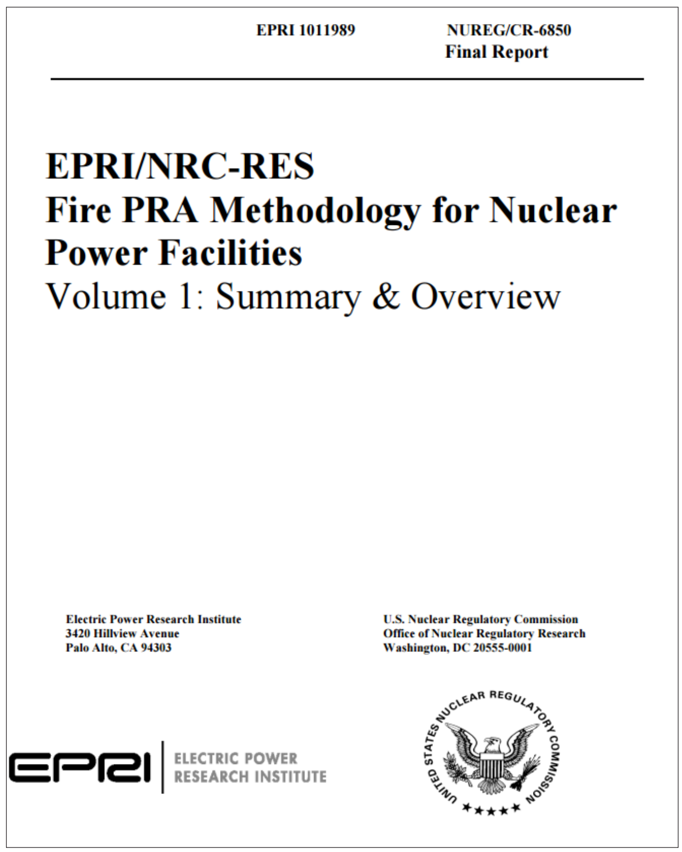
In addition, RES and EPRI have worked jointly to update and improve the fire events database used for NUREG/CR-6850 (EPRI TR-1011989). NUREG-2169 (EPRI 3002002936), "Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database, United States Fire Event Experience Through 2009" was published January 2015. RES also has developed fire PRA methods for low power and shutdown with EPRI serving as peer reviewers and supporting two tabletop plant exercises (see NUREG/CR-7114, "A Framework for Low Power/Shutdown Fire PRA.) Overall, this joint work is producing a significant convergence of technical approaches.

Status

Supplement 2 to NUREG/CR-6850 is in the working stages, and a revision to the joint report is in the planning stages as the methodology continues to mature and other fire research programs advance the state-of-the-art knowledge.

For More Information

Contact Nicholas Melly, RES/DRA, at Nicholas.Melly@nrc.gov.



EPRI 1011989 NUREG/CR-6850
Final Report

EPRI/NRC-RES
Fire PRA Methodology for Nuclear Power Facilities
Volume 1: Summary & Overview

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

ELECTRIC POWER RESEARCH INSTITUTE

UNITED STATES NUCLEAR REGULATORY COMMISSION

Figure 9.4 NRC/RES and EPRI published Fire PRA Methodology for NPPs in 2005.

Fire Human Reliability Analysis Methods Development

Objective

The overall objective of this effort is to develop fire human reliability analysis (HRA) methods beyond those currently in NUREG/CR-6850 (EPRI TR- 1011989), "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," and to develop an HRA methodology and approach suitable for use in a fire probabilistic risk assessment (PRA).

The intent of the fire HRA guidance developed through this effort is to support plants making the transition to 10 CFR 50.48(c) and NRC reviewers evaluating the adequacy of submittals from licensees making that transition. It may be used more generally for fire HRA in support of PRA.

Research Approach

The Office of Nuclear Regulatory Research (RES) has worked collaboratively with the Electric Power Research Institute (EPRI), under a memorandum of understanding (MOU) on fire risk research to develop a methodology and associated guidance for performing quantitative HRAs for post-fire mitigative human actions modeled in a fire PRA. In July 2012, the NRC and EPRI jointly issued NUREG-1921 (EPRI 1023001), "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines—Final Report."

NUREG-1921 identified several issues or areas requiring further research. One of those areas is treatment of scenarios requiring operators to abandon the main control room (MCR). More recently, industry introduced FPRA FAQ-13-0002, "Modelling of Main Control Room Abandonment" (April 2013), related to scenarios involving both loss of habitability (LOH) and loss of control (LOC) as reasons for MCR Abandonment. To address this FAQ, both NRC and industry worked to develop expanded guidance for addressing fire PRA scenarios that involve abandonment of the MCR. The NRC closeout memo for FPRA FAQ-13-0002 for MCR Abandonment due to LOH provided a version of this guidance. However, in closing FPRA FAQ-13-0002, both industry and NRC recognized that more detailed research is needed on this complex issue.

To address this need, NRC-RES and EPRI began working collaboratively in early 2015 to develop additional guidance for both LOH and LOC scenarios that result in MCR abandonment. This guidance will build upon that already provided in the joint EPRI/NRC-RES Fire Human Reliability Analysis Guidelines (NUREG-1921, EPRI 1023001) and in the closeout of FPRA FAQ-13-0002. The updated guidance is expected to be in the form of a joint NUREG-EPRI report (or reports) and issued as a supplement(s) to NUREG-1921 (EPRI 102 3001).

Status

RES and EPRI have begun efforts toward development of additional HRA guidance for MCR abandonment scenarios. The work is currently ongoing. RES and EPRI will continue to provide training on the use of HRA guidance as part of the NRC-RES/EPRI Fire PRA Workshop.

RES will continue to assist the Office of Nuclear Reactor Regulation (NRR) with the development of responses to NFPA 805 Frequently Asked Questions (FAQs) regarding HRA and will provide expert consulting as needed as NRR performs reviews of licensee submittals as well as support for other future activities that require fire HRA expertise.

For More Information
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Figure 9.5 Reactor Operators in a nuclear power plant main control room.

Fire Modeling Activities

Objective

The objective of this program is to provide methodologies, tools, and support for the use of fire modeling in nuclear power plant (NPP) applications.

Research Approach

In 2004, the NRC amended its fire protection requirements to allow existing reactor licensees to voluntarily adopt the fire protection requirements in National Fire Protection Association (NFPA) Standard 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," which allows licensees to use fire models as part of their fire protection programs. However, the fire models are subject to verification and validation (V&V), and the NRC must find them acceptable to ensure the quality and integrity of the modeling. To this end, the NRC's Office of Nuclear Regulatory Research (RES), the Electric Power Research Institute (EPRI), and the National Institute of Standards and Technology (NIST) conducted an extensive V&V study of fire models used to analyze NPP fire scenarios. This study resulted in the seven-volume report NUREG-1824 (EPRI 101 1999), "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications," issued May 2007.

The NRC and its licensees use the results in NUREG-1824 to provide confidence in the predictive capabilities of the various models evaluated. These insights are valuable to fire model users who are developing analyses to support a transition to NFPA Standard 805 to justify alternatives to existing prescriptive regulatory requirements and to conduct significance determination process reviews under the Reactor Oversight Process.

The NRC completed a phenomena identification and ranking table study of fire modeling (NUREG/CR-6978, "A Phenomena Identification and Ranking Table [PIRT] Exercise for Nuclear Power Plant Fire Modeling Applications," issued November 2008) that identified important fire-modeling capabilities needed to improve the agency's confidence in the results. This study helps define future research priorities in fire modeling.

Fire risk assessments often need to determine when cables will fail during a fire in NPPs. As part of the Cable Response to Live Fire (CAROLFIRE) program, the NRC and NIST have developed a simple cable damage model named Thermally Induced Electrical Failure (THIEF). NUREG/CR-6931, "Cable Response to Live Fire (CAROLFIRE)," issued April 2008, documents the test results and model. Volume 3 of CAROLFIRE describes how the THIEF 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 NRC benchmarked and validated the THIEF model against real cable failure and thermal data acquired during the CAROLFIRE program.

NIST has incorporated the THIEF model in both its two-zone and computational fluid dynamics models. In addition, the NRC incorporated the THIEF model in its fire dynamics tools spreadsheets. (See NUREG-1805, "Fire Dynamics Tools (FDTs) Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program," Supplement 1.) The THIEF spreadsheet is a useful tool for inspectors and licensees to quickly determine the likelihood of cable damage from a fire or to indicate the need for further analysis.

Recently, the NRC completed another joint project with EPRI and NIST to develop technical guidance to assist in the conduct of fire-modeling analyses of NPPs. NUREG-1934 (EPRI 1023259), "Nuclear Power Plant Fire Modeling Analysis Guidelines (NPP FIRE MAG)," issued November 2012, expands on NUREG-1824 by providing users with best practices from experts in fire modeling and NPP fire safety.

This application guide contains five commonly available fire modeling tools (FDTs, Fire-Induced Vulnerability Evaluation [Revision 1], Consolidated Fire Growth and Smoke Transport Model, MAGIC, and Fire Dynamics Simulator [FDS]) that were developed by nuclear power stakeholders or that were applied to NPP fire scenarios. Previously, RES, EPRI, and NIST used these same models in the V&V study documented in NUREG-1824.

NUREG-1934 will assist both the user performing the calculation and the reviewers. The report includes guidance on selecting appropriate models for a given fire scenario and on understanding the levels of confidence that can be attributed to the model results. The report also will form the foundation for future fire model training under development by RES and EPRI.

Status

A supplement to NUREG-1824 that evaluates the latest versions of the fire models and incorporated additional test data has been published for public comment. In addition, NUREG-1824, Supplement 1 includes V&V information for the THIEF model and other fire phenomena submodels that were not included in the original NUREG-1824 report. The NRC is continuing to update the fire modeling tools, expand V&V efforts, and develop additional model input data.

For More Information
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Cable Heat Release, Ignition, and Spread in Tray Installations during Fire

Objective

The Cable Heat Release, Ignition, and Spread in Tray Installations during Fire (CHRISTIFIRE) experimental program is an effort to quantify the mass and energy released from burning electrical cables. The program includes fire tests on grouped electrical cables to enable better understanding of the fire hazard characteristics including the ignition, heat release rate, and flame spread. The NRC will use this type of quantitative information to develop more realistic models of cable fires for use in fire probabilistic risk assessment (PRA) analyses such as those performed using the methods in NUREG/CR-6850 (Electric Power Research Institute (EPRI) TR-1011989), "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," issued September 2005 in applications under National Fire Protection Association Standard 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants."

Research Approach

Phase 1 of CHRISTIFIRE included experiments ranging from micro-scale to full scale. Small samples of cable jackets and insulation were burned within a calorimeter to measure the heat of combustion, pyrolysis temperature, heat-release capacity, and residue yield. Meter-long cable segments were slowly fed through a small tube furnace, and a variety of spectrometric techniques measured the composition of the effluent. The standard cone calorimeter test measured the heat release rate per unit area for a variety of cable types at several external heat fluxes.

A large radiant panel apparatus, specially designed for this test program, measured the burning rate of cables when installed in ladderback trays. Finally, a series of 26 multiple-tray full-scale experiments assessed the effect of changing the vertical tray spacing, tray width, and tray fill.

During Phase 1, the National Institute of Standards and Technology (NIST) along with the NRC developed a simple model of flame spread in horizontal tray configurations (called Flame Spread over Horizontal Cable Trays (FLASHCAT]) that makes use of semi-empirical estimates of lateral and vertical flame spread and measured values of combustible mass, heat of combustion, heat release rate per unit area, and char yield. NIST and the NRC completed Phase 1 in 2011 and documented the results in NUREG/CR-7010, "Cable Heat Release, Ignition, and Spread in Tray Installations during Fire (CHRISTIFIRE)—Phase 1: Horizontal Trays," Volume 1, issued July 2012.

Phase 2 of the CHRISTIFIRE project examined flame spread on cables in trays oriented in the vertical direction and the impact of an enclosure on cable flame spread in multiple horizontal trays. A series of 17 experiments were conducted using 2 vertical cable trays that were installed adjacent to each other. A series of 10 experiments were conducted using multiple horizontal trays located in a simulated hallway relatively close to the wall and ceiling. The results of these experiments, along with additional cone calorimeter measurements, will be used to extend application of the FLASHCAT model. The Phase 2 test results have been published in Volume 2 of NUREG/CR-7010, "Cable Heat Release, Ignition, and Spread in Tray Installations during Fire (CHRISTIFIRE)—Phase 2: Vertical Shafts and Corridors."

Status

CHRISTIFIRE was the first attempt in recent years at developing a more realistic understanding of the burning behavior of grouped cables. Based on its success, future phases of the project will examine the effectiveness of various methods of protection for electrical cables. The FLASHCAT model will be validated and extended to other configurations. The first two phases are complete. The third phase intended to identify minimum criteria necessary for cable ignition is currently underway. Additional phases of the project are currently under development.

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Figure 9.6 Burning cables during cable tray fire test (side view of burning cables in trays during a multiple-tray test after ignition using a small gas burner).

Fire Effects on Electrical Cables and Impact on Nuclear Power Plant System Performance

Objective

The NRC has conducted testing and sponsored several expert panels to support an enhanced understanding of fire-induced spurious operations and impacts on plant safety.

Research Approach

The NRC performed fire testing of dc circuits using representative configurations of safety-significant circuits and components used in nuclear power plants (NPPs) to better understand the probability of spurious actuations and the duration of those actuations in dc circuits. The DESIREEFIRE testing program used small- and intermediate-scale tests to evaluate the response of dc circuits to fire conditions. The tests included several different circuits as follows:

- Direct current motor starters.
- Pilot solenoid-operated valve coils.
- Medium-voltage circuit breaker control

The DESIREEFIRE project is another Office of Nuclear Regulatory Research (RES) fire research project established under a memorandum of understanding (MOU) to perform collaborative research with the Electric Power and Research Institute (EPRI). This agreement has provided various components and cabling to the DESIREEFIRE testing program at little or no cost to the NRC. It also provided industry expert advice on the various aspects of the dc power system and circuit design. Testing is complete, and NUREG/CR-7100, "Direct Current Electrical Shorting in Response to Exposure Fire (DESIREEFIRE)," issued in April 2012 documents the results.

Following the testing, the NRC, working with EPRI under the MOU, convened two separate expert panels. The first panel comprised several electrical engineering experts who reviewed all currently available testing data. This panel followed the NRC's phenomena identification and ranking table (PIRT) process to determine the state-of-the-art in predicting hot short-inducted cable failures when exposed to fire conditions. The results of this work are documented in NUREG/CR-7150, Vol. 1, "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE)," issued October 2012.

The second expert panel, again under the EPRI MOU, convened fire probabilistic risk assessment (PRA) experts to explore and advance the state-of-the-art in determining realistic conditional probabilities of hot short-induced spurious operations when cables are exposed to fire conditions. The results from this work are documented in Volume 2 of JACQUE-FIRE, issued May 2014.

In support of the phenomena PIRT exercise on fire-induced damage to electrical cables, the NRC, in collaboration with EPRI and Sandia National Laboratory, performed a comprehensive review of the three major fire-induced cable damage testing programs. The work used a graphical analysis approach to display the data in a manner that would identify trends on spurious operation likelihood and spurious operation duration. The analysis also shows that multiple cable shorts to ground can cause spurious operations resulting from an ungrounded and compatible power supply. NUREG-2128, "Electrical Cable Test Results and Analysis during Fire Exposure (ELECTRAFIRE)," was issued in February 2013.

Status

Preliminary areas for future research identified by the PIRT panel include evaluating the fire-induced effects on instrumentation circuits, electrical panel/cabinet wiring, surrogate ground path failure mode, current transformers, and high-conductor count trunk cables.

In addition, the results of the PIRT and expert elicitation projects will be used to update the state-of-the-art fire PRA methods and data in NUREG/CR-6850 (EPRI TR-1011989). A third phase of the JACQUE-Fire expert elicitation is currently in progress and will evaluate deterministic circuit analysis methods in addition to the PRA methods. The results of the PIRT report were published in October 2012 in NUREG/CR-7150, "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUEFIRE)—Final Report," Volume 1, "Phenomena Identification and Ranking Table (PIRT) Exercise for Nuclear Power Plant Fire-Induced Electrical Circuit Failure," issued October 2012. The expert elicitation results are documented in Volume 2 of NUREG/CR-7150, issued May 2014. Volume 3 of NUREG/CR-7150 is under development. Volume 3 documents the use of risk insights presented in Volume 2 to harmonize the results presented in Volume 1. Additional topics covered in Volume 3 include: design criteria for shorting switches; technical justification for final circuit failure positions; limit on hot short-induced spurious operation duration; limits on teh number of cable interactions considered credible for multiple spurious operation scenarios; and clarification of statements made in Volume 1.

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Beyond Design Basis Fires for Spent Fuel Transportation: Shipping Cask Seal Performance Testing

Objective

The objective of this test program is to explore the performance envelope of O-ring seals under beyond-design-basis thermal excursions (fire conditions) and to estimate package leakage rates under these conditions. The data can be used in the evaluation and analysis of finite-element computer models of spent fuel transportation packages during extra-regulatory fires such as those analyzed in NUREG/CR-6886, "Spent Fuel Transportation Package Response to the Baltimore Tunnel Fire Scenario."

Research Approach

In 2010, the Office of Nuclear Regulatory Research (RES) and the National Institute of Standards and Technology (NIST) started to perform small-scale thermal tests to gather data on the performance of O-ring seals used in spent nuclear fuel transportation packages. The tests described below were designed to explore how O-ring seals of different materials (e.g., metallic and polymeric) in different configurations (e.g., single O-ring and double O-ring) perform during these hypothetical fully engulfing fire conditions.

Pressure and temperature were monitored for several days before the test to ensure that the vessel had no leaks during the test to monitor for leakage. After the test, pressure and temperature were monitored for several days to achieve cool down and pressure stability.

Tests results have shown no catastrophic vessel leakage (e.g., loss of all vessel pressure) has occurred. Seal performance varied and small leaks occurred in several of the tests. The metallic seals (see Figure 9.7) were tested at about 800 degrees Celsius and experienced a small leak several hours into the test; however, the seals did not lose all pressure even after several days of cool down and pressure monitoring. The ethylene propylene seals were typically tested around 450 degrees Celsius. Even when a leak was detected, they also continued to hold pressure after cool down although the integrity of the ethylene propylene seal was clearly compromised (i.e., the seal had transformed into a powder-like material). The pressure boundary was most likely maintained because of tight clearances between the test vessel body and head. Other polymeric seals currently being tested include silicone based and fluoro-carbon polymer O-ring seals. Seals are also being tested in single and double O-ring configurations as found in typical spent fuel transportation packages.

Figure 9.7 Pictures of the small-scale test vessel after 800 degrees C exposure for 9 hours (small-scale test vessel [top left], vessel head after disassembly [top right], and vessel body and metallic seal after disassembly (bottom left and bottom right]).

Status

The results of the first phase of testing were published in April 2012 as NUREG/CR-7115, "Performance of Metal and Polymeric O-ring Seals in Beyond-design-basis Temperature Excursions." The results of the second phase of testing will be published in a revision to NUREG/CR-7115 in 2015.

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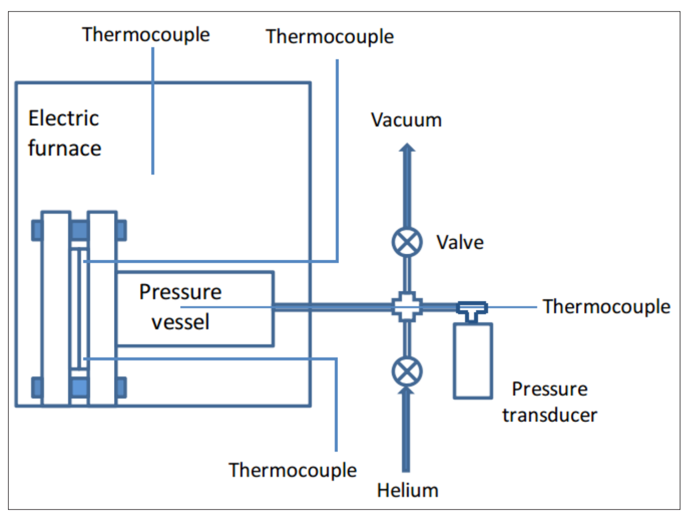

Figure 9.8 A schematic illustration of the experimental apparatus.

Evaluation of Very Early Warning Fire Detection System Performance

Objective

The research effort is related to testing and evaluating the relative performance of smoke-detection systems, including very early warning fire-detection (VEWFD) systems. The test data, operating experience, and human response supports a risk scoping study to allow the fire protection community to better understand how these systems can be used to rapidly detect actual and potential fire sources in nuclear power plant (NPP) applications.

Research Approach

The NRC staff elected to sponsor testing, conduct literature reviews, and visit both U.S. and foreign nuclear and nonnuclear sites to support its evaluation of this technology.

The testing included evaluating conventional spot-type detectors (ionization and photoelectric) and aspirated smoke detectors (ASDs) configured as VEWFD systems tested in three different scales (laboratory bench scale, small room, and large open areas). Variables in test parameters that influence detector response such as smoke source, ventilation rate, device location, and system configuration were evaluated during each scale of testing.

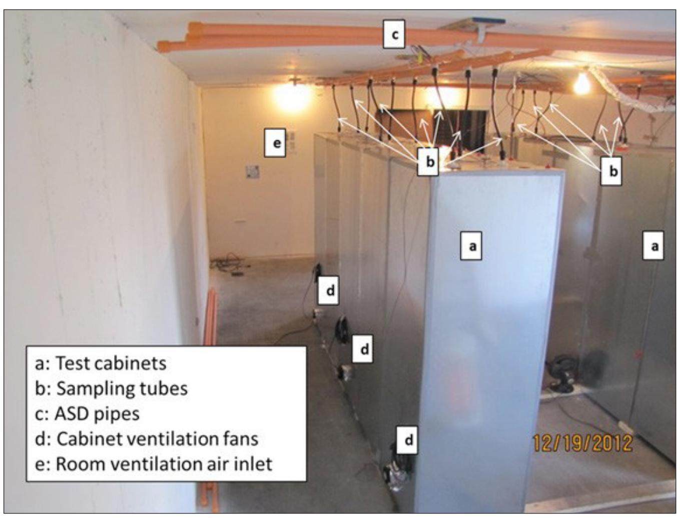

Figure 9.9 Fire test room configuration.

In addition to the confirmatory testing, site visits, operating experience reviews, and a comprehensive literature search were conducted to support an evaluation of the factors that affect the performance of ASD VEWFD system technology and any associated values assigned to the systems in fire probabilistic risk assessment (PRA) to evaluate preventing or detecting and suppressing fires.

The specific values used in the fire PRA as presented in the interim guidance makes an assumption that these systems will detect fires in their incipient stages prior to flaming combustion. This allows additional time for operators to locate the potential fire source and to remove power prior to a fire becoming a potential threat to reactor safety. Because of the human involvement in the fire PRA success scenario, human factors and human reliability engineering experts have been supporting this project and will provide guidance in the final NUREG report concerning system design and estimates on the human failure probability of preventing fire damage.

Status

A draft report has been prepared and issued in 2015 for public comment.

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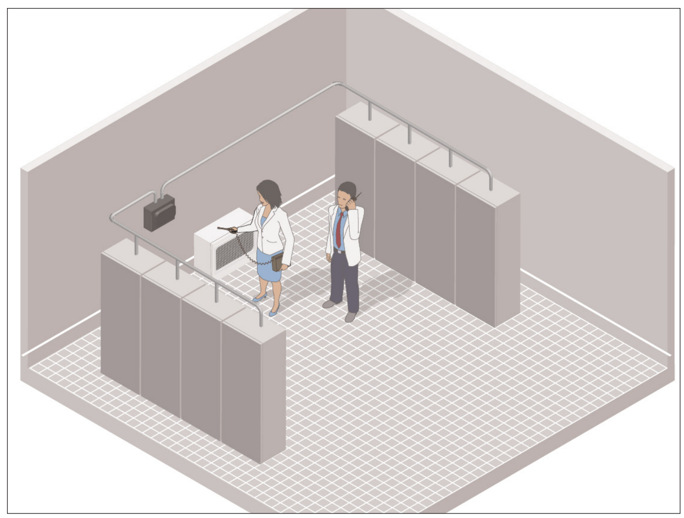

Figure 9.10 Illustration of operator response to aspirated smoke detection within an electrical enclosure.

OECD International Testing Program for High Energy Arc Faults (HEAF)

Objective

This project, originally called the Joint Analysis of Arc Faults (JOAN of ARC) Testing Program, was identified as part of the Organisation for Economic Co-operation and Development (OECD) fire events database program. Catastrophic failures of energized electrical equipment referred to as high-energy arcing faults (HEAFs) have occurred in nuclear power plant (NPP) components throughout the world. HEAF typically occur in 480V and higher electrical equipment and cause large pressure and temperature increases in the component electrical enclosure. These increases in pressure and temperature could ultimately lead to serious equipment failure and secondary fires and could put the NPP at risk. Figure 9.11 shows an example of HEAF damage.

Most recently, the United States has experienced events at Palo Verde Nuclear Generating Station in 2013, H.B. Robinson Steam Electric Plant in 2010, and Columbia Generating Station in 2009. Discussions at the OECD Fire Incidents records exchange meetings indicate similar HEAF events have recently occurred in Canada, France, Germany, and most recently at Japan's Onagawa NPP during the earthquake and tsunami of 2011. OECD Fire Project – Topical Report No.1, "Analysis of High Energy Arcing Fault (HEAF) Fire Events," NEA/CSNI/R (2013) published in June 2013 documents these international events.

HEAFs have the potential to cause extensive damage to the failed electrical component and electrical distribution system along with adjacent equipment and cables located in close proximity. This area is identified as the zone of influence (ZOI). The significant electrical energy released during a HEAF event can act as an ignition source to other components in this ZOI.

The primary objective of this project is to perform experiments to obtain scientific fire data on the HEAF phenomenon known to occur in NPPs through carefully designed experiments. The goal is to use the data from these experiments and past actual NPP events to develop an improved mechanistic model to account for the failure modes and consequence portions of HEAFs. These experiments have been designed to improve the state of knowledge and to provide better characterization of HEAF in the fire probabilistic risk assessment and National Fire Protection Association 805 license amendment request applications.

Initial impact of the arc to primary equipment and the subsequent damage created by the initiation of an arc (e.g., secondary fires) will also be examined.

Figure 9.11 HEAF damage.

Research Approach

To meet the goals of this test program, experiments will be conducted to explore the basic configurations, failure modes, and effects of HEAF events. The equipment to be tested in this study consists of electrical power equipment such as switchgears, breakers, and bussing components, provided by participating countries.

The project is being performed as part of a larger international OECD/Nuclear Energy Agency (NEA) effort. The NRC will be leading the physical testing and instrumentation of equipment with support from the National Institute of Standards and Technology at the designated test laboratory. International member countries participating in the project are providing electrical equipment to be tested as well as technical expertise in the experiment setup and post test data analysis.

Status

Currently, testing is being performed at the KEMA Power test Lab facilities in Chalfont, PA. The first series of tests were performed in fall 2014. The second series of tests are expected to be performed in fall 2015. The data analysis and written report are expected to be started in 2016 by the OECD working group.

For More Information
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Electrical Enclosure Heat Release Rate

Objective

Electrical enclosures are found throughout nuclear power plants (NPPs). These enclosures are typically constructed of metal, and the geometries range from small wall-mounted cabinets to large racks of multiple sections with various ventilation and opening sizes. Electrical components (wires, relays, circuit breakers, transformers, etc.) are installed inside the enclosures. Fire in electrical enclosures has been identified as a significant contributor to fire risk in NPPs. The combination of combustible materials and live electrical energy within the electrical enclosure can lead to fires and high-energy arcing faults (HEAFs). These fires have the potential to disrupt electrical power, instrumentation, and control in the plant.

