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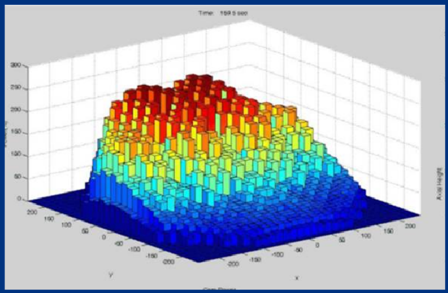
Protecting People and the Environment






Research Activities
FY 2018-2020

Office of Nuclear Regulatory Research

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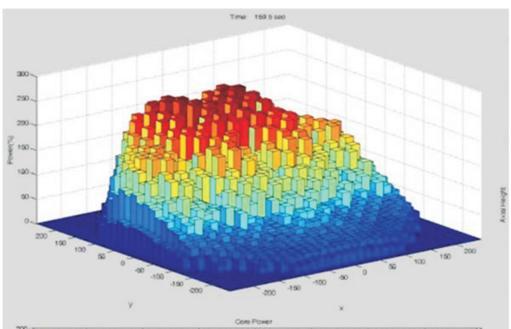
Research Activities

FY 2018-2020



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ABSTRACT

The Office of Nuclear Regulatory Research (RES) supports the mission of the U.S. Nuclear Regulatory Commission (NRC) by providing technical advice, tools, and information to identify potential safety and security issues and resolve them as appropriate, assessing risk and other nuclear safety and security issues, and developing and coordinating regulatory 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 RES staff uses its own expertise and collaborates with partner offices at the NRC, commercial entities, national laboratories, other Federal agencies, universities, and international organizations.

This NUREG describes research being conducted by RES 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. RES has organized this collection of information sheets by topical areas that summarize projects currently in progress. Each sheet provides the name(s) 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.

Some of the highlighted FY 2018–2020 projects include Level 3 probabilistic risk assessment (Chapter 6), human reliability analysis activities (Chapter 7), seismic and flooding research (Chapter 10), material aging and degradation analysis (Chapter 11), and international and domestic cooperative research (Chapter 15). The NRC also continues to focus on other issues such as new and advanced reactor designs, enhancing the reality of probabilistic risk assessments, ground-water monitoring and remediation at operating and decommissioning sites, and emerging radiation protection topics.

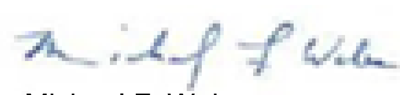
We conduct research both in-house and with the use of contractors and interagency agreements. The office's annual budget for contracted work is typically several tens of millions of dollars (direct contract funds do not include staff or full cost allocations), which is broken down as follows:

- User needs from the NRC's regulatory offices drive over three-fourths of RES activities.
- The Commission drives about 10 percent of RES activities through agency-mandated programs.
- A small amount is used to conduct feasibility studies of emerging technologies that may have regulatory applications in the future.

RES has made steady progress enhancing the planning, prioritizing, tracking, and reporting on research projects, which provides greater transparency on the use of its resources and improves stakeholder confidence. We work daily to enhance our effectiveness, efficiency, and agility.

Currently, RES has about 200 staff members. This staff reflects diversity in academic degrees, demographics, and technical disciplines. About 34 percent have PhDs, and another 36 percent have Master's degrees. The wide range of engineering and scientific disciplines includes expertise in thermal-hydraulics, materials science and engineering, instrumentation and controls, severe accident progression, human factors and human reliability, health physics, fire protection, seismology, and probabilistic risk assessment.

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 of Nuclear Regulatory Research

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FIGURES

ABBREVIATIONS AND ACRONYMS

Numerals

%OLTP	percent of originally licensed thermal power
%RCF	percent of rated core flow
10 CFR	Title 10 of the Code of Federal Regulations
1D	one-dimensional
2D	two-dimensional
3D	three-dimensional

A

AAR	after action review or report
ABWR	advanced boiling water reactor (General Electric and Toshiba)
AC	alternating current
ACRS	Advisory Committee on Reactor Safeguards
ADAMS	Agencywide Documents Access and Management System (NRC)
AEP	annual exceedance probability
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
AMT	Aging Management Tables
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
APEX	Advanced Power Extraction
APR	Advanced Power Reactor (Korea)
APWR	U.S. Advanced Pressurized-Water Reactor (Mitsubishi)
ARC-F	Analysis of Information from RB and CV and water sampling of Fukushima
ASCE	American Society of Civil Engineers
ASCET	Assessment of Structures Subject to Concrete
ASD	aspirating smoke detectors
ASME	American Society of Mechanical Engineers
ASLB	Atomic Safety and Licensing Board
ASP	accident sequence precursor
ASR	alkali-silica reaction
ASTM	American Society for Testing and Materials
ATF	accident tolerant fuel
ATWS	anticipated transient without scram
ATWS-I	ATWS (anticipated transient without scram) with instability

B

BADGER	Boron Areal Density Gage for Evaluating Racks
BAM	Federal Institute for Materials Research and Testing (Germany)
BEIR	Biological Effects of Ionizing Radiation
BETHSY	loop for the study of T/H systems (France)
BIP	Behavior of Iodine Project (Canada)
BNL	Brookhaven National Laboratory
Bq	becquerel
BSAF	Benchmark Study of the Accident at Fukushima
BTP	branch technical position
BWR	boiling-water reactor

C

CADAK	Cable Aging Data and Knowledge
CAMP	Code Application and Maintenance Program (NRC program)
CAPS	CSNI Action Proposal Sheet
CAROLFIRE	Cable Response to Live Fire
CCF	common-cause failure
CEA	Commissariat l'Energie Atomique aux Alternatives (France)
CEUS	Central and Eastern United States
CFAST	Consolidated Model of Fire and Smoke Transport
CFD	computational fluid dynamics
CFP	Cable Fire Propagation
CFR	Code of Federal Regulations
CFSR	Climate Forecast System Reanalysis
CIP	Critical Infrastructure Protection
CIRFT	Cyclic Integrated Reversible-bending Fatigue Tester
CISCC	chloride-induced stress corrosion cracking
CHRISTIFIRE	Cable Heat Release, Ignition, and Spread inTray Installations during Fire
CLI	Commission issued Order
CODAP	Component Operational Experience, Degradation and Aging Program
COL	combined license
COMTE	Complementary Tests
CONTAIN	containment analysis code
CoP	community of practice
CPRR	containment overpressure protection and release reduction
CR	control room
CRIEPI	Central Research Institute of Electric Power Industry (Japan)
CRPPH	International Commission on Radiation Protection and Public Health

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
CV	containment vessel

D

DAKOTA	Design Analysis Kit for Optimization and Terascale Applications
DandD	Decontamination and Decommissioning code
DBA	design-basis accident
DBE	design-basis earthquake
DC	direct current
DCA	design certification application
DCD	design certification documents
DCPD	direct current potential drop
DCSS	dry cask storage system
DE	RES Division of Engineering
DELORES-VEWFIRE	Determining the Effectiveness, Limitations, and Operator Response for Very Early Warning Fire Detection Systems in Nuclear Facilities
DESIREE-FIRE	Direct Current Electrical Shorting in Response to Exposure Fire
DG	draft (regulatory) guide
DI&C	digital instrumentation and control
DIL	Derived Intervention Level
DOD	U.S. Department of Defense
DOE	U.S. Department of Energy
DRA	RES Division of Risk Analysis
DSA	RES Division of Systems Analysis
DSI	direction setting issue
DVD	digital versatile disc

E

EAB	exclusion area boundary
EBP	extra budgetary program
ECCS	emergency core cooling system
ECFS	Electric Cabinet Fire Spread
ECI	exterior communications interface
EDF	Electricité de France
EESS	External Event Safety Section
ELAP	extended loss of alternating power
ELECTRA-FIRE	Electrical Cable Test Results and Analysis during Fire Exposure
EM	electromagnetic
EMDA	Expanded Materials Degradation Assessment
EMUG	European MELCOR User Group
ENDF	Evaluated Nuclear Data File
EO	Executive Order
EPA	U.S. Environmental Protection Agency
EPICUR	Experimental Program for Iodine Chemistry Under Radiation (France)
EPR	U.S. Evolutionary Power Reactor
EPRI	Electric Power Research Institute
EPZ	emergency planning zone
ETE	evacuation time estimate

F

F	Fahrenheit
FAA	U.S. Federal Aviation Administration
FAQ	frequently-asked-questions
FAST	Fuel Analysis under Steady-state and Transients code (FRAPCON/FRAPTRAN merged code
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
FEMA	U.S. Federal Emergency Management Agency
FHID	Flood Hazard Information Digest
FIRE	Fire Incident Record Exchange
FFA	flood frequency analysis
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)
FRTR	Federal Remediation Technologies Roundtable
FSEIS	Final Supplemental Environmental Impact Statements
ft	feet
FY	fiscal year

G

GAIN	Gateway for Accelerated Innovation in Nuclear (DOE program)
GALE	Gaseous And Liquid Effluent code
GALL	Generic Aging Lessons Learned
GDC	general design criterion

GENII	Hanford Environmental Dosimetry System (Generation II).
GI	generic issue
GL	Generic Letter
GMPE	ground motion prediction equations
GMRS	ground motion response spectrum
GRS	Gesellschaft für Anlagen und Reaktorsicherheit (Germany)
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
HEAF	high energy arcing faults
HELEN-FIRE	Heat Release Rates from Electrical Enclosure Fires
HEP	human error probability
HFE	human factors engineering
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 (Japan)
IAEA	International Atomic Energy Agency
IAP	implementation action plan
IBMB	Institute for Building Materials, Solid Construction; and Fire Protection
ICDE	International Common-cause Data Exchange
ICES	Institute for Nuclear Power Operations Consolidated Events System
ICRP	International Commission on Radiological Protection
IDCCS	Integrated Data Collection and Coding System
IDHEAS	Integrated Human Event Analysis System
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronics Engineers
IFE	Institutt for Energiteknikk (Norwegian Institute for Energy Technology)
IMPRINT	Improved Performance Research Integration Tool (U.S. Army Research Laboratory)
IMUG	International MACCS Users Group
In	inch
INL	Idaho National Laboratory
INPO	Institute of Nuclear Power Operations
IPEEE	individual plant examination of external events
iPWR	integral pressurized water reactor
IRSN	Institut de Radioprotection et de Surete Nucleaire (French Institute for Radiological Protection and Nuclear Safety)
ISFSI	independent spent fuel storage installation
ISI	inservice inspection
ISO	International Standards Organization
ISOE	Information System on Occupational Exposure
ISR	in situ recovery
ISSC-EBP	International Seismic Safety Centre's Extra Budgetary Project (IAEA)

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 (Russia)
JNRA	Japan Nuclear Regulatory Authority
JOAN of ARC	Joint Analysis of Arc Faults

K

KAERI	Korea Atomic Energy Research Institute
KATEFIRE	Kerite Analysis in Thermal Environment of Fire
Keff	k-effective reactivity coefficient
KENO	multigroup Monte Carlo criticality code
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
LER	licensee event report
LERF	large early release frequency
LIP	Local Intense Precipitation
LOC	loss of control
LOCA	loss-of-coolant accident
LOH	loss of habitability

LOOP	loss of offsite power
LPZ	low population zone
LSDYNA	Livermore Software Technology Corporation for dynamic explicit finite-element analysis
LTSBO	long-term station blackout
LTRP	Long-Term Research Program
LWR	light-water reactor
LWRS	Light-Water Reactor Sustainability Research (DOE)

M

MACCS	MELCOR Accident Consequence Code System
MAPS	Managing Aging Processes for Storage
MARIAFIRES	Methods for Applying Risk Analysis to Fire Scenarios
MASS	MELCOR Accident Simulation Using SNAP
MASLWR	Multi-Application Small Light Water Reactor
MATLAB	MATrix LABoratory code
MCAP	MELCOR Code Assessment Program
MCCI	melt coolability and concrete interaction
MCNP	Monte Carlo N-Particle code
MCS	mesoscale convective systems
MD	management directive (NRC)
MELCOR	computer code for analyzing severe accidents in NPPs
MELLLA+	maximum extended load line limit analysis plus
MNA	Monitored Natural Attenuation
MODARIA	Modelling and Data for Radiological Impact Assessment (IAEA)
MOU	memorandum of understanding
MOX	mixed oxide
MSPI	mitigating system performance indicator
MSR	molten salt reactor
MTO	Man-Technology-Organization

N

NAM	neutron-absorbing materials
NAS	U.S. National Academy of Sciences
NASA	U.S. National Aeronautics and Space Administration
NCRP	National Council of Radiation Protection and Measurements
NDE	nondestructive examination
NEA	Nuclear Energy Agency
NEI	Nuclear Energy Institute
NEPA	National Environmental Protection Act
NFPA	National Fire Protection Association
NGA	next generation attenuation
NIST	U.S. National Institute of Standards and Technology
NIST	NuScale integrated system test
NMSS	Office of Nuclear Material Safety and Safeguards
NOAA	U.S. National Oceanic and Atmospheric Administration
NPP	U.S. Nuclear Regulatory Commission
NROD	NRC Reactor Operating Experience Database
NRR	Office of Nuclear Reactor Regulation
NSIR	Office of Nuclear Security and Incident Response
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

ODOBA	Observatory for the durability of reinforced concrete structures (France)
OECD	Organization for Economic Cooperation and Development
OMB	Office of Management and Budget
ONR	Office for Nuclear Regulation
1D	one-dimensional
OpE	operating experience
ORNL	Oak Ridge National Laboratory
ORO	offsite response organization
OSX	operating system

P

PAG	Protection Action Guides
PANDA	Passive Non-Destructive Assay of Nuclear Materials
PARCS	Purdue's Advanced Reactor Core Simulator code
PARENT	Program to Assess the Reliability of Emerging Nondestructive Techniques
PARTRIDGE	Probabilistic Analysis as a Regulatory Tool for Risk-Informed Decision GuidancE
Pb	Lead
PCCV	prestressed concrete containment vessel
PCV	primary containment vessel
PEO	period of extended operation
%OLTP	percent of originally licensed thermal power
%RCF	percent of rated core flow
PFA	precipitation frequency analysis
PFHA	probabilistic flood hazard assessment
PFM	probabilistic fracture mechanics
Phebus-FP	Phebus-Fission Products

Phebus-ISTP	Phebus-International Source Term Program
PIE	post-irradiation examination
PIMAL	phantom with moving arms and legs code
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
PreADES	Preparatory Study on Analysis of Fuel Debris
PRM	Petition for Rulemaking
PSA	probabilistic safety assessment
PSHA	probabilistic seismic hazard assessment
PUMA	Purdue University Multi-Dimensional Integral Test Assembly
PWR	pressurized-water reactor
PWSCC	primary water stress-corrosion cracking

Q

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
R&D	research and development
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 code
RB	reactor building
RBHT	Rod Bundle Heat Transfer program
RCF	rated core flow
RCS	reactor coolant system
RDD	radiological dispersal device
REAcct	Regional Economic Accounting Tool
REAP	Reactor Embrittlement Archive Project
REIRS	Radiation Exposure Information and Reporting System
RELAP	Reactor Excursion and Leak Analysis Program code
Rem	Roentgen equivalent man
REMIX	Regional Mixing Model code
RES	Office of Nuclear Regulatory Research
RESRAD	code to determine allowable RESidual RADioactivity in site cleanup
RG	regulatory guide
RIA	reactivity initiated accident
RIC	Regulatory Information Conference
Rn	radon
ROP	Reactor Oversight Process
ROSA	Rig of Safety Assessment test facility (Japan)
RPV	reactor pressure vessel

S

SACADA	Scenario Authoring, Categorization, and Debriefing Application
SAMA	severe accident mitigation alternative
SAMG	severe accident mitigation guideline
SAPHIRE	Systems Analysis Programs for Hands-on Integrated Reliability Evaluation
SAREF	Safety Research post Fukushima
SBO	station blackout
SC	steel plate and concrete composite modular construction
SC	Office of Science (DOE)
SCALE	Standardized Computer Analysis for Licensing Evaluation code
SCAP	Cable Aging Project
SCC	stress-corrosion cracking
SCIP	Studsvik Cladding Integrity Project
SDP	Significance Determination Process
SDO	standards development organization
SEASET	Separate Effects and Systems Effects Tests
SecPop	Sector Population, Land Fraction, and Economic Estimation Program
SECY	Office of the Secretary (NRC staff paper for the Commission)
SER	Safety Evaluation Report
SFP	spent fuel pool
SG	steam generator
SGTR	steam generator tube rupture
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
SRM	staff requirements memorandum (Commission direction to NRC staff)
SRP	Standard Review Plan
SSC	structures, systems, and components
SSHAC	Senior Seismic Hazard Analysis Committee
SSI	seismic soil-structure interaction
STAR CCM+	computer code used for CFD
STCP	Source Term Code Package
STEM	Source Term Evaluation and Mitigation (CSNI experimental program)
S3	Smoke Stratification and Spread short-
STSBO	term station blackout
Sv	Sievert
SW	Space Weather

T

TC	tropical cyclones
TEDE	total effective dose equivalent thermal-
T/H	hydraulic
THI	Thermal-Hydraulics Institute
3D	three-dimensional
TID	Technical Information Document (Atomic Energy Commission document)
TLR	technical letter report
TMI	Three Mile Island (Nuclear Power Plant)
TRAC	Transient Reactor Analysis Code
TRACE	TRAC/RELAP Advanced Computational Engine code
TRITON	Transport Rigor Implemented with Time-dependent Operation for Neutronic depletion code module
TSUNAMI	sensitivity/uncertainty tools in SCALE code
2D	two-dimensional

U

UA	uncertainty analysis
UCF	University of Central Florida
UNSCEAR	United Nations Scientific Committee on Exposure to Atomic Radiation
U.S.	United States
USACE	U.S. Army Corps of Engineers
U.S. APWR	U.S. Advanced Pressurized-Water Reactor (Mitsubishi)
USBR	U.S. Bureau of Reclamation
USGCRP	U.S. Global Change Research Program
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 (realistic verification of the reactor containment)
VERDON	French test program
VEWFD	very early warning fire detection
WLA	Wildlife Liquefaction Array (CA)
VSL	value of statistical life
VSP	Visual Sampling Plan code
VTT	Valtion Teknillinen Tutkimuskeskus (Finland)

W

WGAMA	Work Group on Analysis and Management of Accidents
WGHOF	Working Group on Human and Organizational 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

ZOI	zone of influence

Chapter 1: Agency Programs Support

The function 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 safety issues; performing the research necessary to support regulatory decisions; and issuing regulations and guidance. RES's principal product is knowledge; therefore, knowledge management (KM) is an integral part of the RES mission. RES interfaces with all NRC offices and applies its breadth and depth of technical expertise to coordinating agencywide programs in support of the NRC's regulatory infrastructure. In particular, RES coordinates the development of regulatory guides for nuclear industry use. The NRC issues regulatory guides for public use to present approaches that the staff considers acceptable in implementing the agency's regulations. RES provides the tools and methods used by NRC program offices to issue and maintain the set of about 400 regulatory guides.

In addition, RES coordinates the NRC's use of consensus codes and standards, many of which are endorsed by the NRC through regulatory guides. In this capacity, RES facilitates the appointment of agency staff to the consensus codes and standards committees to offer technical expertise. The NRC cooperates with professional organizations that develop consensus standards associated with systems, structures, equipment, or materials that the nuclear industry uses. 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. The NRC uses these standards in a variety of regulatory applications, particularly in its rules and regulatory guidance.

RES recommends regulatory actions to resolve ongoing and potential safety 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 GI Program enables the public and NRC staff to raise issues with potential significant generic safety or security implications. The program has identified more than 850 GIs to date, resulting in safety improvements at NRC licensees and a variety of regulatory products such as generic communications and regulatory guidance.

RES provides independent analysis of operational data and assessment of operational experience through the review, analysis, and evaluation of the safety performance of facilities licensed by the NRC. The results of the data collection efforts are primarily used to estimate and monitor the risk of accidents at U.S. commercial nuclear power plants. Data and information reported to the NRC are reviewed, evaluated, and coded into databases that form the basis for estimates of reliability parameters used in probabilistic risk assessment (PRA) models. Notably, the Accident Sequence Precursor (ASP) Program systematically evaluates U.S. nuclear power plant operating experience to identify, document, and rank the operating events most likely to lead to inadequate core cooling and severe core damage (precursors) given the 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 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 identify deficiencies in the NRC's regulatory process and to 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.

The NRC conducts feasibility studies of various topics to assess if future research on the topic should be pursued with additional research. In addition, the NRC is making preparations to review and regulate a new generation of non-light water reactors (non-LWRs) including the identification of computer codes and tools needed for non-LWR review and regulation.

Regulatory Guide Program

Objective

Regulatory guides are issued by the U.S. Nuclear Regulatory Commission (NRC) to describe approaches that the staff considers acceptable for implementing Federal regulations. The Office of Nuclear Regulatory Research (RES) provides the program management to enable the NRC program offices to issue and update regulatory guides.

Research Approach

Program Management

RES is responsible for program management of the regulatory guides. The office coordinates the development, review, revision, and withdrawal of regulatory guides with the other program offices as needed. NRC Management Directive 6.6, “Regulatory Guides,” formalizes the regulatory guide development, review, revision, and withdrawal process.

Review and Update Process

New regulatory guides are developed and existing regulatory guides are revised or withdrawn on an as-needed basis. Each regulatory guide is reviewed every 10 years to determine its suitability for continued use. Changes to consensus standards or modifications of NRC staff positions can cause more frequent reviews, revisions, or withdrawal of regulatory guides.

Withdrawal of a regulatory guide means that it should not be used for future NRC licensing activities. Although a regulatory guide may have been withdrawn, licensees can continue to use if it is already part of a facility’s licensing basis.

Status

The Regulatory Guide Update Project was initiated in 2006 at the direction of the Commission to review, prioritize, and update all regulatory guides. As of June 2017, the agency had completed 416 of the 426 regulatory guides identified in the original 2006 update program, 6 additional regulatory guides are in the process of being updated, and 4 regulatory guides are being withdrawn. Figure 1.1 depicts the current status of the Regulatory Guide Update Project.

The agency continues to update its guides on an ongoing basis. About 80 guides are in active development at any given time. The staff issues and updates about 10-20 guides each year, and about half of these are implementing guidance for new or revised rules. In addition, the staff reviews all guides every 10 years on a rotating basis.

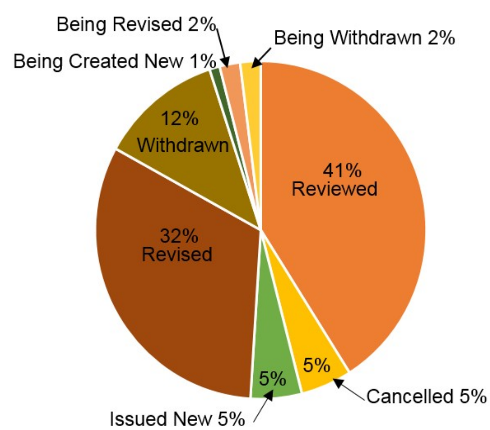

Figure 1.1 Status of the Regulatory Guide Update Project as of June 2017.

For More Information: Contact Tom Boyce, RES/DE, at Tom.Boyce@nrc.gov.

NRC Codes and Standards Program

Objective

The U.S. Nuclear Regulatory Commission (NRC) staff participates in Standards Development Organizations (SDOs) to enable the effective and efficient use of standards in support of the NRC's mission. Advantages of the use of standards for NRC regulatory activities include resource savings, improved efficiency and transparency, and regulatory requirements of higher technical quality than could be achieved through NRC development of government unique standards. Use of SDO standards can also provide greater acceptance of NRC regulatory activities by external stakeholders.

The NRC's use of standards and participation on SDO standards committees is in conformance with Public Law 104-113, the "National Technology Transfer and Advancement Act of 1995," and 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."

Research Approach

In accordance with U.S. Office of Management and Budget (OMB) Circular A-119, the NRC Standards Executive in the Office of Nuclear Regulatory Research (RES) serves as the agency point of contact for standards activities and coordinates the NRC's participation in SDOs and use of consensus codes and standards. For example, the RES staff supporting the Standards Executive coordinate standards committee representation facilitate staff sharing of information needed to support standards meetings, disseminate documents to other NRC offices for input, and promote awareness of safety standards. The standards that are developed with the participation of the NRC staff are generally endorsed or referenced in rules and regulatory guidance. Information on the NRC Codes and Standards program can be found on the NRC public Web site at https://www.nrc.gov/about-nrc/regulatory/standards-dev.html.

The NRC conducts outreach to external stakeholders in the standards community by hosting an annual NRC Standards Forum. The goal of the forum is to identify needed standards in support of all aspects of the nuclear industry and to collaborate to develop them. The participants include private industry, SDOs, academia, government, utilities, and the public. The forum replaced the Nuclear Energy Standards Coordination Collaborative in 2016, which had a similar purpose and was co-sponsored by the NRC and the Department of Energy.

In addition to participation in domestic SDOs, the NRC participates in the development of multiple international standards, particularly those from the International Atomic Energy Agency. When developing NRC regulations and guidance, the NRC considers harmonization of its regulations and guidance with these international standards.

Status

The NRC Codes and Standards Program is an ongoing program that coordinates agency participation in SDOs and standards development activities. Currently, over 200 NRC staff members participate on about 150 committees managed by 16 SDOs.

For More Information: Contact Kurt Cozens, RES/DE, at Kurt.Cozens@nrc.gov.

Generic Issues Program

Objective

The U.S. Nuclear Regulatory Commission's (NRC's) Generic Issues (GI) Program enables the public and the 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 the potential generic issues to determine whether additional regulatory requirements are necessary to ensure continued safe and secure operation of the licensed facilities and to disseminate pertinent information regarding generic issues.

Research Approach

The Office of Nuclear Regulatory Research (RES) coordinates and manages the GI Program described in Management Directive (MD) 6.4, "Generic Issues Program." MD 6.4 describes the process used to resolve the generic issues and provides the staff with a framework for receiving, tracking, and processing generic issues. The GI Program consists of a three-stage process: screening, assessment, and regulatory implementation (refer to the figure below). In the screening stage, the NRC staff uses seven screening criteria to assess whether the program can effectively evaluate the potential generic issues to determine if additional regulatory requirements are necessary. In the assessment stage, the staff performs technical, safety, and regulatory analyses to determine whether a regulatory action is necessary. In the regulatory implementation stage, the agency takes appropriate regulatory action to address the issue with its licensees.

Status

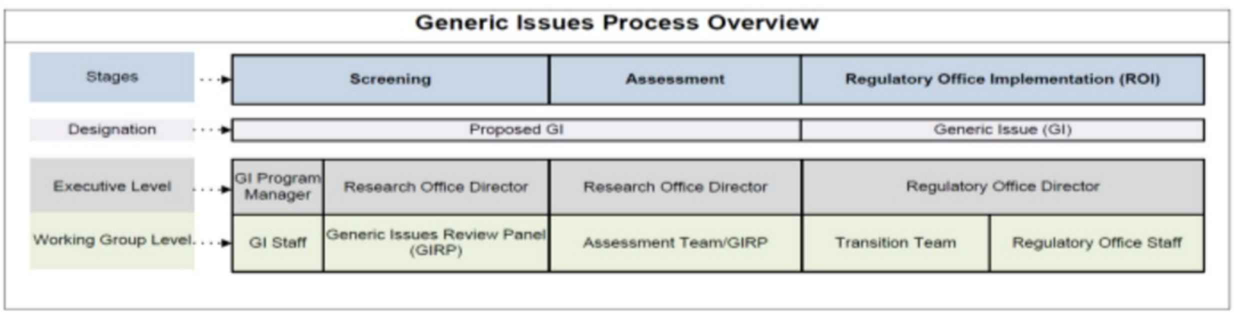

Figure 1.2 Generic Issues Process Overview.

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, and updates to existing regulations.

Information on the status, activities, and milestones of active generic issues is maintained on a Web site called the GI Dashboard that is available on the NRC public Web site at https://www.nrc.gov/about-nrc/regulatory/gen-issues/dashboard.html.

Historical information on the resolution of generic issues is captured in NUREG-0933, "Resolution of Generic Safety Issues." NUREG-0933 is now a user friendly, Web-based, accessible, and searchable document available on the NRC public Web site at: http://nureg.nrc.gov/sr0933/.

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

Feasibility Studies

The U.S. Nuclear Regulatory Commission (NRC) conducts feasibility studies to identify and assess research projects that are potentially needed to prepare the NRC for future technical or regulatory needs, particularly those needs that are not part of ongoing research activities by the agency. These projects are not typically associated with near-term regulatory actions. The staff conducts feasibility studies of selected projects to determine the research activities that are necessary to address the future needs. 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.

Approach

A feasibility study typically determines the merits, scope, regulatory needs, major tasks, deliverables, proposed schedule, likely costs, and identification of possible facilities that would actually perform the research. The outcome of a feasibility study is a recommendation whether to perform a research project.

The Office of Nuclear Regulatory Research (RES) staff has developed the process for initiating and performing feasibility studies. Two methods are used to initiate a feasibility study. The first method is for a program office to directly request the performance of a feasibility study to support a need that they identified. The second method is for any member of staff to identify an idea that may warrant the performance of a feasibility study. Using limited resources, the staff idea is reviewed by a committee of senior technical staff and possibly other subject matter experts. The outcome of this review is a recommendation whether or not a feasibility study should be performed and, if appropriate, a refinement of the study's parameters. If NRC management agrees with the recommendation, the study is performed after resources are made available, depending on the priority of the work.

Status

The Energy Reorganization Act of 1974 established the fundamental role of RES to engage in or contract for research to develop recommendations necessary for the performance of the NRC's licensing and related regulatory functions. The Commission, in a Staff Requirements Memorandum in 1996 regarding Direction Setting Issue (DSI) 22: "Research," directed the staff to continue a balance of confirmatory and anticipatory research. RES performs feasibility studies to support anticipatory research.

The NRC has conducted studies to assess potential future technical and regulatory needs for many years. The RES staff had initiated a Long-Term Research Program (LTRP) in 2007 as one of its activities in support of anticipatory research. However, in 2016, the staff discontinued the LTRP as a program and replaced it by using feasibility studies. The agency intended to use feasibility studies as part of the continuum of research activities performed by RES rather than devoting resources to a standalone program. The staff is incorporating the feasibility studies into its research activities and began the use of feasibility studies in late 2017.

For More Information: Contact Kurt Cozens, RES/DE, at Kurt.Cozens@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 U.S. Nuclear Regulatory Commission (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 to 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.

The annual volumes of the Abnormal Occurrence Report can be found at https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0090/.

Research 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.

Status

RES has led an agency effort to revise the current AO criteria to ensure current regulatory framework and technology is included in the determination of events that meet the definition of an AO described in Section 208 of the Energy Reorganization Act of 1974. The revised AO criteria has been finalized in September 2017, and the new criteria has been implemented starting fiscal year 2018.

For More Information: Contact Vered Shaffer, RES/DSA, at Vered.Shaffer@nrc.gov.

Operating Experience Program

Objective

The objective of the U.S. Nuclear Regulatory Commission's (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 occurring at operating commercial U.S. NPPs.

Research Approach

As noted in the "SPAR Model Development Program" discussion in this report, the NRC has probabilistic risk assessment (PRA) models for all operating commercial U.S. NPPs. Keeping NRC's PRA models current requires up-to-date reliability parameters developed using the operating experience data NRC collects and analyzes.

OpE data sources include licensees' monthly operating reports, the proprietary Institute for Nuclear Power Operations' Consolidated Events System, and licensee event reports (LERs). Most LERs are publicly available through a searchable database, LER-Search (https://nrcoe.inel.gov/secure/lersearch/index.cfm).

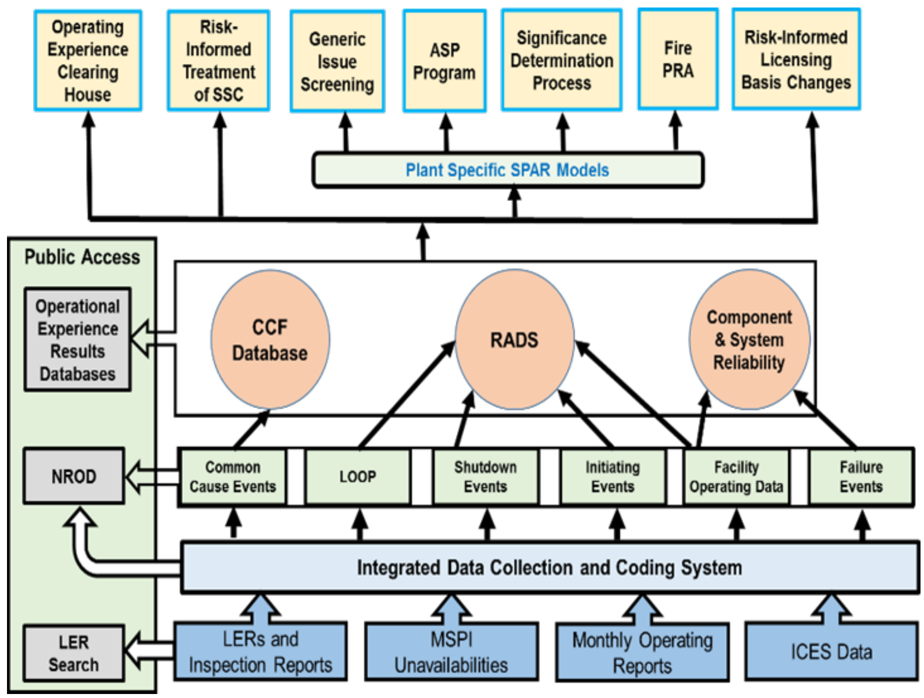

Figure 1.3 Risk-Informed Applications Using NRC Operational Data.

Under the Reactor Operating Experience Data for Risk Applications Project, the Integrated Data Collection and Coding System (IDCCS) database stores the data. IDCCS includes a proprietary search capability, the NRC Reactor Operating Experience Database (NROD), to search this data. IDCCS also includes applications for OpE data analysis, such as the Reliability and Availability Data System (RADS), Common-Cause Failure (CCF) Database, and the Accident Sequence Precursor (ASP) Events Database.

The Computational Support for Risk Applications Project routinely updates the data used in PRA models to include the latest reliability parameters. This enables the NRC to perform state-of-practice risk assessments of operating events and conditions, assess licensee risk-related performance, and conduct special studies of risk-related issues. Moreover, the data supports identifying trends, developing operating experience-based performance indicators, and performing reliability studies for risk-significant systems and equipment. NRC assessment and confirmation of information reported by licensees as part of the Mitigating Systems Performance Index Program may also use this data.

Status

NRC Management Directive 8.7, "Reactor Operating Experience Program," directs the Office of Nuclear Regulatory Research to collect and analyze nuclear plant operational data. Publicly available annual updates on operational data analyses results are available on the Reactor Operational Experience Results and Databases Web site (http://nrcoe.inel.gov/results/).

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

Accident Sequence Precursor Program

Objective

A primary objective of the Accident Sequence Precursor (ASP) Program is to help the NRC monitor its performance, thereby helping to ensure that the NRC meets Safety Objective 1 (see NRC Strategic Plan) to prevent and mitigate accidents and to ensure radiation safety. The NRC also uses the program to contribute to Safety Strategy 1 by evaluating domestic operating events and trends for risk significance and generic applicability. Moreover, the program assists in fulfilling NRC's Safety Performance Goal 4 to identify and help prevent accident precursors and reductions in safety margins at operating commercial NPPs that may be of high safety significance. Information developed from the ASP Program also supports the assessment of existing NRC oversight and licensing programs (Appendix B in the NRC Strategic Plan) and helps shape the NRC's objectives and strategies for operating reactors. In addition, through the ASP Program, the NRC reviews and evaluates operating experience to identify precursors to potential core damage as required by Management Directive 8.7, "Reactor Operating Experience Program."