The classification of electrical enclosures and the determination of their corresponding heat release rate (HRR) probability distributions is currently available in Appendices E and G of NUREG/CR-6850 (EPRI 1011989). These distributions are applied to a given electrical enclosure based on three factors: (1) qualified versus unqualified cable, (2) open versus closed doors, and (3) single versus multiple cable bundles. Refinements are necessary because a comparison of fire modeling results and resulting risk contribution of electrical enclosure fires compared with the fire experience in the U.S. commercial nuclear industry suggests that current methods may be not be realistic for certain fire scenarios.

Research Approach

To better quantify the HRR and burning behavior of electrical enclosures, the NRC initiated the Heat Release Rates from Electrical Enclosure Fires (HELEN-FIRE), NUREG/CR-7179 project with the National Institute of Standards and Technology (NIST). Eight electrical enclosures were acquired from Bellefonte Nuclear Generating Station, a plant owned by the Tennessee Valley Authority located in Hollywood, Alabama. The enclosures were installed in the early 1980s, but the plant was never operated. The enclosures were originally low-voltage control cabinets but, in the experiments, they were reconfigured with various amounts and types of electrical cable to represent other kinds of enclosures that would be found in a typical plant. NIST conducted 112 full-scale experiments at the Chesapeake Bay Detachment of the Naval Research Laboratory using these electrical enclosures.

Subsequent to the completion of the HELEN-FIRE test program, the NRC and the Electric Power Research Institute (EPRI) initiated the Refining and Characterizing Heat Release Rates from Electrical Enclosures during Fire (RACHELLE-FIRE), NUREG/CR-2178 program. The RACHELLE-FIRE program involved a working group of experienced fire protection and fire probabilistic risk assessment researchers and practitioners focused on reaching a consensus in estimating the peak HRR distributions for electrical enclosures used in NPPs. Based on the efforts of the working group, new methods and data have been developed in three specific areas: (1) classification of electrical enclosures in terms of function, size, contents, and ventilation; (2) determination of peak HRR probability distributions considering specific electrical enclosure characteristics; and (3) development of a correction method to the vertical thermal zone of influence (ZOI) above the enclosure during fire.

The new electrical enclosure classifications are based on their electrical function, size, and content. Most power enclosures such as switchgear, load centers, motor control centers, battery charges, and power inverters are grouped based on function. Other applicable electrical enclosures are classified as small, medium, or large based on their volumetric size. The classification is primarily based on the size because it can easily be assessed by visual inspection during walkdowns without the need for opening the electrical enclosure. The "Large" and "Medium" volumetric classifications can be refined to account for the amount of combustible fuel load, type of cable insulation material, and ventilation configuration. These refinements can result in more accurate HRR values based on visual inspection of the enclosure internals.

In practice, the classification described above is intended to work as follows. Electrical enclosures are first classified based on function and size. This classification should be a quick determination since it only requires external visual inspection and knowledge of the enclosure function. A "default" peak HRR distribution is assigned to this initial classification. This default distribution is intended to be conservative as no visual inspection of the enclosure internals is necessary. Based on visual inspection of the enclosure internals, the initial classification can be refined with one of two sub-groups: "low" and "very low" loading. These low and very low categories would allow analysts additional flexibility to reflect actual plant conditions identified through plant walkdowns and the examination of enclosure internals.

The revised peak HRR probability distributions (i.e., gamma distributions) for each of the new enclosure classification groups were developed based on the following factors:

- Review of experimental factors and configurations in testing programs intended to assess the HRR generated by electrical enclosure fires. Both domestic and international test programs were included within the scope of this research.
- Statistical analysis of the applicable experimental results.

- Extensive review and comparison of existing electrical enclosure configurations and operating experience in commercial NPPs and the influencing experimental factors.

Consistent with current Appendices E and G of NUREG/CR-6850 (EPRI 1011989), the probability distributions are defined based on the 75th and 98th percentile values with the 98th percentile value intended for use as the maximum (or peak) HRR to be assumed for any enclosure in a given type/ function classification group. The 98th percentile value also is the value used during initial ignition source screening.

Current practice for determining the vertical component of the ZOI includes a relatively simple process for establishing the elevation and diameter of the fire source. Typically, fire modeling uses the closed-form correlations to predict plume temperatures given a fire located within the enclosure. This practice is conservative because the fire source is positioned assuming that the enclosure does not exist. That is, the fire plume is modeled as if the fire were out in an open location, not inside an enclosure. In reality, the enclosure itself, and especially the enclosure's top cover, disrupts the plume development as compared to open unobstructed plumes. A more realistic treatment of the fire plume calculation is provided in the RACHELLE-FIRE report to account for the dispersion of the plume as it interacts with the top plate of a steel enclosure. The resulting approach is intended to be used in plume temperature calculations supporting the characterization of the ZOI in the early stages of the fire (i.e., before significant room temperature increases).

Figures 9.12 and 9.13 Characterizing Heat Release Rates from Electrical Enclosures.

Status

The HELEN-FIRE report and RACHELLE-FIRE volume 1 report have been published for public comment. Public comments are currently being resolved and the final joint NRC/EPRI reports will be published by the end of 2015. The working group will continue efforts to refine other areas related to characterization of peak HRRs for use in fire modeling and fire PRAs.

For More Information

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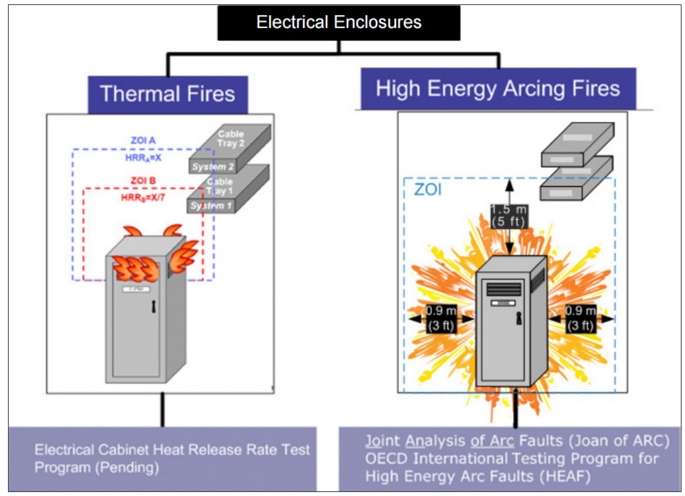

Figure 9.14 Typical electrical enclosures failure modes - Thermal Fire or High Energy Arc Fault.

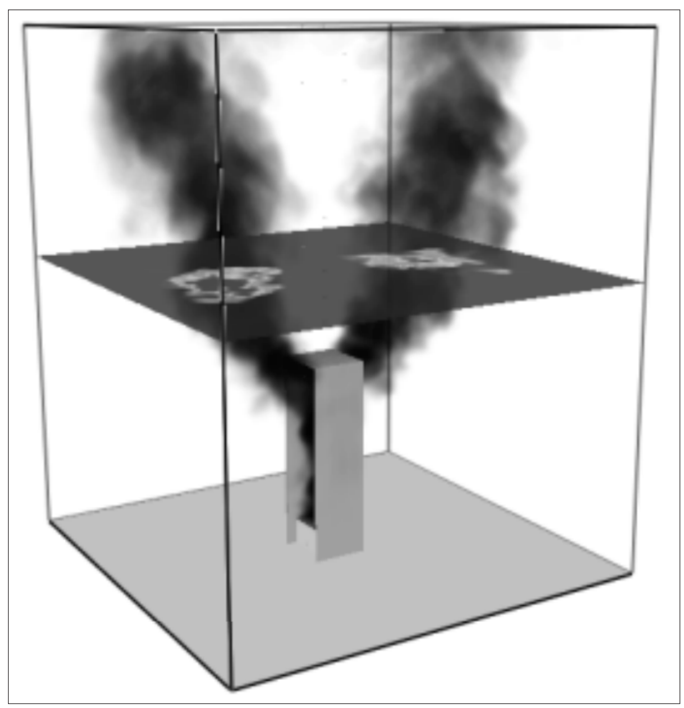

Figure 9.15 Graphic representation of obstruction plume calculation.

Training Programs for Fire Probabilistic Risk Assessment, Human Reliability Analysis, and Advanced Fire Modeling

Objective

This program supports the NRC's policy to increase the use of probabilistic risk assessment (PRA) technology by providing training for 10 CFR 50.48(c) and other fire protection programs in fire PRA, circuit analysis, fire analysis, HRA, and advanced fire modeling.

Research Approach

Since 2005, the NRC and the Electric Power Research Institute (EPRI) have jointly conducted training sessions in fire PRA. These sessions are available at no charge to all interested stakeholders. In 2005 and 2006, three days of general training covered fire PRA topical areas, including PRA, fire models, and fire circuit analysis. In 2007, training was expanded to 2 weeks per year. The courses offered detailed discussions and hands-on examples for each topical area in parallel for four days per week. In 2009, the NRC endorsed the American Society of Mechanical Engineers/American Nuclear Society PRA standard in Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-informed Activities." The 2010 training was updated to include the relationship between NUREG/CR-6850 (EPRI TR-1011989) and the fire PRA standard. In 2010, the training was also expanded to include a module on fire HRA and in 2011, a 5th module entitled "Advanced Fire Modeling" was added.

In 2008, 2010, and 2012, the training sessions were also video recorded and documented along with their training materials in a series of NUREG/CPs. NUREG/CP-0194, "Methods for Applying Risk Analysis to Fire Scenarios (MARIAFIRES)," issued July 2010, documents the 2008 training, NUREG/CP-0301 documents the 2010 training, and a NUREG/CP documenting the 2012 training is expected to be released in 2015. The MARIAFIRES NUREG/CP series is intended to enable self-study for persons unable to attend the course or wanting a refresher course on the material.

In 2015, EPRI and the NRC's Office of Nuclear Regulatory Research will split hosting responsibilities such that EPRI will host a session that includes Module 1 PRA and Module 4 HRA at an EPRI facility in Charlotte, NC. The NRC will host Module 2 Electrical Analysis, Module 3 Fire Analysis, and Module 5 Advanced Fire Modeling at NRC Headquarters. Unlike previous sessions of this workshop, in 2015, the modules will not all be offered in parallel and each training module will only be offered once. These training sessions will be offered during different weeks, which will give participants with an interest in more than one subject area an opportunity to attend more than one module.

Status

The fire PRA, HRA, circuit analysis, fire analysis, and fire-modeling programs are scheduled to continue into the near future. MARIAFIRES-2012 and MARIAFIRES 2014 are in the finalization stages and will be released in 2015. MARIAFIRES 2012 will include five volumes and an updated video of the training that was offered that year. This training continues to be in high demand and attracts participants from a diverse range of backgrounds including NRC headquarters and regional staff; NPP industry employees and consultants; international regulators and power plant operators; national research laboratories; universities and other Federal agencies, such as the Bureau of Alcohol, Tobacco, Firearms and Explosives; National Institute of Standards and Technology; National Aeronautics and Space Administration; and Defense Nuclear Facilities Safety Board.

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Figure 9.16 Photo from NRC-RES/EPRI fire PRA workshop.

Fire Research and Regulation Knowledge Management

Objective

The objective of this research is to support the NRC's knowledge management initiative in the fire protection and fire safety area by collecting relevant historic regulatory and scientific information to preserve, share, and promote a community of practice in a user-friendly format.

Research Approach

NUREG/KM-0003, "Fire Protection and Fire Research Knowledge Management Digest, 2013" was issued January 2014.

The Fire Research and Regulation Knowledge Base is a user-friendly database that provides information needed during such activities as inspections and reviews. The database consolidates all publicly available fire protection documents, such as Title 10 of the Code of Federal Regulations (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," guidelines for fire protection in NPPs, fire inspection manuals, fire inspection procedures, generic letters, bulletins, information notices, circulars, administrative letters, regulatory issue summaries, and regulatory guides. The technical knowledge includes NRC fire research technical publications (i.e., NUREGs) that serve as background information to the regulatory documents. It includes reports of NRC-sponsored fire experiments, studies, and probabilistic risk assessments (PRAs). These documents often provide the technical bases and insights for fire protection requirements and guidelines. The DVD document collection will be expanded to include fire safety standards for International Atomic Energy Agency in the next revision.

This digest supersedes previous fire protection digests and DVDs provided at the Regulatory Information Conference, NUREG/BR-0465, "Fire Protection and Fire Research Knowledge Management Digest," in their entirety. The knowledge management program, NUREG/BR-0364, "A Short History of Fire Safety Research Sponsored by the U.S. Nuclear Regulatory Commission, 1975-2008", is divided into four separate areas:

1. 1975–1987. The Fire Protection Research Program investigated the effectiveness of changes made to the NRC's fire protection regulations after the 1975 BFN fire.
2. 1987–1993. Early fire PRAs were conducted (e.g., the LaSalle Risk Methods Integration and Evaluation Program [RMIEP]).

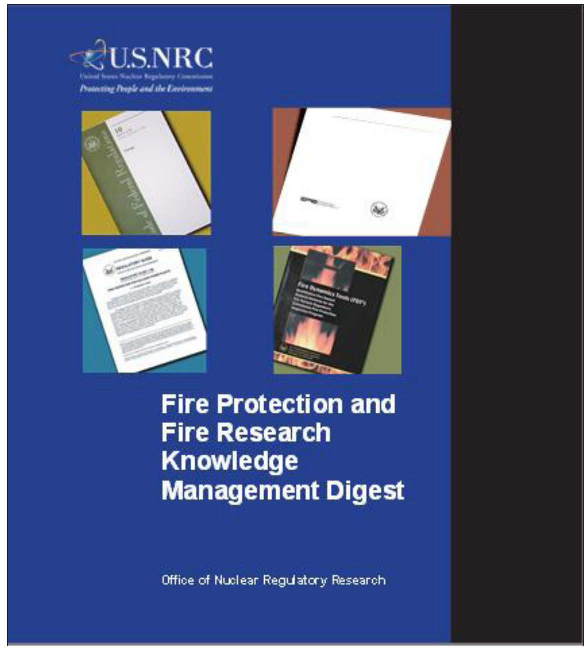

Figure 9.17 NUREG/KM-0003 Cover.

3. 1993–1998. Incremental improvements were made to the RMIEP methods.
4. 1998–present. Methods were developed to better apply the Commission's PRA technology policy to fire risk technology (to be used, where practical, in all regulatory matters).

Status

NUREG/KM-0003 is being revised to expand currently available information and update the programming to improve the user interface of the DVD. The NRC conducted this work in 2014 and plans to release an updated digest every 2 to 3 years.

For More Information
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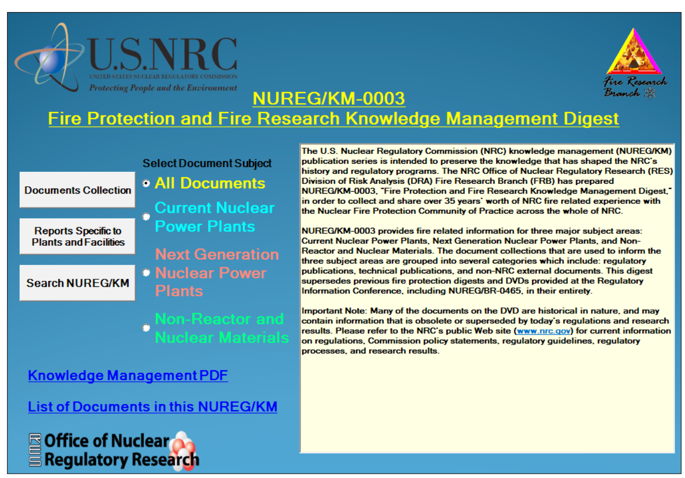

Figure 9.18 NUREG/KM-0003 Supplement 1 database user interface window.

Fire Safety Cooperative Research

Objective

One of the key objectives of the Office of Nuclear Regulatory Research (RES) Fire Research Program is to develop state-of-the-art methods, tools, data, and technical information to support the agency's safety mission. To ensure this unique research is performed by the most knowledgeable researchers in a cost-effective manner, RES has developed numerous fire research cooperative partnerships.

Research Approach

RES staff routinely works with both national and international fire research organizations to assess and improve the agency's fire research program and to maintain a high level of expertise in the field. This work and cooperation provide a robust infrastructure for nuclear power plant fire research.

One of the key partnerships in the United States is with the Electric Power Research Institute (EPRI). Since 1998, RES and EPRI have worked together under a Memorandum of Understanding (MOU) performing cooperative research and development (R&D) in the area of nuclear power plant (NPP) fire risk assessment (FRA). The Fire Risk agreement is one of the oldest long-standing agreements between the two organizations. This MOU allows the NRC and EPRI to draw from the best resources and expertise within the government and the NPP industry. Working under this agreement, both organizations cooperate by exchanging information on planned and ongoing fire risk R&D, sharing technical data, and collaborating on method development and mutually beneficial experimental programs. Recent successes from this program include development of NPP Fire probabilistic risk assessment (PRA) and human reliability analysis methods, fire model application guides and model verification and validation programs, electrical cable functional performance experimental programs, operating experience and fire event data, and unique fire risk training classes.

RES is also closely aligned with the National Institute of Standards and Technology (NIST) Fire Research Division and the Department of Energy (DOE) laboratories, such as Sandia National Laboratories and Brookhaven National Laboratory. Through this partnership with the NIST and DOE National Laboratories, the NRC has access to some of the Nation's most respected technical experts and finest testing facilities.

In the international fire research arena, RES currently has two different types of partnerships. One type is a MOU with an individual country such as the MOU with Japan's Nuclear Regulatory Authority to work together and share results of fire research related to fire PRA, fire modeling, and laboratory fire testing. The second type of international cooperation in fire research is working with the Organization for Economic Cooperation and Development (OECD), Nuclear Energy Agency (NEA) group. RES is leading the OECD/NEA High-Energy Arc Faults (HEAF), experimental testing program and is also a member of the OECD/NEA Fire Incident Record Exchange (FIRE) program.

Status

RES continues to develop and foster strong fire research alliances, both nationally and internationally, to support the development of state-of-the-art methods, tools, data, and technical information to support the agency's safety mission.

For More Information
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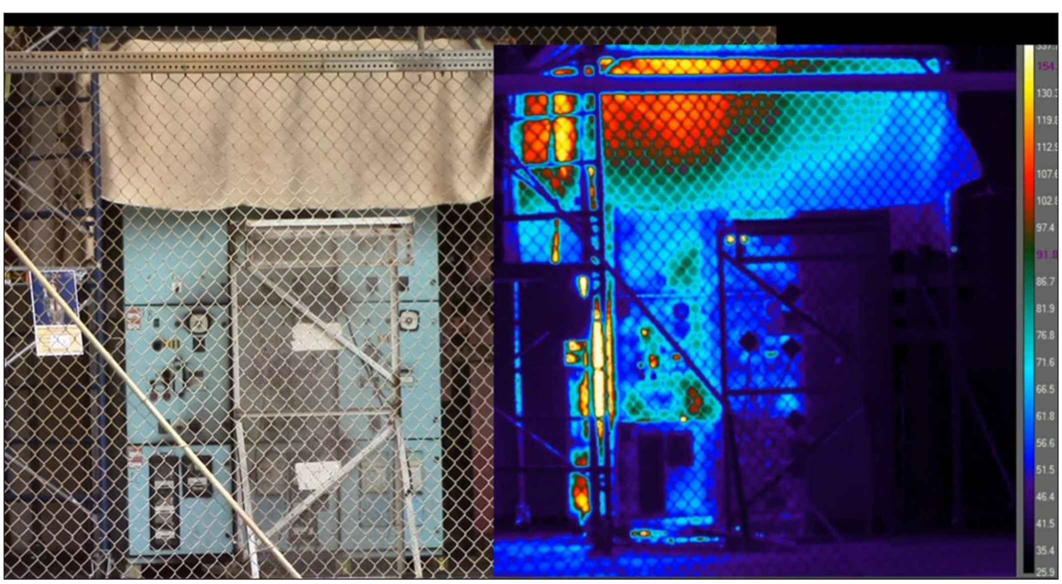

Figure 9.19 High Energy Arc Fault International Test Program – Thermal Camera Imaging.

Chapter 10: External Events Research

External events can have significant impacts on the safe operation of NPPs as the accident at the Fukushima-Daiichi Nuclear Power Plant (NPP) showed. External events cover a broad range of natural hazards such as earthquakes, high winds, floods, etc. RES is currently undertaking several projects addressing external events under the following areas:

Advances in Seismic Hazard Assessment for the Central and Eastern United States (CEUS): RES is continuing research for sites located in the CEUS with the Next Generation Attenuation (NGA) East project. The goal of this cooperative agreement between the NRC, U.S. Department of Energy (DOE), Electric Power Research Institute, and the U.S. Geological Survey is to produce the most up-to-date ground motion prediction equations (GMPEs) to be used in probabilistic seismic hazard analyses (PSHA). Research is also being conducted to develop updated software tools for calculating site-specific PSHA results and for refinement of the guidance for performing structured hazard studies following the Senior Seismic Hazard Analysis Committee (SSHAC) guidelines.

Local Effects on Ground Motion Estimation: Driven by the lessons learned from the reviews of the updated seismic hazards for operating or proposed reactors in the CEUS, research on specific topics that influence the prediction of site response are being explored in detail. Some topics include: development of software for performing 2-D site response, application of 1-dimensional site response analysis in complex geologic environments, over-damping and reduction in shear modulus at large strains for high-frequency ground motions, and selection of dynamic properties for rock-like materials to be used in analyses.

Seismic Induced Ground Failures: Soil liquefaction is a seismic hazard that is assessed in siting new reactors and may be assessed in re-evaluating seismic risk at existing NPP sites. A technical basis for applying risk-based methods is needed to update regulatory guidance on liquefaction evaluation. The NRC has funded a liquefaction study by the National Research Council to assist in developing this technical basis and to identify issues that need additional research for updating our regulatory guidance. NRC research on post-liquefaction residual strength was recently completed and provides staff with probabilistic methods for evaluating soil strength after an earthquake with application to assessing the stability of earth fill embankments.

Seismic Soil Structure Interaction: Future nuclear reactors may be embedded deep below the ground surface. The NRC is conducting research to evaluate methods for calculating pressure applied by the ground on these deeply embedded structures during seismic shaking. Tools are also being developed to improve NRC staff capabilities in performing non-linear soil structure interaction when soil volumetric strains may impact seismic structural performance. NRC is also performing research to develop guidance that links the results of probabilistic seismic hazard analyses with the soil structure interaction analyses. This work consists of developing guidance on developing probabilistic strain compatible properties for use in soil structure interaction analyses.

Tsunami Hazard Assessment: Ongoing tsunami research projects focus on the development of probabilistic methods to evaluate potential hazard to existing NPP sites from seismic- and submarine landslide-induced tsunamis along the U.S. Atlantic and Gulf of Mexico coasts. The study makes use of the Pacific Marine Environmental Laboratory's pre-computed database of over a thousand synthetic tsunami sources to identify potentially hazardous tsunami events for the eastern U.S. coastline. The historical Lisbon 1755 tsunami event is used to validate the simulations by comparing the computed results with the evidence of tsunami impact along the Caribbean arc. The research has also created an NRC-customized version of the ComMIT (Community Model Interface for Tsunami) tool that can be used to increase in-house capabilities for performing sitespecific tsunami hazard assessments at NPP sites in the event of a tsunami warning that could impact a U.S. plant.

Probabilistic Flood Hazard Assessment: NRC has recently initiated a multi-year, multi-project research program on probabilistic flood hazard assessment (PFHA). The objective, research themes, and specific research topics are described in a PFHA Research Plan delivered to the Commission in November 2014 (ADAMS Accession No. ML14296A442). This program is designed to support development of regulatory tools (e.g., regulatory guidance, standard review plans) for permitting new nuclear sites, licensing of new nuclear facilities, and oversight of operating facilities. The probabilistic technical basis developed will provide a risk-informed approach for future regulatory decisions and, as needed, rulemaking. The main focus areas of the PFHA research program are: (1) leverage available frequency information on flooding hazards at operating nuclear facilities and develop guidance on its use, (2) develop and demonstrate PFHA framework for flood hazard curve estimation, (3) assess and evaluate application of improved mechanistic and probabilistic modeling techniques for key flood-generating processes and flooding scenarios, (4) assess and evaluate methods for quantifying reliability of flood protection and plant response to flooding events, and (5) assess potential impacts of dynamic and nonstationary processes on flood hazard assessments and flood protection at nuclear facilities.

Advances in Seismic Hazard Estimation for the Central and Eastern United States

Objective

The prediction of ground motions for a given magnitude and distance (ground motion prediction equations or GMPEs) is an integral part of performing a probabilistic seismic hazard analysis (PSHA). The development of GMPEs continues to be a significant source of uncertainty in seismic hazard results. The NRC research in this area is focused on developing a new set of GMPEs for the central and eastern United States (CEUS) and refining the process and framework by which earth science models (including GMPEs) are developed. Research is also being conducted to develop updated software tools for calculating site-specific PSHA results.

Research Approach

In an effort to standardize approaches to probabilistic seismic hazard analyses (PSHA), the NRC sponsored the development of NUREG/CR-6372, "Senior Seismic Hazard Analysis Committee (SSHAC) Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts." That document (referred to as the "SSHAC guidelines") describes a formal, structured process for conducting expert assessments that could be applied using four different levels of rigor. NUREG-2117 ("Practical Implementation Guidelines for SSHAC Level 3 and 4 Hazard Studies") was written to complement the original SSHAC guidelines report. Since the time of its issuance, the guidance in NUREG-2117 has been applied in numerous seismic hazard studies at a number of critical facilities around the world. Based on this experience, the NRC is currently undertaking a revision to NUREG-2117 to capture the insights from these SSHAC studies and to provide additional guidance on the application of the process to other natural hazards (such as flooding) and for the conduct of Levels 1 and 2 studies.

The NRC is continuing research for sites located in the CEUS with the Next Generation Attenuation (NGA) East project. The goal of this cooperative agreement between the NRC, U.S. Department of Energy, Electric Power Research Institute, and the U.S. Geological Survey is to produce a comprehensive, state-of-the-art set of GMPEs for the CEUS that appropriately capture the inherent uncertainties in the ground motion prediction problem. These GMPEs will be used in future PSHA studies for nuclear facilities located in the CEUS. The NGA-East project is being conducted as a SSHAC Level 3 project following the guidance in NUREG-2117. The project is being managed by the Pacific Earthquake Engineering Research Center at the University of California-Berkeley and has dozens of individual researchers contributing to the project. This project is augmenting the rather sparse empirical data in the CEUS with extensive ground motion simulations.

The NRC continues research to develop software tools to be used in seismic hazard calculations. To capture the inherent epistemic uncertainty in earthquake processes, the latest seismic source characterization and ground motion models (such as NUREG-2115: "Central and Eastern U.S. Seismic Source Characterization Model" and the NGA-East model) have become very complex. Implementing these complex models in PSHA calculations requires the modification of existing codes and benchmarking the results in a series of verification tests. The NRC is also supporting research at the U.S. Bureau of Reclamation to develop software that will provide ground motion simulation results for complex two- and three-dimensional near-surface geological structures. These software tools will be used to evaluate the potential site-specific impact of these complicated geometries on ground motion estimates.

Status

The NGA-East project began in 2009, the final workshop was held in March 2015, and the project will be completed by the end of 2015. The project to update NUREG-2117 started in early 2015 and is expected to be completed in early 2017. The development of 2-dimensional ground motion simulation software will be completed in 2015.