Research Approach

To identify potential accident precursors, the staff reviews and performs screening analyses of operational events described in LERs submitted to the NRC. LERs that do not meet established qualitative criteria screen out of the ASP Program.

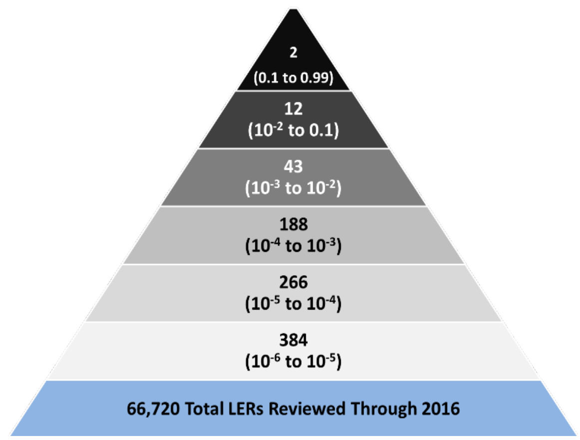

Figure 1.4 Historical ASP Program Results.

This figure is updated to include missing precursor events from the initial precursor study, NUREG-2697, "Precursors to Potential Severe Core Damage Accidents: 1969–1979." The event at Three Mile Island (Unit 2) is considered an accident (i.e. not a precursor).

The staff performs precursor analyses for LERs screened in due to an initiating event such as a loss of offsite power, for LERs that have other concurrent degraded condition(s), or for screened-in LERs with no licensee performance deficiency and, therefore, no Significance Determination Process (SDP) risk evaluation. The results of these analyses determine if these kinds of LERs are precursors. For LERs screened in due to a degraded condition related to an identified performance deficiency, the staff uses SDP risk evaluations in accordance with Regulatory Issue Summary 2006-24, "Revised Review and Transmittal Process for Accident Sequence Precursor Analyses," as the ASP Program result. Detailed ASP analyses utilize the standardized plant analysis risk (SPAR) models and the Systems Analysis Programs for Hands on Integrated Reliability Evaluations (SAPHIRE) software. Analysis results provide an estimate of the probability of core damage, or a change in the probability, considering the conditions experienced during the event (i.e., initiating events and/or degraded conditions). Typically, precursors include events with a conditional probability of core damage or an increase in the probability of core damage of greater than or equal to 1×10^{-6}. Significant precursors are defined as greater than or equal to 1×10^{-3}, of which one event (Davis-Besse, 2002) was observed in the last 20 years.

In addition, the staff evaluates and analyzes precursor data to identify risk significant trends in fleet-wide performance as part of the NRC's long-term operating experience program. Insights from this trending support independent monitoring of the effectiveness of NRC's oversight and licensing programs. Moreover, trending analysis of the ASP Program data can provide insights on the risk impact from NRC's regulatory activities, including initiatives to risk-inform regulations and regulatory processes.

Status

The NRC makes each ASP Program analysis and the ASP Program annual report publicly available in ADAMS. The latest ASP Program annual report is available in ADAMS under ML17153A366.

For More Information: Contact Ian Gifford, RES/DRA, at Ian.Gifford@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 safety issues and by performing the research necessary to support regulatory decisions. RES's principal product is knowledge; therefore, knowledge management (KM) is an integral part of the RES mission. The RES 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. RES directly supports the agency KM program through the NRC KM steering committee and its Communities of Practice (CoP) and by developing seminars, NUREG/KM series reports, and Regulatory Guides among other activities.

Research Approach

In 2006, the NRC formally established a KM program through the issuance of a formal KM policy (SECY-06-0164 "The Knowledge Management Program": Internal NRC Policy Issue information). This policy established the NRC KM steering committee that includes senior managers from each office. The committee implements the program, evaluates its performance, considers new ideas, and discusses future plans. The committee meets routinely and cultivates an awareness of the value of KM initiatives and support staff with KM-oriented projects and goals.

Figure 1.5 NUREG / KM front covers.

RES routinely sponsors 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, the Fukushima Dai-ichi NPP accident, and the 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, during the TMI seminar in 2009, speakers included Former Governor Richard Thornburgh of Pennsylvania and Ed Frederick, an operator on shift at the time of the accident at TMI in 1979. Some of these seminars are also captured in KM reports (e.g., NUREG/KM).

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

Status

The NRC has published 10 NUREG/KMs under this program. The latest, NUREG/KM-0011, "WASH-1400-The Reactor Safety Study – The Introduction of Risk Assessment to the Regulation of Nuclear Reactors," was published in August 2016. KM at the NRC is an ongoing activity with a NUREG/KM on the accident at Chernobyl currently in development. RES will continue its efforts under this program to capture, preserve, and transfer knowledge among employees and stakeholders.

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

NRC Non-Light Water Reactor (Non-LWR) Research

Objectives

The U.S. Nuclear Regulatory Commission (NRC) is making preparations to review and regulate a new generation of non-light water reactors (non-LWRs).

Research Approach

A two-volume vision and strategy staff report has been developed to assure NRC readiness to efficiently and effectively conduct its mission for non-LWR technologies, including fuel cycles and waste forms. The report provides a connection to other NRC mission, vision, and strategic planning activities and describes the objectives, strategies, and contributing activities necessary to achieve non-LWR mission readiness (Figure 1.6). The project has been organized into two phases. Phase 1 is the conceptual planning phase used to lay out the vison and strategy, gather public feedback, and finalize the NRC's approach. Phase 2 includes detailed work planning efforts and task execution. Phase 1 was completed at the end of 2016, and Phase 2 is scheduled to be completed not later than 2025 (including execution of planned tasks). The planning process for this work is broken down into three periods: near-term (0-5 years), mid-term (5-10 years), and long-term (greater than 10 years). The staff report covers Phase 2 actions to be taken in the first five years and will be supplemented with the mid-term and long-term plans later.

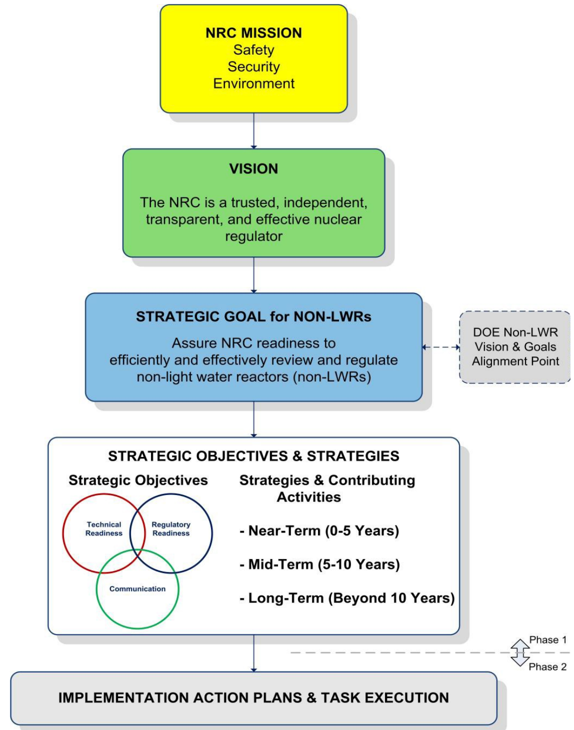

Figure 1.6 NRC Non-LWR Mission Readiness Roadmap.

The staff is pursuing six strategies to prepare for non-LWR licensing:

1. Acquire/develop sufficient knowledge, technical skills, and capacity to perform non-LWR regulatory reviews.
2. Acquire/develop sufficient computer codes and tools to perform non-LWR regulatory reviews.
3. Develop guidance for a flexible non-LWR regulatory review process within the bounds of existing regulations including the use of conceptual design reviews and staged-review processes.
4. Facilitate industry codes and standards needed to support the non-LWR life cycle (including fuels and materials).
5. Identify and resolve technology-inclusive policy issues that impact the regulatory reviews, siting, permitting, and/or licensing of non-LWR nuclear power plants (NPPs).
6. Develop and implement a structured, integrated strategy to communicate with internal and external stakeholders having interests in non-LWR technologies.

Status

Near-term implementation action plans (IAPs) have been prepared for each strategy. RES staff are participating in Molten Salt Reactor training (Strategy 1); working with DOE to acquire, develop, and test analytical tools and high-performance computing systems for evaluating non-LWR designs (Strategy 2); participating in the development of the non-LWR design criteria (Strategy 3); participating in the development of the prototype reactor guidance (Strategy 3); developing consensus codes and standards (Strategy 4); participating in the development of insurance and liability policy (Strategy 5); and participating in many workshops and meetings (Strategy 6).

For More Information: Contact Steve Bajorek, RES/DSA, at Stephen.Bajorek@nrc.gov.

NRC Non-Light Water Reactor (Non-LWR) Strategy 2 – Acquire/develop sufficient computer codes and tools to perform non-LWR regulatory reviews

Objectives

The U.S. Nuclear Regulatory Commission (NRC) staff must have adequate computer models and analytical tools to conduct its review of non-LWR designs in an independent manner. The objective of Strategy 2 includes the identification of computer codes and tools needed for non-LWR review and regulation. The research will provide the staff with a basis for new regulations and guidance, and it will enable the staff to assess and confirm the safety margins inherent to a particular non-LWR design and its type of fuel.

Research Approach

Although the NRC has historically developed many of the computer codes it uses independently, the approach taken for Strategy 2 is to consider the use of established non-LWR codes developed by others where possible and appropriate. The NRC will also leverage its participation in the U.S. Department of Energy's (DOE's) Gateway for Accelerated Innovation in Nuclear (GAIN) initiative to acquire early insights for new or revised non-LWR codes being developed. For the purpose of this strategy, the staff has considered high-temperature gas-cooled reactors, liquid metal reactors, and molten salt reactors (where the fuel may or may not be dissolved in the coolant) as the designs of interest in the near term. Near-term implementation action plans (IAPs) have been developed to identify the needed computer codes and tools for the following functional research area topics: reactor kinetics and criticality, fuel performance, thermal-fluids, severe accidents and offsite consequences, materials, seismic and structural, instrumentation and controls, and probabilistic risk assessment.

Status

The staff is leveraging existing collaborations and research in fast reactor technology and design to initiate code development activities by incorporating known fast reactor models into the existing RES neutronics simulation platforms (SCALE/PARCS). In addition, steps have begun to identify an appropriate cross section methodology (relative effect of state parameters on lattice/core reactivity) for U-Zr fuel, to routinely model many energy groups at the core level, and to upgrade tools to be able to capture the reactivity effect of core expansions. The staff is modifying the fuel performance code FAST (FRAPCON/FRAPTRAN merged code) for ease of changes, maintenance, and implementation of new reactor physics and geometries for non-LWR designs. Helium and liquid sodium coolant properties have been incorporated into FAST, and review of metallic fuel properties (U/Pu-Zr) has started. The staff is evaluating modifications to MACCS (MELCOR Accident Consequence Code System) to improve its capabilities to model atmospheric transport, deposition, and offsite consequences for non-LWR designs. The staff is also evaluating and improving the MACCS capability for probabilistic offsite dose exceedance calculations that can be used to inform emergency planning zone (EPZ) size determinations. Building on the past efforts for high-temperature gas reactors, the staff is identifying technical and regulatory gaps for materials and component integrity Issues for sodium fast reactors and molten salt reactors. The assessment will support the development of a draft regulatory framework for materials-related issues. In addition, the staff is actively participating in ASME Codes and Standards Sections III and XI activities related to materials selection and performance. Instrumentation and controls is a functional area that was not considered as the highest priority. No project was started in FY2017. For probabilistic risk assessment (PRA), the staff is first investigating operating experience and previous PRA work for non-LWR designs and then determining the gaps that exist in PRA policy, guidance, and technology. The staff is also identifying relevant technological trends in PRA methods, models, and tools to determine how these emergent technologies may be used in the future for non-LWR PRAs.

For More Information: Contact Steve Bajorek, RES/DSA, at Stephen.Bajorek@nrc.gov.

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) uses to review licensee submittals and to evaluate and resolve 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 extended power uprates and license renewals.
- Exploratory calculations to establish the technical bases for rule changes.
- Exploratory calculations for the resolution of generic issues.
- Confirmatory calculations in support of design certification and combined operating license reviews for new reactors and advanced reactors.

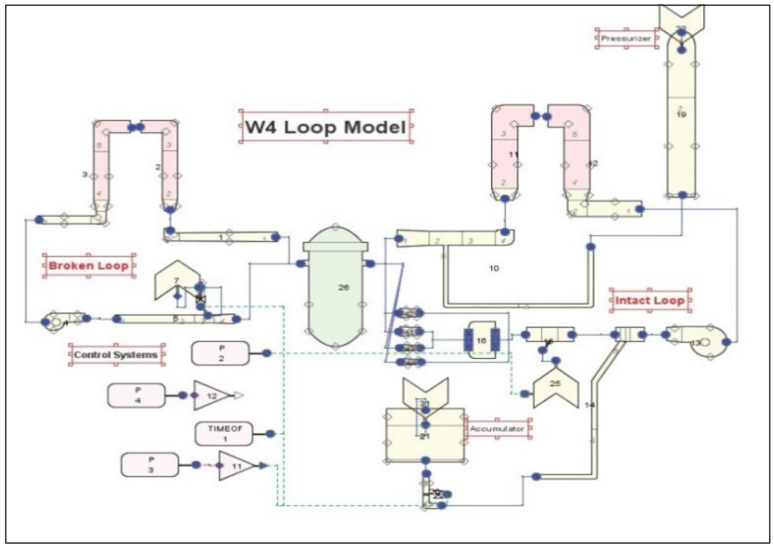

Figure 2.1 Simplified plant model nodalization.

New reactor designs include evolutionary advances in light-water reactor technology and modeling challenges 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. Advanced reactor efforts are focused on molten salt and sodium fast reactor issues. Molten salt reactors (MSRs) received emphasis because of a shortage of information on physical processes and the relative uniqueness of MSRs. Obtaining information on (and in some cases testing) new thermal-fluids codes developed outside of the NRC also received significant attention. Modeling of some of the novel systems and operating conditions of new and advanced reactors requires further code development and additional assessments against specific experimental data. Figure 2.1 shows an example of a simplified reactor system nodalization for TRACE.

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.

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 and thus plays a role in reducing the uncertainty in the technical bases for licensing decisions. 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.

The NRC conducts the Code Application and Maintenance Program (CAMP) with 30 nations through bilateral cooperative agreements. This program provides international contributions to 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.

Research Approach

The TRACE code features a two-fluid, compressible, non-equilibrium hydrodynamics model that can be solved across a one-, two-, or three-dimensional mesh topology. It also features built-in point reactor kinetics and a three-dimensional reactor kinetics capability through coupling with the Purdue Advanced Reactor Core Simulator (PARCS). The code has component models and mesh connectivity that allows a full reactor and containment system to be easily modeled.

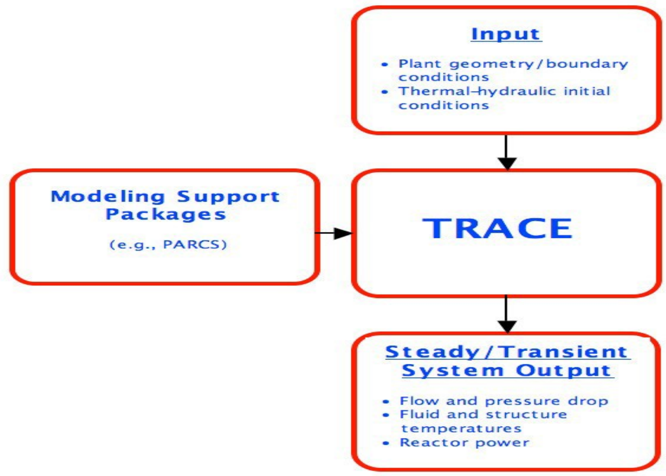

Figure 2.2 TRACE, an advanced, best-estimate reactor system code used to model the thermal-hydraulic performance of nuclear power plants.

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. The 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 other special purpose codes in the future. TRACE has also been coupled to the STAR CCM+ CFD code using an overlapping grid method. TRACE uses a modern code architecture that is portable, easy to maintain, and easy to extend with new models to address future safety issues (Figure 2.2 depicts a graphical representation of TRACE). TRACE has run successfully on multiple operating systems including Windows, Linux, and Mac OSX.

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.

Status

TRACE code development and assessment is an ongoing process. Improvements underway for future versions of TRACE are focused on enhancing capabilities related to the simulation of operating and new light-water reactor designs. Some members of the NRC's Code Application and Maintenance Program (CAMP) international cooperative research program use TRACE for heavy-water reactors and advanced non-LWR designs. TRACE has the flexibility to 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. SNAP removes the need for analysts to use a text-based entry method 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 among the codes.

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 provides a library of common components and physical systems that can be stored and reused in future model development. Integrated modeling guidelines enable the analyst to quickly build models that follow best practices. SNAP includes verification tools and automated model checking tools that correct input model errors. SNAP's interactive and post-processing capabilities allow the results of a running or completed calculation to be animated and visualized.

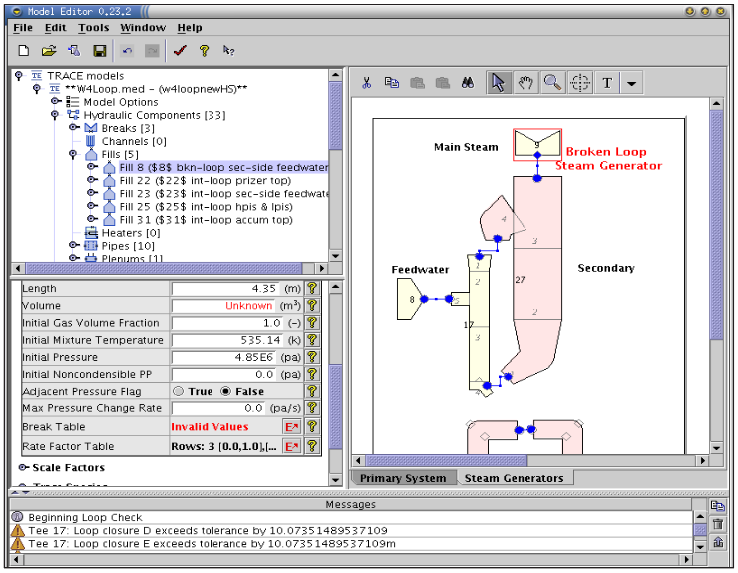

Figure 2.3 Creating input models using SNAP.

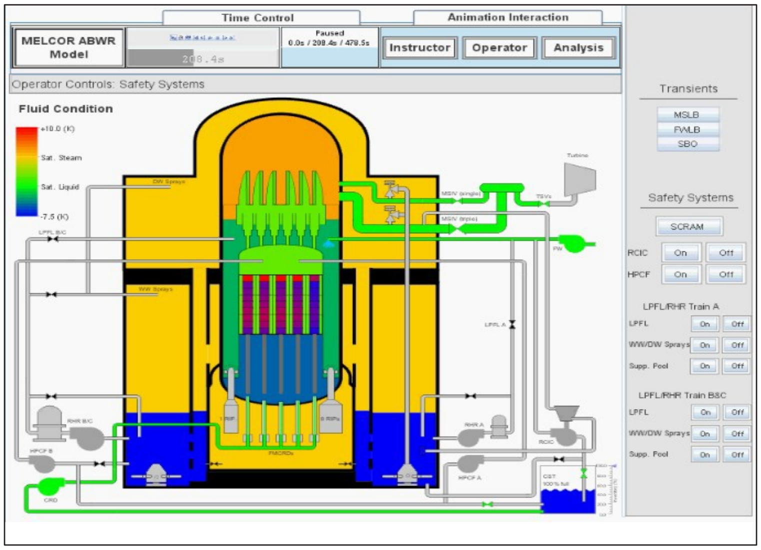

Figure 2.4 Animating analysis results using SNAP.

SNAP also provides for uncertainty quantification by automatic submission of data to the 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

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.

For More Information: Contact Chester Gingrich, RES/DSA, at Chester.Gingrich@nrc.gov.

Thermal-Hydraulic Simulations of Operating Reactors, New Reactors, and Integral Pressurized-Water Reactors

Objectives

The U.S. Nuclear Regulatory Commission (NRC) uses the TRACE code to confirm industry calculations submitted to the NRC to meet regulatory requirements. 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, the Advanced Boiling-Water Reactor, APR-1400, and NuScale.

Research Approach

TRACE plant input decks are developed for specific simulations (Figure 2.5). 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. 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.

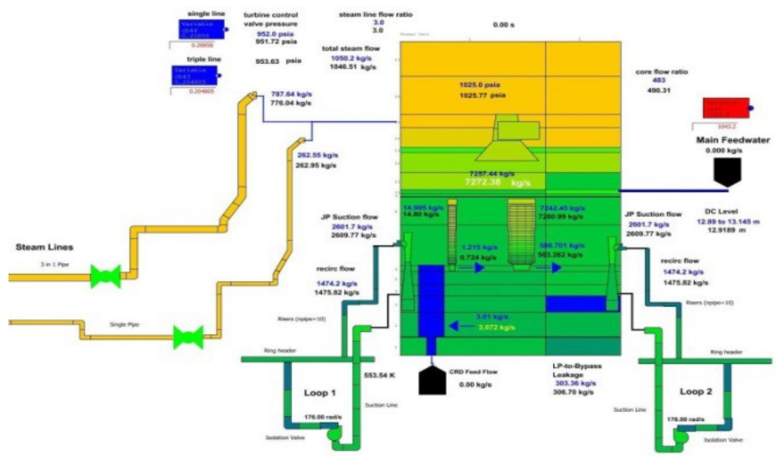

Figure 2.5 Steady-state conditions in a boiling-water reactor.

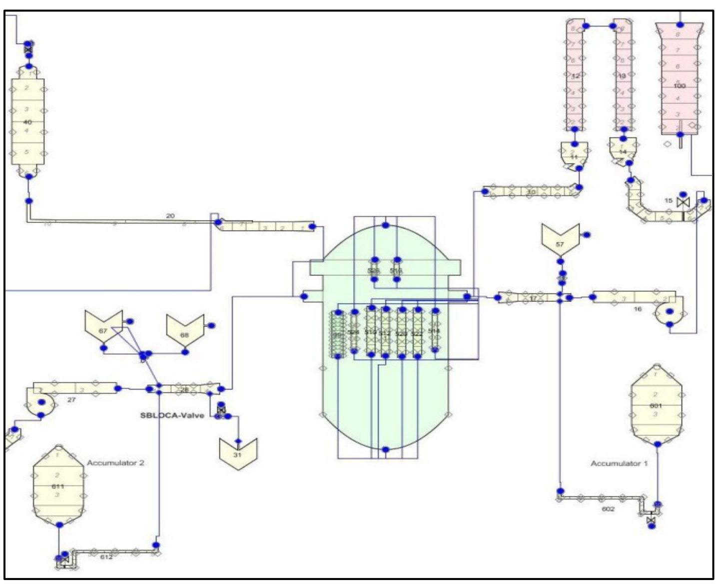

Figure 2.6 Key primary coolant T/H components including reactor vessel, pumps, and steam generator for a two-loop pressurized- water reactor.

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.

Status

RES is developing a library of TRACE input decks for simulating pressurized-water reactors and boiling-water reactors. Building a comprehensive library of plant input decks will enhance the NRC staff’s ability to efficiently perform confirmatory analyses to support regulatory decisions.

For More Information: Contact Chris Hoxie, RES/DSA, at Chris.Hoxie@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 that have extended power uprates. 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. 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.

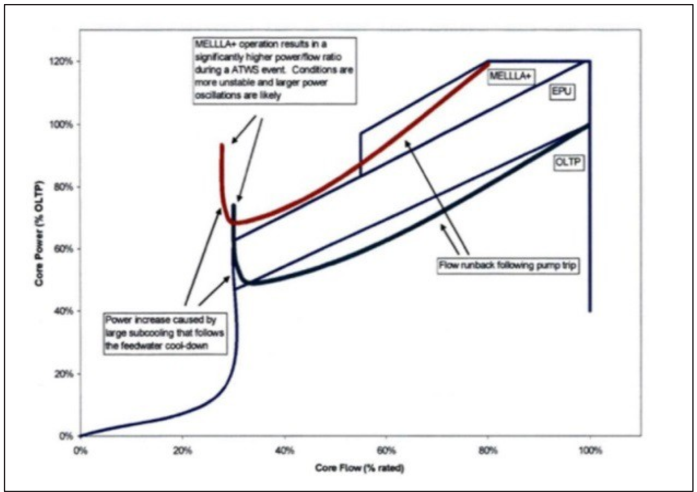

Figure 2.7 Operating state evolutions during ATWS for different operating domains.

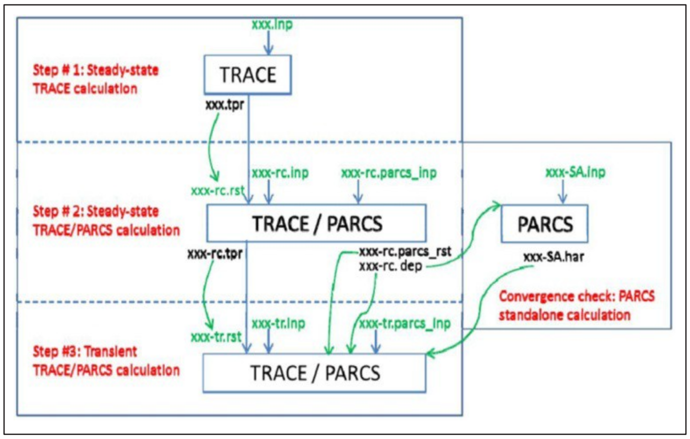

Figure 2.8 TRACE/PARCS coupled methodology.

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. 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. Calculations were performed using TRACE and PARCS in a coupled manner (Figure 2.8). 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.

Status

Visualization tools were used to analyze the evolution of the power oscillations during ATWS-I and to study the oscillation contour (Figure 2.9). TRACE/PARCS predicts the onset of large amplitude power oscillations and the evolution of an out-of-phase oscillation contour in this example.

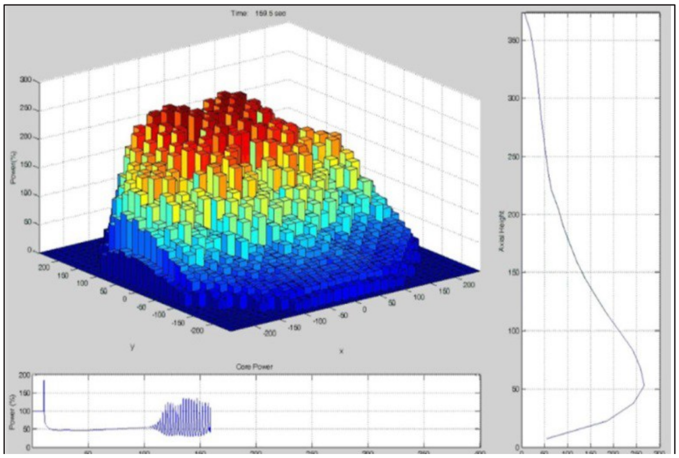

Figure 2.9 Power oscillation visualization during simulated ATWS-I.

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

Computational Fluid Dynamics in Regulatory Applications

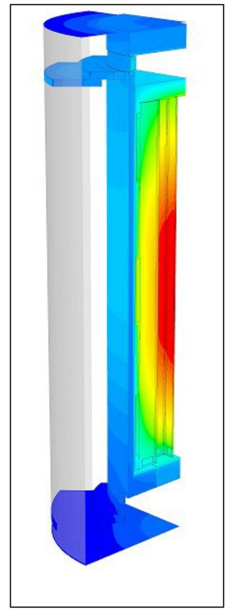

Figure 2.10 Temperature contours of a ventilated dry cask.

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

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.

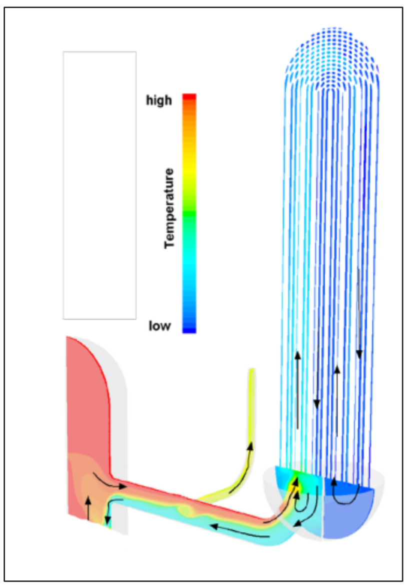

Figure 2.11 During a particular severe accident scenario, hot gases from the core circulate through the hot legs and steam generator in a counter-current flow pattern. The risk of induced failures is considered.

Status

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).

RES used CFD to confirm the distribution of injected boron in the economic simplified boiling-water reactor. 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.

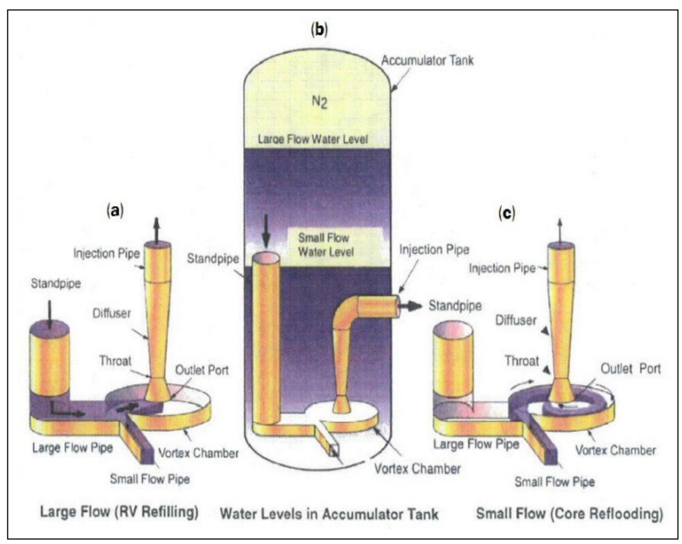

Figure 2.12 The advanced accumulator (b) is a water-storage tank with a flow damper in it that switches the flow rate of cooling water injected into a reactor vessel from a large (a) to small (c) flow rate.

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

Code Application and Maintenance Program (CAMP)

Purpose

The Code Application and Maintenance Program (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 TRACE, PARCS, and 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 TRAC/ RELAPS Advanced Computational Engine (TRACE), Purdue's Advanced Reactor Core Simulator (PARCS), Symbolic Nuclear Analysis Package (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.

Two CAMP meetings per year are held where members present their technical findings and share their experience using NRC T/H computer 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.

Status

In several recent cases, contributions to the CAMP program provided important code improvements and efficiencies for the NRC's regulatory programs. For example, the Republic of Korea's modeling of the advanced accumulator in the proposed APR-1400 reactor design for their own analysis and licensing efforts helped guide the NRC to model the advanced accumulator of both the APR-1400 design and 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 the 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 multi-group 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.

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

Thermal-Hydraulic Cooperative Programs

Objectives

The U.S. Nuclear Regulatory Commission (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.

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. The tests included interfacial area transport in pipes, annuli, and rod bundles; and 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 high-temperature facility is heavily instrumented and uses 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.
- 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.
- International Experimental Programs: 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. The 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 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. 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 economic simplified boiling-water reactor design. The Multi-Application Small Light Water Reactor (MASLWR) and NuScale Integrated System Test (NIST) facilities are being used to assess TRACE for applicability for use in analysis of NuScale.

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

Chapter 3: Fuel and Core Research

The U.S. Nuclear Regulatory Commission (NRC) 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.

RES 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 agencywide, 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 in the 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 U.S. 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. Additionally, 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 undergoing post-irradiation examination (PIE). The NRC relies on fuel property data from Halden to validate its steady-state and transient fuel performance codes including steady-state fission 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 release. 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. The NRC 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. The 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 U.S. Nuclear Regulatory Commission's (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.

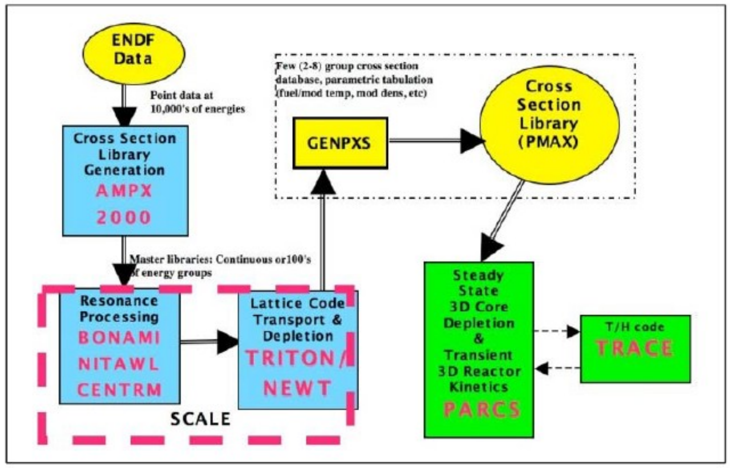

Figure 3.1 NRC nuclear analysis codes for reactor physics.

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. 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 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.

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 the development of advanced and non-LWR reactors.

For More Information: Contact: Don Algama, RES/DSA, at Don.Algama@nrc.gov, and Mourad Aissa, RES/DSA, at Mourad.Aissa@nrc.gov.

High-Burnup Light-Water Reactor Fuel

Objective

Current U.S. Nuclear Regulatory Commission (NRC) research on high-burnup light-water reactor fuel is focused in the following general areas:

- 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.
- Fuel rod computer code updates to prepare for analysis of proposed Accident Tolerant Fuels (ATF).

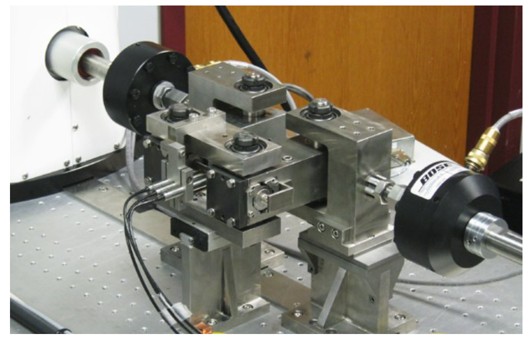

Figure 3.2 The CIRFT device at ORNL. A sister device is installed in a hot-cell to allow for testing of irradiated materials.

Research Approach

The NRC's 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 high-burnup (HBU) fuel were investigated using an innovative system, the Cyclic Integrated Reversible-bending Fatigue Tester (CIRFT), shown in Figure 3.2.

Status

The NRC recently published the results of the CIRFT testing program in NUREG/CR-7198, Revision 1. 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). The results reported in NUREG/CR-7198, Revision 1 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: Contact Michelle Bales, RES/DSA, at Michelle.Bales@nrc.gov.

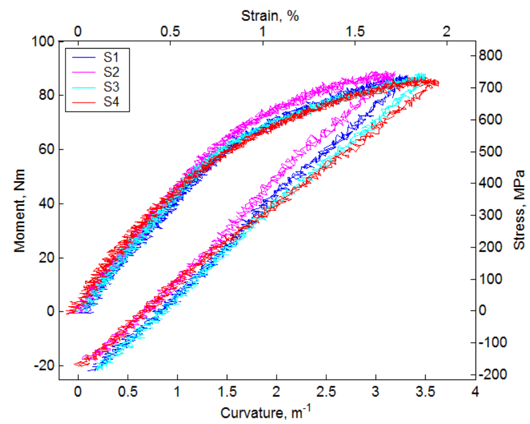

Figure 3.3 Moment-curvature measurements in static tests showing loading and unloading response. The corresponding stress/strain is displayed on right/top axes, respectively.

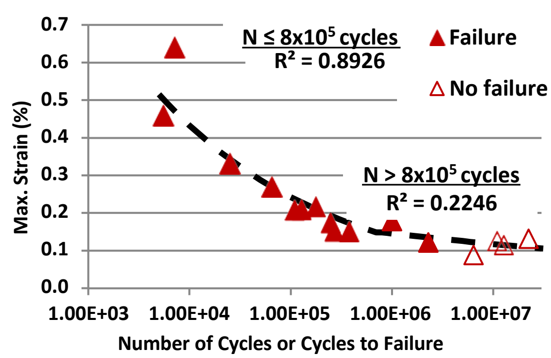

Figure 3.4 Maxima of absolute strain as a function of number of cycles to failure with curve-fitting extended to include the no-failure data.

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 ability to perform qualitative analysis of fuel rod behavior is an essential part of the U.S. Nuclear Regulatory Commission's (NRC's) assessment of safety in reactor operations and spent fuel transportation and storage.

The NRC has developed a new computer code, FAST (Fuel Analysis under Steady-state and Transients), to reliably predict fuel rod thermal and mechanical behavior under steady-state and transient conditions. FAST is the result of merging the NRC's previous fuel performance codes FRAPCON and FRAPTRAN (for steady-state and transient thermal-mechanical analysis, respectively) while introducing greater code flexibility by modernizing the source to the latest code-development standards. The NRC fuel behavior codes must be able to model current fuel designs deployed in the United States, and two ongoing developments in the industry are accident tolerant fuels (ATF) and advanced non-light water reactors (non-LWRs). FAST was designed to allow the implementation of new material properties to support future ATF licensing activities and is under development to allow for non-cylindrical geometries and new coolants to support advanced reactor fuel analysis.

Research Approach

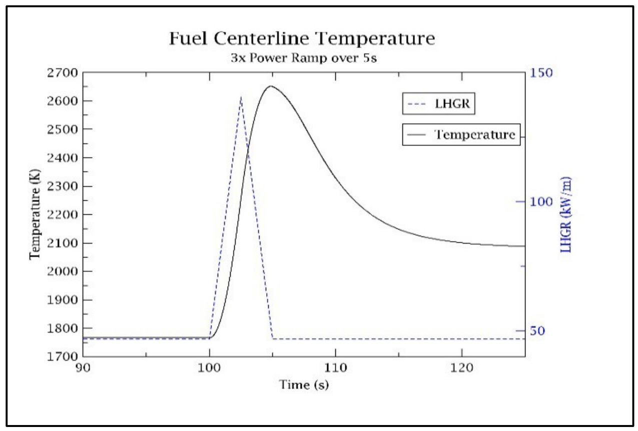

Figure 3.5: Fuel temperature prediction of a power spike.

The FAST code models the thermal-mechanical response of fuel rods including conduction, clad-to-coolant heat transfer, fission gas release, pellet-cladding contact, rod internal pressure, and cladding corrosion. The major improvement of FAST over FRAPCON is the ability to analyze transient response (such as an anticipated operational occurrence as shown in Figure 3.5, loss-of-coolant accident [LOCA]), and reactivity-initiated accident [RIA]). This includes time-dependent conduction through the rod, improved clad-to-coolant heat transfer correlations over multiple flow regimes, CHF correlations, ballooning, high-temperature oxidation, transient fission gas release, coated claddings, and cladding failure. A three-volume NUREG that details the Code Description, Integral Assessment, and Material Library (MatLib) will be available in 2018.

Status

Ongoing development efforts for FAST are focused on improving the inter-operability of FAST with other NRC-supported codes, building upon the code validation database, and preparing the code for the introduction of advanced fuel (uranium silicide (U_3Si_2) and metallic (U/Zr) fuel) and cladding (FeCrAl, SiC) materials, coatings, coolants (helium, sodium), and geometries. The NRC tool SNAP is now used as the graphical interface for FAST and facilitates data transfer between FAST and TRACE to allow for best-estimate LOCA analyses. FAST is under development to be coupled with PARCS and SCALE to assist with ATF and non-LWR analysis where models do not exist within FAST (e.g., radial power distribution as a function of burnup). The solutions to the physics solved within FAST are being updated beyond using the solid right cylinder geometry to allow for analysis of TRISO particles, plate fuels, and non-symmetric geometries. The NRC continues to participate in code benchmarking exercises, such as the OECD/NEA RIA benchmark Phase 2 and the Studsvik Cladding Integrity Project (SCIP) Phase III LOCA benchmarks. Best-practice modeling methods and model improvements are continuously derived from these exercises.

For more information: Contact Ian Porter, RES/DSA, at Ian.Porter@nrc.gov.

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, primarily for transportation and storage casks. This research is intended to ensure an integrated approach to criticality analysis among various U.S. Nuclear Regulatory Commission (NRC) applications and is also applicable to spent fuel pool storage.

Research Approach

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 cask system needs to ensure sub-criticality, thereby preventing criticality accidents. The research approach to burnup credit 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 and, for this reason, 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 cask 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.

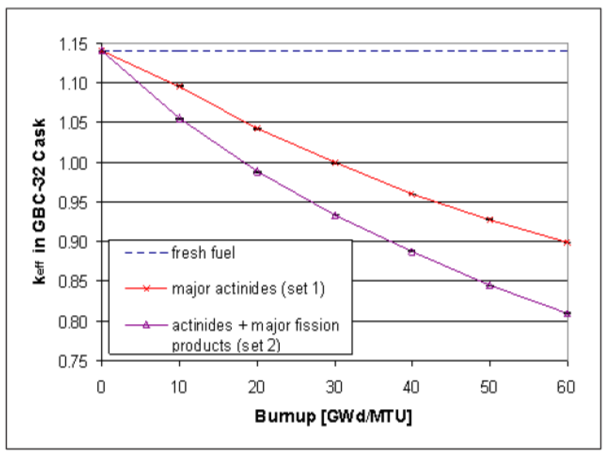

Figure 3.6 Comparison of typical reactivity decrements associated with burnup credit allowance.

Status

The NRC recently completed the work associated with pressurized-water reactor burnup credit. This work supported the release of Revision 3 of Interim Staff Guidance 8 (ISG-8) of the Office of Nuclear Material Safety and Safeguards. 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 bases for BWR burnup credit allowance.

For More Information: Contact Mourad Aissa, RES/DSA, at Mourad.Aissa@nrc.gov.

Fuel Cooperative Research

Objective

The U.S. Nuclear Regulatory Commission (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 at a reduced cost to the NRC. 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 U.S. 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.

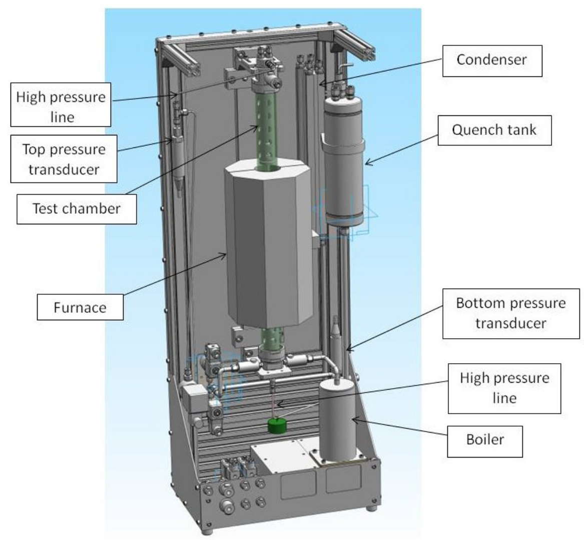

Figure 3.7 Integral LOCA test device. Test segments up to 12 inches long can be tested in this device.

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 release. 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 allows for greater understanding of the phenomena of fuel fragmentation, relocation, and dispersal through separate effects tests. The NRC relies on 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.

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Chapter 4: Severe Accidents and Accident Consequences Research

The Office of Nuclear Regulatory Research (RES) develops, maintains, and applies computational tools and methods to evaluate and resolve 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 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 and offsite consequences that provide quantitative predictive capabilities for simulating nuclear power plant response to severe accidents and offsite consequences. The role of such expertise and analytical capability is wide-ranging in the regulatory environment.

The U.S. Nuclear Regulatory Commission (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 NRC and international research programs 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. 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 potential atmospheric transport and deposition, emergency response andlong-term protective actions, exposure pathways, health effects, and economic costs. MACCS is used to evaluate the consequences of radiological releases for environmental reports and environmental impact statements and to support plant-specific evaluation of severe accident mitigation alternatives required as part of the environmental assessment for license renewal. In addition, MACCS is used in evaluating emergency planning and development of evacuation time estimates. It also provides input for cost-benefit analyses and is used in regulatory analysis. MELCOR and MACCS are and have been 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) analysis of offsite consequences from spent fuel pool accidents; (4) analysis of new and advanced reactors for design certification review, including small modular reactors; and (5) analysis of the accidents 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.

The NRC also sponsors and collaborates in experimental research programs to support the development and validation of models in its severe accident codes. Cooperative international experimental programs such as the Phébus-Fission Product experiments, Committee on the Safety of Nuclear Installations (CSNI) Source Term Evaluation and Mitigation (STEM), and CSNI Behaviour of Iodine Project Phase 3 (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 supports international collaboration on severe accident and offsite consequence modeling through the Cooperative Severe Accident Research Program (CSARP). CSARP includes 28 member nations that focus on the analysis of severe accidents and offsite consequences using the MELCOR and MACCS codes. CSARP also 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 U.S. Nuclear Regulatory Commission (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. MELCOR represents the current state-of-the-art in severe accident analysis and containment thermal-hydraulics.

Research Approach

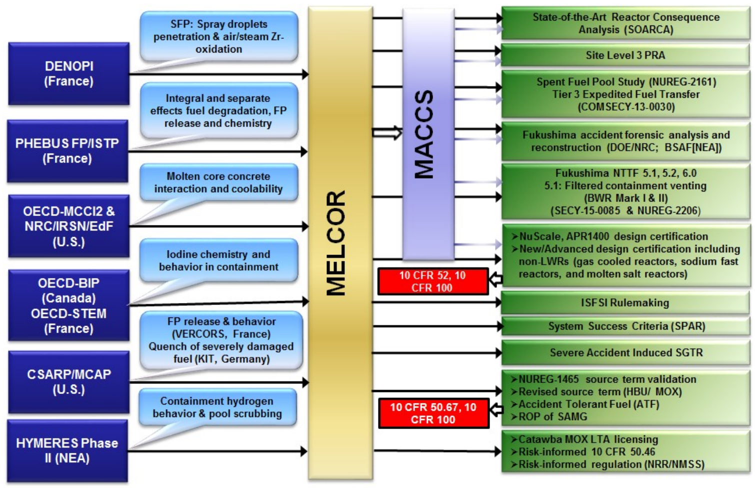

Figure 4.1 Severe accident experimental programs and MELCOR regulatory applications.

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 flow paths, 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. 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).

Status

MELCOR has been under continuous development by the NRC and Sandia National Laboratories. The most recent version of the code (MELCOR 2.2) was released in February 2017, and the code documentation was improved and updated. Current activities include development and implementation of new and improved models to predict the severe accident behavior of various reactor (both light water and non-light water) and spent fuel pool designs and to reduce modeling uncertainties. For non-light water reactor designs, current capabilities exist for gas-cooled reactors and models for sodium fires, and efforts are being made to use the code for prediction of source term for these designs. In addition, enhancements and more flexibility need to be added to the code to evaluate the safety of accident-tolerant fuel designs. Plans are underway to revise the code to improve stability and efficiency of explicit coupling and time integration. The improvements in the code numerics involve casting all implicit equations in residual form and enabling the 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.

MACCS (MELCOR Accident Consequence Code System)

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, along with its Windows-based graphical user interface, WinMACCS, is used to evaluate severe accident consequences as part of the environmental reports and environmental impact statements as well as to assist in emergency planning. These analyses also provide input to cost-benefit analysis for regulatory analyses and support plant-specific evaluation of severe accident mitigation alternatives that are required as part of the environmental assessment.

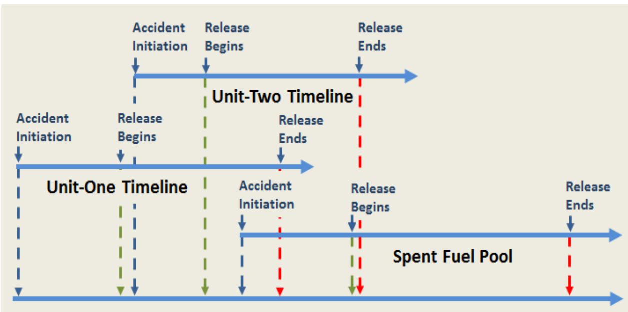

Figure 4.2 Example source release timeline for multiple releases at a single site with multiple units.

Research Approach

The MACCS code 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.

Version 3.10 of the code incorporates the following improvements:

- Added capability to model releases from multiple sources as shown in Figure 4.2 together with the MelMACCS 2.0 preprocessor code.
- Extended allowable phase durations to facilitate modeling of prolonged releases.
- Added a keyhole evacuation model that can simulate preferential evacuation of downwind areas as shown in Figure 4.3.
- Added outputs to evaluate cohort movement during the emergency phase and to evaluate size of population affected by both short- and long-term countermeasures.
- Implemented numerous usability improvements.

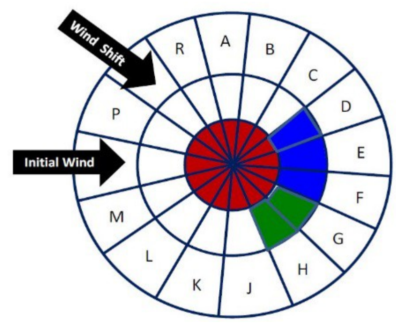

Figure 4.3 Example of the expansion of a keyhole evacuation area as a result of a wind shift.

Sector Population, Land Fraction, and Economic Estimation Program (SecPop) databases used to develop site-specific inputs to MACCS have also been updated to include population data from the 2010 U.S. Census and economic data from 2007 and 2012.

Status

Work is ongoing to update the MACCS code to include additional state-of-practice modeling approaches. Alternate atmospheric transport models are being introduced to MACCS by adding the capability to use results from the National Oceanic and Atmospheric Administration's Hybrid Single-Particle Lagrangian Integrated Trajectory (HYSPLIT) code. This will allow the use of Gaussian puff or Lagrangian particle tracking models, which may provide a better representation of atmospheric transport, dispersion, and deposition at longer ranges or in complex windfields. A new alternate economic model based on the existing Regional Economic Accounting Tool that Sandia National Laboratories developed for the U.S. Department of Homeland Security is under development. This model uses input-output models to capture the upstream supply chain impacts of affected industries outside areas directly affected by radiological releases. Feedback from an external peer review is currently being incorporated in the new economic model.

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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.

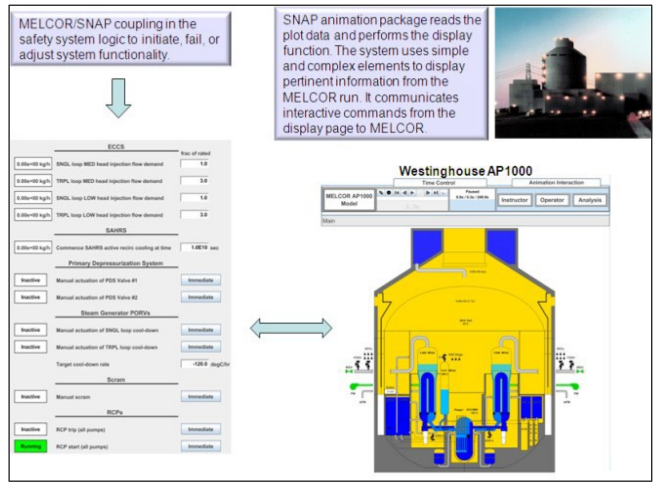

Figure 4.4 MASS user interface for AP1000.

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.

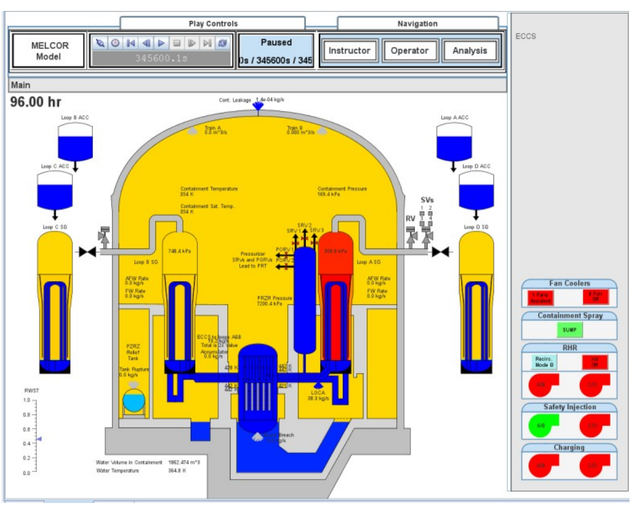

Figure 4.5 Accident progression for a PWR.

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 reactor, and Economic Simplified Boiling-Water Reactor, have been developed. The models run in design-basis and severe 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 operating 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 and spent fuel pools.

For More Information: Contact Hossein Esmaili, RES/DSA, at Hossein.Esmaili@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. The accident source term, which represents the release of radioactive materials to the containment, is being updated. To meet the siting criteria, the accident source term is used as input to simplified calculations that accounts for containment radionuclide removal processes and atmospheric dispersion as shown in Figure 4.6.

The accident source term provides a prescription of fission product release magnitude and timings that represent a broad range of accident scenarios. 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.

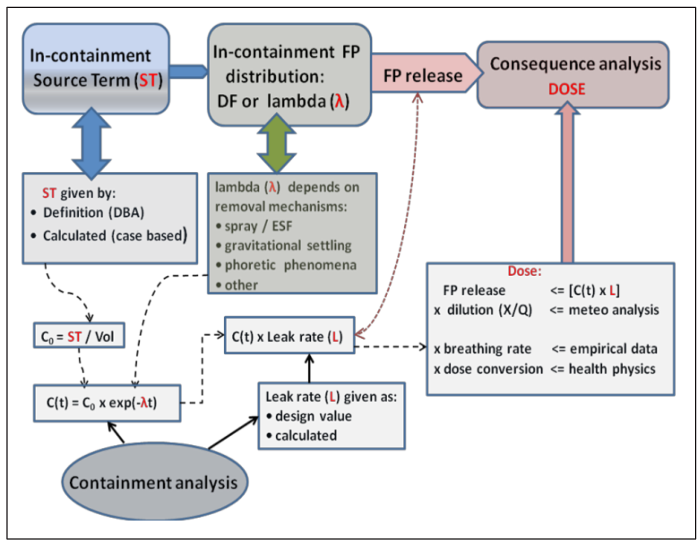

Figure 4.6 Use of source term and relation to other factors in dose calculation

Most 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. In the 1980s and 90s, new data and calculation tools (Source Term Code Package - STCP) were developed to define a new, mechanistic source term that more realistically represents the release of fission products from fuel and transport to the containment, NUREG 1465. The current work extends the containment source term applicability to high-burnup and mixed-oxide fuel based on additional experiments (VERCORS and VERDON) and calculations using the state-of-the-art integrated system level analysis MELCOR code.

Status

The work has been peer reviewed, and the comments and additional analysis have been completed. The NRC is in the process of completing the documentation.

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

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-Fission-Product (FP) experiments and to scale this observed behavior to operating power reactors.

Approach

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, pressurized-water reactor (PWR) sumps are buffered to keep the sump water alkaline. This prevents the iodine that reaches the sump from converting to volatile forms that can then be released to the containment atmosphere. Earlier pure-water benchtop experiments suggested that preventing 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.

The integral Phébus-FP experiments provided an opportunity to test code predictions of containment iodine behavior based on this model. However, the observations of the Phébus-FP experiments, the complexity of which more closely matches prototypic severe accident behavior, show that the expected behavior from the earlier iodine models may not necessarily be the case for power reactors. Instead of a continuously decreasing airborne iodine concentration as predicted by existing models in the presence of a basic sump, the containment iodine concentration developed a steady-state concentration. This steady-state concentration developed independent of whether the sump was acidic or basic.

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.

For a steady-state concentration of gaseous iodine to exist, sources of gaseous iodine must balance the sinks of gaseous iodine. 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 as part of an international experimental programs, primarily in the Behavior of Iodine Project (BIP) conducted in Canada and the Experimental Program for Iodine Chemistry Under Radiation (EPICUR) conducted in France.

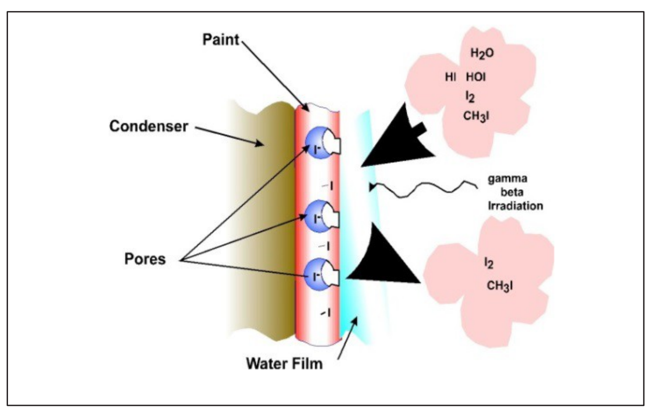

Figure 4.7 Hypothesized mechanism for gaseous iodine source in the Phébus-FP tests.

Status

The third series of BIP and EPICUR experiments are ongoing.

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

Severe Accident Progression in New and Advanced Nuclear Reactors

Objectives

The U.S. Nuclear Regulatory Commission (NRC) has received Design Certification Applications (DCA) for new reactor designs in the past two years. 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 received the DCA 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). The NRC is performing confirmatory analyses to verify the statements of fact set forth in the DCA in regards to design and beyond-design basis accidents.

The NRC received the DCA for the NuScale small modular reactor (SMR) in early 2017. SMRs, or Integral Pressurized-Water Reactors (iPWRs), are advanced reactor concept designs that use 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. The NuScale SMR is designed such that the primary coolant flows via natural circulation at the operating power level, and the reactor coolant pumps are eliminated. Although the frequency of core damage events in iPWRs 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 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 results. The unique design of iPWRs 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 iPWRs and to provide an assessment of the applicability of the MELCOR computer code to those analyses. In addition, confirmatory calculations will be performed to confirm the analyses provided in the applicant's DCA submission for severe accidents and containment response.

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 and NuScale. These analyses are used to determine the applicability of MELCOR to perform confirmatory analyses. A MELCOR model of the APR-1400 was developed, and confirmatory calculations of limiting containment response, hydrogen combustion, and severe accident response were performed. Confirmatory calculations supported the APR1400 Safety Evaluation Report (SER), which is in the final stages of completion. MELCOR models have been developed for the NuScale plant module and testing facility. Planned benchmarking calculation of the testing facility will ensure MELCOR can adequately predict potentially unique behavior of the NuScale thermal-hydraulic response. Also, new MELCOR models, which are unique to the NuScale design, have been incorporated into the plant model. Confirmatory calculations using MELCOR will again be performed for design-basis and beyond-design-basis events to aid in the completion of the NuScale SER.

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Technical Basis for the Containment Protection and Release Reduction (CPRR) Rulemaking for Boiling-Water Reactors with Mark I and Mark II Containments

Objectives

The objective of this study was to evaluate various containment protection and release reduction (CPRR) alternatives for boiling-water reactors (BWRs) with Mark I and II containments following an extended loss of ac power (ELAP) accident to support a proposed rulemaking directed by the Commission in SRM-SECY-12-0157. These analyses addressed the NRC's Fukushima Near-Term Task Force Tier 1 recommendation related to containment venting for BWRs with Mark I and II containments.

Research Approach

The research approach consisted of three primary components: (1) accident sequence analysis (event tree development) to identify accident sequences initiated by ELAP due to internal events and seismic events deemed to be the most significant risk contributors, (2) accident progression analysis of these sequences and assessment of radiological source terms using the MELCOR code, and (3) analysis of offsite consequences from these sequences using the MELCOR Accident Consequence Code System (MACCS). The calculated offsite consequences were weighted by accident frequency to assess relative public health risk reduction associated with various CPRR alternatives.

The probabilistic risk assessment (PRA) covered development of core damage event trees and accident progression event trees for an ELAP event, binning of a large number of possible end states to a manageable fewer categories, and an assessment of risks for these categories. The PRA also covered an assessment of risk reduction attributable to various accident management measures. MELCOR calculations consisted of a large number of accident sequences for a representative BWR Mark I containment and also a smaller subset of these sequences for a representative BWR Mark II containment. Mitigation measures modeled included both pre- and post-core damage venting, reactor pressure vessel (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.) were captured through sensitivity studies. For each MELCOR source term (i.e., fission product release into environment), MACCS calculations were performed for a representative plant site with specified site characteristics, population density, weather conditions, emergency response, and other aspects. MACCS calculations also included the potential influence of an external engineered filter on relevant figures of merit related to health risk, land contamination, and economic consequences.

Status

This analysis supported the draft regulatory basis enclosed with SECY-15-0085, and the complete technical analysis is being published as NUREG-2206. Key insights from the analysis include (1) a combination of venting and water addition is required to prevent containment failure, and water addition is a beneficial strategy for mitigating radiological releases; and (2) the proposed mitigation measures would result in reductions in offsite consequences in the event the accident occurs. However, these reductions would not meet the quantitative threshold for a substantial safety enhancement because the individual early fatality risk and the individual latent cancer fatality risk are significantly below the quantitative health objectives even without any of the proposed modifications.

For More Information: Contact Jonathan Barr or Hossein Esmaili, RES/DSA, at Jonathan.Barr@nrc.gov and Hossein.Esmaili@nrc.gov.

Fukushima Dai-ichi Accident Study with MELCOR 2.2

Objective

The U.S. Nuclear Regulatory Commission (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 II.

Research Approach

The NRC uses the MELCOR code to perform analyses of the Fukushima accidents. The BSAF Phase II study focuses on analyses covering three weeks (to April 1, 2011) from the initiation of the seismic event at the Fukushima plants on March 11, 2011. The analyses will include thermal-hydraulics calculations and estimation of the distribution of degraded core materials and their composition as well as fission product releases and transport within the primary containment vessel (PCV), reactor buildings, and offsite. A key objective of these analyses is to support the safe and timely decommissioning of the reactors at the Fukushima Dai-ichi site.

The Operating Agent for this NEA project is the Institute of Applied Energy (IAE) in Japan that serves as the technical coordinator for the study. Eleven countries—Japan, Canada, China, Finland, France, Germany, Korea, Russia, Spain, Switzerland, and the United States—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 Units 1, 2 and 3. Figure 4.8 shows an example of a MELCOR 2.2 prediction vs. measured data of Fukushima Unit 3. Among the participants, the NRC is the first in completing MELCOR analyses for all three Fukushima units covering three weeks (from March 11 to April 1, 2011) of the event. A final report on BSAF Phase II is expected by end of 2018.

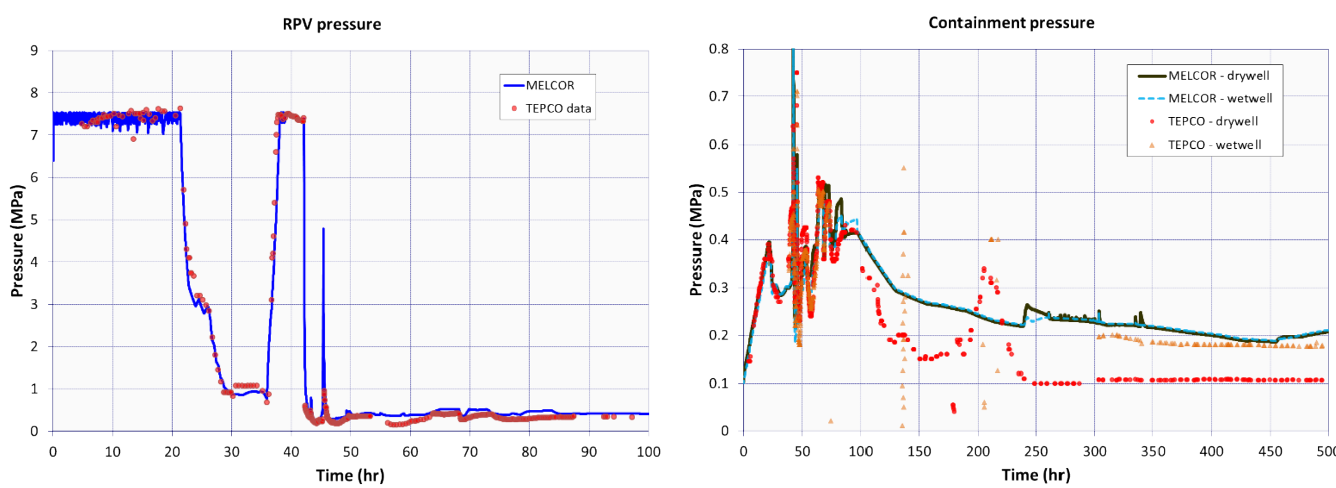

Figure 4.8 MELCOR-predicted reactor and containment pressures compared to TEPCO data (Unit 3).

For More Information: Contact Richard Lee, RES/DSA, at Richard.Lee@nrc.gov.

Sequoyah State-of-the-Art Reactor Consequence Analyses (SOARCA)

Objectives

The U.S. Nuclear Regulatory Commission (NRC) conducted the State-of-the-Art Reactor Consequence Analyses (SOARCA) project to evolve our understanding of realistic outcomes of severe reactor accidents. Following analyses of the Peach Bottom Atomic Power Station, a boiling-water reactor with a Mark I containment, and the Surry Power Station, a pressurized-water reactor (PWR) with a subatmospheric large, dry containment, the Commission approved additional analyses of a PWR with an ice condenser containment, the Sequoyah Nuclear Generating Station. The objective of the Sequoyah SOARCA was to reveal accident progression insights specific to the ice condenser containment and associated offsite consequences for station blackout (SBO) scenarios. Because ice condenser containments have a smaller volume and lower design pressure than other U.S. containment designs, they are potentially more susceptible to early failure from hydrogen combustion during a severe accident. The Sequoyah SOARCA project was also conducted to support the NRC's post-Fukushima evaluation of containment venting and hydrogen mitigation and control.

Research Approach

The Sequoyah SOARCA focused on SBOs including the unmitigated short-term SBO (STSBO) scenario postulated to be initiated by a large, beyond-design-basis earthquake. In the unmitigated STSBO, AC power is lost, and the turbine-driven auxiliary feedwater system is assumed inoperable. The NRC used updated plant- and site-specific information in its analysis and the latest computer codes: MELCOR 2.2 for accident progression calculations and MELCOR Accident Consequence Code System (MACCS) 3.10 for offsite consequence calculations. The Sequoyah SOARCA integrated consideration of uncertainty in parallel with deterministic calculations (see SOARCA uncertainty analyses discussion next). Results of interest include cesium and iodine release magnitude, hydrogen generation, and individual early and latent cancer fatality risks (See Figure 4.9).

Status

Sequoyah SOARCA calculations were generally consistent with those of past studies of ice condenser containments. Specific conclusions from the Sequoyah SOARCA include the following:

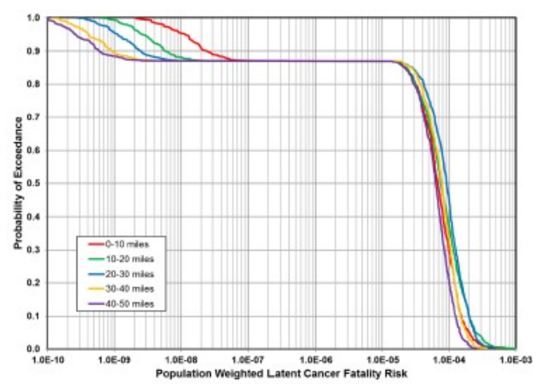

Figure 4.9 Example: Sequoyah SOARCA Results.

- For the unmitigated STSBO (without igniters), the two potential containment outcomes are either late failure (most likely) or early failure (much less likely).
- Successful use of igniters averts early containment failure.
- Essentially zero individual early fatality risk was calculated for Sequoyah STSBO.
- Even for cases resulting in early release to environment, the conditional (if the accident occurs) individual latent cancer fatality (LCF) risk is small.
- Conditional individual LCF risk results for Sequoyah are similar in magnitude to those from other SOARCA analyses.

Staff presented updated the Sequoyah SOARCA to the NRC's Advisory Committee for Reactor Safeguards subcommittee in June 2017 and is completing documentation in the NUREG/CR report series. This study's results and insights supported SECY-15-0137 and SECY-16-0041, which closed the NRC's evaluation of post-Fukushima recommendations related to containment vents and hydrogen control and mitigation.

For More Information: Contact Tina Ghosh, RES/DSA, at Tina.Ghosh@nrc.gov.

State-of-the-Art Reactor Consequence Analyses: Uncertainty Analyses

Objectives

The U.S. Nuclear Regulatory Commission (NRC) initiated the State-of-the-Art Reactor Consequence Analyses (SOARCA) project to evolve our understanding of realistic outcomes of severe reactor accidents. As part of SOARCA, the NRC undertook three uncertainty analyses (UAs): (1) to develop insights into the overall sensitivity of SOARCA deterministic 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 studies.

Research Approach

The NRC conducted a UA for one accident scenario at each of the three SOARCA pilot plants—the unmitigated long-term station blackout (SBO) at Peach Bottom, the unmitigated short-term (ST) SBO at Surry including an induced steam generator tube rupture variation, and the unmitigated STSBO at Sequoyah. For Sequoyah, staff conducted the UA in parallel with the overall deterministic analysis. The UAs involved perturbing numerous uncertain accident progression (MELCOR) model parameters and offsite consequence (MACCS, MELCOR Accident Consequence Code System) model parameters using Monte Carlo sampling of parameter probability distributions. Expert judgment was used to identify the most important parameters to include in each of the UAs. Hundreds of individual calculations were completed, and statistical regressions were performed to quantify uncertainty and to determine which parameters had the greatest influence on the results. Results of interest included cesium and iodine release to the environment, hydrogen generation, and individual early and long-term cancer fatality risks. An example of a result of interest is shown in Figure 4.10 for the Peach Bottom UA.

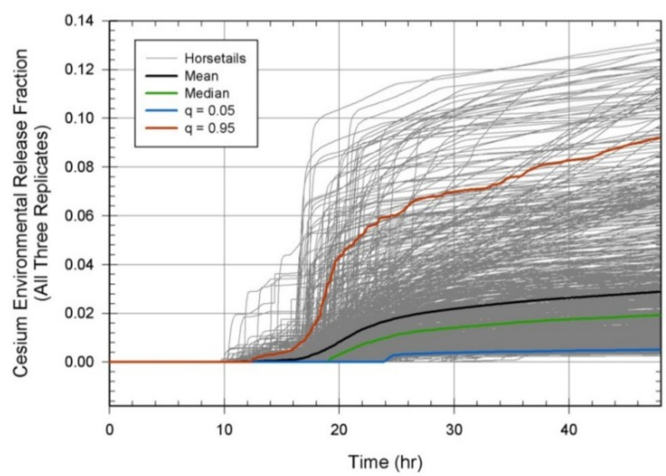

Figure 4.10 Example: Sequoyah SOARCA UA Results.

Status

All three SOARCA uncertainty analyses have corroborated the following SOARCA project conclusions:

- Modeled accidents progress more slowly and release smaller amounts of radioactivity than calculated in previous studies.
- Essentially zero individual early fatality risk was calculated for the accident scenarios analyzed.
- Longer term individual cancer fatality risks for the accident scenarios analyzed are millions of times lower than the general U.S. cancer fatality risk from all causes.