For More Information
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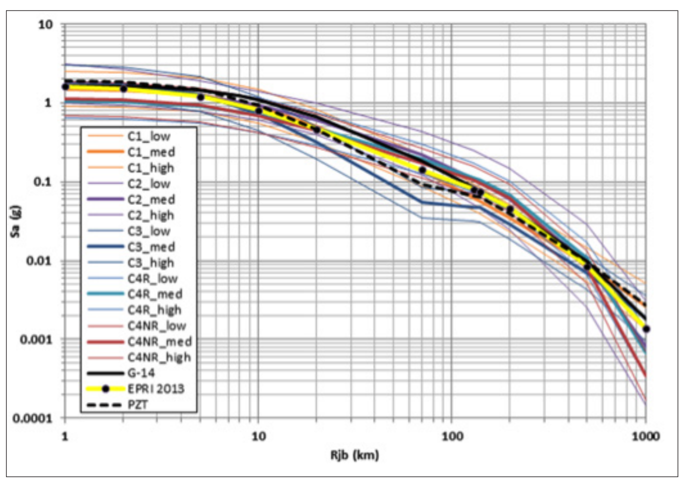

Figure 10.1 A comparison of the variability of predicted ground motions for a magnitude 7.5 earthquake as a function of distance for currently available GMPEs at a frequency of 100 Hz.

Local Effects on Ground Motion Estimation

Objective

The effects of site-specific soil or rock conditions on ground shaking is an important consideration in the development of the site-specific ground motion response spectrum (GMRS) used in the seismic design and evaluation of nuclear facilities. These effects may be quantified by a suite of site response analyses to define the median site amplification and uncertainty for the site specific soil properties at the site.

As a result of staff experience from reviews of early site permits and combined operating license applications conducted by the NRC staff since 2007 and reviews of the operating licensee submittals in response to Recommendation 2.1 of the Fukushima Near Term Task Force, several site response-related research topics have been identified that will support revisions to Regulatory Guide 1.208 "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion" and the continued development of NRC's confirmatory site response analysis tools.

Research Approach

Research in site response includes topics on the selection of shear modulus reduction and damping curves or the appropriate level of low strain damping for various rock materials, investigations of site conditions appropriate for one-dimensional (1D) site response, and large-strain site response analyses.

Various rock types (i.e., soft to firm rock including various degrees of weathered rock) may need to be incorporated into site response analyses; however, these rock materials usually extend beyond the depth range where materials can be retrieved for dynamic testing in the laboratory. In such cases, it may be necessary to rely on existing published curves (Figure 10.2) or to estimate low-strain damping values for these materials, if they are assumed to behave linearly. This research will use available data from additional testing if necessary to develop a technical basis for selecting shear modulus reduction and damping curves or the appropriate level of low-strain damping for various rock materials in the absence of site-specific dynamic laboratory test results. Almost all site response analyses performed for seismic hazard studies assume a 1D layered system. Research using small-strain downhole array recordings has shown that the 1D approach can accurately predict site amplification for some sites, but at other sites the 1D assumption may not be accurate. At this time, it is not possible to identify a priori the sites for which the 1-dimensional assumption will produce an accurate estimate of site amplification. This research will use sites from the Japanese Kik-net network and investigate the site characteristics that discriminate between sites that can and cannot be modeled accurately with the 1D approach.

Most site response analyses are performed using the equivalent-linear approach, which uses shear modulus reduction and damping curves to determine the dynamic properties that are compatible with the strain levels induced by the earthquake input motion. The appropriate development of a GMRS requires site response analyses to be performed for a large range of input motion intensities, which may induce appreciable shear strains. Two important issues need to be considered when performing equivalent-linear site response at large strains: (1) the underestimation of the high-frequency components of shaking due to the large damping ratios associated with larger strains and (2) the shear strength implied by the modulus reduction curve at large strains. This research involves using recordings from downhole array sites to fully develop and validate approaches used to incorporate frequency-dependent soil properties and the soil shear strength into site response analyses.

The NRC is continuing to develop a site response software package through a commercial contract that will allow it to perform computation of site response using random vibration theory. Eventually, this software will implement the developed and validated approaches to incorporate frequency-dependent soil properties and the soil shear strength.

Status

The project to conduct research on the site response topics described above was initiated in 2015. The results of this project will form, in part, the technical basis to update RG 1.208.

For More Information
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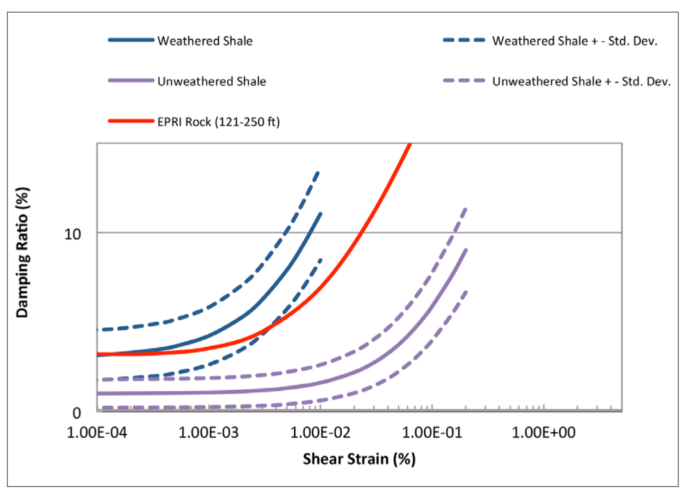

Figure 10.2 Example of damping values used for rock materials in site response analysis. Both weathered and unweathered shales were sampled at similar depth and within range of EPRI rock.

Seismic-Induced Ground Failure

Objective

Seismic safety in the design and operation of nuclear facilities has been evolving since the development of the first rules and guidance for seismic design by the Atomic Energy Commission. In 1998, the NRC issued a policy decision to move toward a risk-informed and performance-based regulatory framework. Risk-informed frameworks use probabilistic methods to assess the likelihood of failure. Significant advances have been made over the past two decades in the ability to assess hazards associated with seismic-induced ground failures such as liquefaction and slope failure. These advances allow for developing and implementing risk-informed and performance based methods into NRC regulatory guidance.

The objective of NRC research on seismic-induced ground failure consists of developing the technical basis for updating regulatory guidance on risk-informed procedures and criteria for assessing seismic soil liquefaction and development of risk-informed guidance on the assessment of slope stability.

Research Approach

Research to develop risk-informed procedures includes a project on the evaluation of post liquefaction residual strength, a study on the state of the art and practice in earthquake-induced soil liquefaction assessment, and monitoring pore-water pressure generation at non-nuclear sites in areas of high seismicity having a high liquefaction potential.

The evaluation of post liquefaction residual strength consists of identifying and evaluating case histories of slope failures that were induced by liquefaction-induced strength loss. The study of these case histories allows for back calculating the shear strength of that experienced liquefaction. Findings reported to the NRC in January 2015 provide a probabilistic relationship between shear strength of liquefied soil and the standard penetration resistance of the soil prior to liquefaction. The median of this probabilistic relationship is shown in Figure 10.3.

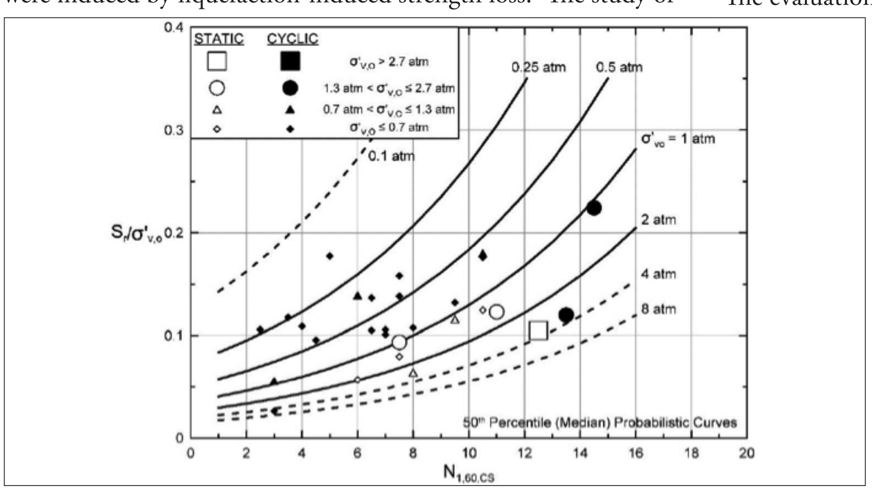

Figure 10.3 Results of probabilistic regression showing the median value of undrained soil shear strength Sr normalized by the initial vertical effective stress s'v,o as a function of penetration resistance N1,60,CS.

This study on the state of the art and practice in earthquake-induced soil liquefaction assessment consists of evaluating (1) the sufficiency, quality, and uncertainties associated with laboratory and in situ field tests, case histories, and physical model tests to develop and assess methods for determining excess pore pressure build-up, liquefaction triggering, and resulting loss of soil strength and its consequences; (2) methods to conduct and analyze laboratory and physical model testing and to collect and analyze field case history data to determine excess pore pressure build-up, the triggering of liquefaction, and post liquefaction soil behavior; and (3) methods and associated data gaps and uncertainties for evaluating the consequences of liquefaction including assessment of residual shear strength. The findings will aid in developing risk-informed regulatory guidance and provide focus for further research efforts that will have the greatest impact on developing updated guidance.

Data used to develop semi-empirical relationships on the triggering of liquefaction and subsequent soil behavior are from post-earthquake investigations at sites where ground motion and pore pressure are not measured during the earthquake. The NRC is funding a confirmatory study that monitors ground motion and pore pressure development at select non-nuclear sites. Observations from in situ recordings of liquefaction taking place can be added to the case history database and aid to better constrain uncertainty in the semi-empirical methods currently implemented in engineering practice.

Status

The evaluation of post liquefaction residual strength was just completed in early 2015 with a project report submitted to the NRC. The study on the state of the art and practice in earthquake-induced soil liquefaction assessment will be complete in August 2015 with a report published by the National Academies of Science. The project to monitor development of pore water pressure during an earthquake was initiated in 2015.

For More Information
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Seismic Soil-Structure Interaction

Objective

The objective of the NRC research on seismic soil-structure interaction is to develop, enhance, benchmark, and then exercise analysis tools for evaluations and confirmatory analyses. These analyses address lessons-learned from design reviews for new reactor licensing, needs for seismic reevaluations, and emerging technologies such as small modular reactors and seismic base isolation.

Of special interest to the research are non-traditional seismic load inputs, specifically incoherent and inclined waves, and nonlinear effects such as foundation uplifting and sliding including buoyancy effects. Also of interest are studies of in-structure response spectra for ground motions with strong high- frequency (greater than 10 Hz) response spectra content and guidance on consistent treatment of uncertainties in soil properties for site amplification and soil-structure interaction analyses.

Results of this research will support updates of design and review guidance for licensing and seismic reevaluations. They will be especially useful to analyze effects of beyond design basis seismic events to assess the margins implicit in the design guidance and for seismic reevaluations of operating facilities.

Research Approach

To address nonlinear effects and non-traditional seismic inputs, the NRC sponsors research at the Lawrence Berkeley National Laboratory (LBNL) and the University of California, Davis that is developing a high-fidelity finite element tool with parallel processing for nonlinear, time domain seismic soil-structure interaction analysis. This tool is called the NRC Earthquake Soil-Structure Interaction Simulator (ESSI). Capabilities of this tool include (1) three-dimensional, inclined, body and surface, uncorrelated seismic waves; (2) material nonlinear behavior of the rock, soil, and structure; and (3) algorithms that address interfaces, contact, and base-isolation systems.

The NRC ESSI simulator project also includes training of the NRC staff on the soil-structure methodology used in the ESSI simulator and on the uses of the tool. The staff plans to use this software to study the significance of nonlinear phenomena in seismic soil-structure interaction to derive insights for guidance update.

Typical seismic soil-structure interaction analyses do not consider the potential for volumetric strains and associated structural response. In addition, most soil models available in numerical analysis software that can capture volumetric strains are calibrated to soil tests experiencing 1-dimensional shaking. NRC-sponsored research is underway at the University of Illinois at Urbana-Champaign to develop a soil model that is calibrated to soil experiencing multi-dimensional shaking. This model will be implemented in a finite element analysis code for application in seismic soil-structure interaction analyses. This project consists of performing element-level laboratory tests on soil and scaled model soil-structure interaction tests using a centrifuge, both under multi-directional seismic loading. Data from these experiments will be used to calibrate the numerical soil model.

Some small modular reactor designs may include significant embedment of safety-related structures below site grade. The numerical modeling methods used to assess soil structure interaction and, in particular, soil pressures exerted on the structure during seismic shaking are being evaluated. The NRC sponsors research at the University of California, San Diego that collects, summarizes, and assesses existing tools for predicting seismic-induced earth pressures, collects and reviews experimental studies and critically reviews analytical work on seismic-induced earth pressures.

New research will study approaches to generate data and insights that support guidance on the use of probabilistic strain compatible properties in soil structure interaction. A goal is to use properties consistent with those from the site-specific ground motion amplification analysis for consistent consideration of uncertainties in seismic risk assessments.

Status

The NRC ESSI simulator has been completed. Ongoing research for this project includes providing the software with pre- and post-processing tools, illustrating the capabilities of the tool for problems of interest, and using available field data for continued validation work. The staff anticipates completion of the current phase of the NRC ESSI project in 2015. The next phase will likely concentrate on continued validation and use of the tool to generate results and insights that support guidance updates.

Both element-level laboratory and centrifuge experiments are currently being performed to provide data for development of a multi-dimensional soil model. Experiments and model development will be completed in 2016.

The first phase of the project to evaluate numerical modeling methods for seismic-induced earth pressures will be completed in 2015. Additional project tasks will be completed in 2017.

For More Information
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Tsunami Research Program

Objective

Since the 2004 Indian Ocean tsunami, significant advances have been made in the ability to assess tsunami hazards globally. The NRC's current tsunami research program was initiated in 2006, and it focuses on bringing the latest technical advances to the regulatory process and exploring topics unique to nuclear facilities. The tsunami research program focuses on several key areas: submarine landslide-induced tsunami hazard assessments, support activities for the licensing of new nuclear power plants in the United States, development of probabilistic methods, and development of the technical basis for new NRC guidance.

Research Approach

Tsunamigenic Source Characterization

The NRC tsunami research program includes assessment of both seismic- and submarine landslide-based tsunamigenic sources in both the near and the far fields. The inclusion of tsunamigenic submarine landslides, an important category of sources that impact tsunami hazard levels for the Atlantic and Gulf Coasts, is a key difference between this program and most previous tsunami hazard assessment programs. The USGS conducted the initial phase of work related to source characterization, which consisted of collection, interpretation, and analysis of available offshore data with significant effort focused on characterizing offshore nearfield landslides and analyzing their tsunamigenic potential and properties. A publicly available USGS report to the NRC titled, "Evaluation of Tsunami Sources with the Potential to Impact the U.S. Atlantic and Gulf Coasts," ten Brink et al., 2008 (Agencywide Documents Accession and Management System (ADAMS) Accession No. ML082960196), which is currently being used by both NRC staff and industry, summarizes this work. In addition, eight papers have been published in a special edition of Marine Geology dedicated to the results of the NRC research program ("Tsunami Hazard along the U.S. Atlantic Coast," Marine Geology, Volume 264, Issues 1-2, 2009). Recently, a review article entitled, "Assessment of tsunami hazard to the U.S. Atlantic margin," was published in Marine Geology (ten Brink et al., 2014, vol. 353).

Tsunami Generation and Propagation Modeling

The USGS database is being used for both reviews of individual plant applications and as input for tsunami generation and propagation modeling being conducted by the experts at USGS, NOAA, and the Joint Institute for the Study of the Atmosphere and Ocean at the University of Washington. The goal of this modeling is to better understand the possible impacts that the identified sources could have on the coasts.

The study uses NOAA's Method of Splitting Tsunami (MOST) numerical model to simulate tsunami generation, propagation, and coastal inundation. The MOST model, coupled with the impact Simplified Arbitrary Lagrangian-Eulerian (iSALE) code, can be used for modeling landslide-based tsunamigenic mechanisms. MOST also is being used to investigate the impact of seismic tsunamigenic sources identified and characterized by the USGS. It uses the Pacific Marine Environmental Laboratory pre-computed database of over a thousand synthetic tsunami sources to identify potentially hazardous tsunami events for the U.S. coastline. As an example, Figure 10.4 shows computed maximum tsunami wave amplitude using the MOST forecast model for the pacific basin for the 11 March 2011 Tohoku, Japan, earthquake.

Status

This program, which includes cooperative work with the U.S. Geological Survey (USGS) and the National Oceanic and Atmospheric Administration (NOAA), has resulted in several important publications on tsunami hazard assessments on the Atlantic Coast of the United States. The current phase of research includes development of probabilistic methods to evaluate landslide-based tsunami sources, analyses of typical sources in selected areas with the potential to impact existing and proposed power plants using probabilistic methods, implementation of NOAA's tsunami warning tools within the NRC, and development of a NUREG/CR describing acceptable tsunami modeling tools. The research also created a NRC-customized version of ComMIT (Community Model Interface for Tsunami) tool, which could be used by the NRC staff to increase the in-house capabilities in performing site-specific tsunami hazard assessments.

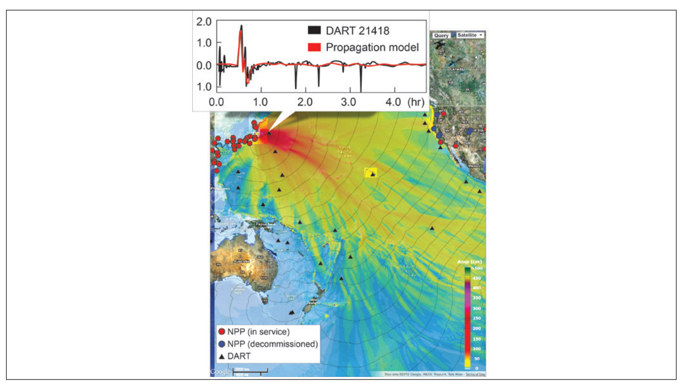

Figure 10.4 Computed maximum tsunami wave amplitude as calculated by MOST, NOAA's tsunami forecast system, for the Pacific Basin during the 11 March 2011 Tohoku event. DART (Deep-ocean Assessment and Reporting of Tsunamis) sensor locations are indicated by black triangles, and the global power plant locations are indicated by red circles. The inset shows the comparison between the observed and computed wave amplitudes at a DART station.

For More Information

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Probabilistic Flood Hazard Assessment (PFHA) Research Program

Objective

The objective of the PFHA research program is to provide guidance and tools to support: 1) review of early site permit (ESP) and combined license (COL) applications; 2) inspection findings under the reactor oversight program (ROP); and 3) risk assessments under the significance determination process (SDP). Specifically, this program includes necessary research in the area of probabilistic flood hazard assessment, including: 1) site-scale flooding hazards due to local intense precipitation; 2) riverine flooding due rainfall and/or snowmelt in the contributing upstream watershed; 3) coastal flooding due to storm surge and tsunami; and 4) flooding due to combined events. The research program also supports risk assessment needs by including work to assess and evaluate methods for quantifying the reliability of flood protection features and procedures, flood mitigation strategies and total plant response to flooding events.

Research Approach

The main focus areas of the PFHA research program are: (1) leverage available frequency information on flooding hazards at operating nuclear facilities and develop guidance on its use; (2) develop and demonstrate PFHA framework for flood hazard curve estimation; (3) assess and evaluate application of improved mechanistic and probabilistic modeling techniques for key flood generating processes and flooding scenarios; (4) assess and evaluate methods for quantifying reliability of flood protection and plant response to flooding events; and (5) assess potential impacts of dynamic and nonstationary processes on flood hazard assessments and flood protection at nuclear facilities. The PFHA research program will be implemented in three phases. Phase 1 will focus mainly on the probabilistic hazard assessment element of risk analysis, but include work on reliability of flood protection features and procedures, flood mitigation strategies, and initial work on quantitative assessment of total plant response to a flooding event. Phase 2 will develop and perform pilot studies to gain real-world experience in applying the methods developed in Phase 1. This phase will also include work to fill in gaps or deficiencies identified during the pilot studies. This phase will include significant interactions with external stakeholders (e.g. one or more licensees, industry research organizations). Phase 3 will develop guidance for conducting a complete flooding PRA. The focus will be on integrating flooding hazards (and other associated external and internal hazards) with PRA models of plant internal performance. This phase will also include significant interactions with internal and external stakeholders, as well as standards-development organizations.

Leverage Available Frequency Information on Flooding Hazards at Operating Nuclear Facilities and Develop Guidance on its Use

There is a near-term need for probabilistic information in operating reactor oversight, where the use of hazard information and insights is already an on-going input in the determination for follow-up inspection actions and resource allocation, and the evaluation of risk-informed licensing actions. For many actions such as the Significance Determination Process (SDP) is relatively short (typically a few months). Thus, there is a need to proactively collect and organize as much information as possible. It is envisioned that building a database of currently available flood hazard frequency information will be prioritized according to anticipated need and level of perceived flooding risk. Where information is already being collected and maintained by other entities (e.g. NOAA/NWS databases on precipitation frequency and hurricane storm tracks), the focus will be on providing guidance on accessing and then using the information in NRC's risk-informed decision making process. Projects initiated to support this research theme will:

- organize flooding information and build database of currently available flood hazard frequency information, prioritized according to anticipated need and level of perceived flooding risk;
- develop guidance on use of currently accepted extrapolation methods for river flooding hazard information;
- develop guidance on use of currently available extrapolation methods beyond the current consensus limits.

Develop and Demonstrate PFHA Framework for Flood Hazard Curve Estimation

Research carried out under the PFHA framework focus area will include development of a formal PFHA framework as well as efforts concentrating on framework application for key flooding scenarios and the use of expert judgment[1]. The use of expert judgment has been studied extensively in the probabilistic seismic hazard assessment (PSHA) field, and a structured process called the Senior Seismic Hazard Analysis Committee (SSHAC) process has been developed (under NRC sponsorship) and applied to numerous NPP projects. It is very likely that ideas, elements, and procedures used in the SSHAC process can be used and/or adapted to develop a structured process for the use of expert judgment in PFHA studies, which we have chosen to call the Structured Hazard Assessment Committee Process for Flooding (SHAC-F).

1 Expert judgment will be required to address questions related to appropriate process models and uncertainty characterization and quantification for very low probability events.

Projects initiated in the PFHA framework focus will:

- develop a formal framework that is applicable to multiple flooding mechanisms as well as combined events;
- investigate formal approaches for assessing uncertainty and the use of experts;
- develop example applications of the framework (e.g. site-scale flooding due local intense precipitation, river flooding, coastal flooding) with cooperation of stakeholders and other federal agencies where feasible and appropriate.

Application of Improved Modeling Techniques for Key Flood Generating Processes and Flooding Scenarios

This research program will also address application of improved computational resources and modeling techniques to key flood generating processes and flooding scenarios for NRC use. The following topics will be addressed:

- assessment and evaluation of numerical modeling methods for estimating extreme precipitation events and processes;
- assessment and evaluation of probabilistic methods for estimating inland (riverine) flood events and processes;
- assessment of paleoflood study methods for extending flood records;
- assessment and evaluation of methods for estimating probability of dam failure;
- assessment and evaluation of methods for modeling dam breach and developing dam breach hydrographs;
- probabilistic modeling of tsunamis due to submarine landslides;
- practical issues in application of joint probability methods to coastal flooding;
- evaluation of methods for estimation of flooding due to combined events.

Assess and Evaluate Reliability of Flood Protection and Plant Response to Flooding Events

The following research topics are aimed at developing the basis for quantitative evaluation of “flood fragility curves” that need to be convolved with the hazard curve to arrive at quantitative risk insights:

- compile available information on reliability of active and passive flood protection features, including lessons learned from implementation of related Fukushima NTTF recommendations;
- develop guidance for the application of human factors and human reliability analysis methods to flood protection and mitigation procedures;
- develop methods for evaluating total plant response to flooding events using PRA and/or margins analysis approaches.

Assess Potential Impacts of Dynamic and Nonstationary Processes on Site Characteristics, Flood Hazard Assessments and Flood Protection

There is a need to evaluate how new information and methods can best be applied to licensing and oversight of nuclear facilities.

Processes and mechanisms related to site parameters and external hazards that may be impacted by climate change include: 1) magnitude, distribution and frequency of precipitation events; 2) magnitude, distribution and frequency of surge generating storms (e.g., tropical and extra- tropical cyclones); 3) antecedent conditions important to flood generation (e.g. snowpack, soil moisture, land use); 4) extremes in temperature and humidity; 5) extremes in snow and ice loads on structures; and 6) magnitude, distribution and frequency of tornado and hurricane winds.

Land use and land cover within watersheds are important factors in evaluating runoff and subsequent flooding hazards. Land use and land cover change (LULCC) over the expected lifetime of the nuclear facility may be a significant source of uncertainty in flood hazard assessments.

Research topics in this focus will:

- produce periodic reports that 1) summarize recent scientific findings on climate change; 2) report on activities of federal agencies with direct responsibility for climate science and policy; and 3) analyze the potential impacts relevant to NRC regulatory activities;
- assess and evaluate the modern state of practice in LULCC modeling as it can provide insights on flood risk over the expected life of nuclear facilities.

Status

Research in this program was initiated in the first quarter of FY2015. The PFHA research plan has been endorsed by the licensing offices, was provided to the Commission in the first quarter of FY 2015, and is available at ML14296A442.

For More Information
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Cooperative Research on External Events

Objective

The NRC has cooperative agreements with Japan Nuclear Regulatory Authority (JNRA) and others in the area of seismic engineering research. The intent and purpose for these collaborations is to maximize the overall benefits of each party's individual programs in the area of seismic safety research. The research performed and information exchanged under the collaborative program has expanded the data and knowledge base in the area of seismic testing and analysis, seismic risk assessments. These research programs also provide the opportunity to interact with international organizations, ensuring NRC cognizance of ongoing seismic research in Japan and other countries.

In particular, the 2011 Fukushima nuclear accident, the only nuclear accident caused by a natural disaster, shows the importance of seismic engineering research in enhancing the understanding of how nuclear power plants (NPPs) perform during rare but very large earthquakes and in improving the safety of NPPs. This collaboration program provides an avenue to access the actual data for postFukushima research.

Research Approach and Status

Collaborative Research on Seismic Issues

The goal of the program with JNRA is to better understand the seismic behavior of NPP structures and components, obtain largescale seismic test data to benchmark analytical techniques, assess equipment fragility test data to reduce uncertainty associated with seismic probabilistic risk assessment (PRA) and seismic margin assessments, confirm and advance current seismic design and analysis methods, and provide the basis for regulatory positions for use in the evaluation of new reactor applications. The exchange of seismic information with Japan is beneficial to the NRC in supplementing the agency's knowledge and in obtaining technically sound earthquake impact data.

The scope of the program includes analyses of various structures, systems, and components (SSC) for which JNRA performs seismic tests and provides test results data to the NRC. These SSC include equipment such as pumps, valves, fans, tanks, and electric panels; degraded piping; concretefilled steel members; and baseisolated structures and components.

On a periodic basis, information exchange meetings are held in the United States and Japan to discuss the findings related to the above collaboration activities, as well as other information that each side may have developed related to the seismic safety research being performed in either country. At least one information exchange meeting is held each year in either the United States or Japan.

IAEA's International Seismic Safety Centre's Extra Budgetary Project (ISSC-EBP)

The International Seismic Safety Centre (ISSC) is part of the International Atomic Energy Agency (IAEA), Department of Nuclear Safety and Security. ISSC develops international standards on siting and seismic safety for NNPs. ISSC has an extra-budgetary program that facilitates collaboration on the development technical documents that provide a basis for international standards. Technical documents also provide details to facilitate application of the standards. RES leads the NRC collaboration with ISSC.

The ISSC-EBP Phase 1 was started in 2010, and completed in 2015. During this period, 13 Safety Reports and 8 TECDOCs were prepared and are under the publication process. In addition, the transition from ISSC-EBP Phase 1 to Phase 2(the second 5 year period) had started in late 2014. The planning meeting for International Seismic Safety Centre's Extra Budgetary Project (ISSC-EBP) Phase 2 was completed in early 2015. The projects in Phase 2 will include Testing and Updating PSHA Results, Ground Motion Simulation, Fault Displacement Hazard Assessment, Soil-Structure Interaction Methodologies, Slope Stability, and Hybrid Simulation to Assess Performance of Seismic Isolation in NPP.