The Peach Bottom UA is complete and was published as NUREG/CR-7155 (2016). This UA 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. Staff is currently updating the draft Surry and Sequoyah UA reports, which were presented at Advisory Committee on Reactor Safeguards (ACRS) subcommittee meetings in February 2016 and June 2017, respectively. Lastly, the staff is developing a summary of important insights from the three SOARCA UAs to serve as a reference document for future regulatory applications.

For More Information: Contact Tina Ghosh, RES/DSA, at Tina.Ghosh@nrc.gov.

Research to Support Regulatory and Cost-Benefit Guidance Updates

Objectives

The Office of Nuclear Regulatory Research (RES) has an ongoing multiyear research program to develop improved guidance and tools to support the technical basis for regulatory and cost-benefit analyses. The U.S. Nuclear Regulatory Commission (NRC) conducts cost-benefit analyses as part of the regulatory review of substantial safety enhancements (i.e., backfit analysis), regulatory analysis, and plant-specific evaluation of severe accident mitigation alternatives and severe accident mitigation design alternatives for environmental impact statements.

Research Approach

The accident at Fukushima Dai-ichi initiated a discussion of how the NRC's regulatory framework considers economic consequences caused by a significant unintended radiological release. In response to this discussion, NRC staff issued Commission paper SECY-12-0110, "Consideration of Economic Consequences Within the U.S. Nuclear Regulatory Commission's Regulatory Framework," to provide the Commission with information and options to address to what extent, if any, the NRC's regulatory framework should be modified regarding its consideration of economic consequences of an unintended release of licensed nuclear materials to the environment. In response to SECY-12-0110, the staff was directed to perform a regulatory gap analysis to identify areas to potentially change or enhance cost-benefit practices.

The staff identified potential changes to current methods and tools related to performing cost-benefit analyses and recommended a two-phased approach to revise the cost-benefit guidance documents:

- Phase 1: Restructure cost-benefit guidance focusing on incorporation of cost-estimating best practices.
- Phase 2: Identify and discuss potential policy issues for the Commission.

As part of phase 1, the staff revised the dollar per person-rem conversion factor using an updated Value of Statistical Life (VSL) and cancer risk coefficient in draft NUREG-1530, Revision 1. A new VSL was selected based on literature review to benchmark the values other Federal agencies have used taking into account recent VSL studies and new meta-analyses. In addition, this revision adopted the Environmental Protection Agency's most recent estimate of the cancer mortality risk coefficient that used models recommended in the National Academy of Science's Biological Effects of Ionizing Radiation (BEIR) VII report. Also, the cost-benefit guidance documents, NUREG/BR-0184, "Regulatory Analysis Technical Handbook," and NUREG/BR-0058, "Regulatory Analysis Guidelines of the U.S. NRC," are being consolidated and restructured to enhance the currency and consistency of this guidance. RES is developing appendices to this consolidated document that will serve as detailed technical references for regulatory analysts.

Status

Revision 1 of NUREG-1530 provides the staff's basis for proposing the new dollar per person-rem conversion factor ($5,200 from $2,000) as a monetary value of the cancer mortality risk resulting from radiation exposure. This value incorporates an updated cancer mortality risk coefficient and a revised value of statistical life. As part of phase 2, the staff continues to develop appendices to NUREG/BR-0058 that provide guidance and best practices for performing severe accident risk analyses supporting backfit and regulatory analyses. The guidance will address quantifying benefits and characterizing the uncertainties associated with these calculations. The staff is also developing a basis for the monetary valuation of morbidity risks as part of cost-benefit analysis. In addition, the staff is conducting research and outreach to other Federal agencies on their process for valuing morbidity effects as part of cost-benefit analysis.

For More Information: Contact Amy Sharp, RES/DSA, at Amy.Sharp@nrc.gov.

Enhancing Guidance for Evacuation Time Estimate Studies

Objectives

An evacuation time estimate (ETE) is a calculation of the time to evacuate the plume exposure pathway emergency planning zone (EPZ), which is an area with a radius of about 10 miles around a nuclear power plant. The ETE is primarily used to inform protective action decisionmaking and may also be used to assist in development of traffic management plans to support an evacuation.

The U.S. Nuclear Regulatory Commission (NRC) updated the regulatory requirements for ETEs in a 2011 rulemaking requiring that licensees update ETEs after every decennial census and in between censuses as necessary. Prior to the rulemaking, a wide variation of ETEs existed in terms of vintage, methodology, and content. The NRC published NUREG/CR-7002, "Criteria for Development of Evacuation Time Estimate Studies," to support standardization of ETEs.

Although the study of evacuations has matured, it remains an area of active research. Improvements in traffic modeling and research in behavioral responses to evacuations can lead to better estimates of EPZ clearance times. The NRC is sponsoring a research study to examine technical topics associated with the modeling and simulation of evacuations. The results of this study are expected to strengthen the technical basis for ETEs and to enhance the NRC's regulatory guidance.

Research Approach

Three highly-detailed traffic simulation models are being developed that model representative small, medium, and large population sites and the associated roadway networks around nuclear power plants (See Figures 4.11 and 4.12). These are considered microscopic traffic simulation models because they simulate microscopic properties like the position and velocity of single vehicles. These models are being used to perform the following task analyses:

- *Task 1: Assess the Impact of Shadow Evacuation*: A shadow evacuation occurs when people evacuate from areas outside of a declared evacuation zone. This task will assess the sensitivity of the ETE to the impact of shadow evacuations.
- *Task 2: Distance of Evacuation Travel*: ETEs provide the estimated clearance time when vehicles exit the EPZ. This task will assess the travel time to specific safe destinations or a set distance beyond the EPZ. In addition, this task may inform the extent of the roadway network beyond the EPZ that needs to be modeled to provide the proper boundary conditions for the ETE.
- *Task 3: Manual Traffic Control:* This task will study the impact of manual traffic control of intersections and its effect on ETEs.
- *Task 4: Determination of Variable Importance:* A sensitivity analysis is being conducted to quantify the sensitivity of clearance times to parameter inputs to determine their relative importance.

Status

Development of the microsimulation models is complete. Work has started on the task analyses.

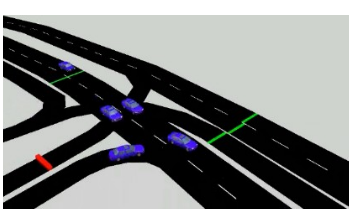

Figure 4.11 Microscopic Traffic Simulation.

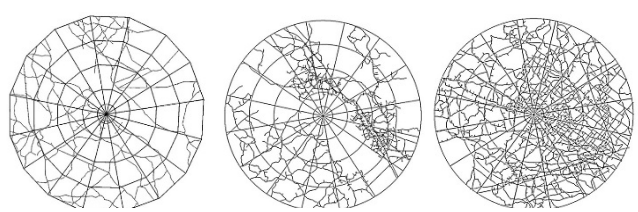

Figure 4.12 Small, Medium, and Large Population Roadway Networks.

For More Information: Contact Sergio Gonzalez, RES/DSA, at Sergio.Gonzalez@nrc.gov.

Offsite Response Organization Capabilities and Practices for Protective Actions in the Intermediate Phase of Emergency Response to a Nuclear Power Plant Accident

Objectives

Emergency planners divide responses to radiological incidents into three phases: early, intermediate, and late. The intermediate phase refers to the period after the immediate response has subsided, and releases are controlled or understood. Offsite response organizations (OROs) have a variety of capabilities and practices in place for actions during the intermediate phase. However, the U.S. Nuclear Regulatory Commission's (NRC's) understanding of ORO decisionmaking regarding protective actions in the intermediate phase is less comprehensive than in the early phase of emergency response. Therefore, the NRC initiated a study of ORO capabilities and practices in the intermediate phase to inform the modeling of protective actions in offsite consequence analyses using the MELCOR Accident Consequence Code System (MACCS).

Research Approach

The study was conducted in two phases. Phase I involved the examination of After Action Reports (AARs) from evaluated ingestion pathway exercises and a sample of state radiological emergency response plans. Phase II involved interviews with State-level ORO decisionmakers and staffs. Both phases of this study focused on the following six topics:

1. Identification of radiological hot spots outside of initially evacuated areas.
2. Relaxation of evacuation and relocation orders.
3. Food condemnation or embargo.
4. Drinking water safety.
5. Evacuation/relocation beyond the 10-mile emergency planning zone (EPZ).
6. Best practices or alternative practices in the intermediate phase.

Status

This study has been completed and is being published in the NUREG/CR report series. The aggregate findings identified several key trends across the six study topics. Overall trends regarding protective action decisionmaking in the intermediate phase included a focus on cooperation within States, regionally, and with the Federal Government; use of Federal guidance; situation-dependent decisionmaking methods; conservative approaches to contamination and condemnation; and the use of technological tools to enhance actions taken in the intermediate phase. In addition, OROs generally rely on Environmental Protection Agency (EPA) Protective Action Guides (PAGs) and Food and Drug Administration (FDA) Derived Intervention Levels (DILs) in their protective action decisionmaking. Therefore, the NRC models can continue to use these values with confidence. A detailed analysis of the study's results can be found in the final NUREG.

For More Information: Contact Sergio Gonzalez, RES/DSA, at Sergio.Gonzalez@nrc.gov.

Research to Support Severe Accident Mitigation Alternatives Analyses for Reactor License Renewal

Objectives

Site-specific Severe Accident Mitigation Alternative (SAMA) analyses are a National Environmental Protection Act (NEPA) requirement implemented in 10 CFR Part 51 for nuclear power plant license renewal applications and are documented in the U.S. Nuclear Regulatory Commission's (NRC's) Final Supplemental Environmental Impact Statements (FSEIS, NUREG-1437 supplements). The NRC's Office of Nuclear Regulatory Research (RES) has been conducting research to ensure that SAMA analyses adequately meet NEPA requirements where contentions have been admitted by the Atomic Safety Licensing Board (ASLB) and ASLB decisions have been reviewed by the Commission.

Research Approach

The staff's focus is to ensure that the MELCOR Accident Consequence Code System (MACCS) analyses supporting SAMA evaluations follow established and accepted practices and address any unique aspects of the site. Two recent examples of research support are the SAMA evaluations for Indian Point 2 and 3, and Fermi 2 license renewals. In the Indian Point 2 and 3 license renewal proceeding, the Commission issued Order CLI-16-07 on May 4, 2016, directing the staff to supplement the Indian Point SAMA analysis with further sensitivity analyses. Specifically, the Commission held that documentation was lacking for two MACCS input parameters related to decontamination modeling. The State of New York had shown in the ASLB proceeding that uncertainty in these two parameters could affect the SAMA analysis cost-benefit conclusions. RES staff requested specific sensitivity analyses from the applicant to address CLI-16-07 and documented its evaluation in Supplement 2 to the Indian Point FSEIS.

The Fermi site has a unique location because it includes significant foreign land and population in Canada within its 50-mile calculational area (as shown in Figure 4.13), which adds a level of complexity to the offsite consequence analysis. MACCS calculations typically use site files generated by the SecPop preprocessor code that uses U.S. population, land use, and economic databases but does not use any foreign data. Therefore, an alternative approach was needed to reasonably account for the Canadian population and land. RES staff conducted sensitivity analyses to verify that the applicant's calculations were reasonable.

Figure 4.13 Concentric rings 10-50 miles from the Fermi 2 site showing both U.S. and Canada within 50 miles.

Status

The staff is documenting its additional SAMA evaluations in the Indian Point 2 and 3 FSEIS, Supplement 2. With the exception of one SAMA candidate for Indian Point 3 for a 20-year license renewal period, the increased averted-accident-cost benefits from the sensitivity analyses were within the existing bounds of the "Benefit with Uncertainty" representing uncertainty in the analysis in the current FSEIS. For the Fermi 2 SAMA evaluation, the staff conducted MACCS sensitivity calculations with the higher decontamination costs and times that were indicated in CLI-16-07 and with a reasonable approach for modeling Canadian population and land. The staff found that the increased averted-accident-cost benefit was still within the existing bounds of the "Benefit with Uncertainty" representing uncertainty in the analysis. The sensitivity calculations allowed the NRC to complete its review in a timely manner with more confidence in the results.

For More Information: Contact Tina Ghosh, RES/DSA, at Tina.Ghosh@nrc.gov.

Modeling of Radionuclide Transport in Freshwater Systems Associated with Nuclear Power Plants

Objective

The Fukushima Dai-ichi nuclear accident leaked more than 1,000 m^3 of highly contaminated water directly into the ocean. This sort of direct leakage had not been analyzed by the U.S. Nuclear Regulatory Commission (NRC). A key objective of this research was to model the potential behavior of a hypothetical aqueous release from a reactor accident to improve the agency's understanding of potential consequences of water-borne releases to three types of freshwater bodies.

Research Approach

This study provides state-of-the-art hydrologic transport modeling results of a hypothetical nuclear power plant accident that leaks waterborne radionuclides directly into a freshwater body. The study postulates releases to three types of freshwater settings typical of nuclear power plants within the United States: a large river, a small river, and a small lake. Two- and three-dimensional modeling is used to explore the relative concentrations of radionuclides as they are transported through the three freshwater bodies. The approach is to determine how advection, dilution/dispersion, radioactive decay, and adsorption/desorption processes affect concentrations of each of the transported radionuclides under the hypothetical conditions in each freshwater setting.

The eight radionuclides studied are associated with aqueous releases expected from damaged light-water reactor fuel. Results are presented for a set of hypothetical locations as Bq/m^3 based on the initial release of a 1 Bq source term. Because these results scale linearly, these fractional concentrations can be multiplied by any source term of interest. Results show that small lakes would be the most impacted, retaining high concentrations for long times. In rivers, the initial pulse of high concentration contaminants moves downstream but remains intact with high concentrations for long distances (almost 300 miles in this study). A key feature of the river transport model is the downstream persistence of higher concentrations along the near shore of the radionuclide release. There can be orders of magnitude differences in near-shore versus far-shore concentrations. Effects of dams on concentrations and travel time are discussed.

Sediment interactions with contaminants results in about 5 percent of radionuclides retained on sediment. After the passage of the waterborne pulse, desorption from the bed sediments becomes a longer-term, widespread source as the sorbed contaminants are driven to re-equilibrate with the cleaner overlying water. Depending on the magnitude of radioactivity released, concentrations of radionuclides sorbed to bed sediments, especially at low-water stages, could have long-term impacts on shoreline use.

Results cannot be taken as representative of any particular nuclear power reactor facility as the modeling parameters and data are highly site specific. However, the observations of the movement of a relatively high concentration pulse of contaminants, the long distances it can be transported in rivers, and the possibility of long-term sediment/water/contaminant interactions are important general insights from this study.

Status

This project has been completed, and the final report, NUREG/CR-7231, "Modeling of Radionuclide Transport in Freshwater Systems Associated with Nuclear Power Plants", was published in April 2017.

For More Information: Contact Mark Fuhrmann, RES/DRA, at Mark.Fuhrmann@nrc.gov.

Cooperative Severe Accident Research Program (CSARP)

Objective

The U.S. Nuclear Regulatory Commission (NRC) has invested heavily in the investigation of severe reactor accidents and has developed computer codes for the analysis of severe accident phenomena, progression, and offsite consequences. CSARP and its technical review meetings provide a forum to exchange technical information on severe accident research to gain insight into regulatory and 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. The NRC coordinates CSARP activities with participation from 28 member nations that focus on the analysis of severe accidents and their offsite consequences using state-of-the-art computer models. The MELCOR computer code is used for modeling accident progression, and the MELCOR Accident Consequence Code System (MACCS) is used to evaluate offsite consequences from a hypothetical release of radioactive material into the atmosphere. The NRC supports and hosts a number of meetings annually to share progress in severe accident research and to report on the status of code development and assessment. Through CSARP, the NRC has access to substantial international severe accident research.

Status

Figure 4.14 CSARP participation and technical meetings.

Topics that are discussed at the CSARP technical review meeting held in September include recent advances in severe accident research programs such as (1) latest information and analysis from the Fukushima accident and status of code modeling, (2) French Institute for Radiological Protection and Nuclear Safety (IRSN) Phebus-Fission Products experiments (France), (3) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI) Behavior of Iodine Project (BIP) experiments, (4) IRSN mixed oxide and high-burnup fuel fission product release experiments (France), (5) Karlsruhe Institute of Technology QUENCH experiments investigating overheated fuel (Germany), and (6) Organization for Economic Co-operation and Development (OECD) molten core-concrete interaction and debris coolability experiments conducted at Argonne National Laboratory (USA).

Other CSARP program meetings include: (1) MELCOR Code Assessment Program (MCAP), which focuses on MELCOR code development and assessment and provides a forum for the presentation and discussion of the user's experience; and (2) International MACCS User Group (IMUG) technical review meeting, which is a forum for the exchange of information and research experience among users of the MACCS code. The Asian MELCOR/MACCS User Group (AMUG) and European MELCOR User Group (EMUG) meetings are held annually to provide an opportunity for more code users to interact with the code development teams. The most recent AMUG meeting was held in South Korea in November 2017 and the next EMUG meeting is planned for Croatia in spring 2018. MELCOR and MACCS User Workshops are included at the meetings and are hands-on training sessions on the use of the codes.

For More Information: Contact Richard Lee, RES/DSA, at Richard.Lee@nrc.gov and Patricia Santiago, RES/DSA, at Patricia.Santiago@nrc.gov.

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 assessing 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. After the Fukushima Daiichi accidents in March 2011, the molten core-concrete interaction (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 (specifically in boiling-water reactors) 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 Organization for Economic Co-operation and Development (OECD) MCCI program. The Commissariat l'Energie Atomique aux Alternatives (CEA) is designing and constructing the PLINIUS 2 facility to conduct large-scale MCCI testing. The commissioning of PLINIUS 2 facility is expected in 2023.

The hydrogen stratification issue is continued under the NEA HYMERES-2 project. The study of fission products behavior especially for iodine continued as part of the NEA STEM-2 and BIP-3 projects. Concurrently, analytical work is performed to supplement the experimental activities, and such work involves analysis of experimental data and development of phenomenological models.

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 such as the NRC's MELCOR code. The Fukushima event, however, pointed to the need for additional MCCI data that are more representative of boiling-water reactors core melt composition.

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 completed work by the OECD Senior Expert Group Safety Research Opportunities Post Fukushima has identified near-term activities to be pursued to assist in the Fukushima Dai-ichi decommissioning.

For More Information Contact: Richard Lee, RES/DSA, at Richard.Lee@nrc.gov.

Fukushima Cooperative Research

Objective

The Nuclear Regulatory Commission (NRC) participates in the Organization for Economic Co-operation and Development (OECD)/Committee on the Safety of Nuclear Installations (CSNI)/Nuclear Energy Agency (NEA)-led activities following the Fukushima Dai-ichi Accident.

Research Approach

Since NEA issued its report, "Nuclear Safety Response and Lessons Learnt," (NEA 7161, 2013) in response to the Fukushima Daiichi Nuclear Power Plant Accident in March 11, 2011, the NRC has participated in the OECD/NEA sponsored joint nuclear safety research projects including:

- **B**enchmark **S**tudy of the **A**ccident at the **F**ukushima Daiichi Nuclear Power Station (BSAF) Phase I and Phase II.
- NEA Senior Expert Group on **Sa**fety **Re**search Opportunities Post-**F**ukushima (SAREF).
- **Hy**drogen **M**itigation **E**xperiments for **Re**actor **S**afety (HYMERES) II.
- **S**ource **T**erm **E**valuation and **M**itigation Project Phase 2 (STEM-2).
- **B**ehaviour of **I**odine **P**roject Phase 3 (BIP-3).

The NRC has also participating in report writing groups of the following **C**SNI **A**ction **P**roposal **S**heet (CAPS):

- CAPS on "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.
- CAPS on "Long-term management of a severe accident in a nuclear power plant"—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 nuclear power plant.
- CAPS on "Report on Phenomena Identification and Ranking Table on Spent Fuel Pool Loss of Cooling Accident"—to conduct a Phenomena Identification and Ranking Table (PIRT) based on expert opinion elicitations to study the reactor fuel degradation that takes place in an atmosphere of steam and air such as a partial drain down scenario in a spent fuel pool, determine from the PIRT whether the accidents in question pose sufficient risk to merit comprehensive (analytical and/or experimental) study, and determine the adequacy of existing severe accident codes to perform reactor fuel degradation under steam and air oxidation.

Status

Besides the NRC, the U.S. Department of Energy and the Electric Power Research Institute participate in BSAF. The BSAF Phase I project has been completed, and a summary report (NEA/CSNI/R(2015)18) was published by NEA in February 2016. The BSAF Phase II project will be completed at the end of 2018. NEA has almost completed the work on all three CAPS, and reports will be published in 2018.

The NEA will launch two SAREF near-term projects (a) **Pre**paratory Study on **A**nalysis of Fuel **De**bri**s** (PreADES) and (b) **A**nalysis of Information from **R**B and **C**V and water sampling of **F**ukushima Daiichi NPS (ARC-F). The near-term projects will provide continued interaction between NEA safety research experts and representatives from Japanese organizations involved in the Fukushima decommission and decontamination activities. PreADES will be launched in January 2018 for a duration of three years (January 2018 – December 2020), while the ARC-F project will be launched in late 2018. Both projects will be led by the Japan Atomic Energy Agency.

For More Information: Contact Richard Lee, RES/DSA, at Richard.Lee@nrc.gov.

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 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, policymaking, 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 that result in developing guidelines and publications for public consumption. RES develops and maintains radiation protection and health physics 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; research on health effects and advanced dosimetry; participation in and monitoring of radiation research activities by National and International scientific and standard setting organizations; exposure and abnormal occurrence reporting; support decommissioning research; and administering the 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.

Environmental Transport Research

The mission of the NRC's environmental transport research 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. Recent projects within this area have addressed the long-term behavior of engineered barriers (specifically rates of release of sequestered radon) and effects of aqueous radionuclide releases to freshwater bodies from a hypothetical severe reactor accident.

NRC Standards for Protection Against Ionizing Radiation and ALARA for Radioactive Material in LWR 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 agency 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).

Figure 5.1 Biokinetic model.

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 effort also supports the preparation of revised Federal Guidance reports on radiation protection and the development of analytical tools for possible revision of 10 CFR Part 20 and 10 CFR Part 50.

For More Information: Contact Minh-Thuy Nguyen, RES/DSA, at Minh-Thuy.Nguyen@nrc.gov.

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), access the scope of the issue by using a survey tool distributed to nine private medical institutions along with Federal partners, 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 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 study is expected to be complete by 2019. The staff plans to use the collected data and use the most up-to-date science for dose calculations to update the Regulatory Guide. Oak Ridge National Lab will be helping staff complete this work.

For More Information: Contact Vered Shaffer, RES/RPB, at Vered.Shaffer@nrc.gov.

The Million Person Study

Objective

The objective of the Million Person 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 U.S. Nuclear Regulatory Commission (NRC) had 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. The NRC supported DOE and this multiagency effort to 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.

Figure 5.3 Radiation worker taking measurements.

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 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 CTscans 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).

Status

Research on the NRC early nuclear power plant and industrial radiographer worker has started. The cohorts are now established; however, current funding shortfalls have slowed progress on completing the analysis.

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

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 U.S. Nuclear Regulatory Commission (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 about routine occupational exposures in licensee performance in radiation protection and for research and epidemiological studies.

Approach

To maintain compliance with 10 CFR 20.2206, NRC licensees must submit their occupational radiation exposure data to the NRC.

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.

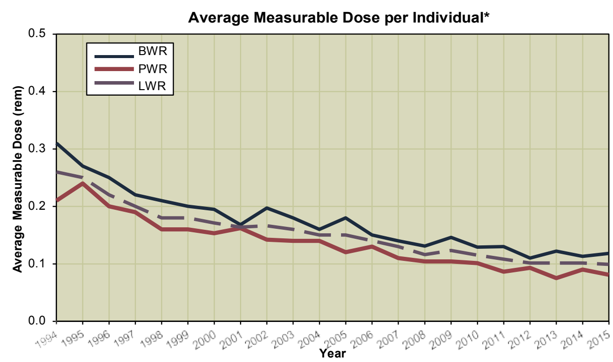

Figure 5.4 Annual Occupational Radiation Dose for PWR/BWR/LWR Reactors.

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 licensees’ as low as is reasonably achievable (ALARA) programs.
- 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 Minh-Thuy Nguyen, RES/DSA, at Minh-Thuy.Nguyen@nrc.gov.

Radiation Protection Computer Code Analysis and Maintenance Program (RAMP)

Objective

The Radiation Protection Computer Code Analysis and Maintenance Program (RAMP) (Figure 5.5) is a program for developing, maintaining, and distributing the U.S. Nuclear Regulatory Commission's (NRC's) radiation protection, dose assessment, and emergency response computer codes. The codes in RAMP include RASCAL, SNAP/RADTRAD, VARSKIN, GALE, DandD, HABIT, GENII, Atmospheric Codes (XOQDOQ, ARCON, PAVAN), and a graphic user interface (GUI) called PIMAL, and Radiological Toolbox.

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.

Figure 5.5 RAMP Logo.

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.
- Periodic training and meeting on the codes to share experiences and discuss code development.

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 to discuss state-of-the-art models, emerging technologies, and various other radiation protection safety issues.

Status

RAMP has over 1,000 domestic and international members including other Federal Agencies such as EPA, DOE, FEMA, NIST and DOD. Members also include U.S State agencies and a number of countries and international partners in North America, Europe, Asia, The Middle East, Africa, South America, and Australia. For the current status of RAMP including User Meeting information, updates on codes, and the latest information, visit us at https://www.usnrc-ramp.com or contact us at ramp@nrc.gov.

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

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 U.S. Nuclear Regulatory Commission's (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).

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.

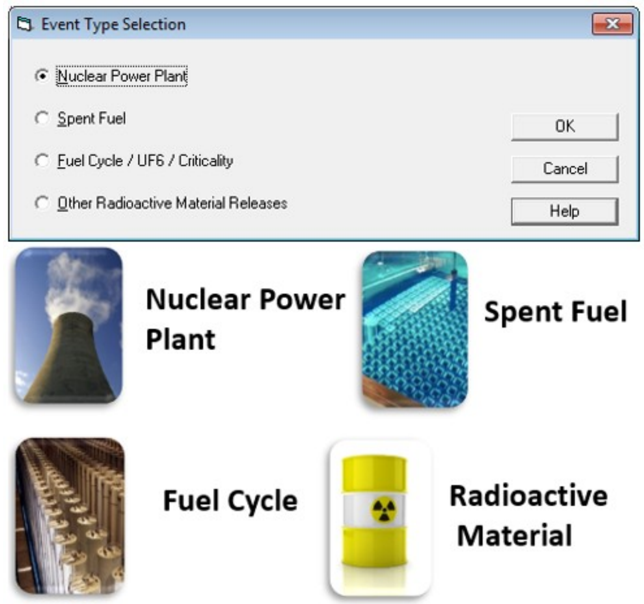

Figure 5.6 RASCAL v4.3.2 Source Term Event Types.

Status

In July 2016, the Office of Nuclear Regulatory Research (RES) released RASCAL version 4.3.2 (update) to resolve coding issues identified by user feedback (see Figure 5.7). Specifically, the update resolves RASCAL 4.3.1 coding issues with the UF6 and coolant release source terms for various fuel cycle and nuclear power plant scenarios. In addition, this update resolved issues with the MetFetch Tool's ability to retrieve and process meteorological data (observations and forecasts) due to changes in the security settings of new servers at the National Weather Service. Finally, the update included the nuclear power plant site-specific data for international nuclear power plant sites from the Radiation Protection Code Analysis and Maintenance Program (RAMP) agreement members such as South Africa.

RASCAL version 4.3.2 is one of the computer codes available through the NRC's RAMP.

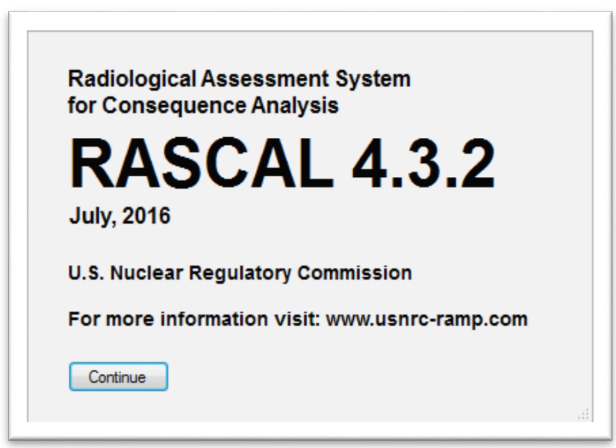

Figure 5.7 RASCAL v4.3.2 Welcome Screen.

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 (EAB), the low-population zone (LPZ), 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 the RADTRAD-Analytical Code (RADTRAD-AC) version 4.5. In addition, the Microsoft Visual Basic graphic user interface (GUI) was replaced with the Symbolic Nuclear Analysis Package (SNAP) GUI (see Figure 5.8). 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 and is referred to as SNAP/RADTRAD version 4.0. Finally, the SNAP framework also allows users to conveniently and easily design production quality plots of numerical data and perform data analysis using the tools in the program AptPlot (see Figure 5.9).

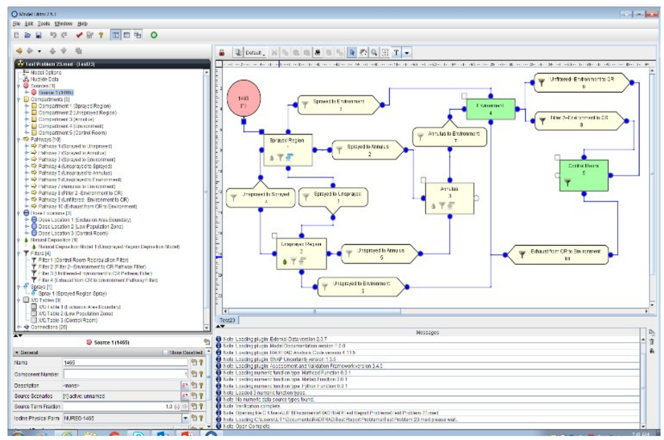

Figure 5.8 Creating RADTRAD input model using SNAP GUI.

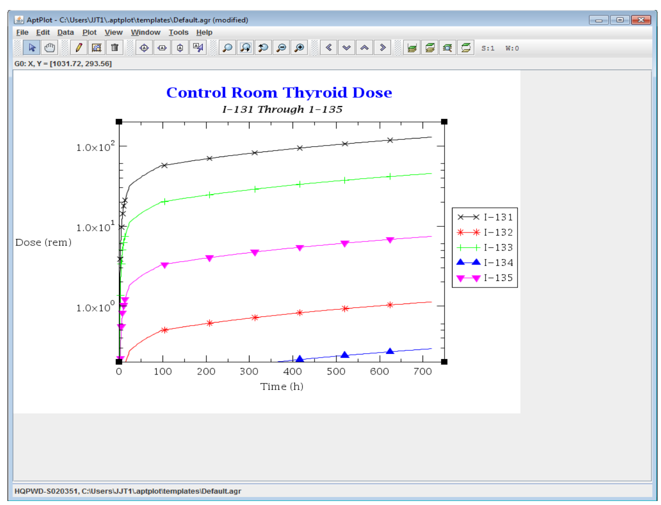

Figure 5.9 SNAP/RADTRAD Control Room Dose Using AptPlot.

Status

The Office of Nuclear Regulatory Research continues to maintain and correct coding errors in SNAP/RADTRAD version 4.0, which consists of the RADTRAD-AC version 4.5.6, the SNAP Model Editor version 2.5.6 with the RADTRAD plug-in version 4.11.5, and AptPlot version 6.7.3. SNAP/RADTRAD version 4.0 is available through the NRC's Radiation Protection Code Analysis and Maintenance Program (RAMP).

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

VARSKIN: Computer Code for Skin Contamination Dosimetry

Objective

The computer code VARSKIN is used to model and calculate skin dose from skin or protective clothing contamination for regulatory requirements under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, “Standards for Protection against Radiation.” The U.S. Nuclear Regulatory Commission (NRC) sponsored the development of the VARSKIN code (see Figure 5.10) to assist licensees in demonstrating compliance with 10 CFR 20.1201(c). NRC inspectors and license reviewers also use VARSKIN to determine compliance of a licensee in licensing actions or in case of a contamination. This regulation requires licensees to have an approved radiation protection program that includes established protocols for calculating and documenting the dose attributable to radioactive contamination of the skin.

Research Approach

VARSKIN has been continually upgraded and improved upon to include five different predefined source configuration simulations of point, disk, cylinder, sphere, and slab sources (see Figure 5.11). The code accounts for air gap and cover materials for photon dosimetry and has been updated to better predict beta dosimetry in shallow skin targets. Although the user can choose any dose-averaging area, the default area for skin dose calculations in VARSKIN is 10 square centimeters to conform to regulatory requirements pursuant to Title 10 of the *Code of Federal Regulations*, Section 20.1201(c).

Status

In January 2017, the Office of Nuclear Regulatory Research (RES) released VARSKIN version 5.3 (update), which addressed errors related to calculated dose for relatively long exposure times from very short-lived nuclides, addressed an off axis angle error when a significant difference existed between the radius of a volumetric source and the radius of the dose-averaging disk, increased the air-gap limit from 5 cm to 20 cm, and applied a new color scheme to VARSKIN to make it easier to read. In January 2018, VARSKIN 6.0 was released with changes to include the ability to use the ICRP 107 database and for the code to automatically include daughter radiations with parent nuclides. Future development for VARSKIN includes adding uncertainty/sensitivity capability and updating the operating platform to an application format so inspectors can use the code in the field.

VARSKIN is one of the computer codes available through the NRC’s Radiation Protection Code Analysis and Maintenance Program (RAMP).

Figure 5.10 VARSKIN ver. 5.3 Welcome.

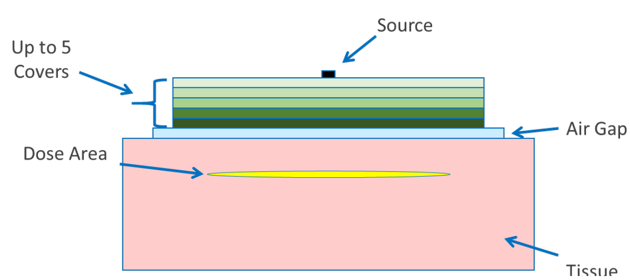

Figure 5.11 VARSKIN Cover Model.

For More Information: Contact Vered Shaffer, RES/DSA, at Vered.Shaffer@nrc.gov.

Phantom with Moving Arms and Legs (PIMAL)

Objective

Modeling scenarios of radiation exposure to the human body, either internal or external, requires an extensive knowledge of fundamental particle physics and complex radionuclide biokinetics. To aid the U.S. Nuclear Regulatory Commission (NRC) staff in developing exposure models and performing the necessary dosimetry calculations for an individual, the Office of Nuclear Regulatory Research (RES) has developed humanoid phantom models (**P**hantom w**I**th **M**oving **A**rms and **L**egs or PIMAL) that are now considered essential tools for radiation dose assessment.

Research Approach

PIMAL has been continually upgraded and improved upon to include a separate male and female stylized phantom with articulated limbs in addition to housing the most recent International Commission on Radiological Protection (ICRP) Publication 110 reference adult male and female voxel phantoms (no articulation). Both internal and external radionuclide sources can be simulated in PIMAL via a dropdown menu in the general user interface. For external sources, the user can select the ICRP's standard external exposure geometries (AP, PA, LLAT, RLAT, or ISO), in addition to a point source, from the menu options. PIMAL 4.1.0 contains an improved user interface. The design of the sliders to control the articulation of the limbs is connected with the textbox input with the bounding conditions of limb articulation included. Source modes (i.e., photon, neutron) for Monte Carlo simulation have been pre-programmed with the source input (photon, neutron, x-ray, radionuclide) to simplify the definition of the radiation source.

Status

PIMAL 4.1.0 software can be employed to adjust the posture of a phantom, generate a corresponding input file for the Monte Carlo N-Particle (MCNP®) radiation transport code, and perform the radiation transport simulations for the dose calculations in MCNP®. The MCNP® code can be run natively from the PIMAL interface or externally in the MCNP® command prompt via the generated MCNP® PIMAL input file. Future development for the PIMAL includes to also couple the code with the international GEANT code instead of MCNP®.

PIMAL is one of the computer codes available through the NRC's Radiation Protection Code Analysis and Maintenance Program (RAMP).

Figure 5.12 PIMAL Phantoms.

For More Information: Contact Vered Shaffer, RES/DSA, at Vered.Shaffer@nrc.gov.

Uranium Milling and Decommissioning Computer Codes

Objective

Licensed milling operations protect public health, safety, and the environment by determining that the total dose to an individual in the public is less than the public dose limit of 100 mrem/y (1 mSv/y). License applicants and U.S. Nuclear Regulatory Commission (NRC) staff use the latest MILDOS-AREA computer code to estimate the radiological impacts of airborne emissions from conventional uranium ore mining and milling and in situ recovery (ISR) facilities.

The NRC has set goals for decommissioning in 10 CFR Part 20, Subpart E, at a fraction of the public dose such that in the case of unrestricted release, public doses attributable to residual contamination after decommissioning do not exceed 25 mrem/y (250 μSv/y).

Research Approach

The NRC continuously updates and improves the codes to ensure they are functional, secure, and compatible with other software. The NRC staff uses feedback from stockholders to plan code development and maintenance.

Status

MILDOS-AREA Computer Code

The Version 4.0 code has the capability to use meteorological data provided by the National Climatic Data Center (Figure 5.13).

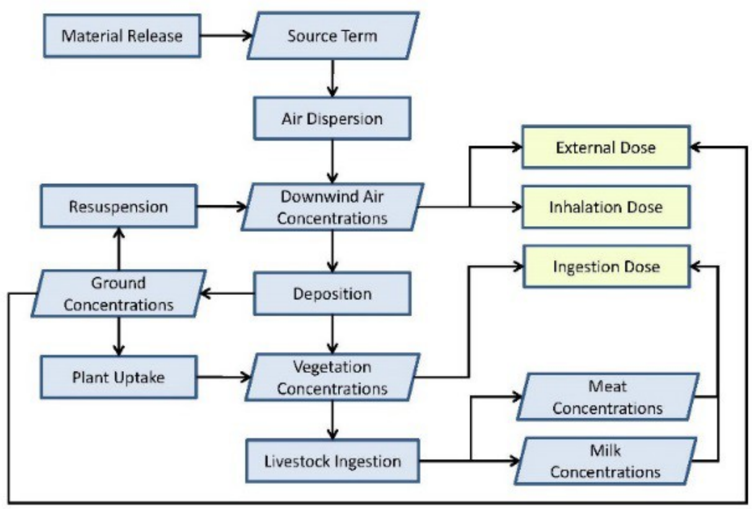

Figure 5.13 MILDOS-AREA Exposure Pathway Calculation.

RESRAD Family of Computer Codes

RESRAD-ONSITE (formerly called RESRAD) is a computer code developed by Argonne National Laboratory for estimating radiation doses and cancer risks to an individual located on top of soils contaminated with radioactive materials. RESRAD-BUILD, RESRAD-OFFSITE, RESRAD-RDD, and RESRAD-BIOTA use pathway analysis to assess radiation exposure and related risks to derive cleanup criteria.

Visual Sampling Plan (VSP) Computer Code

Visual Sample Plan (VSP), developed by Pacific Northwest National Laboratory, is a software tool for choosing the correct number and location of environmental samples in a sampling plan so that the results of statistical tests performed on the data collected have the required confidence for decisionmaking.

Decontamination and Decommissioning (DandD) Code

DandD Version 2.1 allows full probabilistic (i.e., Monte Carlo) treatment of dose assessments and includes a sensitivity analysis module.

For More Information: Contact Tanya Oxenberg, RES/DSA, at Tanya.Oxenberg@nrc.gov.

Ground-Water Monitoring and Remediation at Nuclear Facilities

Objective

Based on presentations made at the 2017 RIC Session TH28, *Ground-Water and Remediation at Operating and Decommissioning NPP Sites* (http://ric.nrc-gateway.gov/docs/abstracts/sessionabstract-39.htm), continued collaboration is needed with Federal and State agencies and industry for assessing onsite abnormal radionuclide releases. The focus is to identify integrated ground-water monitoring and modeling tools to assess the significance of residual radioactivity in the subsurface and to determine the future need, timing, and method for remediation. (Residual radioactivity means radioactivity in structures, materials, soils, ground water, and other media at a site resulting from activities under the licensee's control as specified in 10 CFR 20.1003.) The regulatory issue is to minimize ground-water contamination as specified in 10 CFR 20.1406 (c). The objective of this long-range research study is to identify and evaluate innovative technologies and analysis methods and to develop guidance for monitoring abnormal releases to support remediation decisions.

Research Approach

At operating nuclear facilities, abnormal releases of radioactive contaminants have occurred and were reported to the U.S. Nuclear Regulatory Commission (NRC) per the *NEI 07-07 Ground-Water Protection Initiative*. These reported releases have migrated through the surrounding engineered backfills/structural fills, native soils/bedrock, and into ground-water aquifers. Although most of these releases have been detected prior to migrating offsite, the investigation of their sources involves extensive drilling of boreholes and excavations to locate and characterize the source of the release(s) (e.g., pipe breaks, spent fuel pool and condensate tank leaks). The licensee is required to determine if any offsite discharges to the unrestricted area could exceed 10 CFR Part 20.1302 criteria. At present, the preferred remediation method by the nuclear industry is Monitored Natural Attenuation (MNA).

The research approach is to couple evolving technologies for detecting and surveying radionuclide releases within the nuclear industry's monitoring and modeling programs to assessments of abnormal radionuclide discharges offsite. Regulatory Guide (RG) 4.25 provides a regulatory approach acceptable to the NRC staff for this assessment. RG 4.25 endorses ANSI/ANS 2.17-2010 (R2016), an industry-consensus standard for *Evaluation of Subsurface Radionuclide Transport at Commercial NPPs*. Many of the releases involve complex conceptual site models. These innovative technologies need to be capable of locating the release, identifying the radionuclides, and estimating their concentrations at various downgradient distances to determine if and when more proactive remediation methods are needed to minimize ground-water contamination. For example, the U.S. Department of Energy (DOE) national laboratory scientists shared details on their enhanced MNA studies at the NRC's Regulatory Information Conference (RIC) and in more detail with the NRC staff.

Status

The NRC staff is collaborating with Federal (e.g., DOE, EPA, and DOD), State, and industry through the Federal Remediation Technologies Roundtable (FRTR) to identify and understand the success of these evolving technologies. The NRC staff presented at the EPRI Ground-Water Protection Workshop in September 2017 to further discuss this collaborative venture. The NRC hosted the 2017 Fall Meeting of the FRTR on the subject of "Remediation Technologies for Radionuclide and Heavy Metals in Soil, Ground Water and Sediments." The NRC staff is actively reviewing EPRI's *Groundwater and Soil Remediation Guidelines for Nuclear Power Plants* presented at the 2016 Fall FRTR Meeting.

For More Information: Contact Tom Nicholson, RES/DRA at Thomas.Nicholson@nrc.gov.

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

Objective

The U.S. Nuclear Regulatory Commission (NRC) needs to understand the long-term behavior of the earth materials used as covers to isolate uranium mill tailings waste from the environment. The objective of this study is to determine if changes in soil structure by processes such as wetting, drying, freeze/thaw, and insect and plant intrusion alter hydraulic conductivity and gaseous diffusivity of Radon (Rn) barriers. The study, a collaboration with the U.S. Department of Energy/Legacy Management will determine how soil structure development over time impacts releases of radon and influx of water, and if these changes impact regulatory decisions.

Research Approach

The durability/sustainability of earthen covers used for uranium mill tailings disposal sites will be evaluated with respect to hydraulic flux and radon emissions by measuring a range of cover conditions that include: Rn barrier age, soil structure, soil moisture content, hydraulic conductivity, Lead-210 (Pb-210) profiles, plant rooting depths, vegetation types, and vegetation maturity. Radon barriers at four mill tailings sites are being evaluated: Falls City, TX; Bluewater, NM; Lakeview, OR; and Shirley Basin South, WY. These sites have Rn barriers varying in age, depth, and thickness that are in locations representing a range of vegetation and climates. Field studies were completed at Falls City and Bluewater in which radon fluxes were measured at the top of the Rn barrier and at the top of the waste underlying that barrier using four different size flux chambers. Soil properties were logged in detail, and large (450 mm diameter) block samples were taken for hydraulic conductivity tests. Data from the hydraulic conductivity and other parameters are being be used to construct profiles of hydraulic properties of the Rn barrier as a function of depth. Samples of material from different depths are being measured for Pb-210, a 22-year half-life decay product of Rn-222, to indicate long-term Rn fluxes. The data collected will be used to compare Rn fluxes and water percolation rates for each of the Uranium Mill Tailings Radiation Control Act (UMTRCA) covers relative to the predictions made during design and construction.

Figure 5.14 Radon flux measurement at Falls City.

Figure 5.15 Test Pit at the Bluewater Site.

Status

This project began in 2015 and is anticipated to be completed in 2018. Field work for the final two remaining sites occurred in fall 2017. A paper entitled: *Radon Fluxes from an Earthen Barrier Over Uranium Mill Tailings after Two Decades of Service* was published in the proceedings of the Waste Management 2017 Conference (WM2017). Other papers are being prepared.

For More Information: Contact Mark Fuhrmann, RES/DRA at Mark.Fuhrmann@nrc.gov

Radiation Protection Cooperative Research

Objective

The U.S. Nuclear Regulatory Commission (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, the NRC staff participates in and monitors 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. The 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

The 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 (JCCRER). The NRC is a member of the U.S. delegation to JCCRER 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/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 and operational experience to improve the radiological protection at nuclear power plants. Other national and international outreach includes 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 Minh-Thuy Nguyen, RES/DSA, at Minh-Thuy.Nguyen@nrc.gov.

Modelling and Data for Radiological Impact Assessment (MODARIA)

The NRC staff participates in the IAEA’s Modelling and Data for Radiological Impact Assessment (MODARIA) Program. The program’s mission is to enhance the capabilities of member States to simulate radionuclide transfer in the environment and, thereby, to assess exposure levels of the public and in the environment. The MODARIA II Program comprises seven working groups that cover a wide range of topics. The NRC participates in Working Group 3 – Assessments and Control of Exposure to the Public and Biota for Planned Releases to the Environment. This working group works to ensure an appropriate level of protection from the effects of radiation associated with radionuclide releases from nuclear facilities to adopt an integrated approach to people and biota.

For more information, contact Stephanie Bush-Goddard, RES/DSA, at Stephanie.Bush-Goddard@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?” Another way to assess public safety is to consider risk. According to the traditional definition, risk is the product of the likelihood and consequences of an adverse event. In 1995, the NRC issued a policy statement on the use of probabilistic risk assessment (PRA) encouraging its use in all regulatory matters. 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. The NRC’s PRA 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.” With the development of risk-assessment methods and tools, the NRC can now answer the question, “How likely is it that something will go wrong?” State-of-practice PRAs also incorporate uncertainty analyses to address a fourth question: “How confident are we in our answers to these three questions?” The NRC staff has developed guidance in NUREG-1855, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking,” to address the types of uncertainties reflected in PRAs.

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 added these activities recognizing that PRA methods have evolved to the point where they are useful tools in support of regulatory decisionmaking. PRA methods 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, the acceptability of a PRA and the information derived from a PRA is an important issue. In determining the acceptability of the PRA, the NRC considers the technical adequacy of the PRA, the PRA quality, and whether the PRA is applicable to the regulatory decision under consideration.

Factor	Scoping Options for Operating NPPs
Radiological hazards	Reactor core Spent fuel Other Radioactive Sources
Populatio n 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 and Latent

Figure 6.1 Factors affecting the scope of PRAs for operating NPPs.

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.

The NRC develops and maintains the Systems Analysis Programs for Hands-on Integrated Reliability Evaluation (SAPHIRE) code and SPAR models for performing risk assessments. The NRC uses these tools in its reactor oversight program to perform risk assessments of inspection findings and reactor events to determine their significance for appropriate regulatory response. The NRC also uses these tools to assess the risk of events under the Accident Sequence Precursor (ASP) Program. The ASP Program looks at these events to determine the risk from the event and other conditions that existed at the time of the event. Using insights from the ASP Program, the NRC can monitor changes to industry-wide risk and the effectiveness of the NRC’s oversight and licensing programs.

Full-Scope Site Level 3 Probabilistic Risk Assessment Project

Objectives

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

- Reflect technical advances and plant changes since completion of NUREG-1150 and addresses scope considerations that were not previously considered.
- Extract new risk insights to enhance regulatory decisionmaking.
- Enhance probabilistic risk assessment (PRA) capability, expertise, and documentation.
- 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 where necessary. 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 U.S. Nuclear Regulatory Commission (NRC) tools to perform 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.

Status

The Level 3 PRA project consists of more than a dozen interconnected PRA modeling elements. Completion of each modeling element includes three major activities: development of a draft model, peer review (often in accordance with existing industry consensus standards), and completion of the final model. All major activities have been completed for the reactor, Level 1 PRA for internal events and internal floods. Peer reviews have been completed for the reactor, Level 2 and 3 PRAs for internal events and internal floods, and their final models are expected to be completed in January 2018 and April 2018, respectively. Peer reviews have also been completed for the reactor, Level 1 PRA for high winds, and the screening analysis for other reactor hazards; these models/analyses are expected to be finalized in January 2018. The draft model for the combined Level 1-3 PRA for dry cask storage is expected to be completed in January 2018; the draft models for the reactor, Level 1 PRAs for internal fires and seismic events are both expected to be completed in January 2018; and the draft model for the reactor, low power and shutdown Level 1 PRA for internal events is expected to be completed in February 2018. Additional current work activities include draft models for the Level 2 PRAs for internal fires and seismic events, the draft model for the spent fuel pool PRA, and several pilot applications of a proposed approach for integrated site risk. The full project is expected to be completed in 2020.

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

Probabilistic Risk Assessment Use and Standards

Objective

The objective of this activity is to develop and maintain the framework for the NRC's use of probabilistic risk assessment (PRA) methodologies in regulatory decisionmaking for the oversight and licensing of nuclear power plants.

Research Approach

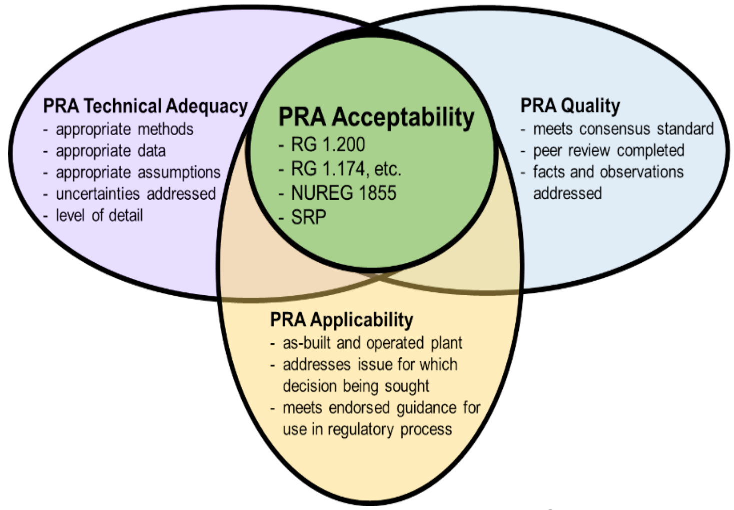

Figure 6.2 The PRA Acceptability Concept.

Under the U.S. Nuclear Regulatory Commission's (NRC's) PRA Policy Statement (60 FR 42622; August 16, 1995), the Commission shared its position on the expanded uses of PRA. This includes, among other elements, the increased use of PRA in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data. To implement this policy, the staff develops and issues guidance on the use of PRA methodologies in regulatory decisionmaking. In developing this guidance, the staff focuses on assuring the PRAs used are acceptable to support a determination that reasonable assurance exists that the risk associated with the proposed action is consistent with the NRC's mandate to protect the public and the environment. Concepts that undergird the ability of a PRA to support regulatory decisionmaking include "PRA Quality," "PRA Technical Adequacy," and "PRA Applicability." "PRA Acceptability" for risk-informed regulatory decisions embodies the concepts of "PRA Technical Adequacy," "PRA Quality," and "PRA Applicability" in that satisfying each of these concepts is necessary to conclude that the PRA is acceptable for use in taking a specific regulatory action or decision.

In RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," the NRC provides guidance on what constitutes an acceptable base PRA. This guidance describes an approach licensees can follow to assure appropriate technical adequacy and quality are included in the PRA to support risk-informed decisionmaking. The NRC also has guidance to determine whether a PRA applies to the intended use (for example, RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." Moreover, the NRC has developed the technical basis and an acceptable approach for treating uncertainty in risk-informed decisionmaking (NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making"). This guidance is central to the NRC staff's determination of whether a PRA is acceptable for use in making the requested risk-informed regulatory decision.

Status

Revision 2 to RG 1.200 endorses ASME/ANS RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications." RG 1.200, Revision 2 also endorses guidance in a number of Nuclear Energy Institute (NEI) peer review documents. Revision 3 to RG 1.200 is under development in parallel with the development and trial use of updated PRA standards and other industry guidance. The NRC published Revision 1 to NUREG-1855 in March 2017 that clarified the approach for the treatment of uncertainty in risk applications. RG 1.174 is currently under revision to clarify the consideration of the defense-in-depth philosophy in risk-informed decisionmaking. The NRC expects to publish Revision 3 to RG 1.174 by March 2018.

For More Information: Contact Mary Drouin, RES/DRA, at Mary.Drouin@nrc.gov.

Treatment of PRA Uncertainties in Risk-Informed Decisionmaking

Objective

The objective of this activity is to provide guidance on how to treat uncertainties in probabilistic risk assessments (PRAs) used to support risk-informed regulatory decisions. This guidance 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. Both NRC staff and the industry can use this guidance.

Research Approach

NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking," provides NRC guidance on the treatment of uncertainties. NUREG-1855 describes an approach the NRC finds acceptable for how licensees or applicants address PRA uncertainties in the context of risk-informed regulatory decisions. It also provides guidance on how the NRC evaluates the impact of those uncertainties. 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 guidance developed by the NRC and EPRI 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 by the licensee or applicant, and how the NRC 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) relates to the random nature of events or phenomena, and increasing the knowledge of the modelled systems does not reduce it. 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. In the first stage (Stage A), the risk-informed activity and associated risk analysis are assessed to determine if NUREG-1855 should be the basis for the treatment of uncertainties. Stage A provides guidance on 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. These stages also include the evaluation of the completeness uncertainties and determining if bounding analyses are acceptable for the missing scope items. Also provided is guidance on 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. Stage G includes guidance for 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

The NRC published Revision 1 to NUREG-1855 in March 2017.

For More Information: Contact Mary Drouin, RES/DRA, at Mary.Drouin@nrc.gov.

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 (ASP) used to assess the risk significance of operational events and conditions.
- Generic Safety Issues screening and prioritization.
- Special system and component studies.

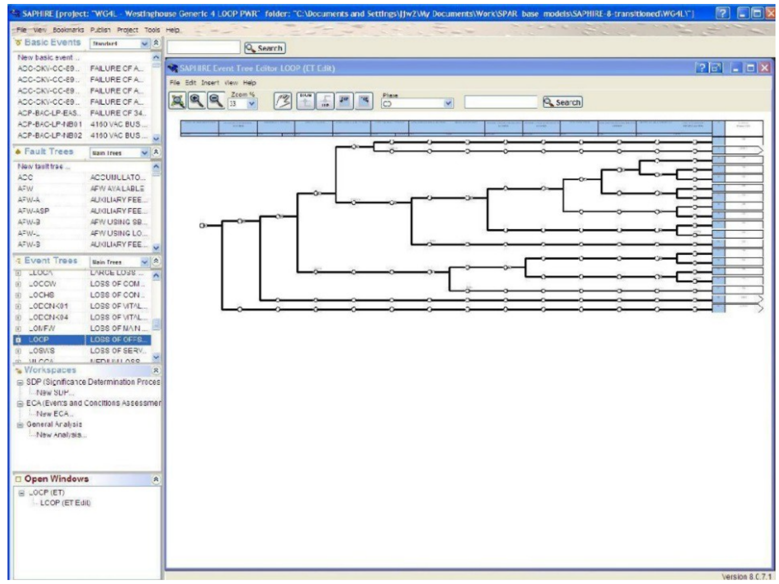

Figure 6.3 Example of loss of offsite power SPAR model event tree display with SAPHIRE.

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. The models also 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, 76 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. In addition, five SPAR models represent new reactors designs (e.g., AP1000 ABWR, [GE and Toshiba], APWR and EPR). The SPAR models include core damage risk resulting from internal events (e.g., general transients, 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 staff has recently updated all SPAR models to reflect the most recent reliability data. For new reactor designs, the staff has been working on expanding the AP1000 SPAR model capabilities (e.g., shutdown and Level 2 model) and is initiating work on plant-specific SPAR models for Vogtle (AP1000).

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

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

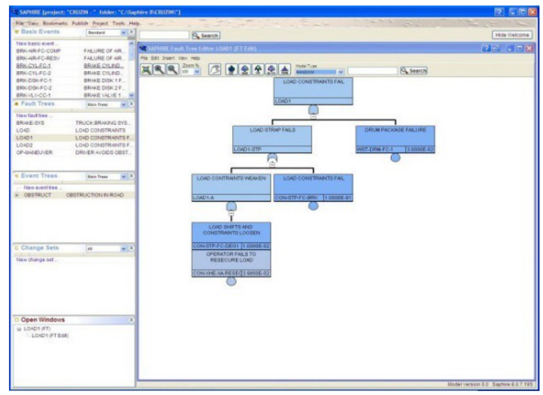

Figure 6.4 A graphical representation of a simple fault tree.

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, planning is 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: Contact Jeffery Wood, RES/DRA, at Jeffery.Wood@nrc.gov.

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 provide off-the-shelf thermal-hydraulic analyses for NRC risk analysts to consult.
- To enhance in-house expertise and knowledge transfer for the purpose of improving the ability to support 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.
- Steam generator tube rupture—time available for operators to mitigate the accident before core damage.
- Station blackout—time available to recover power.

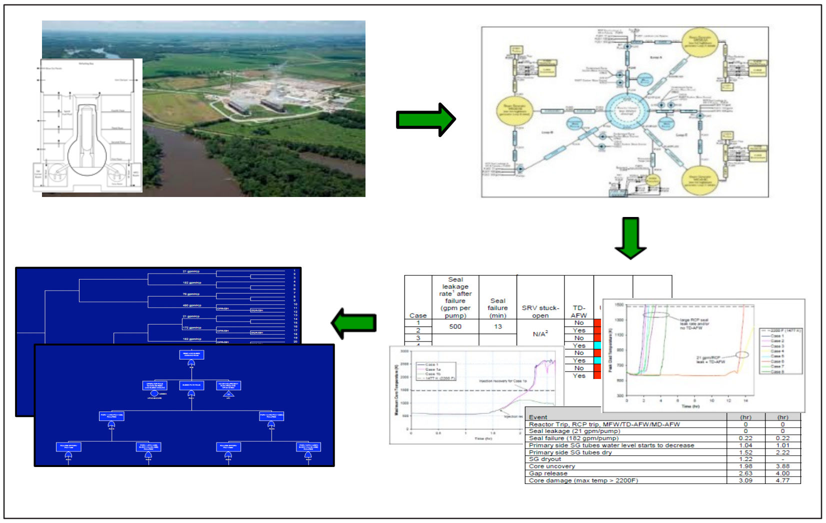

Figure 6.5 Level 1 Success Criteria Analysis.

Analysis for the Surry and Peach Bottom stations can be found in NUREG-1953 including medium- and large-break LOCAs and inadvertent open relief valves. Analysis for the Byron station can be found in NUREG-2187 including small- and medium-break loss-of-coolant accidents, loss of a direct current bus, and loss of decay heat removal during shutdown operations.
In addition, NUREG/CR-7177 investigated modeling assumptions that affect success criteria findings.

Status

Ongoing activities include:

- Analysis for the Duane Arnold Energy station including depressurization during non-ATWS scenarios, ECCS injection following containment failure or venting, safe and stable end-state considerations, and mitigating strategies for loss of ac power. The final NUREG is expected to be issued in 2018.
- Analysis for the Vogtle station (Units 1 and 2) for a mix of issues important to the Vogtle Level 3 PRA project (SECY-11-0089) that will be documented in project documents to be issued at the completion of the Level 3 PRA project.

For More Information: Contact Suzanne Dennis, RES/DRA, at Suzanne.Dennis@nrc.gov.

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 of 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 been 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).

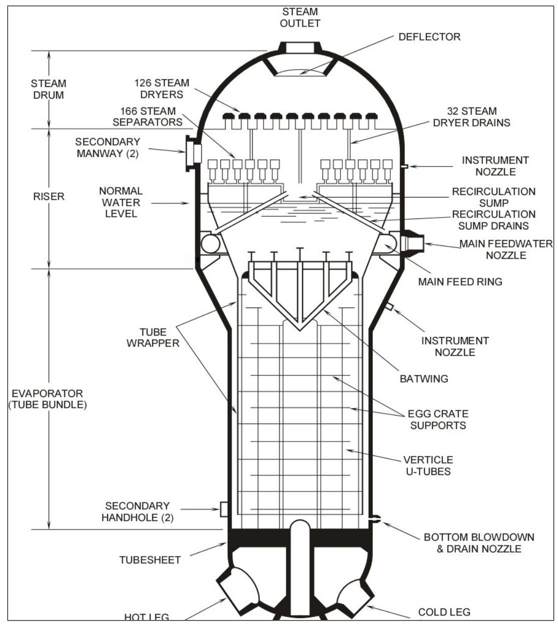

Figure 6.6 Combustion Engineering Steam Generator.

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 pressurized-water reactor (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 screening approach, this method was developed in a manner such that it can be used to 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 decision-making process and also be used to update the PRA standards and guidance.

Status

A draft NUREG-2195 was issued in 2016 for public comments. Comments received from the public, ACRS members, and other NRC offices were addressed. The final NUREG-2195 is expected to be published in early 2018.

For More Information: Contact Selim Sancaktar, RES/DRA, at Selim.Sancaktar@nrc.gov.

Risk Analysis Cooperative Research

Objective

The U.S. Nuclear Regulatory Commission's (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 NRC's safety mission and increasing use of risk-informed regulatory decisionmaking. RES has developed numerous cooperative partnerships to ensure the most knowledgeable researchers performing risk analyses use high-quality tools and data in a cost-effective manner. These cooperative partnerships are with 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 its research programs take advantage of state-of-the-art research results and ensures efficient use of its resources by avoiding overlapping research programs.

Research Approach

Domestically, RES participates with consensus standards development organizations (SDOs) such as the American Society of Mechanical Engineers and the American Nuclear Society to promote the use of consistent guidelines for building and maintaining nuclear power plant PRA models. The NRC staff participates in PRA standard working and writing groups. In addition, NRC has provided grants to SDOs so that recognized independent experts can support consensus standard development. The NRC often endorses these standards in regulatory guidance documents to support risk-informed regulatory decisions.

RES maintains Memoranda of Understanding (MOU) with the Electric Power Research Institute (EPRI) and the National Aeronautics and Space Administration (NASA). The objective of the NRC-EPRI MOU is avoiding unnecessary duplication of effort by sharing of information related to research programs of mutual interest. The NRC works with EPRI whenever such cooperation and cost sharing is appropriate (e.g., does not represent a conflict of interest or compromise the NRC's independence). Examples of cooperation with EPRI include the development of PRA modeling approaches for support system initiating events and offsite power and the development of guidance to address uncertainties. The NASA-NRC MOU supports the development of advanced risk analysis techniques and tools to support risk-informed decisionmaking. Examples of collaboration with NASA include sharing 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, the University of California, Los Angeles; the Ohio State University; and the University of Maryland received grants. These grants support, for example, research in the development of methodological and software enhancements of dynamic PRA platforms, modeling tools for regulatory applications, and automated reliability prediction system software assessment tool.

In the international arena, RES participates on the Organization for Economic Cooperation and Development/Nuclear Energy Agency (NEA) Committee for the Safety of Nuclear Installations Working Group on Risk Assessment (WGRISK). International cooperation through WGRISK fosters continual improvement in the application of risk assessment methods by NEA member countries to improve the safety of nuclear installations.

Status

By engaging in productive cooperative research partnerships, RES takes advantage of state-of-the-art domestic and international research results while efficiently targeting specific research needs. This enables the effective development and maintenance of state-of-the-art methods, tools, data, and technical information in support of the NRC's safety mission.

For More Information: Contact Kevin Coyne, RES/DRA, at Kevin.Coyne@nrc.gov.

Chapter 7: Human Reliability Research

Consistent with the U.S. Nuclear Regulatory Commission's (NRC's) policy statement on the use of probabilistic risk assessment (PRA) and staff requirements memoranda (SRMs) for achieving an appropriate PRA capability for NRC risk-informed regulatory decisionmaking, the NRC has established a phased approach to PRA capability. This phased approach includes an action plan for stabilizing the PRA capability 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. In addition, RES worked with the Electric Power Research Institute to improve HRA methods by developing the Integrated Human Event Analysis System (IDHEAS) HRA Methodology.

Outside of 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]) to collect licensed operator simulator training data to inform human error probability (HEP) estimations in human reliability analysis (HRA). The objective is to acquire a large quantity of licensed operator simulator exercise data to provide statistical indications of the reliability of performing tasks in 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. 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 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 data usability.

A successful data collection program should include high data reliability and a long collection period to acquire enough data for statistical indications. To achieve 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. This type of data includes the operators' performance in meeting the expectations and, if there are performance deficiencies, it includes the information related to the performance deficiency. This data is entered by the plant operating crew during post simulation debriefings to ensure data reliability. For both types of data, the master set of factors is 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.

Finally, to achieve the objective of a long-term data collection, SACADA was designed to mutually benefit the data providers and the NRC. The data providers are the plants' training department and the operations department, and their main interest is to improve human performance. The SACADA tool allows for the plants to replace their current tools in collecting operator simulator performance information. When used to replace the plants' existing tools, SACADA has streamlined data entry that, in turn, has reduced data entry effort for other plant training applications.

Status

SACADA has been collecting operator training data from a U.S. nuclear power station since 2012. The Halden Reactor Project is also using the tool to collect information from operator simulator experiments. A few international research institutes have signed agreements with the NRC to test the SACADA tool, and several U.S. and International utilities are evaluating SACADA for use at their facilities. So far, the SACADA database has collected enough data for demonstrations on how to use the data to inform HRA.

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

Human Reliability Analysis Methods

Objective

This work addresses the issues 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 methodology that can be used to build HRA models for general applications (i.e., external events, shutdown, level-3 PRA, and non-reactor applications). 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. The 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 outcome of the literature review and the cognitive basis framework for HRA were documented in NUREG-2114.

The NRC staff worked with the Electric Power Research Institute (EPRI) under a memorandum of understanding to develop a stand-alone HRA method that reduces analyst-to-analyst variability for internal, at-power scenarios. The method, IDHEAS for Nuclear Power Plant Internal Events at-Power Applications (NUREG-2199, Vol. 1), integrates the strengths of existing methods and addresses key sources contributing to analyst-to-analyst variability. The project team addressed 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.
- Verifying the quantification model and estimating the base Human Error Probabilities (HEPs) through an expert panel that consists of human factors/cognitive engineers, PRA/HRA analysts, and operational personal from U.S. NPPs.

The NRC staff is also developing a version of the IDHEAS method to be used to create HRA methods for general HRA applications. The general 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 conditions and can be used for both reactor and non-reactor applications.

Status

The staff published the cognitive framework report, NUREG-2114, in January 2016 and the IDHEAS method for at-power applications, NUREG-2199, Vol. 1 in May 2016. The staff is currently engaged in testing the method and developing a simplified IDHEAS method for events and condition assessment in support of the NRC's Significant Determination Process and Accident Sequence Precursor program.

For More Information: Contact Jing Xing, RES/DRA, at Jing.Xing@nrc.gov.

Chapter 8: Human Factors Research

Humans are integral to the safe operation of a nuclear power plant (NPP). In the late 1970s, the U.S. Nuclear Regulatory Commission (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 NRC's Office of Nuclear Regulatory Research (RES) continues to look at these and other factors affecting human and organizational performance.

Figure 8.1 Human-System Interface in the Control Room.

The next generation of NPPs 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). For instance, old analog control panels (Figure 8.1) are being replaced by computer-based HSI. The new technologies in NPPs will bring about a host of other changes including new tools to support plant personnel, adjustments to plant staffing configurations, different concepts of operation, and maintenance that is different from currently operating NPPs. 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 may be presented with a different job that includes supervising the automation. The potential benefits 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 impacts of these developments be evaluated and understood by prospective operators and regulators responsible for determining the acceptability of new designs to support human performance and maintain plant safety.

The ongoing maintenance of NPP structures, systems, and components is also critical to continued safe operation. One means of inspecting the integrity of components is through nondestructive examination (NDE). Although rigorous qualification processes exist for NDE equipment, procedures, and personnel, NDE reliability can be affected by a host of human factors related to the design of the task, the people performing the task, and environmental and organizational conditions where the task is performed. RES supports the regulatory offices by identifying how these human factors may influence performance.

All personnel involved in operating an NPP play a role in ensuring safe operation from senior leadership to licensed operators and contract workers. The NRC requires licensees to have a fitness-for-duty (FFD) program to provide reasonable assurance that personnel are able to safely and competently perform their duties. The NRC also expects that licensees will establish and maintain a positive nuclear safety culture by emphasizing safety over competing goals. RES maintains technical expertise to ensure that guidance in these areas is informed by advances in research.

RES supports these activities through various research projects related to human performance in control rooms, human factors in NDE, fitness-for-duty programs, and safety culture. Lastly, RES participates in cooperative research on human factors with international partners through the Working Group on Human and Organisational Factors (WGHOF) of the Nuclear Energy Agency and the Halden Reactor Project.

Human Performance for New and Advanced Control Room Designs

Objective

The objective of this work is to identify and prioritize human performance research that will be needed to support technical basis development and regulatory guidance for review of licensees' implementation of new technology in new and advanced nuclear power plants (NPPs).

Research Approach

The U.S. Nuclear Regulatory Commission's (NRC's) Office of Nuclear Regulatory Research (RES) began this research effort by evaluating current industry trends and organizing them into seven human factors engineering (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, and (7) HFE methods and tools. Next, a panel of independent subject-matter experts representing various disciplines and backgrounds prioritized the areas. NUREG/CR-6947, "Human Factors Considerations with Respect to Emerging Technology in Nuclear Power Plants," documents the results of the study. The findings from the study are being used to develop long-term research plans addressing human performance within these technology areas as follows:

Advances in Human Factors Engineering Methods and Tools

This project has resulted in 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

This study examines the impact of automation on control room design, specifically: (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 3, "Human Factors Engineering Program Review Model," and NUREG-0700, Revision 2, "Human-System Interface Design Review Guidelines," as the acceptance criteria. The guidance in NUREG-0711 and NUREG-0700 is being updated to keep pace with modern I&C systems and advanced levels of automation that will be found in the next generation of NPP control rooms.