Joint Research with U.S. Department of Energy, Electric Power Research Institute and U.S. Geological Survey

The NRC continues collaborative research with the U.S. Department of Energy, the Electric Power Research Institute, and the U.S. Geological Survey (USGS) to produce comprehensive regional seismic source and ground motion models for the central and eastern United States. The current focus is on development of a state-of-the-art ground motion model for eastern North America. This multi-year project involves a large number of participants from various countries, universities, government agencies, and consulting firms. The project is designed to evaluate and represent the uncertainty in predicting ground motions for critical facilities in geological stable regions such as eastern North America. The NRC is also continuing collaborative research with the USGS on several seismic hazard topics including the assessment of the hazard posed by seismicity induced by the deep injection of waste-water produced as a byproduct of oil and gas production.

For More Information
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Chapter 11: Materials Performance Research

The Office of Nuclear Regulatory Research (RES), Division of Engineering provides data, standards, tools, and methods to the NRC's regulatory offices to support their reviews of material performance-related licensing submittals and potential safety issues. The confirmatory research on materials performance focuses on both the development of methodologies needed to support regulatory actions and the work supporting the technical bases for codes and standards developed (e.g., by the American Society of Mechanical Engineers [ASME] and by the American Society of Testing and Materials [ASTM]). This research encompasses a broad scope of materials issues. A common theme in this work is a proactive approach to the management of aging degradation. As interest in license renewal for operation beyond 60 years increases, the staff has begun to assemble technical information on the various aging phenomena that can affect materials in nuclear power plants (NPPs) and develop technical guidance for the staff's review of subsequent license renewal (SLR) applications.

Steam Generator Tube Integrity: Research is currently underway to develop a technical basis for steam generator tube integrity to support regulatory decisions and code applications and to ensure appropriate inspection intervals. To provide this basis, research is focused on the areas of inspection reliability and in-service inspection technology and on the evaluation and experimental validation of tube integrity prediction modeling and degradation modes.

Reactor Pressure Vessel (RPV) Integrity: The safe operation of a NNP relies on maintaining the structural integrity of the RPV during routine operations and postulated accident scenarios. Two key capabilities underpin RPV structural integrity: (1) the ability to predict the behavior of cracked structures under loading, and (2) the ability to predict the effects of irradiation embrittlement on the fracture toughness of RPV steels. Current regulatory procedures depend on empirically based engineering methods that, while generally acknowledged to incorporate large conservatisms, have not necessarily been validated for SLR conditions. Ongoing research is aimed at understanding the adequacy of existing approaches and developing new models and predictive procedures as needed.

RPV Internals: Ongoing research concerning irradiation-assisted degradation (IAD) of RPV internals is focused on assessing the significance of void swelling on the structural and functional integrity of pressurized-water reactor internal components. Research is being conducted on harvested ex-plant materials as well as on representative materials irradiated in test reactors.

Piping Degradation: In response to operating experience with primary water stress corrosion cracking (PWSCC), the NRC has developed programs to conduct confirmatory testing on both crack initiation and crack growth for susceptible materials. Additional research is focused on the development of analyses, computational tools, and experimental testing results to provide support for assessing the impact of PWSCC on the overall safety of piping systems that make up the reactor coolant pressure boundary. The NRC also is assessing the impacts of this active degradation mechanism on the leak-before-break behavior of piping systems through the development of tools and methodologies needed to quantitatively assess compliance with 10 CFR Part 50, Appendix A, General Design Criteria 4.

Buried Piping: High-density polyethylene is a possible replacement material for buried piping. Current research is focused on confirming the ASME-proposed service life, design, fabrication, and inspection requirements for use of this material in NNP applications.

Non-Destructive Evaluation (NDE): Current NDE research is focused on the areas of accuracy, reliability, modeling, and assessment of procedures and requirements. Ongoing research results are used to assess models developed to predict the effects of materials degradation mechanisms and as initial conditions for component-specific fracture mechanics calculations.

Storage and Transportation: Current research supports the NRC's technical bases for review of applications for extended storage and transportation of spent fuel by improving the understanding of potential degradation modes that could affect safety significant structures, systems, and components in dry cask storage systems.

Neutron Absorbers: Current research on the degradation of neutron absorbers involves a complete characterization of the physical condition of Boral®, which has been harvested from the decommissioned Zion NPP. This research provides information on the degradation of Boral® during actual operation and informs the development of in-service inspection guidelines for ensuring the structural integrity of neutron absorbers made of this material.

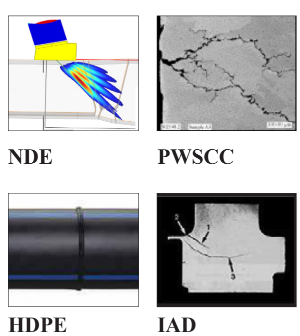

Figure 11.1 Material performance research examples.

Steam Generator Tube Integrity and Inspection Research

Objective

Steam generator (SG) tubes, Figure 11.2, 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. 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 11.2 Steam Generator Tubing.

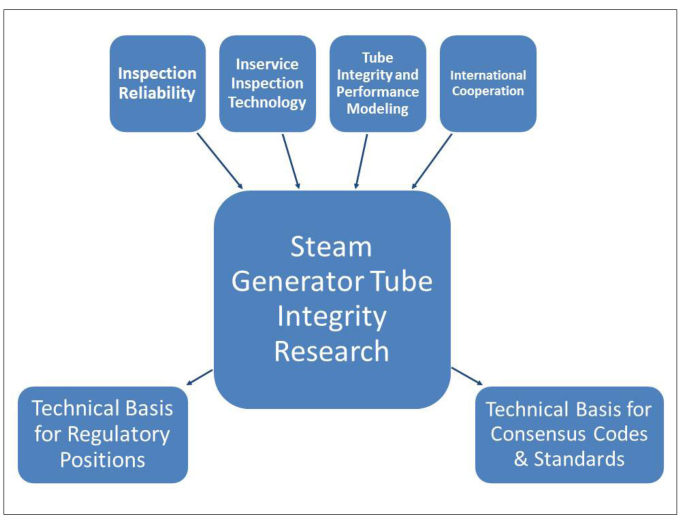

Figure 11.3 Tube Integrity research schematic.

The main objective of this research program is to develop a technical basis for SG tube integrity evaluations to aid in regulatory decisions and to assess code applications as depicted in Figure 11.3.

Research Approach

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 NRC's research addresses the following areas:

- Assessment of inspection reliability.
- Evaluation of in-service inspection technology.
- Evaluation and experimental validation of tube integrity and integrity prediction modeling and degradation modes.

The NRC also administers a collaborative exchange with regulators and researchers from member countries to conduct and share research on tube integrity and inspection technologies materials and test data. Current participants include organizations from Canada, France, Japan, Korea, and the United States.

Status

The NRC tube integrity program has been ongoing for over 20 years and is likely to continue through at least 2019. Laboratory testing is performed at Argonne National Laboratory (ANL) to draw upon unique expertise and facilities at the respective organizations. NUREG/CR reports and technical letter reports are published when sections of the work are completed. Three NUREG/CR reports are expected in 2015 addressing a variety of topics such as automated analysis of eddy current data, once-through steam flaw stability, and tube-to-tube sheet leakage. NRC staff and contractors from ANL meet every 6 months to discuss research findings with the international group involved in the tube integrity program. In addition, research is regularly presented at conferences and workshops to solicit feedback from the technical community and other key stakeholders.

For More Information
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Reactor Pressure Vessel Integrity

Objective

The objective of this work is to ensure the integrity of the reactor pressure vessel (RPV) during both normal operation and postulated accident scenarios. RPV integrity is ensured through two key technologies, both of which are addressed by this research:

1. Prediction of the behavior of cracked structures (in this case the RPV) under loading.
2. Prediction of the effects of radiation embrittlement on the fracture toughness of RPV steels.

Research Approach

Current NRC procedures (i.e., the *Code of Federal Regulations*, NRC Regulatory Guides), the American Society of Mechanical Engineers (ASME) Code, and the Standards of the American Society for Testing and Materials (ASTM) depend on empirically based engineering methods. Although these methods are generally acknowledged to contain large conservatisms, they have not always been validated through periods of extended operations (e.g., operation to 80 years). Ongoing research is therefore focused on understanding the adequacy of existing approaches, quantifying their implicit conservatisms and safety margins, and addressing gaps identified in current predictive procedures to develop better regulations, regulatory guides, codes, and standards.

Conditions that warrant such developments fall into two broad categories: (1) the more common situation where the conservatism of current procedures limits operations with no safety benefit and (2) the less common situation where the conservatism of current procedures is found to be lacking. This work includes a strong focus on ASME Codes and ASTM Standards because these provide the technical underpinnings of numerous regulations that incorporate them by reference and because valuable peer review is obtained through the consensus process. Also this work incorporates a strong emphasis on knowledge management, specifically the development of software tools to ensure that the extensive work on materials characterization accomplished in the past is not lost.

Status

Technology [A], Prediction of RPV Behavior Under Loading: Recent accomplishments include the following:

DG-1299: This draft regulatory guide provides guidance on how licensees can comply with the provisions of the alternate pressurized thermal shock (PTS) rule, 10 CFR 50.61a. DG-1299 and the related technical basis document (draft NUREG-2163) were both issued for public comment.

FAVOR: **F**racture **A**nalysis of **V**essels, **O**ak **R**idge is a computer code that provides a comprehensive probabilistic representation of RPV behavior under routine operating and postulated accident loading. An updated version of FAVOR was released that permits analysis of both accident and normal operations conditions in both boiling-water and pressurized-water reactors for the full range of flaw and embrittlement conditions that are expected in service. The NRC staff used the FAVOR code in developing the technical basis for 10 CFR 50.61a between 2000 and 2010 and is now investigating the possibility of incorporating risk-informed insights into 10 CFR 50 Appendix G. One focus of the work on 10 CFR 50 Appendix G is to assess the structural impact of postulated surface defects that just break through the austenitic stainless steel cladding that is used inside the RPV for corrosion protection. Figure 11.4 below shows a finite element model of such a defect.

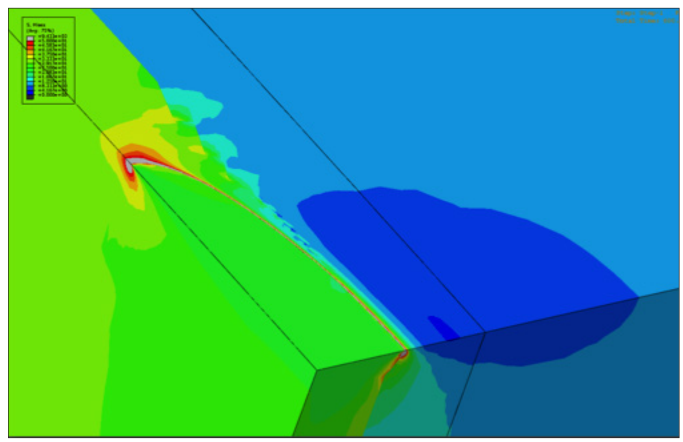


Figure 11.4 Finite element calculated stress contours around a semi-elliptical surface flaw in the stainless steel cladding of a RPV.

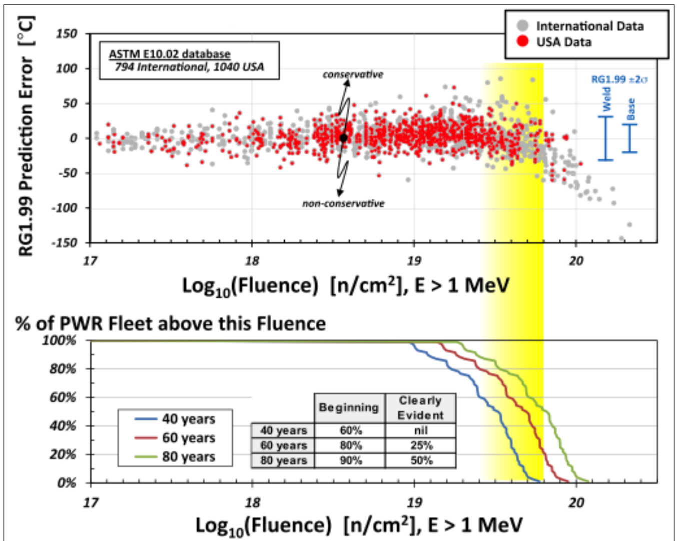
In addition, the NRC staff participates in the development of ASME Codes. These efforts provide the staff with access to current information developed in the technical community, thereby supporting the development regulatory guidance concerning RPVs. Such participation helps to streamline the review and adoption of engineering consensus standards (the use of such standards by the NRC, where possible, is a requirement of the National Technology Transfer and Advancement Act of 1995). An effort is now underway within ASME's Working Group on Flaw Evaluation to adopt the models of reactor steel fracture toughness developed by the industry and the NRC as part of the technical basis work that led to 10 CFR Part 50.61a. This effort, which has been designated by ASME as Revision 1 of Code Case N830, will harmonize the models of reactor steel fracture toughness used by ASME with those used by the NRC.

Technology [B], Prediction of the Effects of Radiation Embrittlement: Recent accomplishments include the following:

REAP: The **R**eactor **E**mbrittlement **A**rchive **P**roject provides open Web-based access to light-water reactor (LWR) surveillance data in the form of both a document archive and a relational database. REAP includes data records from nine countries in addition to the USA; it can be accessed at https://reap.ornl.gov/register. REAP was upgraded to expand its search and recovery capability and is available both as a tool for the NRC (e.g., supporting safety evaluations, developing predictive models) and also as a resource for international safety authorities and researchers. The REAP database provides archival data that can be used in licensing reviews and also provides information from which relationships to predict embrittlement trends such as those of Regulatory Guide 1.99 *(Radiation Embrittlement of Reactor Vessel Materials)*.

BTP 5-3: **B**ranch **T**echnical **P**osition 5-3, which is part of the Standard Review Plan of NUREG-0800, provides methods to estimate transition temperature and upper shelf toughness for early (pre-1972) RPVs. In 2014, the NRC became aware that some of these estimates may be non-conservative. The NRC began an investigation of BTP 5-3 to evaluate this claim and to develop updated conservative estimation methods.

Similar to Topic [A], the NRC staff also participates in the development of ASTM Standards, which provides similar benefits to the ASME Code development work. Recently, ASTM Subcommittee E10.02 (*Behavior and Use of Nuclear Structural Materials*) has developed revised versions of Standard Guides E185 and E2215 that describe, respectively, the design and conduct of RPV surveillance programs and of Standard Guide E900, which provides methods to predict embrittlement trends for RPV steels. This updated guidance incorporates the latest worldwide data and recognizes that RPVs now operate longer than 40 years. These guides will be useful to two NRC efforts: (1) the rulemaking to update the requirements of 10 CFR 50 Appendix H (*Reactor Vessel Material Surveillance Program Requirements*) and (2) the evaluation of the continued adequacy of Revision 2 of Regulatory Guide 1.99 during the period of first and potentially subsequent license renewals. For example, Figure 11.5 shows that during extended operations the predictions of Regulatory Guide 1.99 tend to under-predict the embrittlement trends observed in operating reactors.



Top: Variation of RG1.99 prediction error with fluence. Each symbol represents an individual surveillance measurement. Bottom: Percent of PWRs operating in the USA that will reach the specified fluence after the specified operating lifetime.

Figure 11.5 Variation of RG1.99 prediction error with fluence.

For More Information

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Irradiation-Assisted Degradation of Reactor Vessel Internals

Objective

The internal components of light water reactor (LWR) pressure vessels are fabricated primarily with austenitic stainless steels, which are exposed to high energy neutron irradiation and high temperature reactor coolant. Prolonged exposure to neutron irradiation changes both the microstructure and microchemistry of these stainless steel components: increasing their strength, decreasing their ductility and fracture toughness, and increasing their susceptibility to irradiation-assisted degradation (IAD). Cracks caused by IAD have been found in a number of internal components in LWRs including control rod blades, core shrouds, and bolts (Figure 11.6).

10 CFR Part 54 addresses the requirements for plant license renewal. Specifically, 10 CFR 54.29(a) requires that licensees manage aging effects so that their intended functions will be maintained consistent with the current licensing basis (CLB) for the period of extended operation. Preliminary data suggest that the significance of IAD of LWR vessel internals could increase during both the license renewal period (i.e., 40 to 60 years) and during even longer-term operation of nuclear power plants. The objective of this research is to provide confirmatory technical basis for the performance of reactor vessel internal materials during potential extended operation up to 80 years. Current Office of Nuclear Regulatory Research (RES)-sponsored IAD research focuses on assessing the significance of void swelling on the structural and functional integrity of pressurized-water reactor (PWR) internal components.

Research Approach

The research approach involves harvesting representative ex-plant materials for testing as well as evaluating reactor internals materials irradiated in test reactors. A key aspect of RES's IAD research is leveraging with other organizations to extract maximum value for these expensive, time-consuming, experimental data-gathering efforts. Therefore, RES activities in this area include cooperative research with the Electric Power Research Institute (EPRI) and international partners when appropriate along with targeted confirmatory research funded solely by RES.

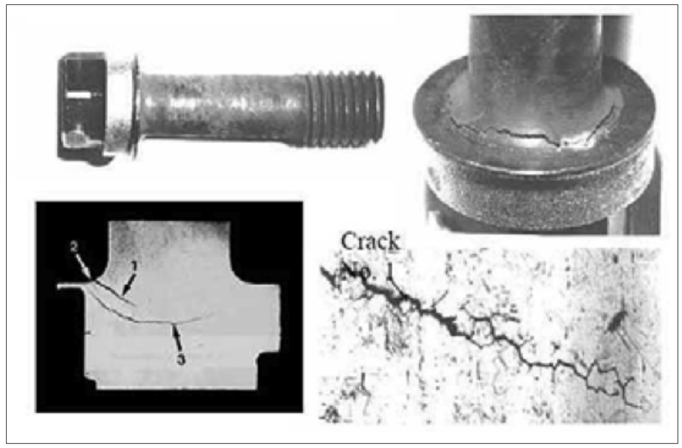

Figure 11.6 Cracking of a baffle bolt in a pressurized water reactor (PWR).

Status

RES is participating in collaborative research on materials harvested from the Zorita reactor in Spain. Materials from the Zorita reactor have very high levels of representative radiation exposure and provide valuable information on the expected behavior of domestic boiling-water reactor and PWR components during long term operation. Zorita materials are being tested in their as-harvested condition at the Studsvik laboratories in Sweden in collaboration with EPRI and several international partners. The results of this research are expected to be available in 2016 for plate materials and 2017 for weld materials. Future plans include further irradiation of weld materials as part of the NRC's participation in the Halden Reactor Project.

In addition to leveraging collaborative research with other organizations, NRC is pursuing independent IAD research. The Halden Reactor facility in Norway performed irradiations of representative reactor internal materials for experimental testing at Argonne National Laboratory. This work focuses on the effects of neutron dose on IAD and the synergistic effects of neutron and thermal embrittlement on fracture toughness in PWR environments.

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Primary Water Stress Corrosion Cracking Growth Rate Testing

Objective

Primary water stresscorrosion cracking (PWSCC) in primary pressure boundary components fabricated from nickelbased alloys is a degradation mechanism that can affect the operational safety of pressurized-water reactors (PWRs). These components include nozzles and dissimilar metal piping welds, among others. In 2001, PWSCC of an Alloy 600 control rod drive mechanism nozzle at the Davis Besse plant allowed primary coolant leakage and significant boric acid corrosion of the low alloy steel reactor pressure vessel head. Figure 11.7 shows leakage from cracks in a steam generator hot leg nozzle weld of Alloys 82 and 182. Alloy 690 and its weld metals, Alloys 52 and 152, which have higher chromium content than Alloys 600, 82, and 182, are now commonly used and are thought to be more resistant to PWSCC.

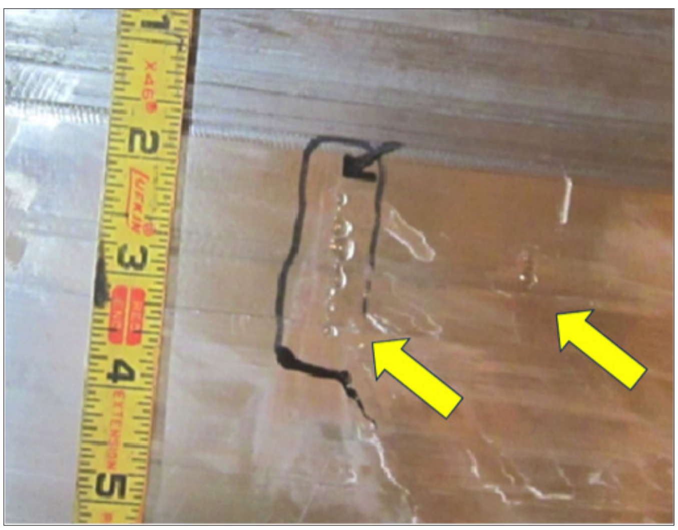

Figure 11.7 Leakage from PWSCC cracks in a steam generator hot leg nozzle.

Because of the positive service history of Alloys 690, 52, and 152 and low PWSCC growth rates measured in industry-sponsored laboratory testing, licensees have requested relief from current inspection requirements in Title 10 of the Code of Federal Regulations, Part 50.55a. To support the reviews of the relief requests and to confirm the industry data, the NRC performs independent testing to measure the PWSCC susceptibility of Alloys 690, 52, and 152.

Research Approach

To measure PWSCC susceptibility, crack growth rate testing is performed on Alloys 690, 52, and 152 in simulated primary water conditions to match the temperature, pressure, and water chemistry used in service. Metallurgical characterization techniques such as mechanical testing, microscopy, and compositional analysis are employed to relate the crack growth behavior to the material properties. Of particular interest are the effects of fabrication processes including rolling, forging, and welding.

The NRC is currently focused on specific testing to address the PWSCC susceptibility of Alloys 690, 52, and 152 in operating reactors and new reactor construction, due to:

- Weld repairs.
- Compositional dilution of chromium in dissimilar metal welds.
- Pre-existing weld defects.
- Warm-worked weld heat-affected zones.
- Variations in weld parameters such as heat input.

NRC also participates in collaborative activities with other organizations that conduct similar research to share materials and test data. Notably, NRC and the Electric Power Research Institute maintain a memorandum of understanding to evaluate the quality of test data and identify best practices for PWSCC testing.

Status

The NRC PWSCC testing program for nickel-based alloys has been ongoing for over the past 10 years and is likely to continue through at least 2018. Laboratory testing is performed at Pacific Northwest National Laboratory (PNNL) and Argonne National Laboratory (ANL) to draw upon unique expertise and facilities at the respective organizations. NUREG/CR reports summarizing key findings are published about every 18 months. A report from PNNL on the effects of cold work on the PWSCC susceptibility of Alloy 690 is expected to be published in 2015. On a more frequent basis, NRC staff and contractors from ANL and PNNL regularly present research findings at conferences and workshops to solicit feedback from the technical community and other key stakeholders.

For More Information
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Primary Water Stress Corrosion Cracking Initiation

Objective

The xLPR (Extremely Low Probability of Rupture) probabilistic code is being developed to evaluate leak-before-break analysis requirements for primary pressure piping systems per NRC Standard Review Plan (SRP) 3.6.3. The goal of the xLPR code is to quantify the probability of rupture of primary water piping systems. More information on the xLPR Code can be found in the leak-before-break summary in this NUREG.

One of the major sources of uncertainty associated with the xLPR code is the time to initiate a PWSCC crack in nickel-base alloys. Efforts by industry are underway to characterize service-induced crack initiation times and the associated uncertainty and account for it in the xLPR code. In addition, the NRC is conducting confirmatory research to provide data to help verify the crack initiation models used in the xLPR code.

The objectives of this project are to develop PWSCC initiation data (1) for nickel alloys 600/182 to help verify the crack initiation models in the xLPR code and (2) for nickel alloys 690/52/152 to develop a relative factor of improvement for crack initiation time.

Research Approach

The NRC and the Electric Power Research Institute are performing cooperative research under a Memorandum of Understanding (MOU) addendum to evaluate PWSCC initiation in nickel alloys. Pacific Northwest National Laboratory (PNNL) is under contract to perform PWSCC initiation testing using the test rig and specimen type shown in Figure 11.8. The testing will be conducted under simulated pressurized-water reactor environmental conditions (i.e., chemistry, temperature, pressure) and at constant load until indications of crack initiation are detected. Direct current potential drop (DCPD) will be used to detect crack initiation, and the DCPD data will be analyzed to estimate crack initiation times.

Figure 11.8 PWSCC initiation testing rig and 1.2-inch-tall specimen developed by PNNL.

Per the MOU, a test plan was developed and reviewed by a panel of PWSCC experts. The PWSCC initiation testing plan includes, but is not limited to, evaluating heat-to-heat variability, within heat variability, the effect to cold work, and the effect of applied stress on time-to-initiation.

Status

Testing on Alloys 600/182 is expected to be completed in 2018, while testing of Alloys 690/52/152 will conclude in 2020.

For More Information
Contact Eric Focht RES/DE at Eric.Focht@nrc.gov.

Primary Water Stress Corrosion Cracking Mitigation

Objective

Weld residual stress (WRS) develops in welded nuclear components during fabrication. These stresses, along with operating loads, contribute to primary water stress corrosion cracking (PWSCC) in dissimilar metal (DM) welds. Finite element analysis (FEA) is a numerical tool that can predict WRS for a given weld geometry (Figure 11.9).

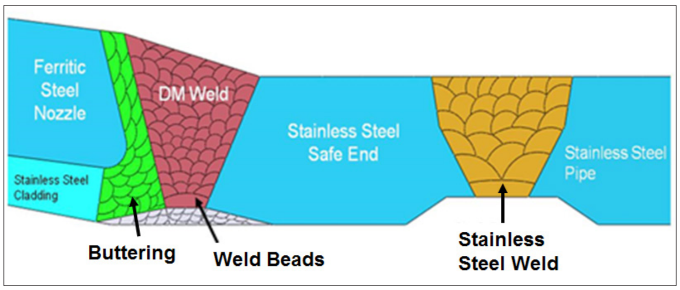

Figure 11.9 Example Mesh Geometry.

The U.S. nuclear industry has proposed various mitigation methods designed to alter the residual stress and decrease the probability of PWSCC in safety-related components. The objectives of this research program include quantifying uncertainties in WRS predictions, developing appropriate guidelines for FEA calculations, and performing confirmatory analysis of industry-proposed mitigation techniques.

The NRC is conducting this research program cooperatively with the Electric Power Research Institute (EPRI) under a Memorandum of Understanding (MOU) addendum.

Research Approach

In calendar year 2014, the NRC and EPRI organized an FEA study for WRS prediction. Ten participants from diverse organizations around the world submitted independent finite element predictions of WRS in a full-scale pressurizer surge line nozzle mockup. Two commercial vendors performed WRS measurements on the mockup (Figure 11.10). The modelers did not have access to the measurement data until after all submissions were received and the round robin study was ended. The data from this study will help NRC staff formulate guidelines for performing FEA estimations of WRS.

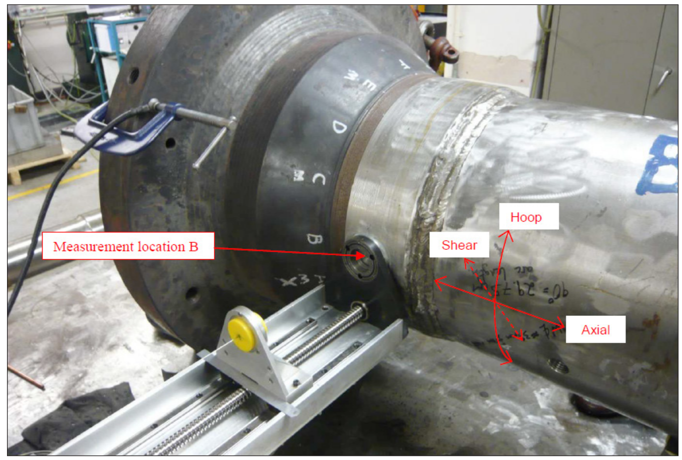

Figure 11.10 WRS Hole-Drilling Measurement on Mockup.

Although industry-proposed PWSCC mitigation methods potentially promote nuclear safety, NRC confirmatory research of industry proposals is an important aspect of the regulatory process. This research program allows NRC staff to develop and maintain the analytical capability to independently assess the effectiveness of these mitigation techniques. Past accomplishments in this area include confirmatory analysis of the optimized weld overlay.

Currently, the industry is researching excavate and weld repair (EWR) as a potential mitigation option for the future. This technique involves grinding material from the outside surface of the welded region and re-welding the resulting cavity. The grinding process may not extend around the entire circumference of the pipe. As such, analytical modeling of this scenario requires a 3-D moving heat source analysis (whereas the model represented in Figure 11.9 is 2-D axisymmetric). In this research program, the NRC staff is extending modeling capabilities to cover EWR.

Status

Previous results from this research program are documented in NUREG-2162 (ML14087A118). More recently, the NRC conducted an FEA study. The results were first made publicly available in an NRC public meeting (ML14352A195). Remaining actions in this project include:

- Statistical analysis of round robin data.
- Development of guidelines for WRS prediction.
- Independent NRC evaluation of the EWR mitigation technique.

For More Information
Contact Michael Benson, RES/DE, at michael.benson@nrc.gov.

Leak-Before-Break

Objective

10 CFR Part 50, Appendix A, General Design Criteria (GDC) 4 states, in part, that the dynamic effects associated with postulated reactor coolant system pipe ruptures may be excluded from the design basis when analyses reviewed and approved by the NRC demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis. The NRC Standard Review Plan (SRP) 3.6.3 describes leak-before-break (LBB) deterministic assessment procedures that have been used to date demonstrate compliance with the GDC-4 requirement.

Currently, SRP 3.6.3 does not allow for assessment of piping systems with active degradation mechanisms such as primary water stress corrosion cracking (PWSCC), which is currently occurring in systems that have been granted LBB exemptions. Even though the piping systems experiencing PWSCC have been shown to be compliant with the regulations through qualitative arguments, a quantitative approach is needed for those systems undergoing active degradation to ensure long-term compliance.

Research Approach

Through a cooperative agreement, the NRC's Office of Nuclear Regulatory Research and the Electric Power Research Institute (EPRI) are developing a computer code, coined xLPR, to calculate rupture probabilities in nuclear piping systems. A prototype version of this code was developed in 2012 as part of a feasibility study (see NUREG-2110), and current research activities seek to build upon the success of that work. The specific activities of this ongoing effort include:

- Completion of a fully verified and validated production version of the xLPR code that has the capability to analyze all materials and degradation mechanisms in piping systems previously shown to comply with the requirements of General Design Criterion 4.
- Conduct of an external review board using experts in fracture mechanics, stress corrosion cracking, uncertainty analysis, probabilistic risk assessment, and software development. The board members are not associated with the development of the xLPR code and not necessarily associated with NRC or EPRI. They will provide an independent review of the code development process.
- Conduct of sensitivity studies to identify which of the code's physical models and input variables contribute most to uncertainty in its outputs.
- Re-evaluation of past LBB analyses with the code to determine rupture probabilities based on the presence of degradation mechanisms and the application of inspection and mitigation strategies.
- Completion of a generalization study to quantify the risks associated with rupture of typical piping system configurations if low rupture probabilities are shown through the re-evaluation of past leak-before-break analyses.
- Development of regulatory guidance to assist licensees with standard approaches for using the code and to support efficient NRC staff reviews of associated licensing actions.

Status

The NRC, in cooperation with EPRI, is currently completing verification and validation of the xLPR computer code and plans to release the production version and related documentation in 2015. The sensitivity studies and re-evaluation of past leak-before-break analyses are slated for completion in 2017. The NRC plans to complete the generalization study, if necessary, and issue regulatory guidance on appropriate use of the code in 2017.

For More Information
Contact David L. Rudland, RES/DE at David.Rudland@nrc.gov.

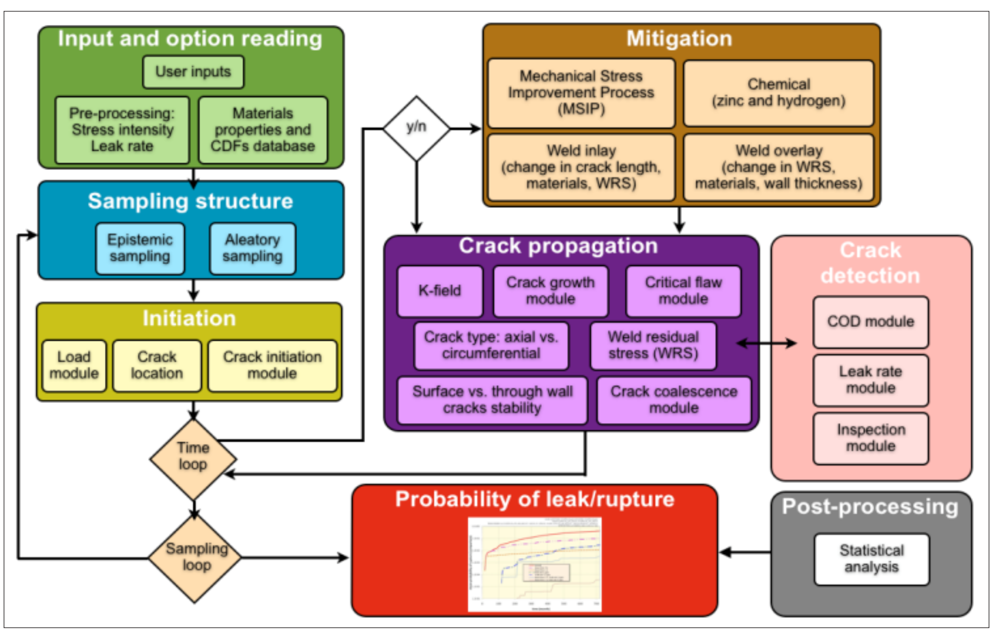

Figure 11.11 xLPR Version 2.0 Module Structure.

High-Density Polyethylene Piping

Objective

Carbon steel piping used for nuclear power plant Class 3 safety-related service water systems (SWS) has experienced general corrosion, microbiologically induced corrosion, and biofouling resulting in leakage and flow restriction. As a result, the nuclear power industry proposed to replace buried carbon steel piping in SWS with highdensity polyethylene (HDPE) piping. The industry uses HDPE, which is immune to corrosion and biofouling and has a service life exceeding 50 years, successfully in nuclear non-safety applications in the United States

Figure 11.12 Corroded carbon steel pipe.

Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code governs the design and installation of Class 3 safety-related SWS. The Section III Special Working Group on Polyethylene Piping passed Appendix nn to provide rules for the design and installation of HDPE piping systems. The objective of this NRC program is to conduct confirmatory research to assess the service life, design, fabrication, and inspection requirements proposed in Appendix nn.

Research Approach

The NRC is performing confirmatory testing and analyses on HDPE piping to evaluate the following:

- Allowable Service Life Conditions for Pipe and Fusion Joints. Slow crack growth (SCG) is the most relevant failure mechanism for HDPE piping in SWS applications, and it is strongly influenced by service temperature and stress. Fullscale pipe testing and smallscale coupon testing are being performed on both parent materials (i.e., no joints) and on fusion joints to verify the resistance of HDPE (specifically components manufactured with PE4710 resin) to SCG.
- Fusion Procedure Qualification Requirements. HDPE pipes are joined together by heat fusion processes developed experimentally for small diameter, thin-walled pipes used for natural gas applications. The essential variables used to qualify the processes for fusing small diameter pipes may not be applicable to large diameter, thick-walled pipes used in nuclear SWS. The NRC is using a combination of analytical modeling of the fusion procedure and long-term pipe testing of fusion joints to identify the critical fusion variables that affect the service life of HDPE fusion joints.
- Nondestructive Testing Methods and Procedure Qualification Requirements. Currently, no procedures exist for volumetric inspections of HDPE piping in Appendix nn. Although industry is working to develop methods for detecting volumetric flaws in HDPE parent pipes and fusion joints, the NRC is performing research to confirm the capability, effectiveness, and reliability of the proposed non-destructive evaluation (NDE) methods.

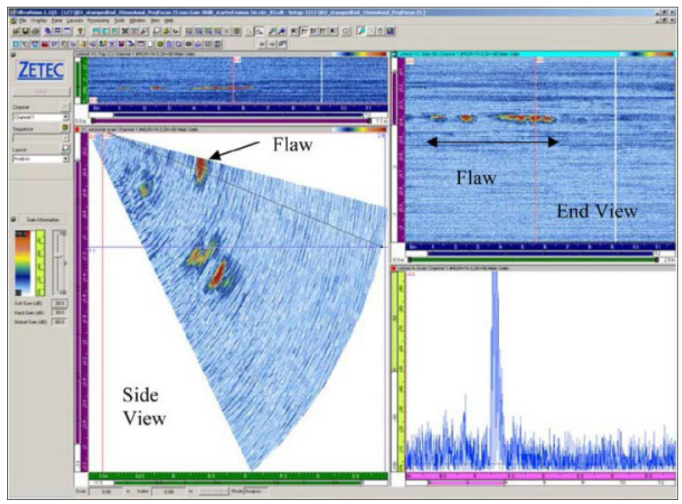

Figure 11.13 Phased Array Ultrasonic Testing (PAUT) image analysis of a HDPE butt fusion joint.

Status

Structural integrity testing thus far has enabled the NRC to validate a fracture-mechanics-based approach for service life prediction in the parent PE4710 material. Testing is ongoing to validate a similar model for fusion joints. The NRC's research has demonstrated that HDPE joints can fail much more quickly than the parent material. NRC NDE research has demonstrated the ability of phased array ultrasonics to find void-like and planar type defects in HDPE piping and joints. Work is ongoing to determine the minimum detectable defect sizes and detectability of fine particulates or incomplete fusion in a joint.

For More Information
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Nondestructive Examination

Objective

In accordance with Title 10 of the Code of Federal Regulations (10 CFR) Part 50.55(a), "Codes and Standards," licensees must inspect structures, systems, and components (SSCs) to ensure that the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) are met and that SSCs can continue to perform their safety functions. Research on nondestructive examination (NDE) of light-water reactor components and structures provides the technical basis for regulatory decisionmaking related to these requirements.

Research Approach

Research activities focus on evaluating the accuracy, effectiveness, and reliability of NDE as currently practiced for the inservice inspection (ISI) of nuclear power plant SSCs. As reactor facilities age, it becomes more important that adequate inspections are conducted to ensure that components are capable of performing their function and, thus, that safety is sufficiently maintained. ISI is one of the primary tools in the management of age-related degradation in nuclear power plants and has been increasingly critical as plants age. Certain materials, configurations, and locations susceptible to degradation are difficult to inspect in the current fleet of reactors and will most likely remain challenging for new reactors. This NRC program 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 11.14). Currently, the ongoing research is focused in the following areas:

- Effectiveness and reliability of advanced/emergent NDE methods and currently applied visual testing (VT) methods.
- Ultrasonic testing (UT) for use in lieu of radiographic testing.
- Adequacy of proposed industry changes to ISI programs.
- Assessment of the capability of UT simulation tools to optimize examination procedure variables.
- Effectiveness of ISI techniques for detecting service degradation, such as:
    - Primary water stress corrosion cracking in Alloy 600, 82, 182 dissimilar metal welds and J-groove penetrations.
    - Potential degradation in cast stainless steel and weldments.
    - Assessment of the reliability of high-density polyethylene (HDPE) for application to ASME Class 3 systems.
    - Assessment of NDE methods for Dry Cask Storage systems.

To help defray costs and to gain access to the expertise of other organizations, the NRC performs some of this work under cooperative agreements with the Electric Power Research Institute (EPRI) and the Institut de Radioprotection et de Surete Nucleaire (IRSN). Moreover, the NRC participates in an international cooperative program, PARENT, aimed at evaluating commercial inspection techniques using blind round robin testing (RRT) and open RRT to assess cracking in dissimilar metal welds in both large and small bore piping and bottom-mounted instrumentation penetrations.

The Office of Nuclear Reactor Regulation and the Office of New Reactors will use the findings from this research program to evaluate licensees' alternatives to ASME Code requirements, new plant submittals, proposed changes to the ASME Code, and ASME Code Cases for NRC endorsement. In addition, results from the NDE of these SSCs are used to assess models developed to predict the effects of materials degradation mechanisms and are used as initial conditions for component-specific fracture mechanics calculations.

Figure 11.14 Components and material that have been removed from canceled plants.

Status

The NRC Research NDE program, with Pacific Northwest National Laboratory serving as the primary contractor, dates back to 1977. Since this time, well over 100 NUREG/CRs have been published. The program continues to address a very broad range of NDE, ISI, and ASME Code related issues essential to support NRC's mission. Publications for the coming year will address NDE modeling, exams of partial penetration welds, assessments of plant-related NDE events, etc.

For More Information
Contact Carol Nove, RES/DE, at carol.nove@nrc.gov.

Subsequent License Renewal Applications Research

Objective

The U.S. commercial nuclear power industry intends to submit subsequent license renewal applications (SLRAs), which will allow nuclear power plant (NPP) operation up to 80 years, in accordance with 10 CFR Part 54.31(d) that "a renewed license may be subsequently renewed." However, the NPPs may need to resolve potential technical challenges from aging effects on passive long-lived systems, structures, and components (SSCs) before the NRC can approve SLRAs.

Aging management programs (AMPs) are developed to anticipate material degradation and to help ensure adequate functionality and safety margins in SSCs. Key technical issues to be addressed in AMPs within subsequent license renewal guidance documents (SLRGDs) as identified by SRM-SECY-14-0016, (ML14241A578) include "reactor pressure vessel neutron embrittlement at high fluence; irradiation-assisted stress corrosion cracking of reactor internals and primary system components; concrete and containment degradation, and electrical cable qualification and condition assessment."

Nuclear reactor components degrade over time via material/environment interactions. The objective of this research is to generate independent and defensible technical data and confirmatory tools and to enable development of regulatory guidance on the aging of SSCs. The Office of Nuclear Regulatory Research (RES) conducts research to generate technical data and to enable the development of confirmatory tools. Such tools support the regulatory review of the licensee's AMPs to ensure their efficacy and adequacy for the subsequent period of extended operation (PEO).

Research Approach

The NRC and industry have conducted extensive research over the past several decades to better understand the safety implications and risk associated with aging of SSCs. RES, through a cooperative research memorandum of understanding (MOU) interfaces with the Department of Energy's (DOE's) Light-Water Reactor Sustainability Research (LWRS) and EPRI's Long-Term Operation (LTO) research. Most recently, the NRC, in cooperation with the DOE LWRS program, has completed research to rank the significance of age-related degradation phenomena that could affect reactor SSCs over 80 years. This research, evaluating the core internals and piping systems, the reactor pressure vessel, electrical cables, and concrete structures, is documented in NUREG/CR-7153, "Expanded Materials Degradation Assessment, Vol. 1-5," 2014. These analyses expanded the scope and horizon of NUREG/CR-6923 "Expert Panel Report on Proactive Materials Degradation Assessment."

An in-depth study of AMP effectiveness at three NPPs, already in the post-40 year PEO, was also recently completed and, among its other findings, identified tuberculation as an aging mechanism leading to fouling (Figure 11.15), previously unidentified in the current license renewal guidance documents (LRGDs) consisting of NUREG-1800, NUREG-1801, and NUREG-1950.

Figure 11.15 Fouling from tubercles in service water system (NRC presentation at NRC/NEI public meeting, Dec 4, 2014, ML14338A376).

RES interfaces with international efforts, such as the International Forum for Reactor Aging Management (IFRAM), and participates in technical meetings focused on some elements of proactive management of materials degradation. These efforts leverage highly skilled resources to support RES goals in SLR research.

Status

RES and NRR staff are working together to develop SLRGDs (rewriting the current LRGDs so that AMPs address issues that may emerge with an 80-year operating horizon). RES and NRR are identifying research vehicles to address any technical gaps. The SRM-SECY-2014-0016 emphasized "the need to strive for satisfactory resolution of these issues prior to the NRC beginning a review of any SLR application."

RES staff continues to interact with the DOE-LWRS Program, EPRI's LTO initiatives, and IFRAM to monitor developments relevant to SLR and, where appropriate, engage in joint research activities.

For More Information
Contact Amy B. Hull, RES/DE, at Amy.Hull@nrc.gov.

Seismic Loading Effects on Reactor Materials Degradation

Objective

The objective of this research is to evaluate the potential cumulative effects of repeated and sudden below-design-basis earthquake (RSBDBE) loading on progressing degradation of nuclear reactor structural materials. It is expected that this program will provide additional information that is valuable for the regulatory guidance related to safety evaluation of structural integrity under sudden unexpected seismic load conditions. The aging degradation mechanisms that are included in this study include uniform and pitting corrosion, flow-accelerated corrosion, microbiologically induced corrosion, irradiation-assisted stress corrosion cracking (IASCC), intergranular stress corrosion cracking (IGSCC), transgranular stress corrosion cracking (TGSCC), primary water stress corrosion cracking (PWSCC), fracture toughness (FT), and fatigue crack initiation and growth. The program will assess the potential cumulative effects of severe or low-level but frequent seismic loading on these aging degradation mechanisms for materials used in the reactor primary pressure boundary components including the reactor core internals and core support structures and those whose functionality is safety-related.

All structures, systems or components important to safety are designed to withstand the effects of the design basis earthquake (DBE). However, these design analyses do not consider either the potential cumulative effects of repeated, sudden, below safe shutdown earthquake loading or the degradation of material properties.

Research Approach

Limited scoping research will be conducted through literature review of the design and inspection code requirements and practices on the effects of dynamic and sudden pulse- type, high-strain-rate loadings, such as those due to below-DBEs, on potentially ongoing degradation of reactor materials. The research shall consider typical pre-existing degradation mechanisms, such as IASCC for reactor vessel and other internals, IGSCC for steam generator materials, and PWSCC for various pressure boundary reactor component materials and associated weldments (particularly at dissimilar metal welds, i.e., at nozzles and other pressure boundary piping components).

Status

The scoping study is ongoing and has identified that RSBDBE loading may impose sudden high-strain rates on reactor materials. The stress-strain behavior under a short-term, high-strain rate may be quite different than under normal loading conditions. These high-strain rates could change the microstructure of reactor materials and thus important properties, such as yield strength of some reactor component materials.

Deformation mechanisms of a material may vary with the rate of strain from creep to wave-propagation and thermal effects (e.g., adiabatic shear banding). At least six potential technical gaps have been identified, which may merit more study and examination. The next steps involve the extension of the scoping study to the effects of seismic loading-induced property changes on progressing material degradation, including stress corrosion cracking and fatigue.

After the completion of the initial scoping study, a recommendation will be made for further research in this area. The recommendation may include for the licensees to further clarify the risk assessment of component degradation due to the cumulative effects of repetitive, below-design-basis seismic loads, for achieving continued plant operation and ensuring adequate public safety. Such research will also provide technical data and information, as necessary, to influence the national codes and bodies of standards used in the reexamination of seismic-loading requirements for the materials of construction for passive components in light-water reactors including the potential cumulative effects of repetitive, below-design-basis seismic loads and the assessment of material degradation during service and its effect on the design safety margin of components.

For More Information
Contact Appajosula S. Rao, RES/DE, at Appajosula.Rao@nrc.gov.

Degradation of Neutron Absorbers in Spent Fuel Pools

Objective

In spent fuel pools (SFPs), a stainless steel rack structure aligns and supports spent fuel assemblies. Assemblies are spaced closely together in such a manner that the distance between fuel assemblies alone may be insufficient to maintain subcriticality in the pool. Therefore, subcriticality assurance is often provided by the use of neutron absorber panels containing boron-10 that are placed within the rack walls.

In the past 15 years, neutron absorber materials, especially Boral® and Boraflex®, have shown various types of degradation such as blistering (shown in Figure 11.16) or matrix degradation. Information Notice 0926, "Degradation of Neutron-Absorbing Materials in the Spent Fuel Pool," dated October 28, 2009, summarizes specific incidents of excessive degradation. Degradation of credited neutron absorber panels may affect criticality calculations and challenge the subcriticality requirement of keff < 0.95 in Title 10 of the Code of Federal Regulations (10 CFR) 50.68, "Criticality Accident Requirements." Currently, plants detect and manage neutron absorber aging and degradation through surveillance programs such as sample coupons, in situ BADGER[1] testing, and RACKLIFE modeling.

In past efforts, the NRC has cataloged the current strategies licensees employ to meet subcriticality requirements and information pertaining to the neutron absorber materials surveillance program information. Current research focuses on evaluating neutron absorber materials surveillance methods and in-situ neutron attenuation measurements using the BADGER system to identify degradation mechanisms and measurement uncertainties associated with BADGER results. The results of this project will be used to evaluate the adequacy of SFP surveillance programs and the bases for nuclear criticality safety analyses.

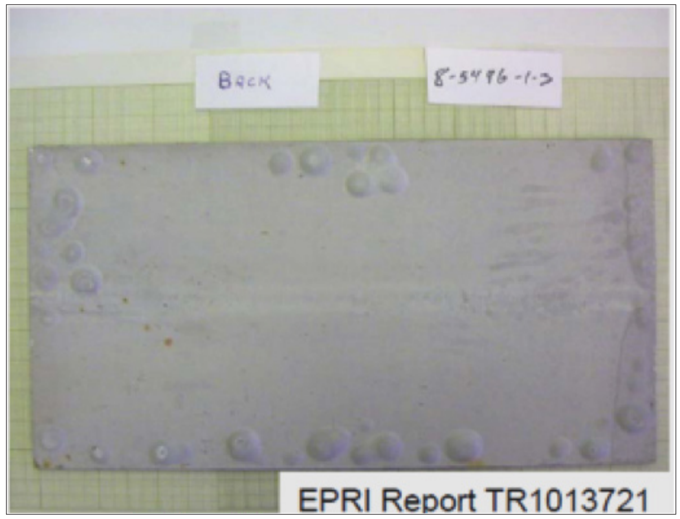


Figure 11.16 Blistering on the aluminum cladding of a boral neutron absorber.

Research Approach

The NRC and the Electric Power Research Institute (EPRI) are conducting cooperative research under a Memorandum of Understanding (MOU) to conduct in-situ testing of the neutron absorber racks at the Zion SFP using BADGER and to extract boral panels from the racks. The NRC's goal is to correlate the BADGER results with the level of degradation observed in panels. The research will help characterize uncertainties associated with BADGER and identify degradation mechanisms associated with boral.

Also, surveillance methods employed for neutron-absorbing materials in SFPs will be evaluated to determine the extent to which they provide representative samples of the materials throughout the SFP and to determine if the prescribed surveillance frequency is adequate.

Status

The boral panels are expected to be removed from the Zion SFP in July 2015, and the evaluations of the panels are expected to be completed in 2017.

For More Information
Contact Eric Focht, RES/DE, at Eric.Focht@nrc.gov.

1 Boron Areal Density Gage for Evaluating Racks (BADGER). The NRC has published two Technical Letter Reports on BADGER: ML12216A307 and ML12254A064.

Extended Storage and Transportation of Spent Nuclear Fuel

Objective

Commercial nuclear power plants use independent spent fuel storage installations (ISFSIs), licensed under Title 10 of the Code of Federal Regulations (10 CFR) Part 72, when spent fuel pools have reached capacity. ISFSIs are initially licensed for 20 years and may receive license renewals for up to 40 years. Extended storage at current or future ISFSI locations is necessary until a permanent solution for spent fuel disposal is available.

The objective of this research is to develop the necessary regulatory technical bases for the extended storage and transportation (EST) of spent nuclear fuel. This effort involves an enhanced understanding of the time dependencies and environmental conditions that affect the possible degradation modes of safety significant structures, systems, and components (SSCs) in dry cask storage systems (DCSSs) as seen in Figure 11.17. Significant operational parameters include fuel burnup, material composition, dry cask design, thermal loading, ISFSI location, and the age of the systems. The NRC will use the information obtained in this program to evaluate ISFSI license renewals and determine the need for aging management through inspections or monitoring of the condition of DCSSs.

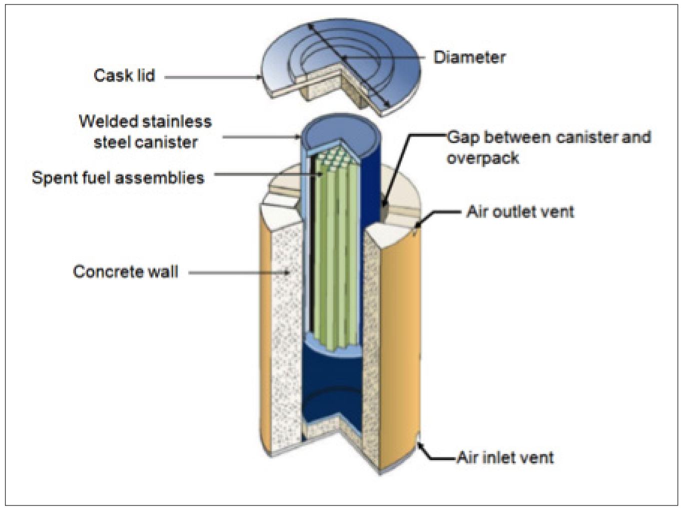
Research Approach

The NRC developed a multitask approach to identify the technical information needs and to conduct focused investigations on significant technical issues. An assessment of aging and degradation phenomena that affect DCSS SSCs was performed and used to identify areas for additional. This assessment was published in May 2014 as the final EST Technical Information Needs (TIN) report and can be found in ADAMS at ML14043A402. The EST TIN report is used to prioritize EST research efforts based on the assessment of identified technical issues.

Status

Research has been completed on several high-priority items including:

- Chloride-induced stress corrosion cracking (CISCC) of stainless steel canisters in marine environments (NUREG/CR-7170).
- Non-destructive evaluation (NDE) of DCSSs.
- Thermal analysis of a horizontal DCSS (NUREG/CR-7191).
- Vacuum drying and potential residual moisture in the canister.
- Available monitoring methods for DCSS.



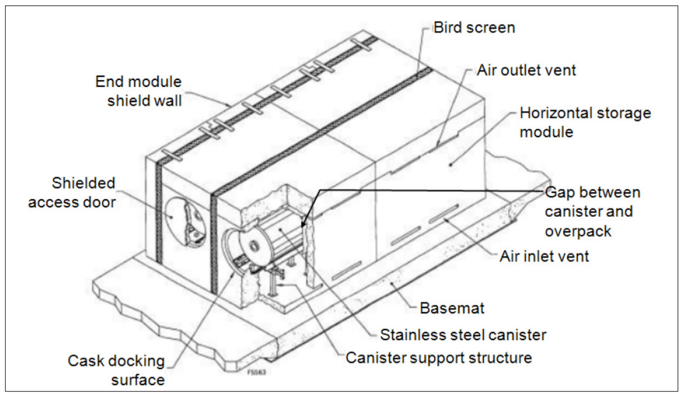


Figure 11.17 Schematics of vertical (top) and horizontal (bottom) DCSS.

A key finding from EST research efforts to date has been to demonstrate that CISCC of stainless steel canisters is plausible for the expected canister environments.

EST research efforts are ongoing in the following areas:

- Stress analysis of high burnup spent fuel cladding during extended storage.
- Concrete degradation modes, inspection, and assessment.
- Aging management of DCSS SSCs.
- Thermal analysis of a vertical DCSS.

One important ongoing research activity is the cladding stress analysis to assess the potential for stress-dependent cladding degradation mechanisms during extended dry storage. This analysis takes into account fuel swelling as well as gas production and release during storage to predict whether sufficient cladding stress will be present.

For More Information
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Material Performance Cooperative Research

Objective

Typically, computer software packages, experimental data, numerical procedures, and other analytical methodologies are needed to fully understand and characterize the performance of materials used in nuclear power plants. The development of these tools and data add to the technical basis needed for safety determinations. Cooperative agreements have been developed in several materials research areas that allow for leveraging resources and minimizing duplication of effort.