Status

In addition to NUREG/CR-6947 mentioned above, two additional reports are under development: (1) Integrated System Validation, and (2) Cognitive Task Analysis. Two technical reports are currently available related to automation and complexity: (1) Human-System Interfaces to Automatic Systems, and (2) The Effects of Degraded Digital Instrumentation and Control Systems on Human-system Interfaces and Operator Performance. A revision to NUREG-0711 was published in 2012. Due to its size, NUREG-0700 is being updated in two phases. The phase 1 update is complete and is expected to be published in 2018, 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, U.S. Nuclear Regulatory Commission's (NRC's) Office of Nuclear Regulatory Research (RES) 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 (Figure 8.2). 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, technical specifications).

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.

Figure 8.2 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 Niav Hughes, RES/DRA, at Niav.Hughes@nrc.gov.

Human Factors in Nondestructive Examination

Objective

The objective of this work is to support the U.S. Nuclear Regulatory Commission (NRC) regulatory offices by conducting research to identify, understand, and prioritize the human factors that are most likely to impact personnel performance during nondestructive examination (NDE) in nuclear power plants.

Research Approach

NDE is a means of testing a specimen or component without damaging or destroying it (Figure 8.3). NDE plays a vital role in ensuring the safety of nuclear power plant operations. The effective use of NDE to find flaws in a component can be dependent on the personnel performing the examination, the design of the task, along with the environmental and organizational conditions within which personnel carry out the task. These human factors issues must be considered to have reasonable assurance that a licensee is meeting the regulatory requirements of the NRC for quality assurance in Appendix B to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.

The NRC's Office of Nuclear Regulatory Research (RES) is currently focused on understanding the human factors that can challenge performance in a specific type of NDE called ultrasonic testing (UT). The first stage of the research is to characterize the current state of human factors research in NDE by conducting a literature review. Next, RES staff plans to conduct a task analysis to develop a thorough understanding of the process of performing NDE in nuclear power plants. Then, the RES staff plans to work with industry to prioritize the human factors identified. This research effort will assist the NRC in planning future research, evaluating whether changes are needed in regulatory requirements, and providing technical justification for regulatory decisions.

Figure 8.3 Examination of Pipe Weld Using Ultrasonic Testing.

Status

The RES staff completed the literature review of human factors considerations in NDE in February 2017. The technical letter report titled, "Review of Human Factors Research in Nondestructive Examination," is available in the Agencywide Document and Management System at accession number ML17059D745. The task analysis is expected to be completed in early 2018, and the prioritization of human factors issues is planned for 2018.

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

Fitness for Duty

Objective

The objective of this work is to support the U.S. Nuclear Regulatory Commission (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 requires certain licensees to have a Fitness for Duty (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 NRC’s Office of Nuclear Regulatory Research (RES) participates in and provides technical support to several working groups engaged in FFD rulemakings and program implementation. Two main initiatives related to 10 CFR Part 26 are described below.

Fatigue

RES has been working to resolve issues related to the 10 CFR Part 26 rulemaking on work hour controls for specific workers at nuclear power plants. In conjunction with the rulemaking, RES is developing guidance for implementing the fatigue management requirements, and RES has been looking at new methods to manage fatigue in the workplace and technologies for 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 10 CFR 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 continues providing support to the regulatory offices in answering implementation questions from inspectors and licensees regarding the NRC's requirements for managing nuclear power plant worker fatigue. The staff is currently working on an update to Regulatory Guide 5.73, "Fatigue Management for Nuclear Power Plant Personnel," that will incorporate implementation guidance based on these questions and lessons learned since the fatigue requirements went into effect in 2011.

For More Information: Contact Valerie Barnes, RES/DRA, at Valerie.Barnes@nrc.gov or Lawrence Criscione, RES/DRA, at Lawrence.Criscione@nrc.gov.

Safety Culture

Objective

The objective of this work is to provide technical expertise related to human and organizational performance to support the U.S. Nuclear Regulatory Commission's (NRC's) safety culture oversight and policymaking 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 safety performance. The NRC has long recognized the importance of maintaining a positive safety culture in nuclear operations. The NRC's Safety Culture Policy Statement (76 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.

The NRC's Office of Nuclear Regulatory Research (RES) provides ongoing technical support for safety culture activities across the NRC. Past projects have included 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. Most recently, the RES staff has been working with the International Atomic Energy Agency (IAEA) to develop safety culture perception questionnaires for use by nuclear power plant licensees and regulatory agencies as one tool in an overall safety culture assessment. The 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 at accession number ML12172A093.

The manual for the IAEA safety culture perception questionnaire for license holders is publicly available as a working document on the IAEA's Web site. The manual includes a copy of the questionnaire and offers high-level guidance for how to conduct a safety culture survey. The questionnaire is currently used by the IAEA to support their independent safety culture assessment reviews.

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.

Human Factors Cooperative Research

Objective

As part of its ongoing participation with international partners, the U.S. Nuclear Regulatory Commission (NRC) participates with and supports the Working Group on Human and Organizational Factors (WGHOF). The objective of this work is to ensure that activities remain aligned with NRC research goals and priorities and address the Organization for Economic Co-operation and Development (OECD), Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI) strategic priorities, safety issues, and topics.

Research Approach

The WGHOF is a working group under the NEA that focuses on human and organizational factors affecting safety at nuclear facilities. 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. 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 accident at the Fukushima Dai-ichi Nuclear Power Plant in Japan.

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 and actual fire events demonstrate that fire can be a significant contributor to nuclear power plant (NPP) risk. In particular, these studies and events show that failures of fire protection defense-in-depth features can lead to risk-significant conditions. Fire protection programs in U.S. NPPs use this concept of defense-in-depth to achieve the required degree of fire safety by using echelons of protection from fire effects. The three echelons 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 U.S. Nuclear Regulatory Commission (NRC). Through this state-of-the-art program, RES supports other NRC offices by developing and validating fire analysis methodologies and supporting data. These methodologies, which include fire probabilistic risk assessment, fire human reliability analysis, and mathematical fire modeling, 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.

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. The 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 Testing of Electrical Components.

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. RES staff are drafting a Fire Research Plan that will encompass projects that focus on specific areas intended to address emerging regulatory needs, advance realism in fire probabilistic risk assessments (PRAs), and support improving and maintaining the knowledge and tools needed to support regulatory oversight activities.

In addition, the 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 on High Energy Arcing Fires (refer to Figure 9.1, a picture captured of the testing) with the Organization for Economic Cooperation and Development (OECD) 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) embarked on a cooperative project to improve the state-of-the-art in fire risk assessment to support this new risk-informed environment in fire protection. This project produced a consensus 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, with participation by 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.

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.

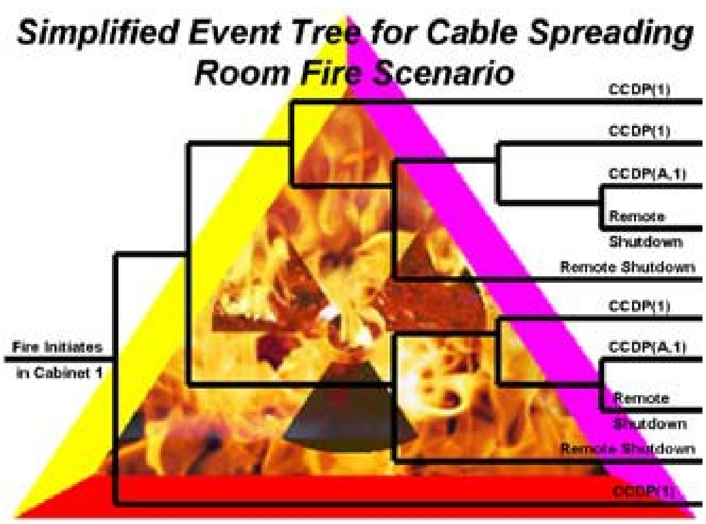

Figure 9.2 Simplified fire PRA event tree representing different sets of fire damage and plant response.

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.

Current research is focused on expanding initial guidance that has been developed in NUREG-1921/EPRI 1023001, “EPRI/NRC-RES Fire Human Reliability Analysis Guidelines—Final Report,” to address main control room abandonment (MCRA) scenarios. Such expanded guidance is expected to improve realism for fire HRA/PRA. In addition, this research will address issues related to command and control and local actions, which are important to other PRA hazards, as well as fire PRA.

Research Approach

The Office of Nuclear Regulatory Research (RES) has worked collaboratively with the Electric Power Research Institute (EPRI) to develop a methodology and associated guidance for performing HRA in support of fire PRA. In July 2012, the U.S. Nuclear Regulatory Commission (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). To address this need, NRC-RES and EPRI began working collaboratively in early 2015 to develop additional guidance for both loss of habitability (LOH) and loss of control (LOC) scenarios that result in MCRA. This guidance builds upon that already provided in the joint EPRI/NRC-RES Fire Human Reliability Analysis Guidelines and interactions between NRC and industry in the Frequently Asked Questions (FAQ) process. The updated guidance will be in the form of a NUREG report (or reports) and issued as a supplement(s) to NUREG-1921.

Figure 9.3 Reactor Operators in a nuclear power plant main control room.

Status

RES and EPRI have begun development of additional HRA guidance for MCR abandonment scenarios. Currently, the first research product, *EPRI/NRC-RES Fire Human Reliability Analysis Guidelines: Qualitative Analysis for Main Control Room Abandonment Scenarios (NUREG-1921, Supplement 1/EPRI 3002009215)*, is scheduled for publication in 2018. Follow-on work to develop HRA quantification guidance has begun and is expected to be published in 2018, including peer review and testing. 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: Contact Susan E. Cooper, RES/DRA, at Susan.Cooper@nrc.gov or Tammie Rivera, RES/DRA, at Tammie.Rivera@nrc.gov.

Fire Modeling Activities

Objective

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

Research Approach

In 2007, the U.S. Nuclear Regulatory Commission's (NRC's) Office of Nuclear Regulatory Research (RES), the Electric Power Research Institute (EPRI), and the National Institute of Standards and Technology (NIST) completed an extensive verification and validation (V&V) study of fire models used to analyze NPP fire scenarios. This study resulted in the seven-volume report NUREG-1824, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications." 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.

Subsequent to the publication of NUREG-1824, the NRC conducted 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. 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, "Nuclear Power Plant Fire Modeling Analysis Guidelines (NPP FIRE MAG)," issued November 2012, supplements NUREG-1824 by providing users with best practices from experts in fire modeling and NPP fire safety. 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. Recently, a supplement to NUREG-1824 that evaluates the latest versions of the fire models and incorporated additional test data was published.

To better quantify the heat release rate (HRR) and burning behavior of electrical enclosures, the NRC conducted the Heat Release Rates from Electrical Enclosure Fires (HELEN-FIRE), NUREG/CR-7179 project with NIST. NIST conducted 112 full-scale tests using eight electrical enclosures, acquired from Bellefonte Nuclear Generating Station and configured with various amount of electrical cable to represent typical NPP electrical enclosures. Subsequent to the completion of the HELEN-FIRE test program, the NRC and EPRI initiated the Refining and Characterizing Heat Release Rates from Electrical Enclosures during Fire (RACHELLE-FIRE), NUREG-2178 program using a working group of experienced fire protection and risk assessment researchers and practitioners. 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 Heat Release Rate 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.

Status

Currently, the NRC is working with EPRI to develop additional improvements in the tools, methods, and data used for fire modeling to support risk-informed applications. A RACHELLE-FIRE 2 working group is developing new heat release rate distributions for pumps and motors, better guidance for determining electrical cabinet to cabinet fire spread, improved timing for fire growth profiles, and extensions to the obstructed plume ZOI methodology. Another joint NRC-industry working group is conducting research to support improved guidance for analyzing transient fires in NPP applications.

For More Information: Contact David Stroup, RES/DRA, at David.Stroup@nrc.gov.

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 U.S. Nuclear Regulatory Commission (NRC) uses 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. 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. Also, 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. The results of this work are documented 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 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

The third phase of the CHRISTIFIRE project is intended to identify minimum criteria necessary for cable ignition. In addition, the use of cable tray tops and bottoms and other techniques for limiting cable tray ignition and flame spread will be examined, and recommendations for credit in fire PRAs will be developed.

For More Information: Contact David Stroup, RES/DRA, at David.Stroup@nrc.gov.

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

Objective

The U.S. Nuclear Regulatory Commission (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 a series of fire tests to better understand the effects of fire on cables and circuits used to power and control safety systems. The testing programs included:

- CAROLFIRE [NUREG/CR-6931]
- DESIREE-FIRE [NUREG/CR-7100]
- KATE FIRE [NUREG/CR-7102]

Figure 9.4 Horizontal electrical cable circuit integrity test.

All available data was later reviewed and analyzed by a team of engineers to graphically identify trends as documented in NUREG-2128, "Electrical Cable Test Results and Analysis during Fire Exposure (ELECTRAFIRE)," issued in February 2013. The analysis also shows that multiple cable shorts to ground can cause spurious operations resulting from an ungrounded and compatible power supply. This information supported multiple groups of experts to better characterize the probability of occurrence and duration of spurious actuations for both deterministic and performance-based applications. These expert panels are commonly referred to as JACQUE-FIRE. This 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.

The first panel comprised several electrical engineering experts who reviewed all currently available testing data and NUREG-2128. This panel followed the NRC's phenomena identification and ranking table (PIRT) process to identify the phenomena that influences the likelihood and duration of hot short-inducted cable failures when exposed to fire conditions. The results of this work are documented in NUREG/CR-7150, Vol. 1, "Phenomena Identification and Ranking Table (PIRT) Exercise for Nuclear Power Plant Fire-Induced Electrical Circuit Failure," issued October 2012. The second panel represented by a group of fire probabilistic risk assessment (PRA) experts used the findings from the first panel, available data, and their judgement to develop numerical estimates for the likelihood and duration of fire-induced spurious operations. This effort is documented in NUREG/CR-7150, Vol. 2, "Expert Elicitation Exercise for Nuclear Power Plant Fire-Induced Electrical Circuit Failure."

Status

The use of expert opinion has advanced the state-of-knowledge in this area since the last investigation that occurred in the early 2000s. This detailed evaluation has identified areas where additional research are needed due to a lack of understanding and/or data. These areas for future research 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).

For More Information: Contact Gabriel Taylor, RES/DRA at Gabriel.Taylor@nrc.gov.

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.

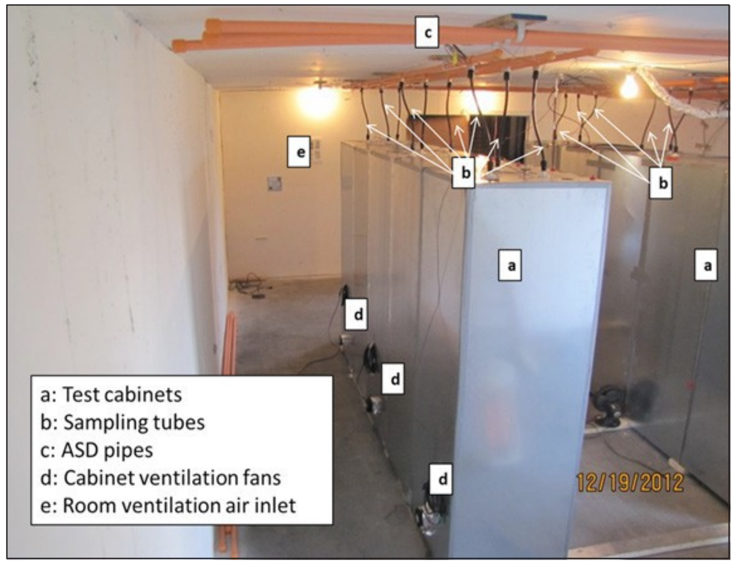

Figure 9.5 Fire test room configuration.

Research Approach

The U.S. Nuclear Regulatory Commission (NRC) staff sponsored testing, conducted literature reviews, and visited 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.

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 this scenario, human factors and human reliability engineering have supported this project and have provided guidance in the final NUREG report concerning system design and estimates on the human failure probability of preventing fire damage.

Status

A final report NUREG-2180, “Determining the Effectiveness, Limitations, and Operator Response for Very Early Warning Fire Detection Systems in Nuclear Facilities (DELORES-VEWFIRE),” was published in December 2016.

For More Information: Contact Gabriel Taylor, RES/DRA, at Gabriel.Taylor@nrc.gov.

International Testing Program for High Energy Arcing Faults (HEAF) Phase 2

Objective

The primary objective of this project is to perform experiments to obtain data on the high-energy arcing fault (HEAF) phenomenon known to occur in nuclear power plants (NPPs) through carefully designed experiments. This phase of testing will explore the influence of aluminum materials on HEAF damage states. The goal is to use the data from these experiments and past actual NPP events to develop a mechanistic model to account for the failure modes and consequence portions of HEAFs.

Research Approach

In June 2013, an Organisation for Economic Co-operation and Development/Nuclear Energy Agency (OECD/NEA) report on international fire operating experience documented 48 HEAF events, accounting for about 10 percent of the total fire events reported. These HEAF events are often accompanied by loss of essential power and complicated shutdowns. To confirm the probabilistic risk assessment (PRA) methodology for HEAF analysis in NUREG/CR-6850 "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities Vol. 2," the NRC led an international experimental campaign from 2014 to 2016. The results identified a previously unidentified failure mechanism posed by aluminum components in or near electrical equipment as well as the potential for unanalyzed equipment failure mechanisms. The phase 1 test report can be downloaded at: https://www.oecd-nea.org/nsd/docs/2017/csni-r2017-7.pdf.

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. The project is being performed as part of a larger international OECD/NEA effort. The U.S. Nuclear Regulatory Commission (NRC) will be leading the physical testing and instrumentation of equipment with support from the National Institute of Standards and Technology (NIST) 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

In February of 2017, a phenomena identification and ranking table (PIRT) exercise was held with participants from the member countries of the first experimental program. The PIRT identified and prioritized the phenomena that require additional study in the upcoming test program. The second phase of testing is scheduled to begin in 2018.

For More Information: Contact:
Nicholas Melly, RES/DRA, at Nicholas.Melly@nrc.gov
Gabriel Taylor, RES/DRA, at Gabriel.Taylor@nrc.gov
David Stroup, RES/DRA, at David.Stroup@nrc.gov or
Kenneth Hamburger, RES/DRA, at
Kenneth.Hamburger@nrc.gov.

Figure 9.6 High Energy Arc Fault Testing of Electrical Components.

Evaluate the Effects of Fire on Electrical Cables Coated with Fire Retardant Coatings

Objective

The U.S. Nuclear Regulatory Commission (NRC) has conducted testing to better understand the performance of fire retardant cable coating materials to support clarifying guidance found in Appendix Q, “Passive Fire Protection Features,” of NUREG/CR-6850 (EPRI 1011989). These materials are typically applied to nonqualified electrical cables to reduce flame propagation characteristics of the cable. Specific objectives of this program include assessing the fire properties (i.e., flame spread, heat release rate, and ignition) and electrical response properties (i.e., loss of cable functionality) of fire retardant cable coatings materials.

Research Approach

The NRC performed several series of fire tests to assess the effects of fire on electrical cables coated with fire retardant cable coating materials. The tests used to evaluate the properties and performance of fire retardant cable coatings include radiant panel tests, micro-combustion calorimetry, thermogravimetric analysis, furnace ignition tests, IEEE-383/1202 vertical flame spread test, IEC-60331 circuit integrity test and horizontal tray tests with a test enclosure similar to that of NUREG/CR-6931 CAROLFIRE. To monitor electrical time to damage, tests (including standardized tests) were modified to include circuit functionality monitoring to determine when the cable fail electrical, which allows for the determination of the delay in time to damage attributed to the application of the cable coating.

Tests included thermoplastic-insulated cables that do not pass the vertical flame spread test requirements of IEEE-383/1202 and thermoset-insulated cables to do pass the IEEE flame spread qualification test. The test included four commercially available cable coating materials; three that are known to be in use at several U.S. nuclear power plants.

Figure 9.7 IEEE-383/1202 modified test with single cable layer tray coated with fire retardant coating.

Status

Scoping tests were performed at Sandia National Laboratories using the radiant exposure and, based on these results, subsequent tests have been completed at the National Institute of Standards and Technology using flaming type fire exposures. The NRC and its contractors are currently finalizing a three-volume NUREG report that documents (1) the historical use of cable flame retardant coatings in nuclear power plants, review of literature, and summary of existing technical guidance; (2) test results on cable coating fire properties; and (3) test results on impact of cable coating to electrical performance under thermally damaging fire conditions.

For More Information: Contact Gabriel Taylor, RES/DRA at Gabriel.Taylor@nrc.gov or Felix Gonzalez at Felix.Gonzalez@nrc.gov.

PRISME 3 – Fire Propagation in Elementary, Multi-Room Scenarios

Objective

The PRISME 3 program is an Organisation for Economic Cooperation and Development (OECD) Nuclear Energy Agency (NEA) experimental fire program that grew out of the international collaborative fire modeling project. The PRISME 3 program objectives are to study a number of fire sources to quantify fire behavior for use in probabilistic risk assessment (PRA) and fire modeling validation and verification (V&V). The program will last from 2017 to 2022, and participants include Institut de Radioprotection et de Sûreté Nucléaire (IRSN); Electricité de France (EDF); Gesellschaft für Anlagen und Reaktorsicherheit (GRS); VTT Finland; the U.S. Nuclear Regulatory Commission (NRC); Office for Nuclear Regulation (ONR); Institute for Building Materials, Solid Construction; and Fire Protection (iBMB); Korea Atomic Energy Research Institute (KAERI); Central Research Institute of Electric Power Industry (CRIEPI); Tractebel; and Bel V.

Research Approach

The PRISME 3 program comprises four experimental campaigns:

1) Smoke Stratification and Spread (S3)

   A total of five tests will be conducted in IRSN's DIVA facility. Two will focus on smoke propagation through openings, two will focus on smoke stratification and propagation from multi-fire sources, and one will focus on an elevated fire source.

2) Electric Cabinet Fire Spread (ECFS)

   A total of eight tests will be conducted. Four tests will be conducted under a calorimeter (SATURNE facility) and four will be conducted in the DIVA facility. In each set of four, two experiments will focus on fire spread from an open-door cabinet to an adjacent cabinet separated by a double wall, and two will focus on fire spread from an open-door cabinet to electrical cables.

3) Cable Fire Propagation (CFP)

   A total of eight tests will be conducted. Two will be conducted under a calorimeter and will study the effect of the cable tray configuration on the rate of fire spread. Three will study the effects of under-ventilated conditions on the fire behavior. The remaining three tests will study the behavior of fire spread in a service gallery configuration.

4) Complementary Tests (COMTE)

   This campaign is reserved to complete any of the previous campaigns, repeat tests as necessary, and perform confirmatory work.

Analytical results will be used to further validate commonly used fire models (FDS, CFAST, etc.) for which the NRC maintains a V&V guide (NUREG-1824). Expanding the repository of experiments incorporated into the V&V guide expands the range of validation and applicability of a model and increases the accuracy of statistical measures used to predict model uncertainty and bias. Probabilistic results will be used to confirm or revise the guidance for fire PRA as specified in NUREG/CR-6850, particularly where cable fire spread and cabinet-to-cabinet fire spread are concerned.

Status

The testing commenced in fall 2017.

For More Information: Contact Kenneth Hamburger RES/DRA, Kenneth.Hamburger@nrc.gov.

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

Objective

This program supports the U.S. Nuclear Regulatory Commission's (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, human reliability 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 two 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 fifth module entitled "Advanced Fire Modeling" was added.

In 2008, 2010, and 2012, the training sessions were also 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 in July 2010 documents the 2008 training; NUREG/CP-0301 documents the 2010 training; and NUREG/CP-0303 documents the 2012 training that was released in 2016. 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. The 2012 training session was the last session recorded of the MARIAFIRES series.

In 2015, EPRI and the NRC's Office of Nuclear Regulatory Research split hosting responsibilities such that EPRI hosted a session that includes Module 1 PRA and Module 4 HRA at an EPRI facility in Charlotte, NC. The NRC hosted Module 2 Electrical Analysis, Module 3 Fire Analysis, and Module 5 Advanced Fire Modeling at NRC Headquarters. Future 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, and fire-modeling programs are scheduled to continue into the future. This training continues to be in 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.

For More Information: Contact Tammie Rivera, RES/DRA, at Tammie.Rivera@nrc.gov for fire HRA content; Nicholas Melly, RES/DRA, at Nicholas.Melly@nrc.gov for fire PRA content; David Stroup, RES/DRA, at David.Stroup@nrc.gov for fire analysis and advanced fire modeling; and Gabriel Taylor, RES/DRA, at Gabriel.Taylor@nrc.gov for electrical analysis.

Fire Research and Regulation Knowledge Management

Objective

The objective of this program is to support the U.S. Nuclear Regulatory Commission's (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.

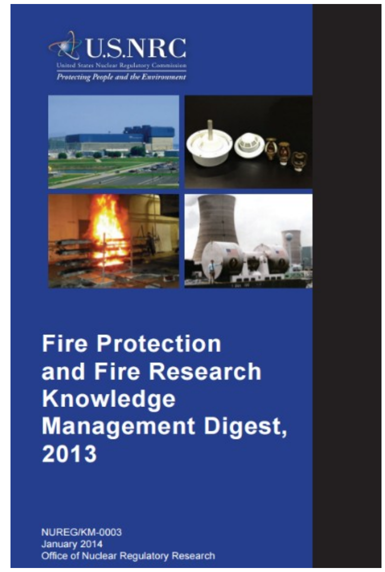

Figure 9.8 NUREG/KM-0003 cover.

Research Approach

Launched in 2012 as part of the NRC knowledge management (KM) program, the NUREG/KM series report is currently an integral part of the Office of Nuclear Regulatory Research (RES) KM activities. The office has published two NUREG/KMs related to fire protection in the nuclear industry: (1) NUREG/KM-0003 "Fire Protection and Fire Research Knowledge Management Digest," January 2014 and (2) NUREG/KM-0002 "The Browns Ferry Nuclear Plant Fire of 1975 Knowledge Management Digest", May 2013.

NUREG/KM-0003 contains a user-friendly DVD that provides information needed during activities such as inspections and reviews. The knowledge base consolidates all publicly available fire protection documents such as Federal regulations, guidelines for fire protection in nuclear power plants, 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.

NUREG/KM-0002 is a compendium of documents related to the most significant fire in a U.S. nuclear power plant. The results of this fire have affected fire regulations since 1975 including the creation of Appendix R "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," of the Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities". NUREG/KM-0002 captures technical and regulatory documents and pictures and videos that resulted from investigation, studies, and knowledge management activities (e.g., seminars) related to the event.

Another important KM-related NUREG is NUREG/BR-0364, "A Short History of Fire Safety Research Sponsored by the U.S. Nuclear Regulatory Commission, 1975-2008." This NUREG explores the history of the U.S. NRC fire protection research program from 1975 up to 2008.

Status

NUREG/KM-0003 is being revised to expand currently available information and to update the programming to improve the user interface of the DVD. The NRC plans to release an updated digest every 3-5 years depending on available resources. RES is also involved in the development of other NUREG/KMs currently being developed. This include NUREG/KM's related to hydrogen events and the Chernobyl Nuclear Power Plant accident of 1986.

For More Information: Contact Tammie Rivera, RES/DRA, at Tammie.Rivera@nrc.gov or Felix Gonzalez, RES/DRA, at Felix.Gonzalez@nrc.gov.

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

The 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 provides 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 U.S. Nuclear Regulatory Commission (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 U.S. 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 an 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). RES is leading the OECD/NEA High-Energy Arcing Faults (HEAF), Joint Analysis of Arc Faults (JOAN of ARC) experimental testing program, and is also a member of the OECD/NEA Fire Incident Record Exchange (FIRE) program.

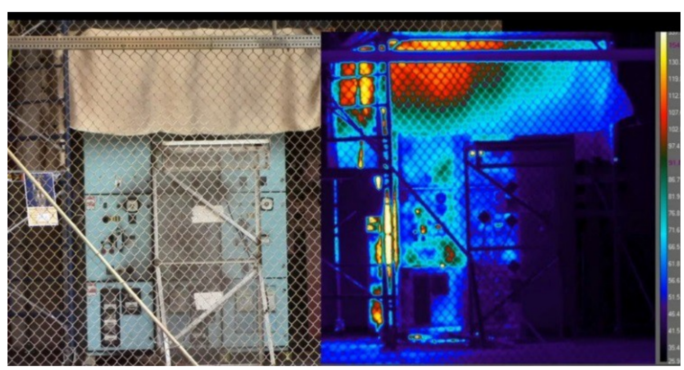

Figure 9.9 High Energy Arc Fault International Test Program – Thermal Camera Imaging.

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: Contact Mark Henry Salley, RES/DRA, at MarkHenry.Salley@nrc.gov.

Chapter 10: External Events Research

External events can have significant impacts on the safe operation of nuclear power plants (NPPs) as the accident at the Fukushima-Daiichi NPP showed. External events cover a broad range of natural hazards including earthquakes, liquefaction, tsunamis, and floods. The Office of Nuclear Regulatory Research (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 in the CEUS with the Next Generation Attenuation (NGA) East project. The goal of this project between the U.S. Nuclear Regulatory Commission (NRC), U.S. Department of Energy (DOE), Electric Power Research Institute (EPRI), and the U.S. Geological Survey (USGS) 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 update software tools for calculating site-specific PSHA results and to refine guidance for performing structured hazard studies using the Senior Seismic Hazard Analysis Committee (SSHAC) guidelines.

Local Effects on Ground Motion Estimation: Driven by the lessons learned from the reviews of 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 at existing NPP sites. A technical basis for applying risk-based methods is needed to update liquefaction evaluation regulatory guidance. The NRC has funded a National Research Council liquefaction study to assist in developing this technical basis and to identify additional research needed to update regulatory guidance. The NRC has also funded collection and analysis of strong ground motion data coupled with pore water pressure measurements at densely instrumented observation sites. Research on post-liquefaction residual strength was recently completed providing staff with probabilistic methods for evaluating soil strength after an earthquake for assessing earth fill embankment stability.

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 co-seismic applied pressure on deeply embedded structures. 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. The NRC is also performing research to develop guidance linking the results of probabilistic seismic hazard analyses with soil structure interaction analyses. This work consists of developing guidance on developing probabilistic strain compatible properties for use in soil structure interaction analyses.

Probabilistic Flood Hazard Assessment: The NRC has 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 will 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 to: (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. Presentations from the January 2017 Probabilistic Flood Hazard Assessment Research Workshop can be viewed online (ADAMS Accession No. ML17040A626).

Important Advances in Seismic Hazard Evaluation for the Central and Eastern United States

Objective

The objective of this program is to focus and refine U.S. Nuclear Regulatory Commission (NRC) research on the process and framework by which earth science models are developed, on developing a new set of ground motion models for the Central and Eastern United States (CEUS), and on effectively implementing uncertainties into site-specific probabilistic hazard calculations.

Research Approach

To standardize 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 "SSHAC guidelines") describes a formal, structured process for conducting 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. NUREG-2117 has been applied in numerous seismic hazard studies at a number of important facilities around the world. The NRC is currently revising NUREG-2117 to capture the insights from these SSHAC studies and to provide additional guidance on processes for updating existing studies, for the conduct of Level 1 and 2 studies, and on application of the process to other natural hazards.

The NRC is continuing research for CEUS sites with the Next Generation Attenuation (NGA)-East project which is a cooperative endeavor between the NRC, U.S. Department of Energy, Electric Power Research Institute, and the U.S. Geological Survey (USGS). The project will produce a comprehensive, state-of-the-art set of ground motion prediction equations (GMPEs) for the CEUS that capture ground motion prediction uncertainties. 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. Pacific Earthquake Engineering Research Center at the University of California-Berkeley manages the project. This project is augmenting sparse empirical CEUS data with ground motion simulations.

The NRC continues research to develop seismic hazard calculation software tools. To capture epistemic uncertainty in earthquake processes, seismic source characterization and ground motion models (such as NUREG-2115: "Central and Eastern U.S. Seismic Source Characterization Model") have become complex. Implementing these complex models in PSHA calculations requires the modification of existing codes and benchmarking results in a series of verification tests.

The NRC supports collaborative research with the USGS to benchmark seismic hazard results at numerous sites in the CEUS using both NUREG-2115 and the current USGS hazard model and to conduct forensic investigations of the differences in results.

Status

The NGA-East project began in 2009 and was completed at the end of 2017. The project to update NUREG-2117 started in early 2015 and was completed in 2017. The evaluation of alternative CEUS hazard models will be completed in 2019.

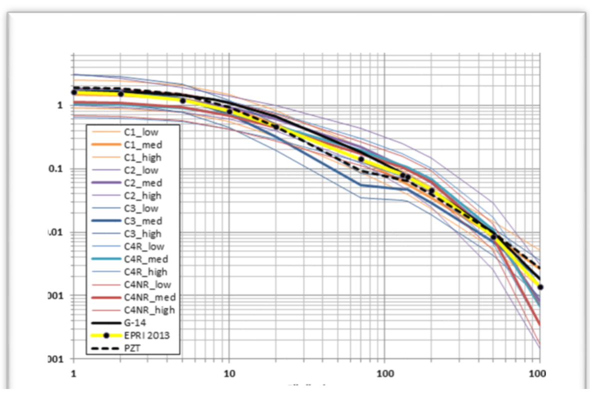

Figure 10.1 Shows a comparison of the variability of predicted ground motions for a magnitude 7.5 earthquake as a function of distance for currently available GMPEs at a frequency of 100 Hz.

For More Information Contact Jon Ake, RES/DE, at Jon.Ake@nrc.gov.

Local Effects on Ground Motion Estimation

Objective

Experience from staff review of early site permits and combined operating license applications conducted by the U.S. Nuclear Regulatory Commission (NRC) staff since 2007 and reviews of the operating licensee submittals in response to Recommendation 2.1 of the Fukushima Near Term Task Force identified site response-related research that will support revision to Regulatory Guide (RG) 1.208, “A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion,” and the continued development of the NRC’s confirmatory site response analysis tools.

Research Approach

The effects of soil or rock conditions on ground shaking is an important consideration in the development of site-specific ground motion response spectrum (GMRS). These effects may be quantified by a suite of response analyses to define the median site amplification and uncertainty for the site-specific soil properties. Research on site response includes the selection of shear modulus reduction and damping curves or the 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 may need to be incorporated into site response analyses; however, these rock types usually extend beyond the depth range where materials can be retrieved for laboratory dynamic testing. It may be necessary to rely on existing published curves (Figure 10.2) or estimated low-strain damping values for these materials, if they are assumed to behave linearly. This research will use data from additional testing to develop a basis for selecting shear modulus reduction and damping curves or the appropriate level of low-strain damping for various rock types in the absence of site-specific data. Almost all site response analyses used for seismic hazard studies assume a 1D layered system. Research using small-strain downhole array recordings showed the 1D approach may not accurately predict amplification for all sites. It is not possible to identify the sites for which the 1D assumption will produce an accurate estimate of site amplification. This research will use Japanese “KiK-net” network data and will investigate characteristics of sites that can and cannot be modeled accurately with the 1D approach.

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

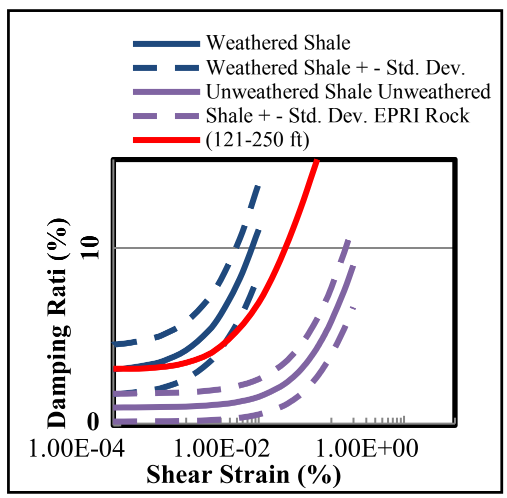

Figure 10.2 Example of damping values used for rock materials in site response analysis. Both weathered and unweathered shales were sampled at similar depth and within range of EPRI rock.

Status

Research on site response topics was initiated in 2015. The results will form, in part, the technical basis to update RG 1.208.

For More Information: Contact Scott Stovall, RES/DE, at Scott.Stovall@nrc.gov.

Seismic-Induced Ground Failure and Deformations

Objective

Seismic-induced ground failure is often associated with liquefaction or slope instability. Seismic-induced ground deformations can cause failure in supported structures, systems, and components. The objective of U.S. Nuclear Regulatory Commission (NRC) research on seismic-induced ground failure and deformations is to develop the technical basis for evaluating these hazards and associated risk.

Research Approach

Research on risk-informed evaluations of seismic-induced ground failure and deformations includes developing a public liquefaction triggering case history database, monitoring ground motion and pore water pressure development at select non-nuclear sites, and experimental work and model development to assess seismic induced settlement.

The case history database and ground motions monitoring work is consistent with recommendations made by the National Academies of Sciences study titled, "State of the Art and Practice in the Assessment of Earthquake-Induced Soil Liquefaction and Its Consequences." Probabilistic liquefaction triggering models will be developed using data from the public database, and these models will allow for assessing liquefaction hazard. Strong motion data coupled with pore water pressure measurements collected at densely instrumented geotechnical observation sites (e.g., Wildlife Liquefaction Array in Southern California, http://www.nees.ucsb.edu/data-portal) can validate models, help constrain uncertainty, and lead to a better understanding of the physics that causes liquefaction and associated ground failure. A cross section of the instrument layout, site geology, and example of data collected during a seismic event at the Wildlife Liquefaction Array (WLA) in Southern California is provided in Figure 10.3.

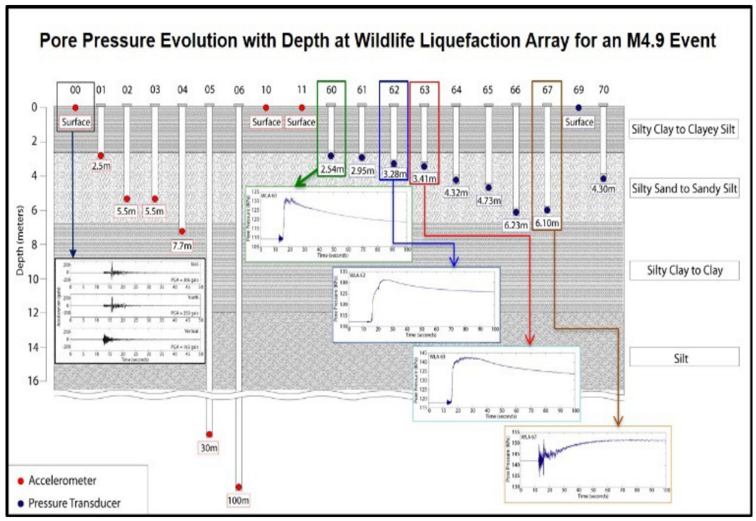

Figure 10.3 ***Insitu*** **observations of pore pressure and acceleration at Wildlife Liquefaction Array (WLA) during the August 2012 Brawley earthquake swarm are shown here.**

Multi-dimensional cyclic shear testing on small soil samples and a larger scale soil/structure system in a geotechnical centrifuge is providing data to develop a simple model for estimating seismic-induced settlement. In addition, a soil constitutive model will be implemented with finite element analysis software for performing soil-structure interaction evaluations that include the effects of soil settlement.

Status

The public liquefaction triggering case history database project was initiated in late 2016. This work will be completed in 2018. The first of two workshops to support the public database development took place in July 2017. The ground motion monitoring project was initiated in 2015 and ended in 2017. Seismic events that induced low levels of ground motion were recorded over this time period. The project on seismic-induced settlement was initiated in 2012 and concluded in 2017. A NUREG/CR documenting this work is pending publication.

For More Information: Contact Thomas Weaver, RES/DE/SGSEB, at Thomas.Weaver@nrc.gov.

Seismic Soil-Structure Interaction

Objective

The objectives of the U.S. Nuclear Regulatory Commission (NRC) seismic soil-structure interaction (SSI) research are to compare and evaluate structural response using multiple acceptable approaches for developing design ground motions and assessing if the objectives of the national standards of achieving an 80th percentile non-exceedance in-structure response are achieved. Results from this research are expected to support NRC staff in their regulatory reviews and provide the technical basis for updates to the standard review plan and regulatory guidance.

Research Approach

SSI analyses require application of seismological, geotechnical, and structural engineering principles. Figure 10.4 provides a schematic that illustrates SSI aspects. Due to the broad range of expertise involved, guidance is required to ensure each discipline provides appropriate data for inputs to SSI analyses. The technical basis for developing hazard consistent design ground motions is provided in NUREG/CR-6728. NUREG/CR-6728 documents multiple acceptable methods for developing design ground motions. However, the potential for inconsistencies in SSI analysis results when using these acceptable methods. This is, in part, due to a lack of guidance on which strain-compatible properties should be provided by seismologists and geotechnical engineers to the structural engineers for the SSI analyses. National standards such as American Society of Civil Engineers (ASCE) 4 and ASCE 43 provide guidance on the analysis and design of safety-related nuclear structures. The objective of these standards is to achieve an 80th percentile non-exceedance probability for in-structure response using input motions at the mean hazard level (design ground motions per NUREG/CR-6728 and strain compatible properties). Documentation is not available to verify that this objective is met.

Research in this area will consist of performing a sufficient number of analyses for a range of hazard levels and site conditions to explicitly characterize the distribution of in-structure response using two approaches for developing design ground motions. Results from these analyses will be compared with deterministic SSI analysis approaches typically used to assess structural behavior. Comparison of analysis results will identify if the approaches used to develop design ground motion and strain compatible soil properties produce significant differences in structural behavior. These analyses will also be used to determine if non-exceedance probabilities expected from following national standards are achieved. The approach will implement seismic hazards that are both greater than and less than the median hazard for U.S. nuclear sites. In addition, at least three site profiles that are consistent with geologic conditions from U.S. nuclear power plant sites will be used in the analyses with reasonable estimates of uncertainty in the site geophysical properties. A series of technical reports will document the research findings.

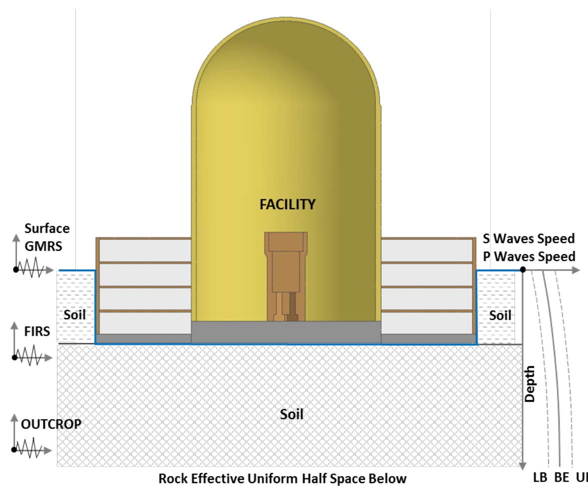

Figure 10.4 A schematic of seismic soil-structure interaction aspects is shown here.

Status

The first of three geologic profiles has been selected for the required SSI analyses that will be performed for this research, a preliminary set of analyses has been completed, and meetings have been held to exchange information between the NRC staff and the contractor assisting with this research. The technical reports documenting results and findings from this research will be completed over the next two years.

For More Information: Contact Thomas Weaver, RES/DE, at Thomas.Weaver@nrc.gov.

Development of Flood Hazard Information Digest for Operating NPP sites

Objective

Flood hazard information and insights are important inputs to the U.S. Nuclear Regulatory Commission's (NRC's) Reactor Oversight Program Significance Determination Process (SDP) for dispositioning flooding-related inspection findings (e.g., follow-up inspection actions, resource allocation, risk-informed regulatory actions). One particular challenge in developing probabilistic flooding hazard estimates within the SDP is that the required flood hazard information is not readily accessible, and it is challenging for NRC staff to assemble and analyze the information within the very limited time available for the SDP. Thus, a need exists to better organize flooding information at operating reactor sites (e.g., flood hazard types, frequency information, flood protection features) and to improve its accessibility for NRC staff performing SDP analyses.

The objective of this project is to develop the Flood Hazard Information Digest (FHID), a database architecture for organizing flood hazard information at operating NPP sites. This project also includes work to populate the FHID with site-specific information and to provide guidance on retrieving information from the database. The FHID is intended for use by the NRC staff and, due to the sensitivity of the information it contains, is not accessible to the general public. This study is part of the Probabilistic Flood Hazard Assessment program.

Research Approach

This research has been organized into several steps:

1. Conduct Flooding Hazard Information Needs Workshop with prospective users to inform FHID database scope, content, and design.
2. Review of existing NRC databases to inform FHID database architecture and implementation.
3. Develop beta implementation of FHID.
4. Perform beta testing with volunteers, conduct a series of demonstration seminars for prospective users, and develop User Manual.
5. Populate the database with site-specific flood hazard information.

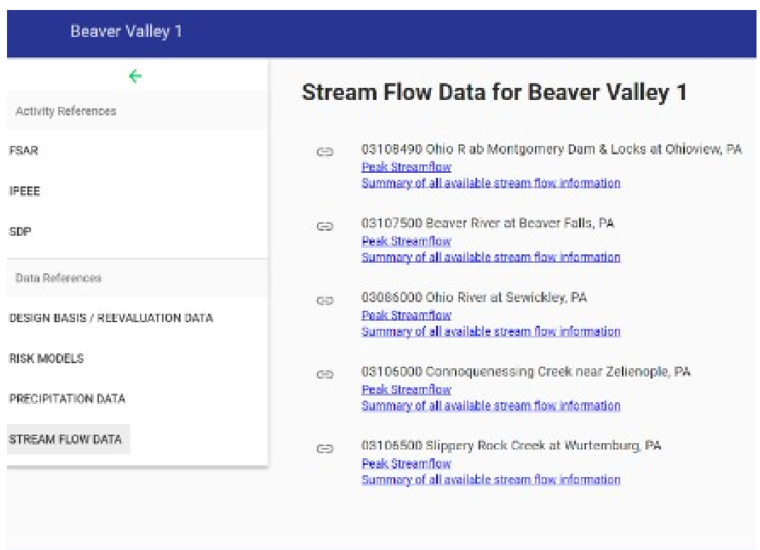

Figure 10.5 Flood Hazard Information Digest.

Status

Steps 1-5 described above are essentially complete. Current work is focused on populating the FHID with site-specific information and incidental modifications to the database taxonomy, architecture, and User Manual. However, to support the recent Commission direction to enhance existing agency processes for ongoing assessment of natural hazards, the scope of this project may be expanded to support other natural hazards in addition to flooding.

For More Information: Contact Joseph Kanney, RES/DRA, at Joseph.Kanney@nrc.gov.

Application of At-Site Peak-Streamflow Frequency Analyses for Very Low Annual Exceedance Probabilities

Objective

The objective of this study is to provide guidance on very low annual exceedance probability (AEP) estimation and the quantification of uncertainties using streamgage-specific data. The term “very low AEP” implies exceptionally rare events defined as those having AEPs less than about one-in-a-thousand. Such low AEPs are of great interest of flood frequency analyses for critical infrastructure, such as nuclear power plants. The current research is part of the U.S. Nuclear Regulatory Commission’s (NRC’s) Probabilistic Flood Hazard Assessment (PFHA) Research Program in support of developing a risk-informed licensing framework.

Research Approach

Flood frequency analyses at streamgages are most commonly based on annual instantaneous peak streamflow data and a probability distribution fit to these data. The fitted distribution provides a means to extrapolate to small AEPs. Within the United States, the Pearson type III probability distribution when fit to the base-10 logarithms of streamflow is widely used, but other distribution choices exist. Parameter estimation methods include product moments, the expected moments algorithm, L-moments, maximum likelihood, and maximum product of spacings.

The first task of this project comprehensively studies multiple distributions and parameter estimation methods for two U.S. Geological Survey (USGS) streamgages (01400500 Raritan River at Manville, New Jersey and 01638500 Potomac River at Point of Rocks, Maryland). The results of this task involve the four parameter estimation techniques and up to nine probability distributions including the generalized extreme value, generalized log-normal, generalized Pareto, and Weibull. Uncertainties in streamflow estimates related to AEP are depicted and quantified as two primary forms: quantile (aleatoric [random sampling] uncertainty) and distribution-choice (epistemic [model] uncertainty). Sampling uncertainties of a given distribution are relatively straightforward to compute from analytical or Monte Carlo-based approaches. Distribution-choice uncertainty stems from choices of potentially applicable probability distributions for which divergence amongst the choices increases as AEP decreases. Conventional goodness-of-fit statistics, such as Cramér–von Mises, and L-moment ratio diagrams are demonstrated to hone distribution choice. The results generally show that distribution choice uncertainty is larger than sampling uncertainty for very low AEP values. For task two, the state of practice for non-standard flood data at streamgage locations, regional information, and non-stationarity in flood frequency analyses are identified. Site-specific flood frequency analyses were done under different scenarios (i.e., comparing systematic peaks only, systematic peaks plus historic peaks, systematic and historic peaks plus paleoflood information). Regional information was assessed by comparing results determined with a site skew, weighted skew, and regional skew. In addition, in one test case, paleo information collected at a site was transferred to a nearby site as an attempt to use regional information. Some methods for dealing with climate non-stationarity were assessed such as separate wet and dry period flood-frequency estimates and using the most recent 30- year period as it is the period most likely (by some theories) to represent future climate.

Status

The report on the first task of the project was published as USGS Scientific Investigations Report 2017-5038. The second task is in progress.

For More Information: Contact Meredith Carr, RES/DRA, at Meredith.Carr@NRC.gov.

Technical Basis for Extending Frequency Analysis Beyond Current Consensus Limits

Objective

For inland nuclear facility sites (i.e., non-coastal sites), onsite flooding due to local intense precipitation or inundation due to flooding on nearby streams are key flooding scenarios that must be assessed. Precipitation frequency analysis (PFA) and flood frequency analysis (FFA) are the most widely applied tools for probabilistic flood hazard assessment (PFHA). The application of these tools to estimate rainfall and floods with annual exceedance probabilities (AEPs) of 0.01 to 0.002 using at-site data is relatively routine. When regional data are included, extension of frequency methods to AEPs on the order of $1x10^{-3}$ is broadly considered to be feasible, although not necessarily routine. For estimating AEPs below $1x10^{-3}$, no single consensus method exists. Instead, different types of data and methods of analysis (including numerical simulations) are generally combined on a case-by-case basis.

The objective of this project is to leverage experience gained by the U.S. Bureau of Reclamation (USBR) in combining multiple methods to extend frequency analysis for rainfall and flooding to low AEPs. This project will develop a technical basis document to assist the NRC staff in developing guidance for extending frequency analysis methods beyond current consensus limits for both rainfall and riverine flooding applications. The focus will be on describing alternative methods and approaches for integration of the characterizations from multiple approaches to estimate rainfall and floods with AEPs $1x10^{-5}$ to $1x10^{-6}$. In addition to describing alternative methods and approaches for combining them, this project will also focus on uncertainty characterization and quantification.

This project is part of the Probabilistic Flood Hazard Assessment Research Program.

Research Approach

To meet the objectives described above, this research has been organized into the following steps:

1. Perform a literature review, including USBR project reports.
2. Compile descriptions of state-of-art methods used by USBR for estimating precipitation and flood frequencies at AEPs of interest for critical infrastructure such as large dams and provide case studies of their application.
3. Perform knowledge management activities, including training for the NRC staff and completion of the NUREG/CR report.

Status

This work is underway, and is expected to be completed by early 2018.

For More Information: Contact Joseph Kanney, RES/DRA, at Joseph.Kanney@nrc.gov.

Stratigraphic Records of Past Floods for Improved Flood Frequency Analysis on the Tennessee River

Objective

Assessing inundation hazards for nuclear power plants (NPPs) is challenging, particularly when considering extreme magnitude, low-frequency events that are of interest for NPP risk assessments. Streamflow observations older than 200 years are practically nonexistent in the United States, so another approach is needed to estimate the flood frequencies. Paleoflood hydrology studies (use of geologic evidence to estimate stage and dates for unrecorded past floods) has provided dramatically improved statistical estimates of flood magnitude and frequency in arid and semi-arid environments. The objective of this project is to apply these techniques to conduct a comprehensive investigation of the flood stratigraphic record for a river in the temperate and humid climate of the eastern United States where most NPPs are located. Though this study will focus on the Tennessee River Gorge below Chattanooga, Tennessee, it will provide greater insight into the feasibility of this type of study to other eastern U.S. drainages.

Research Approach

A preliminary study (the Eastern U.S. Riverine Flood Geomorphology Feasibility Study) has shown that prehistoric flood deposits are present along the Tennessee River Gorge and can be sampled and dated effectively (the significance of this location is that the river channel in the gorge has been stable for long periods of time, allowing estimates of discharge to be derived from paleostage data). One deposit provided evidence of a large, hitherto unknown flood dating to about the 1700s although more work is being done to confirm the occurrence and magnitude of this flood. Other sites revealed evidence of three to four floods similar in size or larger than the “Great 1867” flood within the last 3,000 years, one possibly more than 50 percent larger.

Based on the positive results of the preliminary study, a comprehensive analysis of flood deposits in the Tennessee River Gorge area is underway with the first of three field trips completed in February 2017. The stratigraphy of previously identified paleoflood sites was examined and recorded. Additional potential sites were identified. Organic matter and sediment samples were collected for radiocarbon (C-14) and Optically Stimulated Luminescence dating analysis, respectively. Accurate elevation data for all sites was collected to allow estimates of flood paleostage.

Status

This project is underway. The preliminary study assessing the feasibility of using paleoflood data on the Tennessee River has been published as U.S. Geological Survey Scientific Investigation Report 2017-5052. Complementary studies on flood deposits outside of the gorge are being conducted by university researchers working for the Electric Power Research Institute.

Figure 10.6 A layered paleoflood deposit sample and typical locations of flood deposits under rock overhangs and in caves along the river.

For More Information: Contact Mark Fuhrmann, RES/DRA, at Mark.Fuhrmann@nrc.gov.

Technical Basis for Probabilistic Flood Hazard Assessment - Riverine Flooding

Objective

Knowledge of the magnitude, frequency, and duration of flooding hazards at a nuclear power plant (NPP) site is essential for the proper design and safe operation of the facility. Structures, systems, and components (SSCs) that are important to safety must withstand flooding hazards while performing their safety functions or be protected from such hazards by some appropriate combination of barriers and procedures. This research project will advance the U.S. Nuclear Regulatory Commission’s (NRC's) risk-informed regulatory approach by extending it into the area of riverine flooding hazard assessment, which currently depends primarily on deterministic analyses.

The objective of this project is to assess the technical basis to support development of regulatory guidance for probabilistic assessment of riverine flood hazards, including extreme events, for NPPs. This project is part of the Probabilistic Flood Hazard Assessment (PFHA) Research Program.

Research Approach

The technical focus of this work is to investigate the use of PFHA for the extreme riverine floods that are of interest in the design of NPPs. The work comprises two main components: (1) data availability and statistical analysis and (2) simulation methods. The data component will identify and assess existing databases relevant to hydrologic modeling of riverine flooding. The methods component will identify and assess probabilistic flood modeling methods and tools with respect to suitability for NPP flood risk assessment applications. The state of practice for PFHA at several Federal agencies is reviewed, including the U.S. Army Corps of Engineers, the U.S. Bureau of Reclamation, the U.S. Department of Energy, the Federal Energy Regulatory Commission, and the Federal Emergency Management Agency.

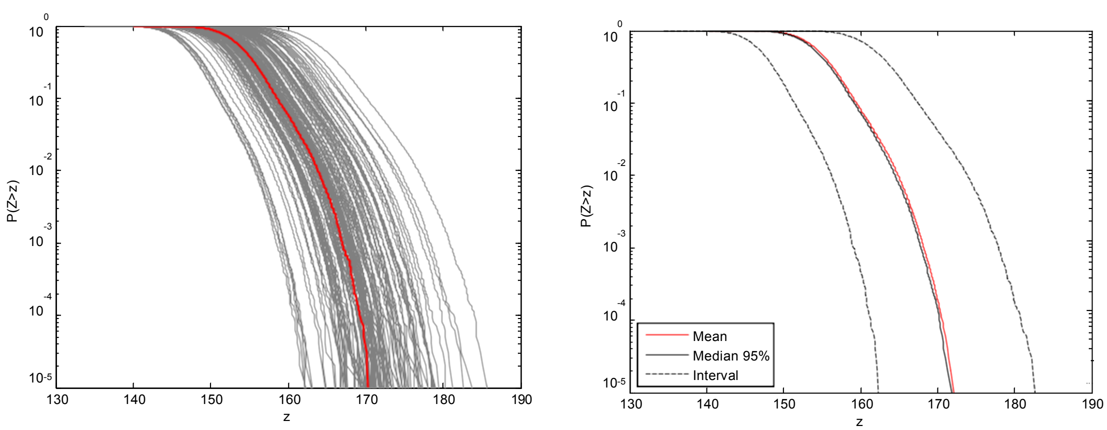

Figure 10.7 Aleatory hazard curve (in red) from constant *θθ* at mean values and ∈= 00.

Status

This work was recently completed. The results of the project are reported in NUREG/CR-7241, “Technical Basis for Probabilistic Flood Hazard Assessment - Riverine Flooding.”

For More Information: Contact Joseph Kanney, RES/DRA, at Joseph.Kanney@nrc.gov.

Probabilistic Flood Hazard Assessment Framework Development

Objective

A widespread interest exists in application of Probabilistic Flood Hazard Assessment (PFHA) methods to nuclear power plant (NPP) external flooding hazard assessments on the part of the U.S. Nuclear Regulatory Commission (NRC) and licensees to better risk-inform regulatory activities related to flooding issues. However, it is also recognized that (1) no consensus framework or standard set of methods exist for performing PFHA, (2) very large uncertainties in estimates for both the magnitude and frequency for extreme floods are of interest for NPP applications, and (3) considerable reliance is placed on professional or expert judgment in several steps of the PFHA process.

The objective of this project is to develop and demonstrate an overall conceptual, mathematical, and logical framework for probabilistic flood hazard assessment for inland and riverine sites (e.g., non-coastal sites). The framework should facilitate construction of site-specific flood hazard curves, and support full characterization of uncertainties in site-specific storm flood hazard estimates for the full range of return periods of interest for critical infrastructure facilities such as NPPs. This project is part of the Probabilistic Flood Hazard Assessment Research Program.

Research Approach

For inland nuclear facility sites (i.e., non-coastal sites), onsite flooding due to local intense precipitation or inundation due to flooding on nearby streams are key flooding scenarios that must be assessed. Therefore, a PFHA must be able to incorporate probabilistic models for a variety of processes (e.g., precipitation, runoff, stream flow, water control structure operation), allow for characterization and quantification of aleatory variability and epistemic uncertainties, facilitate propagation of uncertainties, and facilitate sensitivity analysis. A PFHA for inland flooding must also be able to accommodate models for various types of rainfall or snow events: (1) local intense precipitation (typically summer thunderstorms), (2) larger-scale warm season rainfall via convective or synoptic processes, and (3) cool-season synoptic processes. The framework should be capable of modeling spatial and temporal correlation between and within precipitation events.

This research has been structured according to the following steps:

1. Literature review.
2. Framework for Warm Season Rainfall and Local Intense Precipitation.
3. Framework for Cool Season Rainfall, Snow and Snowpack.
4. Framework for Site-scale Flooding from Local Intense Precipitation.
5. Framework for Riverine Flooding - Rainfall or Rainfall and Snowmelt.
6. Framework for Riverine Flooding - Hydrologic Dam/Levee Failure.

Status

Work on Tasks 1 and 2 has been completed. Work on the remaining tasks is underway.

For More Information: Contact Joseph Kanney, RES/DRA, at Joseph.Kanney@nrc.gov.

Structured Hazard Assessment Committee Process for Flooding

Objective

Widespread interest exists in the application of Probabilistic Flood Hazard Assessment (PFHA) methods to nuclear power plant (NPP) external flooding hazard assessments on the part of the U.S. Nuclear Regulatory Commission (NRC) and licensees to better risk-inform regulatory activities related to flooding issues. However, it is also recognized that (1) no consensus framework or standard set of methods are available for performing PFHA, (2) very large uncertainties exist in estimates for both the magnitude and frequency for extreme floods that are of interest for NPP applications, and (3) considerable reliance is placed on professional or expert judgment in several steps of the PFHA process.

The situation outlined above for PFHA is very similar to the state of affairs in probabilistic seismic hazard assessment (PSHA) several decades ago, so a structured multilevel assessment framework known as the Senior Seismic Hazard Analysis Committee (SSHAC) process developed for PSHA might, with suitable modification and adaptions, have utility for PFHA. In April 2014, the NRC and the Federal Energy Regulatory Commission staff conducted a workshop to discuss development of a structured hazard assessment committee process for flooding (SHAC-F). Riverine flooding and dam failure applications were the focus. Workshop participants generally agreed that SHAC-F should be pursued further, but that additional investigation using case studies to work out implementation details would be required.

The objective of this research project is to further develop and apply the SHAC-F process in PFHAs to provide confidence that all technically defensible data sets, models, and interpretations have been given appropriate consideration and weighting in the flood hazard analysis. This project will concentrate on local intense precipitation flooding and riverine flooding without dam failures. This project is part of the PFHA Research Program.

Research Approach

The research approach adopted is to conduct a series of a series of case studies focused on different types of flood hazard assessments to address practical issues in developing a robust SHAC-F process. Topics being addressed in this research include:

1. Technically defensible data, models, and methods for PFHA applied to:
   a. Local intense precipitation flooding.
   b. Riverine flooding (with and without snowmelt).
2. Adapting the SSHAC process to probabilistic flood hazard assessments
   a. Use of expert assessment.
   b. Participatory peer review.
   c. Appropriate hierarchy of approaches (similar to levels of SSHAC process).

Status

Case studies on Local Intense Precipitation (LIP) flooding have been completed. Case studies for riverine flooding are underway.

For More Information: Contact Joseph Kanney, RES/DRA, at Joseph.Kanney@nrc.gov.

Numerical Modeling of Local Intense Precipitation

Objective

The objective of this project is to assess the suitability of a regional numerical weather model to simulate local intense precipitation processes and then investigate the physical mechanisms of storm systems that lead to extreme precipitation by means of a regional numerical weather model. This study is part of the Probabilistic Flood Hazard Assessment program.

Research Approach

This research has been organized into the following steps:

- Conduct literature review of local intense precipitation climatology, data sets, and numerical modeling approaches.
- Identify a number extreme storm events for which with high-resolution quantitative precipitation estimates are available (e.g., Stage IV bias-corrected radar observations).
- Select a numerical weather simulation code and perform regional configuration and calibration using a subset of the extreme storms.
- Perform detailed simulation of selected storm events using the regional numerical weather model and evaluate its performance in modeling extreme precipitation in these storm systems by comparing the model results to detailed time-space observations.
- Investigate physical mechanisms of storm systems leading to local intense precipitation in a future climate.

Based on the literature review, the Weather Research and Forecasting (WRF) numerical weather simulation code was selected for this project. Moreover, two major storm systems leading to local intense precipitation in the United States were identified: mesoscale convective systems (MCSs) and tropical cyclones (TCs). During work plan development, initial and boundary conditions for numerical model simulation were selected for a historical period and a future period. The Climate Forecast System Reanalysis (CFSR) dataset is used for a historical period, and the Community Climate System Model 4.0 (CCSM4) Global Climate Model (GCM) future projections will be used for a future period. The Stage IV data were selected as observation data for model configuration and validation. Moreover, the historical largest local intense precipitation event was selected for each of the selected storm systems (MCS and TC) to assess the suitability of the WRF regional numerical weather model in modeling local intense precipitation processes (a Midwest U.S. MCS on August 19, 2007, and Hurricane Frances, August-September 2004). In addition, the methodologies for model configuration and its performance evaluation in modeling local extreme precipitation for the analysis of model errors and for the investigation of the mechanisms that produce local extreme precipitation events were proposed.

Status

Work performed to date has shown that one can adequately simulate local intense precipitation with respect to its location and timing for both MCS and TC type storms. As the next step of the project, climate projection of one general circulation model (GCM) during the 21st century will be downscaled using the WRF model at fine grid resolution over the inner domains of selected historical MCS and TC systems. The physical mechanisms of the future storm systems leading to extreme local precipitation will be investigated.

For More Information: Contact Elena Yegorova, RES/DRA, at Elena.Yegorova@nrc.gov.

Quantification of Uncertainty in Probabilistic Storm Surge Models

Objective

Storm surge is typically the dominant flooding hazard for nuclear power plants (NPPs) located in coastal settings (e.g., sites along the along Atlantic and Pacific oceans, Gulf of Mexico, and Great Lakes in the United States). Storm surge is a complex phenomenon involving the interactions of several processes including direct forcing from the storm wind and pressures fields and the effects of wind-driven waves. Storm surge propagation is strongly influenced by bathymetric and topographic features near a site as well as interaction of the storm surge with the astronomical tides. In addition, the effects of wave setup and wave runup need to be included in the storm surge estimate.

Storm surge hazard assessments use the historical record for storm types appropriate to the region (e.g., tropical, extratropical, or hybrid) to determine initial estimates for extreme winds. However, because storm surge events are rare and of limited extent, detailed analysis of historical storm events are augmented by synthetic storms parameterized to account for conditions more severe than those in the historical record but considered to be credible on the basis of climatological and meteorological reasoning. The synthetic storms are then used to drive surge and wave models to estimate the flooding hazard at the NPP site. The collection of synthetic storm/surge events must be constructed very carefully with due consideration of issues such as uncertainty characterization, uncertainty propagation, and sensitivity analysis. Currently, the joint probability method (JPM) is most commonly used for hurricane-driven storm surge.

The objective of this project is to develop and demonstrate an approach to fully characterize uncertainties in site-specific storm surge flood hazard estimates for the full range of return periods of interest for critical infrastructure facilities such as NPPs. This study is part of the Probabilistic Flood Hazard Assessment Research Program.

Research Approach

This research has been organized into the following steps:

1. Perform literature review.
2. Investigate epistemic uncertainties in site-specific storm occurrence rate models.
3. Investigate data, models, and methods for defining joint probability of storm parameters.
4. Investigate models and methods for generating synthetic storm sets.
5. Investigate approaches for characterizing errors arising from factors such as numerical surge simulation and exclusion of parameters from the JPM-OS integral.

Status

Work on Tasks 1 through 4 has been completed. Technical reports summarizing work performed on Tasks 1 and 2 are in publication. A technical report covering work performed in Tasks 3 and 4 is in preparation. Task 5 is in progress.

For More Information: Contact Joseph Kanney, RES/DRA, at Joseph.Kanney@nrc.gov.

Flood Penetration Seal Performance at Nuclear Power Plants

Objective

As part of the flooding protection at nuclear power plants (NPPs), penetrations in external (and some internal) walls that normally allow for the passage of cables, conduits, cable trays, pipes, and/or ducts should be watertight to prevent water from flowing through them and affecting the performance of safety-related components in the NPP. Flood seals for penetrations (FSPs) are installed to seal these openings to ensure watertightness and integrity of the wall penetrations. Currently, no known standard testing protocols, testing methods, or acceptance criteria are available to evaluate the effectiveness and performance of FSPs.

The objective of this project is to develop testing standards and protocols to evaluate the effectiveness and performance of seals for penetrations at NPPs. In addition, a series of tests are to be conducted on the seals to assess their effectiveness to water intrusion based on the developed testing strategy and protocols. This study is part of the Probabilistic Flood Hazard Assessment program.

Research Approach

This research has been organized into the following steps:

- Task 1.1: Perform review of NRC and external information sources to identify the most common flood seal assemblies in current use at U.S. NPPs.
- Task 1.2: Develop standard testing procedures, acceptance criteria, and protocols to assess effectiveness and performance of seals.
- Task 2: Evaluate the developed testing protocol by conducting tests of selected penetration seal designs.
- Task 3: Summary technical report (NUREG/CR).

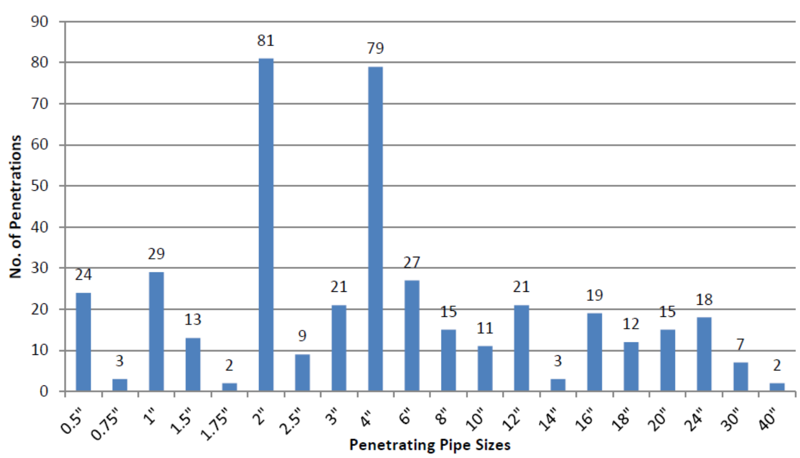

Figure 10.8 Reporting Data from NPPs.

Status

Tasks 1.1 and 1.2 have been completed. Tasks 2 and 3 are in progress.

For More Information: Contact Thomas Aird, RES/DRA, at Thomas.Aird@nrc.gov.

Effects of Environmental Factors on Manual Actions for Flood Protection and Mitigation at Nuclear Power Plants

Objective

Following the Fukushima Dai-ichi nuclear power plant accident, the U.S. Nuclear Regulatory Commission (NRC) identified the need to ensure the manual actions for flood protection and mitigation at nuclear power plants (NPPs) are both feasible and reliable. Environmental factors and conditions associated with floods that trigger manual actions for flood protection and mitigation can adversely affect the operators' ability to perform these actions. The current research is part of the NRC's Probabilistic Flood Hazard Assessment (PFHA) Research Program in support of developing a risk-informed licensing framework.

Research Approach

This project focused on characterizing manual actions from available NPP flood protection and mitigation procedures, developing a conceptual framework for assessment of impacts of environmental conditions on human performance, characterizing environmental conditions that are expected to be associated with floods, and reviewing the research literature related to effects of environmental conditions on human performance.

The first step was to identify and characterize the environmental conditions that might accompany flooding events; these conditions include heat, cold, noise, vibration, lighting, humidity, wind, precipitation, standing and moving water, ice and snowpack, and lightning. Based on a review of NRC staff assessments of flooding walkdown reports from 60 NPP sites and individual NPPs' plant procedures (e.g., Abnormal Operating Procedures) that were available, a set of manual actions that would need to be performed at and around NPP sites in preparation for a flood event were identified and characterized. A method for decomposing the manual actions into simpler units—tasks, subtasks, generic actions, and performance demands—was developed to facilitate assessment of environmental conditions' impacts consistent with approaches in human performance literature.