Research Approach and Status

For the topics described below, the NRC has separate Memoranda of Understanding (MOUs) with the Electric Power Research Institute (EPRI) and the U.S. Department of Energy (DOE) to promote general information sharing and describe the parameters for conducting cooperative research programs between the two organizations.

Extremely Low Probability of Rupture (xLPR) Development

The objective of this research is to develop a robust analysis methodology for evaluating reactor coolant system piping rupture probabilities that uses realistic input data and models and appropriately treats epistemic and aleatory uncertainties. The tool is being verified, validated, and benchmarked to enable its use in support of licensing, rulemaking, design, and regulatory decisions by both the nuclear industry and the NRC. International cooperation is ongoing through the PARTRIDGE program, which is focused on probabilistic fracture mechanics methodologies and has participants from the United States, Canada, Sweden, Korea, and Taiwan.

Non-Destructive Examination (NDE)

The overall objectives of this work are to identify and evaluate the effectiveness of NDE methods in detecting and characterizing flaws, to assess the reliability of NDE methods for selected examinations, and to evaluate aspects of inspector qualifications. The NRC will use the information developed in this effort to form a technical basis on the effectiveness and reliability of NDE and to support the development of guidance within the American Society of Mechanical Engineers (ASME) Section XI code. International cooperation is ongoing through the PARENT program, which is focused on the international inspection techniques for dissimilar metal welds, and has participation from the United States, Sweden, Japan, Finland, and Korea.

Environmental Degradation

The objective of the research is to develop data, methodologies, and impacts of environmental degradation on the integrity of nuclear-grade materials. Current research is focused on active material degradation and impacts of radiation on the material performance. The research generated will support technical bases for inspection requirements and aid in regulatory decisions.

Primary Water Stress Corrosion Cracking (PWSCC)

The objective of this research is to develop PWSCC initiation and crack growth data for nickel-based alloys 600/182 and 690/52/152. The NRC will use the data to help verify crack initiation and growth models used in the probabilistic xLPR code and evaluate inspection programs proposed by the nuclear industry for dissimilar metal weld components. In addition, an expert panel is evaluating the quality of the data.

Neutron-Absorbing Materials (NAM) in Spent Fuel Pools (SFP)

The objective of this addendum is to coordinate the harvesting of Boral® NAM panels from the decommissioned Zion SFP and to conduct cooperative research on degradation mechanisms that may compromise the neutron absorbing capacity of Boral panels.

Irradiation-Assisted Degradation

The objective of this research is to generate data from harvested ex-plant material on the effects of high-fluence neutron irradiation. Parameters tested include tensile properties, crack growth rate, fracture toughness, and microstructural changes such as void swelling.

Steam Generator Tube Integrity and Inspection

The objective of this research is to develop the technical basis for the evaluation of steam generator tube integrity. To provide this basis, the program addresses the assessment of inspection reliability, evaluation of in-service inspection technology, evaluation and experimental validation of tube integrity and integrity prediction modeling, and evaluation and experimental validation of degradation modes.

Subsequent License Renewal

The principal areas of interaction are DOE's Light-Water Reactor Sustainability Research (LWRS) Program and NRC's research, which includes a cooperative program with EPRI, to support subsequent license renewal (SLR). The cooperative program with EPRI ensures the timely exchange of information on planned and ongoing aging management research activities. Through the NRC and DOE programs, materials-related gaps in relation to SLR have been identified and documented in NUREG/CR-7153, "Expanded Materials Degradation Assessment, Vol. 1-5," 2014, that expanded the scope and horizon of NUREG/CR-6923. This research ranked the significance of aging-related degradation phenomena that could affect reactor system and components during SLR. This work was used to evaluate possible age-related material degradation for the second license renewal period. NUREG/CR-7153 was used as one of the technical bases for creation of the SLR guidance documents.

For More Information

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Chapter 12: Structural Performance Research

The Office of Nuclear Regulatory Research (RES) maintains a research program in structural civil engineering to support regulatory needs of multiple offices within the agency. Structural performance in nuclear installations is an essential aspect of their safety and security. It is known from previous studies that the mode and timing of failure of critical structural components such as reactor containments are very important in determining accident consequences. Safety-related structures and equipment in nuclear power plants must be designed to standards and guidance that ensure performance of their intended safety function under design basis events, such as seismic events, with sufficient margin.

Structural performance research reviews the technical bases of codes and standards for the design of safety-related nuclear structures and equipment to inform regulatory guidance document revisions. These guidance documents are used by licensees to prepare license applications or amendment requests. Instances of material degradation and aging such as containment liner degradation, loss of prestress, and alkali silica reaction (ASR) of concrete have been observed in U.S nuclear power plants. Research is underway on the significance of material degradation and aging for structural performance and safety. The research is done in the context of long-term operations (up to 80 years) of nuclear power plants to inform subsequent license renewal guidance, reviews, and related aging management programs.

This chapter provides additional details on five structural performance research areas that address current or anticipated regulatory needs. When appropriate, the research in those areas includes collaborative research with international and U.S. institutions working on the safety of nuclear installations.

<u>Concrete Irradiation Effects on Structural Performance</u> – Concrete structures in nuclear reactor containments in the proximity of the reactor vessel (e.g., the primary and biological shield walls and reactor vessel support structures) can be subjected to high levels of neutron and gamma radiation under sustained operating temperatures up to about 150 degrees F. For long- term operations, the radiation fluence/dose experienced by the concrete in these structures may approach levels that degrade the concrete. RES is starting a confirmatory research program to assesses the structural and safety significance of concrete irradiation for long-term operations.

<u>Chemical Degradation of Concrete and Structural Effects</u> – Concrete at nuclear power plants deteriorates over time due to the effects of several chemical and physical processes including alkali-silica reaction (ASR). Research is underway to assess the structural performance of ASR-affected structures for design basis static and dynamic loading and load combinations through its service life including the 20-year subsequent license renewal period.

<u>Structural Analysis</u> – Research is ongoing to maintain state-of-the-art structural analysis capabilities on nonlinear structural analysis. This research involves in-house activities as well as contracts or grants to national laboratories and research universities. It includes benchmarking existing analysis tools, sensititivity studies to inform best practices, and as needed, development of new modeling capabilities. The research supports confirmatory analyses for safety and security purposes, assessment of safety margins, and studies that inform regulatory actions.

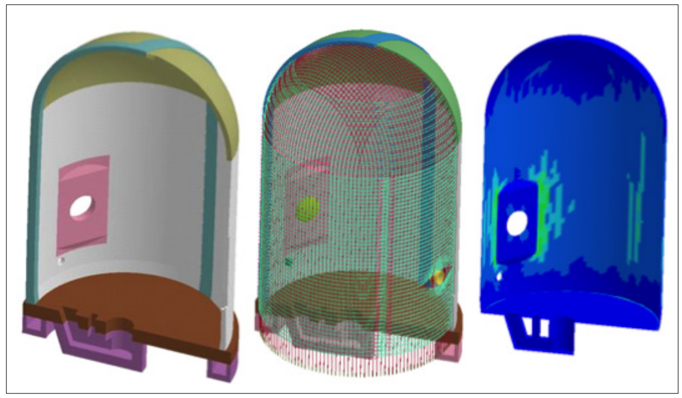

Figure 12.1 Finite element model of a prestressed concrete reactor containment and contours of maximum principal strain in the liner under beyond design basis pressurization.

<u>Steel Plate and Concrete Composite Modular Construction</u> – Some designs in the new generation of nuclear power plants have incorporated the use of steel plate and concrete composite modular (SC) construction in safety-related structures such as structures that support the reactor coolant system. RES sponsors research at Brookhaven National Laboratory to review the technical bases for the first U.S. design standard for SC construction to inform the NRC guidance. A research grant to Purdue University researches missile impact loads on SC walls to support development of design standards and related guidance. A new project will research methods to assess the condition of SC structures following seismic or other high-demand events.

<u>Seismic Isolation Technology Research</u> – Seismic isolation technology can susbtantially reduce the transmission and amplification of seismic ground motion to equipment designed to prevent and mitigate accidents. Research at Lawrence Berkeley National Laboratory and at the University at Buffalo studies the technical bases for the formulation of design performance expectations for isolation systems for use in nuclear power plants. This research involves confirmatory testing of isolators, model development, and sensitivity analyses using design basis and beyond design basis ground motions to gain insights for design and review guidance.

Concrete Irradiation Effects on Structural Performance

Objective

The primary objective of the research on concrete aging issues is to study the structural performance of aged concrete structures for its intended functions for long-term operations (up to 80 years). This research reviews, evaluates, and augments confirmatory analyses and testing the technical basis on the effects of concrete irradiation on structural performance to inform development of regulatory documents such as the Standard Review Plan and the Generic Aging Lessons Learned (GALL).

Research Approach

Neutron and gamma irradiation of concrete structures (reactor supports and shielding structures) can affect dimensional and physical properties of concrete (e.g., aggregate expansion, cement paste micro-cracking, reduction of compressive and tensile strength) that may affect structural performance and shielding capacity. Concrete structures in nuclear reactor containments in the proximity of the reactor vessel (e.g., the primary and biological shield walls and the reactor vessel support structures) can be subjected to high levels of neutron and gamma radiation under sustained operating temperatures up to about 150° F. For long-term operations, the radiation fluence/dose experienced by the concrete in these structures may approach levels that degrade the concrete. RES is starting a confirmatory research program to assess the structural and safety significance of concrete irradiation for long-term operations.

To scope the research efforts, the staff established five goals for the research on irradiation effects on concrete as follows:

- Provide the basis for radiation thresholds that will cause significant concrete degradation.
- Estimate the bounding fluence/dose for long-term operations (up to 80 years).
- Characterize the damage to concrete structures.
- Identify the structural and shielding safety significance of degradation from radiation and temperature.
- Inform aging management and monitoring programs.

Recently, independent research conducted by the U.S. Department of Energy (DOE) significantly augmented the available information on irradiation effects on concrete. The NRC's planned research will review this augmented data and the testing conditions for the data therein to evaluate the use of it in identifying irradiation thresholds that can lead to significant mechanical or physical degradation. This research also explores testing of irradiated concreted harvested from decommissioned nuclear power plants for confirmatory purposes.

Status

A research plan is being developed and implemented to inform the subsequent license renewal process of the safety significance of the combined effects of concrete irradiation and sustained elevated temperatures for structures in the proximity of the reactor vessel to address the five goals listed above.

In 2015, the NRC contracted Oak Ridge National Laboratory to study the possible scope of such testing and to inform the NRC of testing requirements including radiation and temperature environments (e.g., fluence rate). The Office of Nuclear Regulatory Research (RES) established bi-lateral collaborative research agreements with regulators in other countries to explore harvesting of irradiated concrete from decommissioned nuclear power plants in those countries. Also, RES staff participates in the International Committee on Irradiated Concrete (ICIC) in nuclear power plants to gather information on testing facilities and opportunities for testing of irradiated concrete.

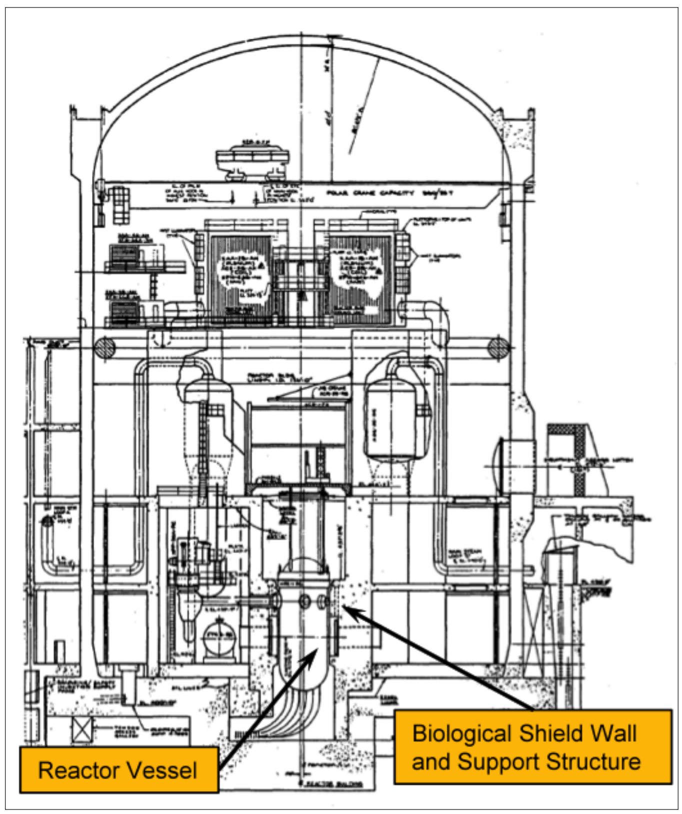

Figure 12.2 Location of the biological shield wall and support structure in a pressurized-water reactor [NUREG/CR-5640].

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Chemical Degradation of Concrete and Structural Effects

Objective

Alkali-Silica Reaction (ASR) is a chemical degradation in concrete that may occur over time as a reaction between the highly alkaline cement paste and reactive non-crystalline (amorphous) silica found in many common aggregates. This reaction causes the expansion of the altered aggregate by the formation of a swelling gel of Calcium Silicate Hydrate (C-S-H). The gel increases in volume with water and exerts an expansive pressure inside the material that may cause spalling and loss of strength of the concrete. The time-dependent structural capacity of ASR-affected concrete structures needs to be assessed for license renewal decisions for nuclear power plants (NPPs).

The objective of the research study is to develop the technical basis and regulatory guidance for NRC staff to evaluate ASR-affected concrete structures. The research assesses the structural performance of ASR-affected concrete structures for design basis static and dynamic loading and load combinations through its service life including the operations for the 20-year license renewal period. The overall research outcome will be a methodology to determine, for an existing ASR-affected structure, (1) its current structural capacity to resist static and dynamic loads and (2) an estimate of future structural capacity to resist static and dynamic loads.

Research Approach

The research study consists of six tasks. Tasks 1 to 3 deal with the ASR effects on the structural properties of reinforced concrete structures. Tasks 4 to 6 deal with identifying and evaluating methods, including microstructural analyses, for determining the state and rate of the ASR reaction and its impacts on concrete design properties and material performance. The effects of the ASR on other degradation mechanisms such as corrosion of the steel reinforcement will also be evaluated. The following sections describe the progress of the research to date. Task 1 will use three large concrete block specimens (3 ft 6 in wide, 6 ft high and 16 ft long) and three compression specimens (2 ft x 2 ft x 4 ft high) all made with three different reactive aggregates. Each block specimen will consist of three regions. Each region will be fabricated with a different amount of hoop stirrups and ties, where Regions 1 and 3 signify, respectively, moderate and heavy confinement, while Region 2 represents minimal or no confinement.

Electric resistance gages will be used to measure strain in the reinforcing bars including longitudinal bars, stirrups, and ties. Tri-axial strain gauges will be embedded in concrete block specimens at selected locations to measure internal expansion of concrete due to ASR. Thermocouples will be imbedded in concrete block specimens to measure the internal temperature of the specimens during the course of the test. Surface expansion of the test specimens will be measured by means of demountable mechanical (DEMAC) gauges and a laser-tracking system.

All test specimens will be cast and kept in a large environmental chamber (about 32 ft wide x 48 ft long x 36 ft high). The chamber is being modified to accommodate the test specimens to maintain proper temperature and humidity over a 4-year period. Cores will be removed from the block specimens, and compressive and tensile tests under confinement pressure will be performed to determine the mechanical properties of the concrete (compressive strength, tensile strength, and modulus of elasticity) throughout the test duration.

In the concrete materials arena (Task 4), the materials group is working with the structures team to identify suitable mixture types for the reinforced structural elements. The primary performance criterion for the structural testing is the minimum 28-day compressive strength: 30-35 MPa (4000-5000 psi). The secondary performance criterion for the reinforced beams is the flow, or workability, properties. Because of the sensitive nature of the sensors and strain gauges, the concrete mixture would have to be placed with a minimum of vibration.

Status

Bulk materials, sensors, and monitoring and measurement equipment have been acquired. Test specimens have been designed and will be poured in late summer 2015. Data acquisition will begin in late summer 2015.

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Steel Plate and Concrete Composite Modular Construction

Objective

The objective of this research is to review, evaluate, and, as needed, augment using confirmatory analyses and testing the technical basis for evaluating the suitability and performance of steel plate and concrete composite modular construction in nuclear power plants. This includes the review of the adequacy of the technical bases for the first U.S. standard for the design of safety-related steel plate and concrete composite modular construction structures developed by the American Institute of Steel Construction (AISC). The information gained in this research will support the development of guidance for design, review, and in-service inspection.

Research Approach

The new generation of nuclear power plants incorporates the steel plate and concrete composite modular (SC) construction in some of their designs. This construction technology consists of two steel plates connected by trusses or tie bars. The steel plates also have shear transfer devices on the interior faces of the plates (Figure 12.3). Concrete is poured between the steel plates forming a composite solid panel. Nuclear reactor designs, such as Westinghouse's AP1000 and Mitsubishi's USAPWR, use SC construction in some of their safety-related structures.

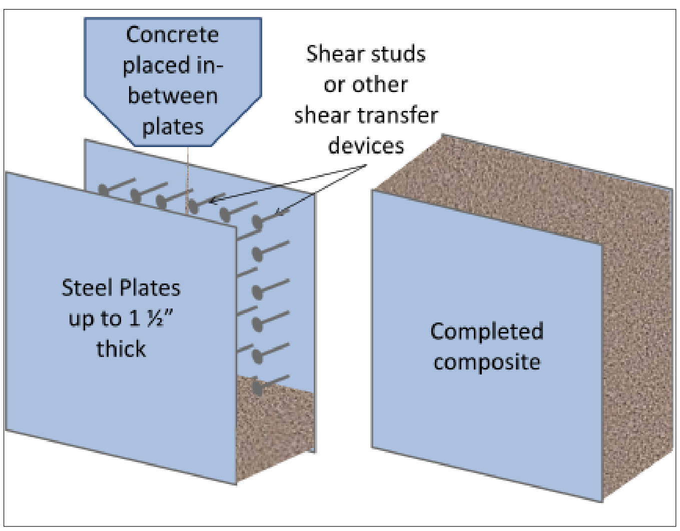

Figure 12.3 Schematic of steel plate and concrete composite modular construction.

Considerable research and testing has been done and continues to be done to understand the structural properties of SC construction for use in safety-related structures. Testing initiated in Japan more than two decades ago led to the 2010 SC design code (JEAC-4618) of the Japan Electric Association (JEA), and research in Korea led to the development of a SC design standard by the Korean Society of Steel Construction. More recent research in the United States (e.g., research sponsored by the U.S. Department of Energy at Purdue University) led to the development of the first U.S. standard for SC construction by the AISC scheduled for publication in 2015.

NRC research activities related to SC construction have been initiated in three areas:

- Review of SC design codes and standards.
- Review of standardized design methodology for impact.
- Condition assessment following a severe loading event.

Status

Supplement 1 to the AISC standard N690-2012, "Specification for Safety-Related Steel Structures for Nuclear Facilities," will include the first U.S. standard for the design of safety-related SC walls. In 2014, the NRC started research at Brookhaven National Laboratory to review the technical basis for this standard. This research will inform the staff how the standard meets the staff expectations and will be used to inform the development of related guidance for design and review. Completion of this research is expected in 2016.

The performance of SC construction under impact loads is expected to differ from that for reinforced concrete structures in a few aspects. One of the primary differences is the confinement of damaged concrete by the steel plates on the face opposite to the impact which tends to improve their impact performance. In 2014, the NRC awarded a 3-year grant to Purdue University for experimental and analytical research to confirm and, as needed, update SC design methodologies for impact in design standards. This research also will inform related NRC guidance for design and review.

There may be a research need to identify viable ways to assess the condition of SC structures after construction and throughout their service life, specifically after a seismic or other potentially damaging event. An assessment or inspection challenge related to SC structures is that the steel plates prevent the visual inspection of the concrete and significantly affect the performance of non-destructive examination techniques.

New research will evaluate the effectiveness of various approaches for structural condition assessment to inform, for example, regulatory actions for possible restart of operations after a potentially damaging event. The research, especially in relation to the condition assessment of SC structures following an event, will take into account an holistic approach to inspection and condition assessment involving a combination of inspection, testing, and analysis similar to the general approach in NRC, Electric Power Research Institute (EPRI), and International Atomic Energy Agency (IAEA) guidance for the restart of nuclear power plants following an earthquake. It will evaluate techniques and approaches therein in conjunction with, for example, NDE techniques.

For More Information

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Seismic Isolation Technology Research

Objective

The work in this area researches the technical bases to formulate design performance expectations for isolation systems for use in nuclear power plants. Seismic isolation technology has the potential to substantially reduce the transmission and amplification of seismic ground motions which results in reduced demands on safety-related structures and equipment. The research addresses design challenges for the possible implementation of base-isolation systems in nuclear installations. Examples of these challenges are the performance of the isolators for beyond-design-basis seismic events and the consideration of vertical seismic ground motions.

Research Approach and Status

NRC-sponsored research at Lawrence Berkeley National Laboratory and at the University at Buffalo studies the technical bases for the formulation of design and review guidance for the possible use of seismic isolation technology in nuclear power plants. The first element of the research consists of (1) review and confirmatory testing of isolator's properties to understand their response and failure mechanisms over a full range of demands of interest (Figure 12.4) including vertical motions, and (2) the development of analytical models and their implementation in software for the calculation of the seismic response of isolators and seismically isolated structures.

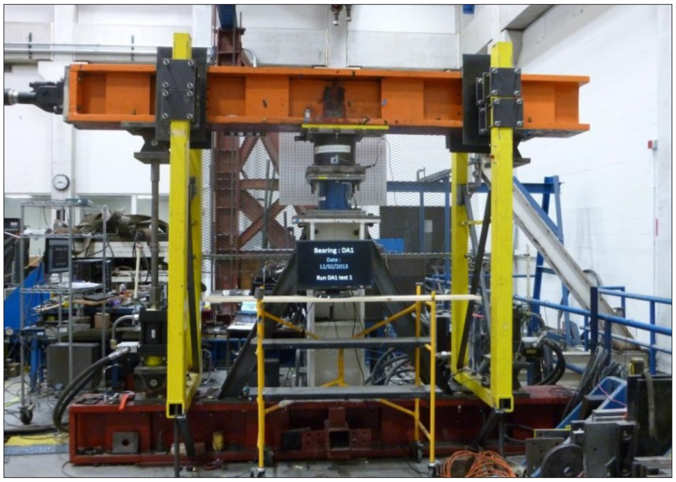

Figure 12.4 Single isolator testing apparatus at the University of Buffalo.

This research then uses those experimental results, models, and software to conduct extensive sensitivity analyses on the performance of a range of designs subjected to design basis and beyond-design-basis ground motions. These are ground motions with mean probabilities of being exceeded in the range of 1 in 10,000 to 1 in 100,000 per year. An important aspect of this research is to identify ground motions with characteristcs that are representative of various regional sites and various local site conditions as well as ground motions that than challenge the full range of isolator performance. Results of these analyses include the horizontal range of motion that the isolators need to accommodate which, together with the deformation capacity of the isolators, is necessary to understand performance criteria for safe arresting mechanisms for the isolation systems. Results of these extensive sentitivity and parametric analyses provide insights on the expected performance of various systems and diretly support the development of design and review guidance. These analyses will also provide a compendium of results that can inform the staff reviews of designs as well as supplement regulatory design guidance.

This research also addresses expectations for the modeling of the seismic response of seismically isolated nuclear power plants that differ from the modeling of non-seismically-isolated structures. An example is consideration of seismic soil-structure interaction methods that can account for the nonlinear behavior of the isolators and isolation systems.

In addition, the NRC remains cognizant of international and other domestic research in this area. Examples are activities in the International Seismic Safety Center of the International Atomic Energy Agency and research that the U.S. Department of Energy sponsors at Idaho National Laboratory.

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Cooperative Structural Performance Research

Objective

To ensure that the structural engineering performance research program accounts for and leverages ongoing relevant research being done by other U.S. and international agencies and institutions working on the safety of nuclear installations, the Office of Nuclear Regulatory Research (RES) maintains collaborative research programs that involve exchange of technical information, round robin analyses, and as needed, collaborative testing and analysis with those organizations.

Research Approach and Status

Concrete Aging Issues

The research in this area reviews, assesses, and as needed, augments using confirmatory analyses and testing the technical basis that informs the development of regulatory documents such as the Standard Review Plan and the Generic Aging Lessons Learned (GALL). Much of the research in this area is conducted under the U.S. Department of Energy (DOE) Light-Water Reactor Sustainability Program (LWRS) and under the Electric Power Research Institute (EPRI) to support the Long-Term Operations (LTO) program for nuclear power plants. RES has separate collaborative research agreements with each of these organizations to exchange technical information. This exchange is essential for the review and assessment of the technical bases for the viability of long-term operations. RES staff has frequent technical interchange meetings with DOE and EPRI staff. These exchanges have concentrated on concrete irradiation effects, aging management, and supporting technologies like non-destructive examination (NDE) and Alkali-Silica Reactions (ASR) effects.

RES also participates in the activities of the International Committee on Irradiated Concrete (ICIC). This interaction is especially relevant to identify opportunities for harvesting and testing irradiated concrete from decommissioned plants for confirmatory purposes. International bi-lateral agreements with Japan, Finland, France, and Spain, who are engaged in similar research programs, complement these activities. Regarding ASR effects, RES participates in activities of the Working Group for Integrity and Aging of Structures and Components of the Nuclear Energy Agency's Committee for the Safety of Nuclear Installations (NEA/CSNI/WIAGE) as well as activities of the Committee on ASR of the International Union of Laboratories and Experts in Construction Materials Systems and Experts (RILEM).

Impact Research and Analysis Benchmarking

Currently, the NRC participates in two international collaborative research programs in this area. One is the IMPACT program with the Technical Research Center of Finland (VTT) and the other is a round-robin benchmarking analysis study within the auspices of the NEA/CSNI/WIAGE. The objectives of these programs are (1) to benchmark computer codes that the NRC staff and its contractors use in impact assessments and (2) to synthesize the results of benchmarking into recommendations for good practices. These studies are relevant for impacts of wind-borne missiles on safety-related structures during tornados and hurricanes. In addition, the NRC believes that it is prudent for nuclear power plant designers to take into account the potential effects of the impact of a large, commercial aircraft on nuclear facilities. Anticipated benefits to the NRC from its participation in these programs include (1) reducing uncertainty associated with confirmatory assessments of impact loads on nuclear installations and (2) ensuring that assessments performed for U.S. reactors represent the state of the art.

The NRC, the VTT, and nuclear regulators and nuclear safety research organizations in other countries participate in a multiyear international experimental research program called IMPACT to collect and analyze new data on the performance of reinforced and prestressed concrete walls subject to impact loads. The VTT provides all testing data for this program using unique testing facilities not readily available elsewhere in the world, while the technical work of the NRC and the other participants focuses on analytical efforts. The program is in its third phase and has tested more than 20 impacts on concrete walls (Figure 12.5) in each phase involving various types of walls and reinforcement, soft missiles, hard missiles, and liquid-filled missiles. The work is documented in draft joint reports and VTT reports and disseminated at technical conferences.

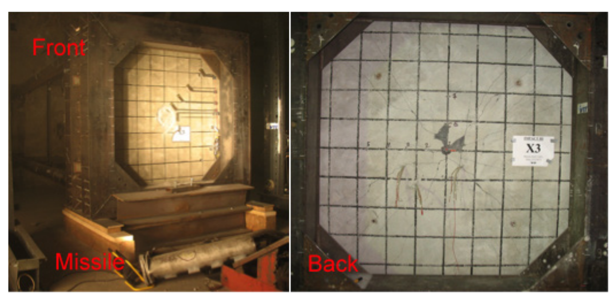

Figure 12.5 Impact of deformable missile on a concrete wall.

The collaborative research within the auspices of the NEA/CSNI/WIAGE entered its third phase that will study propagation of vibrations to structural components away from the impact wall. Results and conclusions of the second phase are in the NEA report NEA/CSNI/R(2014)5 and its addendum.

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Chapter 13: Digital Instrumentation and Control and Electrical Research

Digital Instrumentation and Control Research Program

The digital instrumentation and control (I&C) area continues to evolve as the technology changes and the NRC continues to refine its regulatory approach. As operating nuclear power plants (NPPs) upgrade their control rooms, analog equipment is being replaced with modern digital equipment including flat screen operator interfaces and soft controls. Future plants will have highly integrated control rooms similar to those in Figure 13.1. As a result, the NRC continues to improve applicable licensing criteria and regulatory guidance and to perform research to support licensing these new digital I&C systems. In an effort to continually improve the licensing process, the NRC accepted a recommendation to update the NRC research program balance shortterm regulatory needs and longterm anticipatory research needs from the National Research Council report, "Digital Instrumentation and Control Systems in Nuclear Power Plants." The Advisory Committee on Reactor Safeguards (ACRS) also encouraged research in the digital I&C area to keep pace with the everchanging technology.

Figure 13.1 Highly Integrated Control Room.

Overall Program

The Office of Nuclear Regulatory Research (RES) developed a comprehensive 5year Digital System Research Program Plan that defined the I&C research to support the regulatory needs of the agency. The agency periodically reviews and updates the Digital System Research Plan with input from the Advisory Committee on Reactor Safeguards (ACRS), external stakeholders, and the NRC staff. The updated research plan consists of key research program areas: (1) safety aspects of digital systems, (2) security aspects of digital systems, (3) knowledge management, and (4) projects supporting license office user needs. The products of these research programs include technical review guidance, information to support regulatory-based acceptance criteria, assessment tools and methods, standardization, and knowledge management initiatives.

Analytical Assessment of Digital Safety Systems

Ongoing research is exploring the state of the art in analysis of safety critical digital systems and examining the need for new regulatory review tools such as the use of system hazard analysis, a safety demonstration framework, and guidance for review of software tools. RES has published research studies in Research Information Letters (RIL) – 1001, "Software-related Uncertainties in Assurance of Digital Safety Systems - Expert Clinic Findings, Part 1;" RIL -1002, "Identification of Failure Modes in Digital Safety Systems – Expert Clinic Findings, Part 2;" and RIL -1101, "Technical Basis to Review Hazard Analysis of Digital Safety Systems."

Digital System Probabilistic Risk Assessment

Research supporting the goal of riskinforming digital system reviews is investigating an acceptable method for modeling digital system reliability for use in probabilistic risk assessment. One of the major challenges is developing an acceptable method for modeling digital system reliability. The staff examined a number of reliability and risk methods that have been developed in other industries such as aerospace, defense, and telecommunications. Based on its review of these techniques and available failure data, the staff performed benchmark studies of digital system modeling methods including traditional eventtree, faulttree, and dynamic methods. Internal staff and ACRS reviews of the studies challenged the viability of the methods and the availability of data needed. Further research on the failure modes of digital systems and quantitative software reliability is being pursued.

Security Aspects of Digital Systems

Planned research supporting security aspects of digital systems will investigate improvements in the regulatory framework described in Regulatory Guide 5.71, "Cyber Security Programs for Nuclear Facilities." ACRS has expressed concerns with the integration of cyber security with the safety assessment of digital systems.

Digital Instrumentation and Control Probabilistic Risk Assessment

Objective

The objective of this research is to identify and develop methods, analytical tools, and regulatory guidance for (1) including models of digital systems in nuclear power plant probabilistic risk assessments (PRAs) and (2) incorporating digital systems in the NRC's risk-informed licensing and oversight activities.

Research Approach

The NRC has been investigating reliability modeling of digital systems, which encompasses both hardware and software, for several years. Previous projects identified a set of desirable characteristics for reliability models of digital systems and assessed candidate methods against these attributes. In the area of digital hardware reliability, a simulation-based tool has been developed to determine the combinations and sequencing of component level failures that could impact system functions. Current research efforts are focused on developing methods for quantifying software reliability.

As an initial step in this area, an expert panel was convened to establish a philosophical basis for modeling software failures in a reliability model. After reviewing several quantitative software reliability methods, two methods apply to an example software-based protection system in a proof-of-concept study: the Bayesian Belief Network (BBN) approach and the statistical testing method. These methods are being applied to the Loop Operating Control System (LOCS) of the Idaho National Laboratory (INL) Advanced Testing Reactor (ATR). The work has highlighted several areas needed for additional research for PRA modeling of digital systems including the following:

- 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.
- Data on hardware failures of digital components including addressing the potential issue of double-crediting fault-tolerant features such as self-diagnostics.
- Data and methods for modeling common-cause failures (CCFs) of digital components.
- Methods for addressing human reliability and modeling uncertainties in modeling digital systems.

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. Therefore, a goal of this research program is to assess the practicality and usefulness of including digital systems in nuclear plant PRAs.

Status

Recent accomplishments and near-term objectives include the following:

- Development of a failure mode taxonomy for a digital instrument and control (I&C) system performed by the OECD/NEA Working Group on Risk Assessment (WGRISK) (NEA/CSNI/R(2014)16
- WGRISK Task DIGREL - Failure modes taxonomy for reliability assessment of digital I&C systems for PRA.)
- In collaboration with the Korea Atomic Energy Research Institute, quantify software reliability using BBN-based on software development cycle quality attributes.
- Estimate the reliability, including software, of the ATR LOCS using PRA-based statistical testing.

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Analytical Assessment of Digital Instrumentation and Control Systems

Objective

This research project is driven, in part, by the Commission's Staff Requirements Memoranda (SRM)-M080605B, "Meeting with Advisory Committee on Reactor Safeguards (ACRS), Thursday, June 5, 2008." The SRM directed the staff to investigate the use of digital system failure modes. The NRC's regulatory offices also have expressed needs for additional analytical assessment tools through the fiscal year (FY) 2010–FY 2014 "U.S. Nuclear Regulatory Commission (NRC) Digital Systems Research Plan" (Agencywide Documents Access and Management System [ADAMS] Accession No. ML100541484).

Research Approach

The current NRC regulatory guidance framework was intended for analog instrumentation and control (I&C) technology. Traditional hazard analysis techniques (such as failure modes and effects analysis) that demonstrate satisfaction of requirements for analog I&C have limitations when applied to systemic concerns in digital I&C systems. As a result, the NRC is also examining new hazard analysis methods (Figure 13.2) and a safety demonstration framework. The safety demonstration framework includes research on the application of evidence-argument-claim structures (also known as assurance cases or a safety case). This research will explore mapping the NRC's regulatory guidance framework into a safety-goal oriented, evidence-argument-claim framework.

This research will provide the technical basis for reasonable assurance determinations in digital I&C safety reviews made with hazards analysis methods and safety demonstration framework analytical tools. Using computer science and systems engineering knowledge, this project is researching analytical methods that can improve the review of safety critical digital I&C systems and components. The research primarily involves assessing the state-of-the-art knowledge in assurance of safety critical systems by consultation with experts in software systems engineering. The NRC has established an Interagency Agreement with the Carnegie Mellon Software Engineering Institute and is also collaborating with a number of academic researchers in this area. In a collaborative effort with the Halden Reactor Project, the staff is supporting a research project investigating use of a safety demonstration framework.

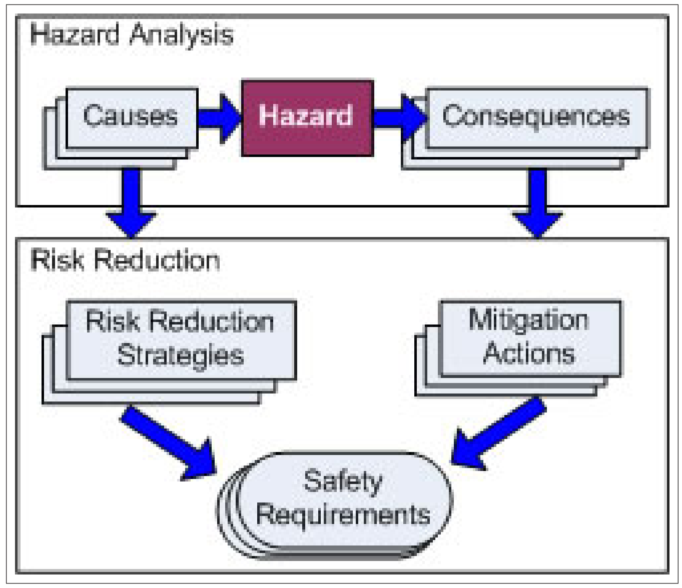

Figure 13.2 Hazard analysis.

Status

Three Research Information Letters (RILs) address the Commission's directions to the staff in SRM - M080605B. RIL-1001, "Software Related Uncertainties in the Assurance of Digital Safety Systems – Expert Clinic Findings, Part 1," dated May 4, 2011, discusses uncertainties that impede reasonable assurance determinations of digital I&C safety systems containing software. RIL-1002, "Identification of Failure Modes in Digital Safety System - Expert Clinic Findings, Part 2" presents a synthesized generic set of digital I&C system failure modes with a discussion of benefits and limitations for use in regulatory reviews. RIL-1003, "Feasibility of Applying Failure Mode Analysis to Quantification of Risk Associated with Digital Safety Systems – Expert Clinic Findings, Part 3," will discuss the feasibility of applying failure mode analysis to quantification of risk associated with digital I&C systems. RIL-1003 is scheduled to be completed in 2015.

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Digital Instrumentation and Control Cooperative Research

Objective

The NRC Strategic Plan discusses the importance of domestic and international collaborations to foster sharing of lessons learned, operational experience, and regulatory experience. The NRC values collaborative research that supports improved safety and effective and efficient licensing of the new digital systems, a key research area for both domestic and international research organizations.

Research Approach

The Digital System Research Plan's research program in the Knowledge Management topic area describes staff cooperative and collaborative initiatives for digital instrumentation and control (digital I&C) that support agency strategic goals. Collaborative research efforts in the United States and internationally support sharing standards and research data for digital systems. The products of these collaborations include technical review guidance, information to support regulatory based acceptance criteria, assessment tools, methods, and standardization.

Status

Domestically, the NRC has a research Memorandum of Understanding with the Electric Power Research Institute (EPRI) in key technical areas that support collaborative research and sharing of research results. Work in the area of safety aspects of digital systems includes analytical assessment research to support safety evaluations of digital I&C systems. Ongoing research in failure modes is examining the need for new regulatory review tools such as the use of system hazard analysis. The NRC and the industry are interested in risk informing digital safety system licensing reviews. EPRI has conducted research and developed a potential process for digital system probabilistic risk assessment development.

The NRC participates in interagency research and development working groups to share experience and analysis techniques with other Federal Government agencies such as the National Aeronautics and Space Administration (NASA), Federal Drug Administration (FDA), Federal Aviation Administration (FAA), and U.S. Department of Defense (DOD).

Figure 13.3 Halden Reactor Project.

Internationally, the NRC provides funding for research conducted by the Halden Research Project (Figure 13.3). Research collaboration for a safety demonstration framework will improve the understanding of criteria to ensure that these systems will not compromise their safety functions and will not adversely affect nuclear power plant (NPP) safety.

Staff work in standard development organizations such as the Institute of Electrical and Electronic Engineers and the International Electrotechnical Commission supports international NPP digital system standards harmonization and NRC knowledge management and regulatory efficiency improvements.

For More Information
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Electrical Research Program

Electrical systems at nuclear power plants (NPPs) range from high-voltage switchyard to medium-voltage power distribution and low-voltage AC/DC control power to the backup and emergency power sources and station DC batteries. Safe and resilient electrical systems are critical to safe nuclear operations. The Fukushima event demonstrated the severe safety impacts of loss of offsite and onsite emergency power.

The NRC staff developed a comprehensive Electrical System Research Program Plan that defined the research needed to support the regulatory needs of the agency. The Electrical System Research Program consists of a number of research projects that are focused on investigation of the critical design and performance aspects of these systems in operating NPPs and new reactors. Research topics are responsive to License Office regulatory needs.

Figure 13.4 Electrical switchgear.

Power Source Reliability

Several projects are assessing the performance of offsite and onsite normal and emergency power sources to review design and maintenance adequacy and ensure system reliability is meeting regulatory requirements. Susceptibility of NPPs to loss of offsite power events was studied and reported in NUREG/CR –7174, "Susceptibility of Nuclear Stations to External Faults." Planned research will investigate reliability of onsite normal and emergency power in plant events.

Station Battery System Testing

Confirmatory research projects on NPP station batteries and DC distribution systems are conducting testing to confirm maintenance and surveillance practices, testing to predict battery performance in extended loss of AC power scenarios related to severe accident response, and testing to resolve concerns observed in industry events. Initial NRC-sponsored testing conducted at Brookhaven National Labs was published in NUREG/CR –7148, "Confirmatory Battery Testing: The Use of Float Current Monitoring to Determine Battery State-of-Charge."

Electrical Cable Qualification and Condition Monitoring

A key issue for the current fleet of operating NPPs is aging management programs for license renewal, and a key technical area is cable qualification and condition monitoring.

Research in this area is investigating methods used for simulated aging of electrical equipment as well as condition monitoring to confirm that past equipment qualification practices were adequate and to determine optimum condition monitoring methods to monitor cable aging in periods of extended license renewal.

Ongoing research projects have obtained new and naturally aged cable samples that will be subjected to synergistic effects of radiation temperature and humidity similar to that seen in operating NPPs. A number of condition-monitoring techniques will be applied during and following aging protocols to determine condition-monitoring method predictive capability. Finally, the synergistically aged cables will be subject to loss-of-coolant accident testing to determine qualification adequacy and margins.

Electrical Cable Qualification and Condition Monitoring

Objective

The NRC confirmed in its review of Generic Letter (GL) 2007-01, "Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients," that electrical cables are often overlooked or ignored in aging analyses and condition monitoring evaluations because they are passive components that are generally considered to require no routine inspection and maintenance. However, electrical cables are very important safety components because they provide power to safety-related equipment and are used for instrumentation and control of safety functions. GL 2007-01 showed that a significant number of failures occurred under normal service conditions within the service interval of 20-30 years, which is before the renewed license period and before the end of the expected life span of the cables.

A variety of environmental stressors in nuclear power plants can influence the aging of electrical cables such as temperature, radiation, submergence/moisture/humidity, vibration, chemical spray, and mechanical stress. Exposure to these stressors over time can lead to degradation that may go undetected unless the aging mechanisms are identified and electrical, mechanical, or physical properties of the cable are monitored.

The objective of this research is to confirm the adequacy of cable qualification methods including the synergistic effects of radiation and temperature aging and condition-monitoring methods. These methods include (1) mechanical condition indicators, (2) dielectric condition indicators, and (3) chemical indicators.

Research Approach

The first phase of the project will focus on assessing condition-monitoring techniques during normal operational aging. Thus, cables will be subjected to normal operating conditions (temperature, radiation, humidity) in both mild and harsh environments to simulate use up to 60 years. For better estimates of cable performance, the aging will be performed synergistically at low dose and low temperature for 18 months to produce homogeneous degradation in the cable samples (i.e., appropriate acceleration factors under oxidative conditions).

The second phase of the project will focus on cables subject to accident conditions in harsh environments. The cable samples will be exposed to simulated accident (temperature, pressure, humidity, radiation, chemical/steam spray) conditions. The condition-monitoring techniques will be evaluated for the capability to predict proper operation of cables during and after the accident (post-accident period). The post-accident period may vary but could be up to 45 days.

Status

The NRC has contracted with the National Institute of Standards and Technology (NIST) to conduct this research. NIST is using University of Maryland facilities for radiation exposure of cable samples. The research is scheduled to be completed in 2017 with publication of a NUREG/CR.

For More Information
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Battery-Testing Program

Objective

The NRC is sponsoring confirmatory nuclear station battery testing at Brookhaven National Laboratory (BNL). The research program will validate if the batteries generally used in the nuclear industry remain in a fully charged condition and operational readiness while in standby, and it will determine if charging current is a suitable indicator of a fully charged condition for leadcalcium batteries. The research also will determine the batteries' ultimate performance capabilities for extended loss of alternating power (ELAP) conditions. Lastly, it will validate whether the individual short-circuit current contributions of a battery and a battery charger(s) are independent of each other in a typical nuclear power plant direct current system configuration.

Research Approach

To ensure that the battery has the capability to perform its design function following discharges or surveillance testing, the staff initiated the research and arranged the testing of batteries to be performed in three phases: (1) evaluation of charging current as a monitoring technique, (2) evaluation of the use of charging current to monitor battery capacity, and (3) evaluation of the criteria for selecting the point when a battery can be returned to service and meets its design requirements.

To evaluate the batteries' response to extended loss of alternating current power conditions, the staff tested plant ELAP profiles from four nuclear power plants (three pressurized-water reactors and one boiling-water reactor).

To determine the short circuit characteristic performance of the two types of battery chargers, the staff will be conducting performance short-circuit tests (1) with the battery disconnected from the battery charger, (2) with the battery charger disconnected from the battery, and (3) with the battery connected to the battery charger and the battery individually.

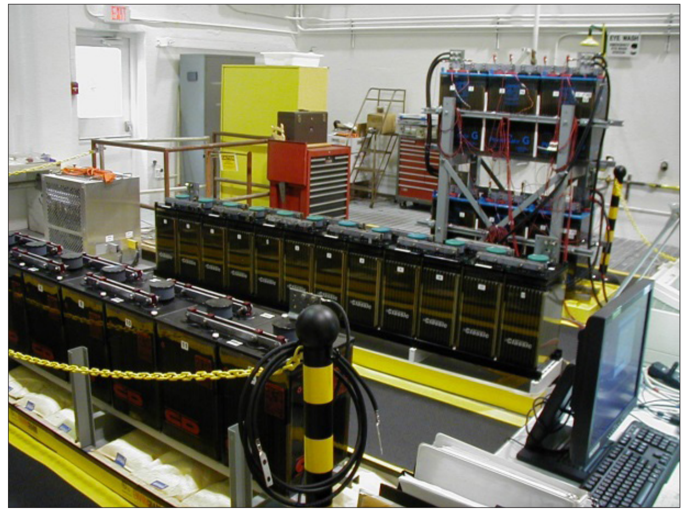

Figure 13.5 BNL battery facility.

Status

Figure 13.5 shows the BNL battery facility. Two tests have been completed, and results for the confirmatory battery testing are documented in NUREG/CR - 7148, "Confirmatory Battery Testing: The Use of Float Current Monitoring to Determine Battery State-of-Charge," and the extended battery testing will be soon be published in NUREG/CR- 7188, "Testing to Evaluate Extended Battery Operation in Nuclear Power Plant."

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Electrical Cooperative Research

Objective

The NRC Strategic Plan discusses the importance of domestic and international collaborations to foster sharing of lessons learned, operational experience, and regulatory experience. The NRC values collaborative research that supports improved safety, and effective and efficient licensing of electrical safety systems. NRC staff actively participates in both domestic and international research collaborations beneficial to the agency's mission.

Figure 13.6 EPRI headquarters.

Research Approach

The Office of Nuclear Regulatory Research (RES) developed a comprehensive Electrical Systems Research Program Plan, which defined the research to support the regulatory needs of the agency. As a key aspect of conducting electrical research, RES seeks beneficial cooperative research arrangements both nationally and internationally that can support NRC research objectives and improve the quality of research projects.

Status

Domestically, RES has a research Memorandum of Understanding with the Electric Power Research Institute (EPRI) supporting two of the electrical research programs—the Battery Testing research and the Electrical Cable Condition Monitoring research. In addition, in both these research program areas, the NRC has cooperated with the U.S. Department of Energy (DOE). DOE partially funded the NRC's battery testing research and is conducting significant research on electrical cables as part of the Light- Water Reactor Sustainability Program. EPRI and DOE have supported NRC research efforts by sharing technical information and expertise.

Internationally, RES has collaborated with International Atomic Energy Agency (IAEA) and Organization for Economic Development/Nuclear Energy Agency (OECD/NEA) initiatives in the area of electrical cable performance and condition monitoring. NRC participated in the IAEA Coordinated Research Project on Electrical Cables and the OECD/NEA Cable Aging Data and Knowledge project to collect international cable performance information.

Staff work in standard development organizations such as the Institute of Electrical and Electronic Engineers and the International Electrotechnical Commission support international nuclear power plant electrical system standards harmonization and NRC knowledge management and regulatory efficiency improvements.

For More Information
Contact Russell Sydnor, RES/DE, at Russell.Sydnor@nrc.gov.

Chapter 14: Fukushima Dai'ichi Accident Research

On Friday, March 11, 2011, a 9.0-magnitude earthquake struck Japan and was soon followed by a tsunami that was estimated to have exceeded 45 feet (14 meters) in height. The incident resulted in extensive damage to the six nuclear power reactors at the Fukushima Dai-ichi site. Since that time, the NRC has been working to understand the events in Japan and to relay important information to U.S. nuclear power plants (NPPs). The Office of Nuclear Regulatory Research (RES) has been supporting the agency's lessons learned effort.

In particular, the NRC established a Near-Term Task Force (NTTF) of senior agency experts to determine lessons learned from the accident and to initiate a review of NRC regulations to determine whether the agency needs to take additional measures to ensure the safety of U.S. plants. The NTTF issued its report entitled, "Recommendations for Enhancing Reactor Safety in the 21st Century," on July 12, 2011 (Agencywide Documents Access and Management System Accession No. ML111861807), which concluded that continued operation and licensing activities pose no imminent risk. The report also concluded that enhancements to safety and emergency preparedness are necessary and presented a dozen recommendations for the Commission's consideration.

The NRC subsequently prioritized and expanded the NTTF recommendations (SECY-11-0137, "Prioritization of Recommended Actions To Be Taken in Response to Fukushima Lessons Learned," dated October 3, 2011, ADAMS ML11272A111), and it continues to make additions and modifications as appropriate. The recommendations were divided into three tiers based on the urgency of the issues as described in SECY-11-0137.

The following major RES efforts resulting from the Fukushima Dai-ichi accident at the time of publication are described in more detail in subsequent sections of this chapter:

- Containment protection and release reduction analysis of Mark I and I boiling-water reactors.
- Potential enhancements to the capability to prevent or mitigate seismically induced fires and floods.
- Hydrogen control and mitigation inside containment and other buildings.
- Fukushima Dai-ichi accident study with MELCOR 2.1.
- Fukushima Cooperative Research.

Figure 14.1 Fukushima Units 1, 2, 3, and 4 after the accident showing extensive damage to the reactor buildings.

Containment Protection and Release Reduction Analysis of Mark I and II Boiling-Water Reactors

Objective

The objective of this study is to evaluate various post-accident containment overpressure protection and release reduction (CPRR) strategies by performing severe accident analysis using the MELCOR code and consequence analysis using the MACCS code and to provide a technical basis for regulatory analysis of these strategies in support of the ongoing CPRR rulemaking activities.

Research Approach

The research approach consists of (1) selection of risk-dominant accident sequences arising from an extended loss of alternating current power (ELAP) and as informed by probabilistic risk analysis (PRA), (2) MELCOR calculations of reactor pressure vessel (RPV) and containment thermal-hydraulics under severe accident conditions and an assessment of containment fission product retention, and (3) MACCS (MELCOR Accident Consequence Code System) calculations of offsite consequences including health risk, land contamination, and economic consequences.

The PRA covers development of core damage event trees (CDETs) and accident progression event trees (APETs) for an ELAP event, binning of a rather large number of possible end states to a manageable fewer categories, and an assessment of risks for these categories. The PRA activity also covers an assessment of risk reduction attributable to various accident management measures.

MELCOR calculations consist of a rather large number of accident sequences for a representative boiling-water reactor (BWR) Mark I containment and also a smaller subset of these sequences for a representative BWR Mark II containment. Mitigation measures accounted for in MELCOR calculations include both pre- and post-core damage venting, RPV pressure control, and water addition into the RPV as well as the drywell. In addition, variations in mitigation actions (e.g., vent cycling, wetwell vs. drywell venting, water management, etc.) and variations in engineered safety systems performance (e.g., reactor core isolation cooling system operation, safety release valve, etc.) are captured through sensitivity studies.

For each MELCOR calculation and its corresponding source term (i.e., fission product release into environment), MACCS calculations are performed for a representative plant site with specified site characteristics, population density, meteorological conditions, emergency management, and other aspects. Variations in site characteristic parameters are accounted for in a large number of MACCS sensitivity calculations, which also include the influence of an external engineered filter on relevant figures of merit related to health risk, land contamination, and economic consequences.

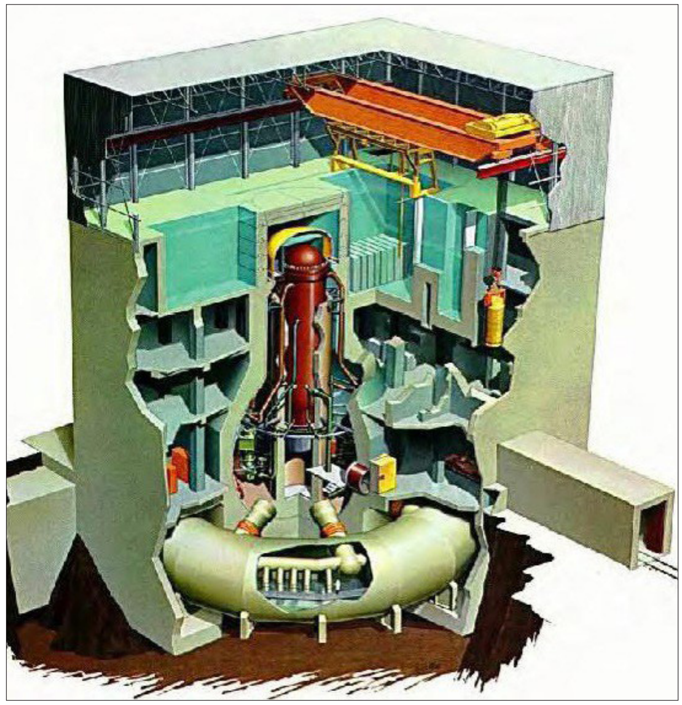

Figure 14.2 Schematic of a boiling-water reactor with Mark I containment.

Status

MELCOR severe accident analysis, MACCS consequence analysis, and PRA activities have all been completed. The draft regulatory basis was documented in SECY-15-0085. Some major findings from the analysis are:

- A combination of venting and water addition is required to prevent containment failure, and water addition is a beneficial strategy for mitigating radiological releases.
- For the accident scenarios considered and source terms calculated, there is zero early fatality risk, and frequency-weighted individual latent cancer fatality (LCF) risk is orders of magnitude below NRC's QHO Safety Goal.
- LCF risk (per event) is dominated by long-term phase exposures to lightly contaminated areas.

For More Information

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Seismically Induced Fires and Floods

Objective

Seismically induced fires have the potential to cause multiple failures of safety-related structures, systems, or components (SSCs) and to induce separate fires in multiple locations at the site. Events, such as pipe ruptures (and subsequent flooding), could also cause such problems in multiple locations simultaneously. In addition, seismic events could degrade the capability of plant SSCs intended to mitigate the effects of fires and floods.

To address this issue, the NRC's Near-Term Task Force (NTTF) concluded that the staff should evaluate potential enhancements to the capability to mitigate seismically induced fires and floods. The NTTF identified this issue as Recommendation 3, "Evaluate Potential Enhancements to the Capability to Prevent or Mitigate Seismically Induced Fires and Floods." Although the staff believes that the use of traditional deterministic design-basis methods can enhance the capability to prevent seismically induced fires and floods, accident sequences and complex dependencies needed to evaluate the mitigation of these events can be done more systematically through probabilistic risk assessments (PRAs). Therefore, the staff initiated the development of an appropriate PRA methodology to support the eventual resolution of this issue.

Research Approach

The following activities are being conducted to resolve this issue:

- Continue development of PRA methods for seismically induced fires and floods. This will include two main subtasks:
  - Engage PRA standards development organizations to develop the technical elements and standards for the PRA method.
  - Complete a feasibility scoping study to evaluate PRA approaches for assessing multiple concurrent events.
- Reevaluate NTTF Recommendation 3 based on information obtained from Tier 1 activities and PRA method development activities as well as recommend further activities.

Following a December 2013 public workshop (ADAMS ML14022A249), work is continuing on completing the feasibility scoping study. The main objective of this scoping study is to better define the objectives and potential approaches for a PRA method suitable for assessing seismically induced fires and floods. Recent activities have resulted in the formation of two expert panels to address several specific PRA modeling issues. The expert panels include subject matter experts (i.e., seismic analysis, PRA, flooding and internal fire analysis) from industry, national labs, and the NRC.

Status

NRC staff has been gathering responses from the expert panel members and is analyzing the inputs. The expert panel results, together with other information, will be used to produce the final feasibility report later in calendar year 2015. NRC staff will explore the possibility of performing a pilot application of the proposed risk assessment approach with industry stakeholders. NRC staff will continue to monitor the progress of other NTTF recommendations related to this issue to factor appropriately more information related to seismic and flooding hazards and mitigation strategies into the eventual resolution of NTTF Recommendation 3.

For More Information
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Hydrogen Control and Mitigation Inside Containment and Other Buildings

Objective

In accordance with Title 10 of the Code of Federal Regulations (10 CFR) 50.44, "Combustible Gas Control for Nuclear Power Reactors," licensees are required to use various hydrogen control and mitigation schemes inside containment buildings depending on their unique design characteristics. As a result of insights and continued post-accident analyses of the Fukushima events, the NRC will reassess (under NTTF Recommendation 6) the hydrogen control rule as it relates to the various containment designs. In addition, the agency will evaluate connected buildings for the potential of combustible gas ingress and will determine what design enhancements may be necessary.

Research Approach

The NRC will reassess hydrogen control while recognizing the various interrelated operating aspects and conditions. For example, the Fukushima accident revealed that the primary containment pressure in boiling-water reactor (BWR) Mark I and II containments significantly exceeded its design limit, particularly as a result of hydrogen gas generated by severe core damage and relocation along with steam buildup. Licensees' severe accident management guidelines (SAMGs) address containment pressure control. However, damage to the equipment and other factors hampered the timely mitigation of increasing pressures in the Fukushima containments. As a result, hydrogen leaked into the associated reactor buildings. Therefore, pressure and hydrogen control for severe accidents in Mark I and II containments should now consider the effect of leakage into the reactor buildings.

Consequently, the NRC is reevaluating the integration of reliable containment venting strategies to follow up on NTTF Recommendation 5.1, which states, "Order licensees to include a reliable hardened vent in BWR Mark I and Mark II containments," and under Recommendation 5.2, which states, "Reevaluate the need for hardened vents for other containment designs, considering the insights from the Fukushima accident. Depending on the outcome of the reevaluation, appropriate regulatory action should be taken for any containment designs requiring hardened vents."

During postulated severe accident events in any containment design, venting containment is a form of hydrogen control; that is, removing a significant fraction of the gas from the primary containment and thus reducing the potential of hydrogen leaking into the adjacent buildings. Figure 14.3 is a pictorial overview that shows the relationship of containment venting and hydrogen control for differing containment designs. Because of the smaller primary containment relative to other designs, pressure control and venting are more strongly coupled to hydrogen control in the Mark I and Mark II containments.

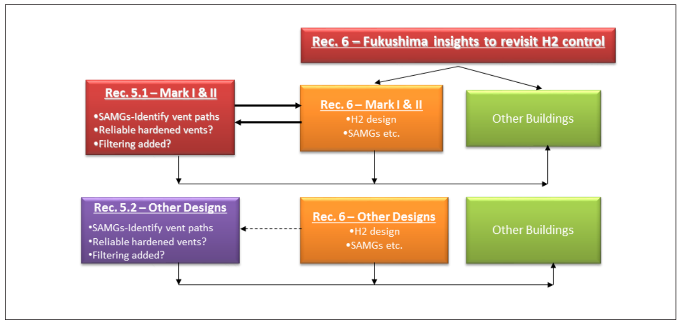

Figure 14.3 Relationships of NTTF Recommendations 5 and 6.

Using the MELCOR code, accident progression insights with respect to generation, transport, and combustion of hydrogen (and consideration of mitigation) are derived from the BWR Mark I and II studies performed under NTTF 5.1, the State-of-the-Art Consequences Analyses (SOARCA) project for the completed studies of Peach Bottom (BWR Mark I), Surry (large dry subatmospheric), and the ongoing effort to analyze Sequoyah (pressurized-water reactor ice condenser plant) will provide a useful foundation for the containment performance of these various containment types. As needed, the NRC will perform additional accident progression studies that will focus on containment performance and the potential adverse consequences on adjacent buildings, and the consideration whether some type of containment venting system is provided.

Status

Currently, the NRC is participating in an Organisation for Economic Co-operation and Development/Nuclear Energy Agency benchmark study of the accident at Fukushima. This effort will place particular emphasis on hydrogen generation from all sources and will compare the information derived to the current understanding used as the basis for existing hydrogen control and mitigation schemes. Also, the NRC participated in a working group connected to the same organization stated above and generated a final report titled, "Status Report on Hydrogen Management and Related Computer Codes."

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Fukushima Dai-ichi Accident Study with MELCOR 2.1

Objective

The NRC participates in the Organisation for Economic Co-operation and Development (OECD)/Nuclear Energy Agency (NEA) Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Station (BSAF) Phase I study.

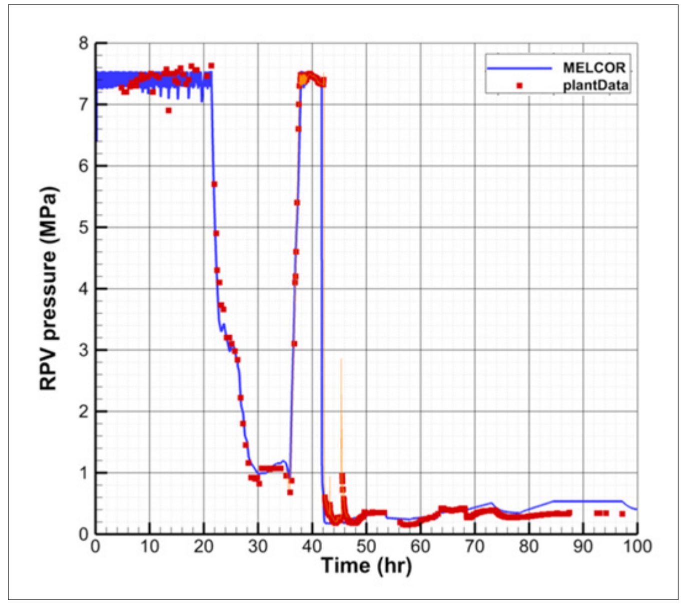
Research Approach

The NRC uses the MELCOR code to perform analysis of the Fukushima accidents. The BSAF Phase I study focuses on analyses covering six days (to March 17, 2011) from the initiation of the seismic event at the Fukushima plants on March 11, 2011. The analyses focus on thermal-hydraulics and an estimation of the distribution of degraded core materials and their composition. The six days duration was chosen because, from that time on, the plants were believed to achieve stable and continuous cooling by alternative water addition and plant parameters were stabilized.

The Operating Agent for this NEA project is the Japan Atomic Energy Agency (JAEA). JAEA is supported by the Institute of Applied Energy (IAE) who serves as the technical coordinator for the study. Eight countries—Japan, United States, Russia, France, Germany, Switzerland, Republic of Korea, and Spain—are participating in this NEA project. Many severe accident codes including MAAP4, MELCOR, SAMPSON, SOCRAT, ASTEC (IRSN), and ATHLET-CD/COCOSYS were used by participants for the analyses.

Status

The NRC has completed the MELCOR analysis of the Fukushima Unit 1 and 3. Figure 14.4 shows an example of MELCOR 2.1 prediction vs. measured data of Fukushima Unit 3. A final report on BSAF Phase I has been completed, and it is under review by participants.



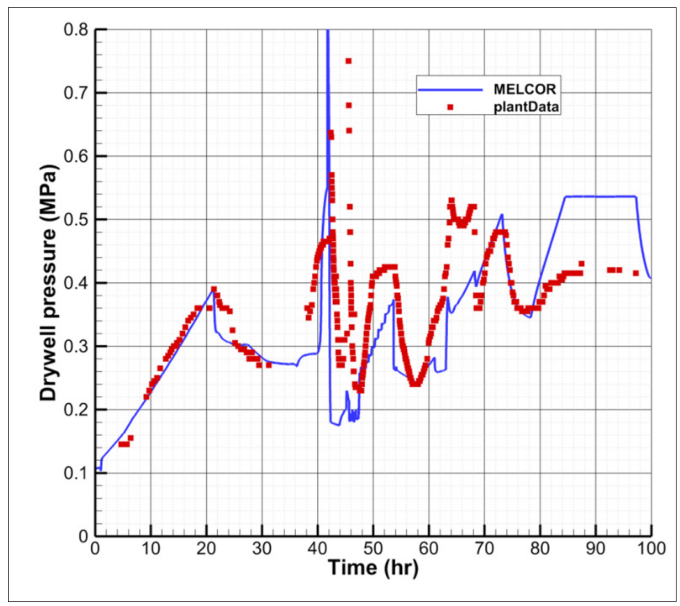


Figure 14.4 MELCOR-predicted reactor (top) and containment (bottom) pressures compared to TEPCO data (Unit 3).

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Fukushima Cooperative Research

Objective

The NRC participates in the Organisation for Economic Co-operation and Development (OECD)/Committee on the Safety of Nuclear Installations (CSNI)/Nuclear Energy Agency (NEA)-led activities follow-up to the Fukushima Daiichi Accident.

Research Approach

OECD/NEA published a report, "The Fukushima Daiichi Nuclear Power Plant Accident – OECD/NEA Nuclear Safety Response and Lessons Learnt," NEA 7161, 2013. The report discussed immediate response by NEA member countries and the NEA actions in follow-up to the Fukushima accident. The NEA follow-up actions described under "nuclear regulation" include actions taken to strengthening accident management, strengthening and implementing of the concept of defense-in-depth, review of precursor events, nuclear site selection, and crisis communication. The report also describes additional "nuclear safety" activities launched by NEA. The activities (known as CSNI Action Proposal Sheet [CAPS]) include:

- Filtered containment venting—a summary of the current status of the technology and venting strategies as well as developments required for possible improvements to filtration technologies (completed - June 2014).
- Hydrogen behavior—a status report providing the current knowledge base of hydrogen behavior, mitigation measures, and computer code validation (completed - June 2014).
- Probabilistic safety assessment (PSA) for natural external events—the proceedings for a workshop to share methods and commendable practices for PSA for natural external events (completed - June 2014).
- Robustness of electrical systems—the proceedings for a workshop describing the technical basis of the provisions already taken or planned after the Fukushima Daiichi NPP accident regarding electrical sources, distribution systems and loads (completed - December 2014).
- Spent fuel pool loss-of-coolant accident (LOCA)—a status report of the knowledge base for spent fuel pool accident phenomenology and mitigation measures and a guide for further research activities (completed - December 2014).
- Metallic margins under high seismic loads—a summary of the technology base and design practices for assessing aged metal component and piping response to high seismic loads (completed - December 2014).
- Human performance under extreme conditions—the proceedings for a workshop summarizing challenges during extreme events, good practices and knowledge gaps, and proposed principles for human performance under extreme conditions. (completed - December 2014).
- Benchmarking of fast-running emergency response codes—an assessment of existing response codes that estimate fission product releases and radiation doses for a range of accident scenarios and reactor designs (to be completed - December 2015).

In addition, NEA has launched several joint nuclear safety research projects:

- Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Station (BSAF).
- Hydrogen Mitigation Experiments for Reactor Safety (HYMERS).
- Pressurized-water reactor transient tests under post accident scenarios (PKL phase 3).
- Advanced Thermal-hydraulic Test Loop for Accident Simulation (ATLAS).

Status

NRC participated in many of the aforementioned CSNI CAPS writing groups, and most of the reports were completed in December 2014. NEA has also launched additional CAPS on (1) informing severe accident management guidance and actions through analytical simulation—to provide an assessment of severe accident management through modeling of operator actions in integral severe accident codes and to prepare a status report on best recommended practices and (2) long-term management of a severe accident in a nuclear power plant (NPP)—to review existing regulatory frameworks, practices, existing knowledge, and issues under consideration in OECD countries with respect to the management on the long term of a severe accident in a NPP.

The NRC as well as the U.S. Department of Energy and the Electric Power Research Institute participate in BSAF (described separately in this NUREG). The first phase of the project is near completion, and discussions are underway for a second phase to begin in 2015.

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Chapter 15: International and Domestic Cooperative Research

The NRC's Office of Nuclear Regulatory Research (RES) has implemented hundreds of cooperative agreements with international and domestic organizations. Experimental data, numerical procedures, and other analytical tools and methodologies are needed to fully understand and characterize the operation of nuclear facilities. The development of these tools and data add to the technical basis needed for safety determinations.

International and domestic cooperative programs have been developed in many research areas that allow for leveraging resources and minimizing duplication of effort. RES applies a set of established criteria when considering the cooperative research programs it agrees to participate in. Considerations include cost, benefit, timeliness of expected results for current and expected regulatory uses, and more. The cooperative programs for each research area are described in the Cooperative Programs sections for each chapter within this NUREG and summarized in this chapter.

RES has implemented over 100 bilateral or multilateral agreements with more than 30 countries and the Organisation for Economic Co-operation and Development (OECD). These agreements cover a wide range of activities and technical disciplines including severe accidents, thermal-hydraulic code assessment and application, digital instrumentation and control, nuclear fuels analysis, seismic safety, fire protection, human reliability, and more.

RES actively seeks international cooperation to obtain technical information on potential safety issues that require test facilities not available domestically that would require substantial resources to duplicate in the United States. RES often will 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 research projects it conducts. Bilateral exchanges with counterparts multiply the amount of information available to RES staff. As an example, RES has developed an extremely beneficial relationship with the Canadian Nuclear Safety Commission in the area of environmental modeling, groundwater monitoring, and more. Similarly, the NRC and the French Institute of Radiation Protection and Nuclear Safety (IRSN) cooperate in dozens of technical areas.

Many of the agreements are established bilaterally with a foreign regulator or research institution for participation in one of the two largest nuclear safety computer code sharing programs. The Code Applications and Maintenance Program includes thermal-hydraulic code analysts from more than 20 member nations. The Cooperative Severe Accident Research Program also includes more than 20 member nations that focus on the analysis of severe accidents using the MELCOR code. The Radiation Protection Computer Code Analysis and Maintenance Program (RAMP) is a new program for developing, maintaining, and distributing the NRC's radiation protection, dose assessment, and emergency response computer codes. These programs include user group meetings at which participants share experience with the NRC codes, identify code errors, perform code assessments, and identify areas for additional improvement, experiments, and model development.

The OECD's Nuclear Energy Agency (NEA) coordinates most of the NRC's multilateral research agreements. The NRC plays a very active role at the OECD/NEA with RES maintaining leadership roles in the Committee on the Safety of Nuclear Installations (CSNI) (including CSNI's seven working groups and joint research projects) and the Committee on Radiation Protection and Public Health. The RES Director is the Chairman of CSNI, and RES senior management represents the NRC on the Halden Reactor Project's Board of Management.

RES also serves as the agency lead on codes and standards. By acting as the agency lead in the International Atomic Energy Agency's (IAEA's) Nuclear Safety Standards Committee, RES coordinates NRC contributions to the many IAEA safety standards guides. RES also participates in two "extra-budgetary programs" within IAEA entitled, "Protection against Tsunamis and Post Earthquake Consideration in the External Zone," and "Seismic Safety of Existing Nuclear Power Plants," which feeds into IAEA's International Seismic Safety Center.

RES has long been a leader in the area of enhancing its resources with international and domestic knowledge, skills, and use of available research facilities worldwide. The staff has worked and continues to work to ensure that the international and domestic activities in which it participates have direct relevance to the NRC's regulatory program. For example, Memoranda of Understanding (MOU) between the NRC and EPRI and the NRC and DOE promote general information sharing and describe the parameters for conducting cooperative research programs between the two organizations. In addition, the NRC has established cooperative agreements, grants, and contracts with U.S. universities, laboratories, and agencies to conduct experiments, studies, and research programs.

NRC participation in these agreements allows broader sharing of experimental and analytical data. Data obtained are used to validate NRC safety codes, to improve analytical methods, to enhance assessments of plant risk, and to develop risk-informed approaches to regulation. As a result, NRC tools and knowledge stay current and are state of the art. This enhances the NRC's ability to make sound regulatory and safety decisions based on worldwide scientific knowledge that promotes the effective and efficient use of agency resources.

Halden Reactor Project

Objective

The NRC and its predecessor, the U.S. Atomic Energy Commission, have been participating in the Organisation for Economic Co-operation and Development/Nuclear Energy Agency (OECD/NEA) Halden Reactor Project (HRP) since its inception in 1958. HRP, which is located in Halden, Norway, is managed by the Norwegian Institute for Energy Technology (IFE) and operates on a 3-year research cycle, with the current program plan running from 2015–2017. The NRC benefits directly from HRP research, which maximizes the use of NRC research funds by leveraging the resources of other HRP participants. In addition, participation in the HRP facilitates cooperation and technical information exchange with the participating countries.

Research Approach

Fuels and Materials Research

The Halden boiling-water reactor (see Figure 15.1) is fully dedicated to instrumented in-reactor testing of fuel and reactor materials. Since its initial startup, the reactor facility has been progressively updated and is now one of the most versatile test reactors in the world. 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.

Recent NRC reviews of industry fuel behavior codes have directly employed data from the HRP fuels program. These data also are essential for updating the NRC's fuel codes and materials properties library, which are used to review and audit industry analyses. The NRC is particularly interested in loss-of-coolant accident (LOCA) tests, which address the effects of burnup, rod pressure, cladding corrosion, and absorbed hydrogen on integral fuel behavior during a LOCA.

The HRP's nuclear reactor materials testing program has provided fundamental technical information to support the understanding of the performance of irradiated reactor pressure vessel materials and supplemented results generated under NRC research programs. There are plans for HRP's materials testing program to investigate the irradiation-assisted stress corrosion cracking of weld materials harvested from the decommissioned Zorita reactor in Spain.

Figure 15.1 Halden boiling-water reactor.

Man-Technology-Organization Research

The Norwegian IFE research facilities also include several labs for Man-Technology-Organization (MTO) research. Among those is the Halden Man Machine Laboratory (HAMMLAB) (see Figure 15.2). HAMMLAB uses a reconfigurable simulator control room that facilitates research into instrumentation and control (I&C), human factors, and human reliability analysis (HRA). 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. Currently, ongoing HRP experiments are addressing a number of topics of interest to the NRC including control room staffing strategies, the role and effects of automation in advanced control room designs, and aids to improve control room teamwork. The NRC expects that this research will contribute to the technical basis for human factors guidance, especially for new reactor designs.

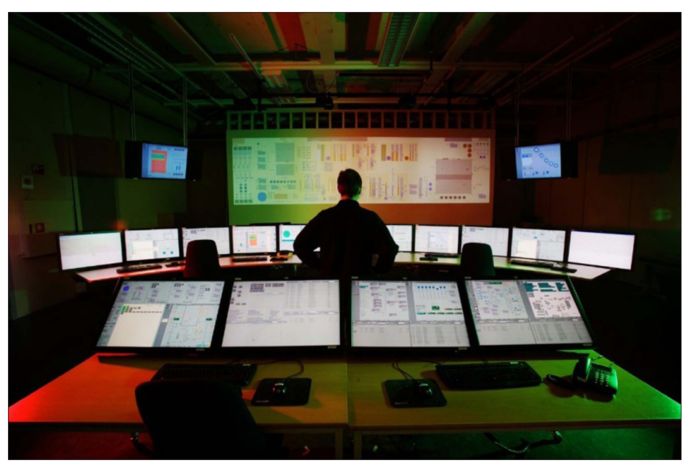

Figure 15.2 HAMMLAB control room simulator.

The MTO laboratory also conducts research in the area of digital I&C. NRC's primary interest in this area is the development of a safety demonstration framework that directly supports NRC's Digital I&C Research Plan and is of high value for developing regulatory guidance. HRP has coordinated international expert elicitations on the topic of developing a safety demonstration framework for digital I&C systems.

Status

More information regarding the NRC's participation in the OECD Halden Reactor Project can be found in SECY-14-0142 in ADAMS at ML14294A008.

For More Information
Contact Matthew Hiser, RES/DE, at Matthew.Hiser@nrc.gov.

International Operating Experience Database

The Organization for Economic Co-operation and Development (OECD) is an intergovernmental organization of industrialized countries. The Nuclear Energy Agency (NEA) is an agency within the OECD with the mission to assist its member countries in developing the scientific, technological and legal bases required for safe use of nuclear energy. The NEA's current membership consists of 31 countries in Europe, North America and the Asia-Pacific region that together account for approximately 86% of the world's installed nuclear capacity.

Within the NEA, the Committee on the Safety of Nuclear Installations (CSNI) consists of representatives for regulatory organizations that are responsible for conducting research to support regulatory decisions. The Office of Nuclear Regulatory Research (RES) of the U.S. Nuclear Regulatory Commission (NRC) is the U.S. representative on CSNI. Under the auspices of CSNI, its member countries conduct joint research projects on safety-significant topics. Currently RES participates in several CSNI-sponsored database projects that aim to capture international operating experience and share knowledge related to cable aging, component degradation, fires, and common-cause failures.

Cable Aging Data and Knowledge (CADAK) Project

Low- and medium-voltage electrical cable systems consist of cables, terminations, and other associated components (such as cable trays, penetrations, and conduit) used to power, control, and monitor various types of electrical apparatus and instrumentation. These cable systems are constructed of materials that are susceptible to age-related degradation and, if the degradation is severe, cable failure can result. CADAK aims to establish the technical basis for assessing the qualified life of electrical cables in light of age-related degradation mechanisms identified subsequent to initial qualification testing. This project intends to investigate the adequacy of the margins and their ability to address age-related degradation.

The following three specific objectives have been targeted to achieve this goal:

1. develop a database on nuclear power plant (NPP) cables that defines the scope of the effort;.
2. develop a database on monitoring and performance prediction related to every unique NPP application of cables; and
3. identify best practices related to equipment qualification and condition monitoring to support long-term operation of NPPs.

Additionally, the expertise developed through CADAK may be extended to other technical equipment such as cable penetrations and pressure/level transmitters that have common elements among the participating countries. Participating countries include Belgium, Canada, France, Japan, Slovak Republic, Spain, Switzerland, and the United States of America. The first 3-year term expired in December 2014. Participants are currently finalizing plans for the second 3-year term to begin in 2015.

Component Operational Experience, Degradation and Aging Program (CODAP)

The objectives of the CODAP project are to:

1. collect information on passive metallic component degradation and failures of the primary system, reactor pressure vessel internals, main process and standby safety systems, as well as non-safety-related components with significant operational impact;
2. develop topical reports on degradation mechanisms; and
3. provide users with tools to apply the database for regulatory decision-making

The relational event database allows users to sort and filter events by a variety of fields, including power plant name, degradation mechanism, and nuclear component.

Among other applications, the database is useful for identifying emerging degradation trends and assessing the generic implications of events. The first three-year term for CODAP ended in December 2014, with 13 participants: Canada, Switzerland, Czech Republic, Germany, Spain, Finland, France, Japan, Korea, Sweden, Slovak Republic, Chinese Taipei, and the United States. The second three-year term will begin in 2015.

Fire Incidents Records Exchange (FIRE) Project

Fire is often an important contributor to core damage and plant damage states but realistic modelling of fire scenarios is difficult due to the scarcity of reliable data for fire analysis. Therefore, the FIRE project was initiated to foster multilateral cooperation in the collection and analysis of data related to fire events in nuclear power plants. The project was formally launched in January 2003 with twelve participating countries: Canada, Czech Republic, Finland, France, Germany, Japan, Korea, The Netherlands, Spain, Sweden, Switzerland and the United States.

The objectives of FIRE include establishing a framework for sharing event information useful to fire risk assessment and collecting and analyzing fire events to better understand these events, their causes, and their prevention. Fire events are captured in all plant operation modes as well as fires during construction and decommissioning.

The database contains fields to describe event descriptions, ignition and root cause information, extinguishment, and comments on consequences and corrective actions to name a few. The classification of events through coded attributes allows for effective searching for events of interest to the U.S. The database facilitates the development of qualitative insights into the root causes of fire events which can then be used to derive approaches for their prevention or mitigation.

This project is also facilitating improvements of existing international reporting systems and indicators for risk based inspections. The database project also provides a valuable link for international communication on other potential fire safety issues and led to the identification of the problem of High Energy Arc Faults (HEAF) in electrical equipment which matured into a separate OECD Project.

International Common-cause Data Exchange (ICDE) Project

Common-cause failures (CCF) can significantly impact the availability of safety systems of nuclear power plants. For this reason, the ICDE project was formally initiated by CSNI in 1997. The purpose of ICDE is to allow countries to collaborate and exchange CCF data to enhance the quality of risk analyses that include CCF modelling. Participating countries include Canada, Czech Republic, Finland, France, Germany, Japan, Korea, Spain, Sweden, Switzerland, United Kingdom, and the United States.

The specific objectives of the ICDE project are to:

1. collect and analyze CCF events to better understand such events, their causes, and their prevention;
2. generate qualitative insights into the root causes of CCF events for subsequent prevention or mitigation of their consequences;
3. establish a mechanism sharing experience gained in connection with CCF phenomena, including the development of prevention measures;
4. generate quantitative insights and record event attributes to facilitate quantification of CCF frequencies in member countries; and
5. estimate CCF parameters.

Qualitative insights gained from the analysis of CCF events are made possible by capturing raw event data in the ICDE database. The confidentiality of the data is a prerequisite of operating the project. The ICDE database is accessible only to those members of the ICDE Project who have actually contributed data to the database. The database covers key components of the main safety systems of nuclear power plants.

Components in the database include centrifugal pumps, diesel generators, motor operated valves, safety and relief valves, check valves, batteries, switchgears and breakers, reactor protection system components, heat exchangers, fans, main steam isolation valves, and digital instrumentation and control equipment. Other items may be added to or deleted from database upon the decision of the participating countries by taking into account their importance in probabilistic safety assessments.

For More Information
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International and Domestic Cooperative Research

The alphabetized list of international and domestic cooperative research found in this document is provided below for quick reference to the associated chapter(s) and page number(s).

NRC FORM 335 (12-2010) NRCMD 3.7	U.S. NUCLEAR REGULATORY COMMISSION	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)
BIBLIOGRAPHIC DATA SHEET *(See instructions on the reverse)*		NUREG-1925, Rev. 3
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same as above

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

The Office of Nuclear Regulatory Research (RES) supports the regulatory mission of the U.S. NRC by providing technical advice, tools, and information to identify and resolve safety issues, make regulatory decisions, and issue regulations and guidance. This includes conducting confirmatory experiments and analyses; developing technical bases that support the NRC's safety decisions; and preparing the agency for the future by evaluating the safety aspects of new technologies.

The NRC focuses its research primarily on near-term needs related to the oversight of operating reactors, as well as to new and advanced reactor designs. RES develops technical tools, analytical models, and experimental data to allow the agency to assess safety and regulatory issues. The RES staff uses its expertise to develop these tools, models, and data or uses contracts with commercial entities, national laboratories, and universities or in collaboration with international organizations.

This NUREG presents research conducted across a wide variety of disciplines, ranging from fuel behavior under accident conditions to seismology to health physics. This research 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 has organized this collection of information sheets by topical areas that summarize projects currently in progress. Each sheet provides the names of the RES technical staff who can be contacted for additional information.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)	13. AVAILABILITY STATEMENT
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