The literature review was structured to integrate the most recent research information that addresses environmental conditions and to present the findings in a format that is most useful for those reviewing and assessing performance impacts from the range and combinations of tasks, generic actions, and performance demands pertinent to outdoor work in varying weather conditions. The literature review summarizes the state of knowledge in terms of mechanisms of action, effects on performance, and potential mitigation measures. Based on this review, the report presents a typology of performance demands that includes detecting and noticing, understanding, decisionmaking, action, and teamwork that provide a basis for applying research findings to estimate performance effects. A conceptual framework and methodology was developed that illustrates the relationships among environmental conditions, manual actions, and performance effects information. The methodology is illustrated using a proof-of-concept method for an example manual action. Also, an example assessment of a manual action was evaluated in the U.S. Army Research Laboratory's Improved Performance Research Integration Tool (IMPRINT) to demonstrate the use of the framework and the literature review results.

Status

This project is underway. A draft NUREG/CR is being reviewed by the staff.

For More Information: Contact Meredith Carr, RES/DRA, at Meredith.Carr@nrc.gov.

A Simulation-Based Dynamic Analysis Approach for Modeling Plant Response to Flooding Events

Objective

All nuclear power plants must consider and evaluate external flooding risks such as local intense precipitation, riverine flooding, dam failure, and coastal flooding due to storm surge and tsunami. These events have the potential to challenge offsite power, threaten many onsite plant safety systems and components, challenge the integrity of plant structures, and limit plant access. Applicants and licensees must evaluate the potential impact of these events, including plant response, to fully comprehend plant risks and to ensure that flood protection features and procedures as well as flood mitigation measures are adequate to ensure plant safety.

The objective of this project is to investigate dynamic analysis approaches to model plant response to a local intense precipitation (LIP) flooding event, but the results can inform other flooding scenarios such as river or dam failure events. Both margins analysis and probabilistic risk assessment (PRA) approaches are considered in the project. The results and lessons learned from this project can be used in the future development of regulatory guidance on assessing plant response to flooding events. This project is part of the Probabilistic Flood Hazard Assessment Research Program.

Research Approach

The analysis of plant systems and accident sequences consists of developing event trees and fault tress in which the initiating event can be the external hazard itself or a transient or loss-of-coolant accident initiating event induced by the external event. Various failure sequences that lead to core damage, containment failure, and a specific release category are identified, and their conditional frequencies of occurrence are calculated. The unconditional frequency of core damage or of radionuclide release for a given release category is obtained by integrating over the entire range of hazard intensities. The steps outlined have to date been implemented almost exclusively by frequency analysis combined with system event and fault trees in a model. Although this approach has proven broadly successful for some external events (e.g., seismic events and loss of offsite power), flooding events present some unique and challenging aspects: (1) the performance of flood protection features may be a function of flooding levels and associated effects, (2) the degree of flooding may influence the rate of random or common cause failures, (3) the planned response to flooding events at many NPPs relies heavily on procedures and manual actions, (4) the feasibility and reliability of manual actions can be impacted by theflooding, (5) the duration of the flooding event can be quite long, and onsite conditions may change throughout the event. Dynamic analysis approaches that depict scenarios through simulation methods offer an alternative for representing flooding processes and highly time-dependent nature of flooding response.

Status

This work was recently completed. The results of the project are reported in NUREG/CR-7242, “A Simulation-Based Dynamic Analysis Approach for Modeling Plant Response to Flooding Events.”

For More Information: Contact Joseph Kanney, RES/DRA, at Joseph.Kanney@nrc.gov.

Potential Impacts of Accelerated Climate Change

Objective

The focus of this project is to follow developments in climate change research and to assess the potential for identified trends to impact U.S. Nuclear Regulatory Commission (NRC) licensing and oversight activities. Processes and mechanisms that may be impacted by climate change and are of interest to the NRC include extremes in air and water temperatures, decreasing water availability, increasing frequency and intensity of storms and flooding, and sea level rise. Moreover, these effects will likely not occur individually and may combine and exhibit compounding effects. For instance, antecedent moisture conditions (i.e., snow pack and soil moisture) are critical to extreme precipitation and flood scenarios at a given site. Potential implications of climate change may vary from reduction in plant efficiency and available generation capacity in the event of high-water temperatures or low-water levels in cooling ponds to increased risk of physical damage to nuclear facilities during high-wind storms or coastal storm-surge events. Of particular interest is addressing the effects of regional climate change variability that may dominate the local cumulative impact of climate change on individual nuclear plant safety. This study is part of the Probabilistic Flood Hazard Assessment Research Program.

Research Approach

This project relies on a review of recent literatures focusing on studies that improve understanding of the mechanisms of how the NRC-relevant climate parameters may change in a warmer climate, including discussions of the robust and uncertain aspects of the changes and future directions for reducing uncertainty in projecting those changes. By providing a high-level review of recent advances in climate change science and synopsis of regional climate change in the United States from recent assessments, this study aims to build the foundation for a more comprehensive review of understanding and assessment of climate changes relevant to NRC needs over project performance period.

The NRC's safety reviews and environmental reviews are fundamentally different processes and have substantially different needs. For the NRC environmental reviews, climate research published in the U.S. Global Change Research Program (USGCRP) National Assessment is used directly. From the NRC safety perspective, climate research is unlikely to provide direct information at annual exceedance probability level of 0.001 or less in the near future. However, an improved understanding of large-scale climate pattern changes (e.g., the occurrence of extreme precipitation events such atmospheric rivers, rain on snow with frozen soil, etc.) can help inform the probabilistic characterization (i.e., likelihood) of extreme events. This research has been organized into yearly focus areas with an annual report issued for each year:

- Year 1 – General assessment of trends in North American climate.
- Year 2 – Focus on climate trends in specific region (Southeastern United States).
- Year 3 – Focus on climate trends in specific region (region to be determined).

Status

The Year 1 report summarizing recent scientific findings on climate change with a focus on climatic elements that are relevant to NRC concerns (i.e., increasing air and water temperatures, decreasing water availability, increasing frequency and intensity of storms and flooding, and sea-level rise) has been completed and is available in ADAMS (ML16208A282). The Year 2 report focused on reviewing scientific findings regarding region-specific climatic extremes for the southeastern United States is in review. Year 3 work is in progress.

For More Information: Contact Elena Yegorova, RES/DRA, at Elena.Yegorova@nrc.gov.

Erosion Processes in Embankment Dams

Objective

An important part of dam breach modelling is to understand and accurately model the erosion processes of the embankment soil materials that occur as the dam breaches. Currently, most breach analysis of dams employ regression equations to predict the breach process. Use of regression analysis for dam breach, though widely used, do not account for the physical erosion of dam materials and its evolution as the breach progresses. Physical model experiments to model the erosion processes have in the past been attempted to accurately model the soil breach process. They have considered only homogenous and non-cohesive embankments and focused on failure through overtopping. However, since internal erosion (piping) is a common failure mode and zoned dams are the norm in large dams, studies of erosion processes in zoned dams that consider both overtopping and internal erosion are important understanding the breach progress of these types of dams. Most critical dams upstream of nuclear power plants are large zoned dams. This study is part of the Probabilistic Flood Hazard Assessment Research Program.

Figure 10.9 Internal Erosion (piping) failure in dam.

Research Approach

The Bureau of Reclamation (Reclamation) will test two physical models of rockfill zoned (clay core/filters/toe drains) embankment dams to study erosion processes as the breach progresses. One physical model each would investigate overtopping and internal erosion in the embankment dams. The physical models of the dams will be instrumented and use made of geophysical methods to detect seepage and internal erosion.

Figure 10.10 Dam breach test facility at Reclamation.

The dam breach test facility in Reclamation's hydraulics laboratory is designed to allow testing of embankments with 2:1 upstream and downstream slopes, a 1-ft crest thickness, and 3-ft dam height. The test embankments will be about 13-ft long, located at the exit of a headbox that is elevated 2 ft above the lab floor. Water discharged through the breached dam will be held in a large volume tailbox that will serve as a sedimentation basin. The test embankments are constructed with a length of up to 13 ft.

The work done for this research will complement research that Reclamation will be initiating on its own at its facilities. It involves geophysical monitoring and embankment physical model testing to study and understand embankment breach progression in real time. Data from the tests will be compared and analyzed against models of erosion processes of dam breach progression such as the Windows Dam Analysis Model (WinDAM) developed for the analysis of overtopped earth embankments and internal erosion.

Status

To date, the Bureau of Reclamation has tested one rockfill dam (with dimensions as noted above) consisting of a well-compacted clay core and surrounded on the upstream and downstream side with compacted gravel fill. The test was a study of water overtopping the dam. Data from the test has been collected and are currently being analyzed.

For More Information: Contact Jacob Philip, RES/DE/SGSEB at Jacob.Philip@nrc.gov.

Cooperative Research on External Flooding

Objective

The U.S. Nuclear Regulatory Commission's (NRC's) Office of Nuclear Regulatory Research continues to explore and engage in cooperative research on flood hazard assessment with both international and domestic partners. This involves establishing protocols and memoranda of understanding with international organizations such as the Committee for the Safety of Nuclear Installations (CSNI) of the Nuclear Energy Agency of the Organization for Economic Cooperation and Development (OECD/NEA) and the French Institute for Radiological Protection and Nuclear Safety (IRSN). Currently, NRC/RES chairs the OECD/NEA Committee on Safety of Nuclear Installations Working Group on External Events. NRC/RES is also pursuing cooperative flooding research with domestic organizations such as the Electric Power Research Institute (EPRI), U.S. Department of Energy (DOE) Laboratories, the U.S. Geological Survey (USGS), the U.S. Army Corps of Engineers (USACE), and the U.S. Bureau of Reclamation (USBR).

These cooperative research programs provide the opportunity to interact with domestic and international organizations ensuring NRC cognizance of ongoing flooding and other external events research in other organizations and other countries. This approach has proven to be cost-effective, avoids duplication of efforts, and ensures awareness of positions or analyses being forwarded by the industry or by domestic and international organizations.

Research Approach and Status

Cooperation with Electric Power Research Institute (EPRI)

The NRC and EPRI have recently entered into an agreement to collaborate on research in external flooding hazards assessment. Areas of mutual interest include:

- Characterization and probabilistic assessment of external flood hazards, including human-made hazards such as dam failure.
- Effectiveness of passive and active flood protection barriers.
- Modeling of certain flood scenarios (e.g., riverine, coastal storm surges).
- Incorporation of external flood effects into PRAs, including:
    - Identification of potential impacts of flooding on plant systems, structures, and components.
    - Treatment of operator response in the context of human reliability analysis including manual actions for flood protection and mitigation.

Initial efforts at collaboration have focused on technical information exchanges information. As research on various projects matures, the NRC/RES and EPRI may also seek to cooperate more closely in identifying and implementing pilot testing of technical approaches and methods in the areas outlined above.

Cooperation with Other Federal Agencies

The NRC/RES has entered into Interagency Agreements to perform much of the flood hazard assessment research described in this section. Flooding research projects have been conducted with the U.S. Army Corps of Engineers, the U.S. Geological Survey, the U.S. Bureau of Reclamation, Pacific Northwest National Laboratory, Oak Ridge National Laboratory, and Idaho National Laboratory.

For More Information: Contact Joseph Kanney, RES/DRA, at Joseph.Kanney@nrc.gov**.**

Cooperative Research on External Events-Seismic

Objective

The U.S. Nuclear Regulatory Commission's (NRC's) Office of Nuclear Regulatory Research (RES) continues to explore and engage in cooperative research with both international and domestic partners. These activities ensure awareness of positions or analyses being forwarded by the industry, domestic partners and international organizations, avoiding duplication of effort and enhancing cost-effectiveness. This involves establishing protocols and Memoranda of Understanding (MOU) with international organizations such as the Committee for the Safety of Nuclear Installations (CSNI) of the Nuclear Energy Agency (NEA), the Japan Nuclear Regulatory Authority (JNRA), and domestic organizations such as the Electric Power Research Institute (EPRI), the U.S. Department of Energy (DOE) and the U.S. Geological Survey (USGS).

Research Approach and Status

Collaborative Research with JNRA

The NRC has a cooperative agreement with JNRA in the area of seismic engineering research. The goal of the program with JNRA is to better understand the seismic behavior of nuclear power plant (NPP) structures, systems, and components (SSCs), obtain large-scale seismic test data to benchmark analytical techniques, assess equipment fragility test data to reduce uncertainty associated with seismic probabilistic risk assessment 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 scope of the program includes analyses of various SSCs for which JNRA performs seismic tests and provides test results to the NRC. 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 related to seismic safety research being performed in either country.

IAEA's External Events Safety Section Extra Budgetary Program (EESS-EBP)

The External Event Safety Section (EESS) of the International Atomic Energy Agency (IAEA), Department of Nuclear Safety and Security develops international standards on siting and seismic safety for NPPs. The EESS has an ongoing extra-budgetary program (EBP) that facilitates collaboration on the development of 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 EESS. The current phase of the EBP was started in 2014 and includes projects on 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 U.S. DOE, EPRI, and USGS to produce comprehensive consensus regional seismic source and ground motion models for the Central and Eastern United States. Currently, active areas of cooperative research include: (1) the development of a state-of-the-art ground motion model for eastern North America, (2) continuing collaborative research with the USGS on multiple seismic hazard topics including seismic hazard comparisons and 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, and (3) collaboration with EPRI on methodologies for site response analyses and compilation of plant-level seismic fragility estimates.

For More Information Contact Scott Stovall, RES/DE, at Scott.Stovall@nrc.gov.

Chapter 11: Materials Performance Research

The Office of Nuclear Regulatory Research (RES) provides data, standards, tools, and methods to the U.S. Nuclear Regulatory Commission's (NRC's) regulatory offices to support their reviews of material performance-related licensing submittals and 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 development. 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 aging phenomena that can affect materials in nuclear power plants (NPPs) and to 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. The goal is to ensure appropriate inspection intervals while still maintaining public safety. To provide this basis, research is focused on two areas: (1) inspection reliability and in-service inspection technology and (2) the evaluation and experimental validation of tube integrity prediction modeling of known degradation modes.

Reactor Pressure Vessel (RPV Integrity: The safe operation of an NPP relies on maintaining the structural integrity of the RPV during routine operations and also postulated accident scenarios. Two key capabilities underpin the assessment of RPV structural integrity: (1) the ability to predict the behavior of cracked components under loading, and (2) the ability to predict the effects of irradiation embrittlement on the fracture toughness of RPV steels. Ongoing research is aimed at understanding the adequacy of existing approaches and developing new models and predictive procedures to address the integrity of the RPV under extended applications.

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 and performing confirmatory analysis of the performance of RPV internal materials during extended operation up to 80 years. Research is being conducted on harvested ex-plant materials as well as on representative materials irradiated in test reactors.

Piping Degradation: To better understand how the reactor is influenced by the phenomenon of primary water stress corrosion cracking (PWSCC), the NRC is conducting confirmatory testing on both crack initiation and crack growth for susceptible materials. Additional research is focused on the development of analysis methods and computational tools, and the performance of experimental testing to assess the impact of PWSCC on the overall safety of piping systems of the reactor coolant pressure boundary. The NRC also is assessing impacts of PWSCC on the leak-before-break behavior of piping systems.

Non-Destructive Evaluation (NDE: Current NDE research is focused on the areas of effectiveness and reliability of NDE methods, assessment of NDE modeling tools, and evaluation of human factors on NDE performance. Results of ongoing research are used to support regulatory decisions associated with NDE and in-service inspection (ISI) of safety-related systems. The research efforts also support review of relief requests and proposed changes to ASME Code requirements.

Storage and Transportation: Current research develops the technical bases for review of license renewal applications for continued storage and transportation of spent fuel. It enables an improved understanding of potential degradation modes that could affect safety significant structures, systems, and components in dry cask storage systems. Particular focus is on the potential for chloride-induced stress corrosion cracking of stainless steel canisters.

Neutron Absorbers: Current research on the degradation of neutron absorbers involves a characterization of the physical condition of Boral®, which has been harvested from the decommissioned Zion NPP. This research evaluates the degradation of Boral® during actual operation and will be used to evaluate the adequacy of spent fuel pool (SFP) surveillance programs and nuclear criticality safety analyses.

Steam Generator Tube Integrity and Inspection Research

Objective

Steam generator (SG) tubes, Figure 11.1, 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. 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 11.2.

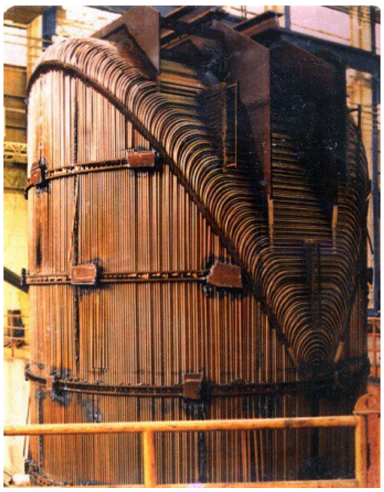

Figure 11.1 Steam Generator Tubing.

Research Approach

To help ensure that SG tubes are inspected appropriately, flaw evaluations are conducted correctly, and repair and tube plugging criteria are implemented adequately, 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 U.S. Nuclear Regulatory Commission (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, Korea, and the United States.

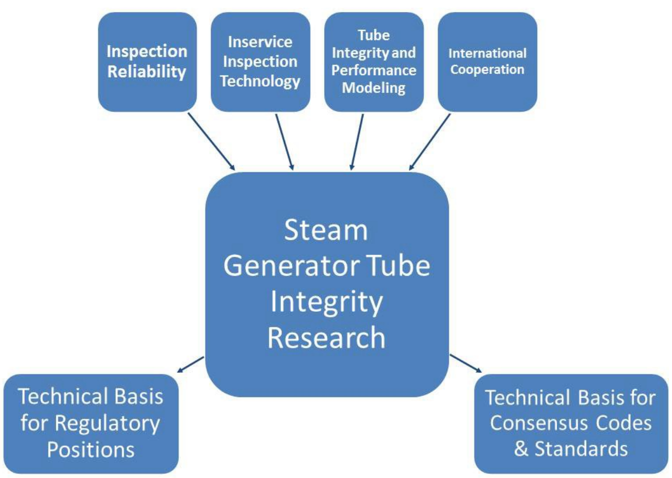

Figure 11.2 Tube Integrity research schematic.

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. A NUREG/CR report was published in 2017 addressing the stability of circumferential flaws in once-through steam generator tubes. The 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: Contact Patrick Purtscher, RES/DE, at Patrick.Purtscher@nrc.gov.

Reactor Pressure Vessel Integrity

Objective

This research is intended to ensure the integrity of the reactor pressure vessel (RPV) during both normal operation and postulated accident scenarios through the development of two key technologies: (1) prediction of the behavior of cracked structures (in this case the RPV) under loading and (2) prediction of the effects of radiation embrittlement on the fracture toughness of RPV steels.

Research Approach

Current U.S. Nuclear Regulatory Commission (NRC) regulatory guidance, 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. While generally acknowledged to contain large conservatisms, such methods have not always been validated through periods of extended operations. Ongoing research focuses on understanding the adequacy of existing approaches, quantifying their implicit conservatisms, and addressing gaps identified to develop better guidance and regulations.

Status

Technology [A] - Prediction of RPV Behavior Under Loading

- **DG-1299**: Provides guidance for implementation of the alternate pressurized thermal shock (PTS) rule, 10 CFR 50.61a. NUREG-2163 provides a companion technical basis document.
- **FAVOR**: **F**racture **A**nalysis of **V**essels, **O**ak **R**idge is a computer code providing a probabilistic representation of RPV behavior under routine operating and postulated accident loading. FAVOR supported 10 CFR 50.61a development and now permits analysis of accident and normal operations conditions in boiling- and pressurized-water reactors. One focus of this work 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 11.3 below shows a finite element model of such a defect. In addition, an extensive verification and validation effort is underway to bring FAVOR up to current software quality standards.

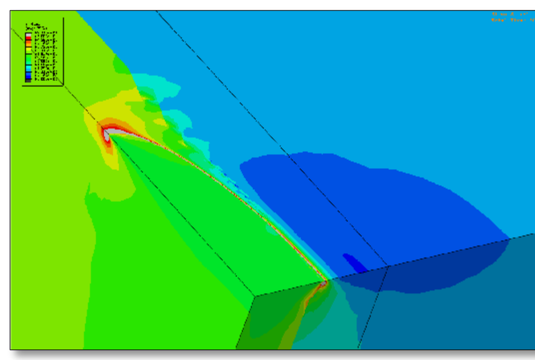

Figure 11.3 Finite element calculated stress contours around a semi-elliptical surface flaw in the stainless steel cladding of a RPV.

The U.S. Nuclear Regulatory Commission (NRC) staff also participates in the development of ASME Codes. These efforts provide access to current technical information, thereby supporting regulatory guidance development.

Technology [B] - Prediction of the Effects of Radiation Embrittlement

- **REAP**: The **R**eactor **E**mbrittlement **A**rchive **P**roject provides open Web-based access to light-water reactor 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. REAP was upgraded to expand its capability and is available as a tool for the NRC and international safety authorities and researchers.
- **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. In 2017 the NRC completed an investigation of BTP 5-3 of this claim, and its potential safety impact.

For More Information: Contact Mark Kirk, RES/DE, at Mark.Kirk@nrc.gov.

Irradiation-Assisted Degradation of Reactor Pressure Vessel Internals

Objective

The internal components of light-water reactor (LWR) pressure vessels are fabricated primarily with austenitic stainless steels that 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 11.4).

The objective of this research is to provide the technical basis for evaluation of 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 and also conducting research on welds and the heat affected zone at the weld metal interface.

Research Approach

The research approach involves harvesting representative ex-plant materials and reactor internals for testing and irradiation 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 and also performing confirmatory research funded solely by RES.

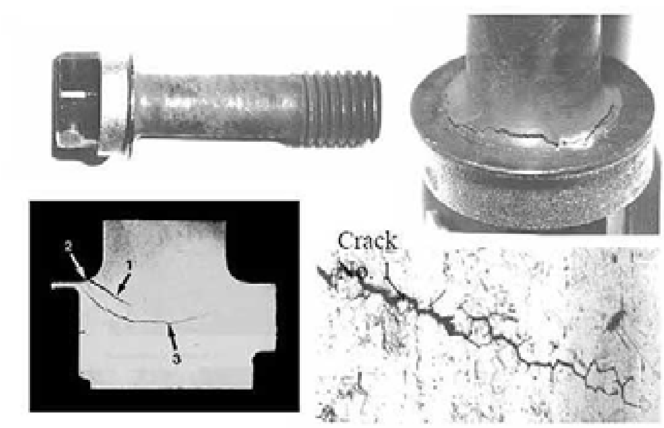

Figure 11.4 Cracking of a baffle bolt in a pressurized-water reactor.

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 2018 for plate materials and 2020 for weld materials. Future plans include further irradiation of weld materials as part of the U.S. Nuclear Regulatory Commission's (NRC's) participation in the Halden Reactor Project.

In addition to leveraging collaborative research with other organizations, the 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.

For More Information: Contact Appajosula S. Rao, RES/DE, at Appajosula.Rao@nrc.gov.

Primary Water Stress Corrosion Cracking Growth Rate Testing

Objective

Primary water stress-corrosion cracking (PWSCC) in primary pressure boundary components fabricated from nickel-based 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 11.5 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.

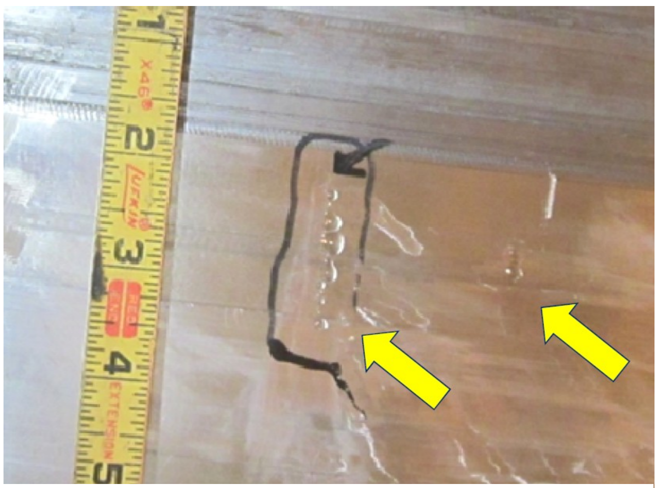

Figure 11.5 Leakage from PWSCC cracks in a steam generator hot leg nozzle.

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 U.S. Nuclear Regulatory Commission (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.

Specific testing to address the PWSCC susceptibility of Alloys 690, 52, and 152 in operating reactors and new reactors includes weld repairs, compositional dilution of chromium in dissimilar metal welds, pre-existing weld defects, warm-worked weld heat-affected zones, and variations in weld parameters such as heat input. The NRC also participates in collaborative activities with the Electric Power Research Institute under 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 forecasted to continue through 2019. 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 ANL on the susceptibility at or near the interface between Alloy 52/152 weld metals and other weld or base metals is expected to be published in 2018. Other staff findings are presented at conferences and workshops.

For More Information: Contact Meg Audrain, RES/DE, at [Margaret.Audrain@nrc.gov.](mailto:Margaret.Audrain@nrc.gov)

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 the U.S. Nuclear Regulatory Commission (NRC) Standard Review Plan 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 summary on leak-before-break analysis in this NUREG.

One of the major sources of uncertainty associated with the xLPR code is the time to initiate a primary water stress corrosion cracking (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 11.6. 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.

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

Two PWSCC initiation systems were constructed at PNNL and made operational in 2015. Initial materials characterizations including tensile testing and PWSCC crack growth rate testing have been completed. PWSCC initiation testing of the Alloy 600/182 and Alloy 690/52/152 is underway and is planned to be completed in 2020 and 2021, respectively.

For More Information:
Contact Eric Focht, RES/DE at Eric.Focht@nrc.gov.

Figure 11.6 PWSCC initiation testing rig and 1.2-inch-tall specimen developed by PNNL.

Weld Residual Stress Validation

Objective

Weld residual stress (WRS) develops in welded nuclear components during fabrication. These stresses can exacerbate degradation in piping welds at nuclear power plants. Finite element analysis (FEA) is a numerical tool that can predict WRS for a given weld geometry (Figure 11.7).

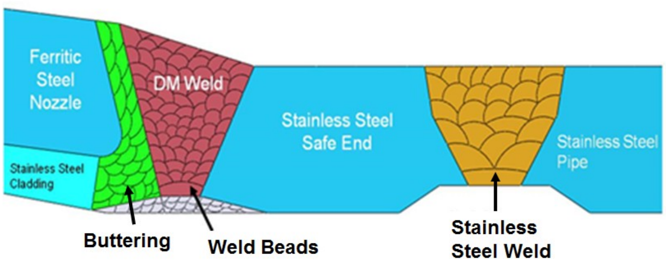

Figure 11.7 Example Weld Geometry.

The U.S. nuclear industry uses FEA to estimate WRS in technical and regulatory submittals to the U.S. Nuclear Regulatory Commission (NRC). The objective of the NRC's research program is to develop a validation methodology to increase confidence in FEA predictions of WRS. Much of this research was conducted under a Memorandum of Understanding (MOU) with the Electric Power Research Institute (EPRI). Based on the research, the NRC is currently developing a NUREG that discusses recommendations on WRS validation.

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. The dataset from this study forms the underlying basis of the proposed validation methodology developed in this program.

In calendar year 2016, probability and statistics experts at Sandia National Laboratory (SNL) developed an uncertainty quantification method for WRS predictions and measurements (ML16301A055). This uncertainty method provides a framework for making engineering judgments about WRS prediction quality. In addition, the NRC staff applied flaw growth calculations in accordance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code to supplement the SNL work. The outcomes of these efforts will be used to develop a validation methodology for WRS predictions. This research impacts regulatory efficiency by providing a method acceptable to the staff for developing WRS predictions.

Status

Previous results from this research program are documented in NUREG-2162 (ML14087A118). Other related research efforts are documented in ML16020A034 and ML16257A523. These research programs have led to in-house FEA capabilities at the NRC that assist NRC staff in reviewing emerging issues at U.S. nuclear power plants. A NUREG documenting the proposed validation methodology is currently under development and is expected to be completed by the end of calendar year 2018.

For More Information: Contact Michael Benson, RES/DE, at Michael.Benson@nrc.gov.

Leak-Before-Break Analysis

Objective

Title 10 of the *Code of Federal Regulations*, 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 U.S. Nuclear Regulatory Commission (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 to 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 occurring in systems that have been granted LBB exemptions. Although the piping systems experiencing PWSCC have been shown to be compliant with the regulations through qualitative arguments, the objective of this research is to establish a quantitative approach 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) have developed the Extremely Low Probability of Rupture code (xLPR), to calculate rupture probabilities in nuclear piping systems. Steps in the development of the code are:

- 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 GDC 4.
- 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 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 LBB 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.

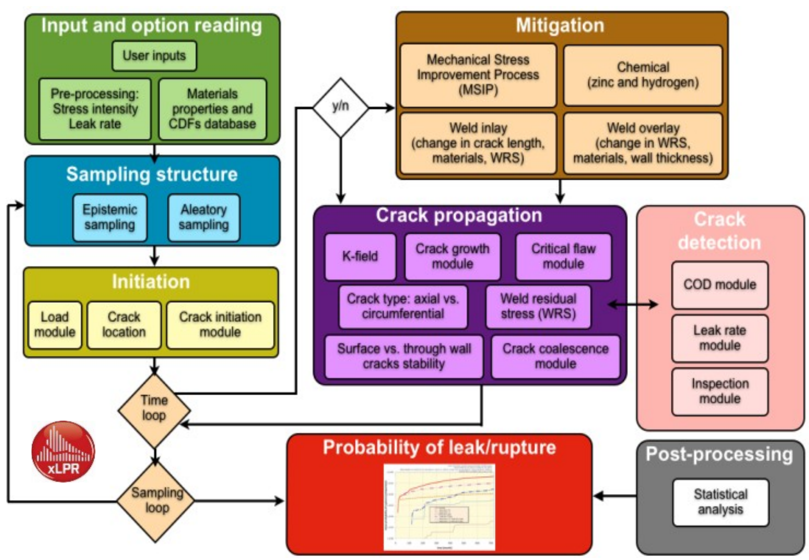

Figure 11.8 xLPR Version 2.0 Module Structure.

Status

The NRC, in cooperation with EPRI, released the production version of the code and related documentation in 2017. The sensitivity studies and re-evaluation of past leak-before-break analyses are slated for completion in 2018. The NRC plans to complete the generalization study, if necessary, and issue regulatory guidance on appropriate use of the code in 2019.

For More Information: Matthew J. Homiack, RES/DE, at Matthew.Homiack@nrc.gov.

Probabilistic Fracture Mechanics

Objective

The U.S. Nuclear Regulatory Commission (NRC) has observed an increase in the number of relief requests, license amendments, and topical reports using probabilistic fracture mechanics (PFM) calculations relying on novel or modified PFM codes as the technical basis for the proposed changes. As a result, regulatory and staff guidance and criteria to review such requests needs to be supported by an adequate technical basis. Accordingly, the staff is performing research to develop methods that will enable more efficient and consistent review of relief requests, license amendment requests, topical reports, American Society of Mechanical Engineers (ASME) Code Actions, and ASME Code Cases where state-of-the-art PFM methods are applied.

Research Approach

Probabilistic fracture mechanics calculations are unique in that they combine a series of deterministic calculations that, instead of using fixed values as inputs, use randomly determined inputs sampled from distributions of possible inputs. Unlike deterministic calculations, it is often not possible for the NRC staff to reproduce or verify the accuracy of the PFM calculations. The Office of Nuclear Regulatory Research is leveraging the experience acquired in the FAVOR (Fracture Analysis of Vessels - Oak Ridge) and xLPR (Extremely Low Probability of Rupture) projects to develop guidance on best practices for the development of PFM codes and PFM analyses.

The PFM guidance project consists of the following steps:

- Development of a technical letter report (TLR) highlighting important aspects of producing a rigorous, defensible, high quality PFM analysis.
- Development of a Regulatory Guide defining the desired steps and characteristics of a PFM analysis used in regulatory decision-making.
- Development of a NUREG technical basis for the proposed Regulatory Guide.
- Definition and conduct of one or more pilot studies to ‘test’ the developed guidance and make adjustments based on lessons-learned from the pilot studies.

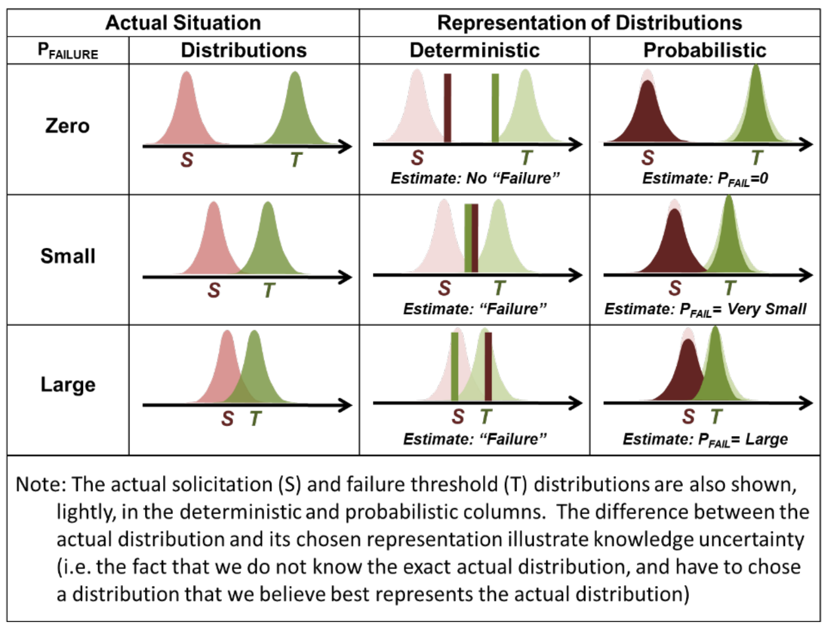

Figure 11.9 Probabilistic fracture mechanics calculations distributions.

Status

The TLR on PFM best practices as well as a draft Regulatory Guide and supporting draft technical basis NUREG are slated for completion in 2018. A public meeting will be held during 2018 to discuss the concepts in these documents. A first round of pilot studies is also scheduled to begin in 2018 to test the draft guidance.

For More Information: Contact Patrick A.C. Raynaud, RES/DE at 301-415-1987 or Patrick.Raynaud@nrc.gov.

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) techniques provides the technical basis for regulatory decisions on how licensees meet these requirements.

Research Approach

Research activities are focused on evaluating the accuracy, effectiveness, and reliability of NDE as currently practiced for the inservice inspection (ISI) of nuclear power plant (NPP) SSCs. As reactor facilities age, it becomes more important that adequate inspections are conducted to ensure that SSCs 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 NPPs 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 U.S. Nuclear Regulatory Commission (NRC) program is using fabricated mockups and components removed from reactors to determine the effectiveness of existing and emerging NDE techniques. Current research is focused in the following areas:

- Assessment of the capability of ultrasonic testing simulation tools (modeling) to optimize examinations (Figure 11.10).
- Effectiveness of ISI techniques for detecting service degradation, such as potential degradation in cast stainless steel and weldments.
- Theoretical and experimental studies related to incomplete examination coverage.
- Human Performance Influences on NDE reliability.
- Assessment of NDE methods for dry cask storage systems.

The NRC NDE research program, with Pacific Northwest National Laboratory serving as the primary contractor, dates back to 1977. The program continues to address a broad range of NDE, ISI, and ASME Code related issues. Publications will address NDE modeling, assessment of visual testing capabilities, and human factors influences on NDE reliability. In addition, NRC research is performed 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, Program to Assess the Reliability of Emergent Nondestructive Techniques (PARENT), aimed at evaluating inspection techniques using blind and open round robin testing to assess cracking in dissimilar metal welds. The findings from this research will be used 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 NPP SSCs are used to assess models developed to predict the effects of materials degradation mechanisms and as initial conditions for component-specific fracture mechanics calculations.

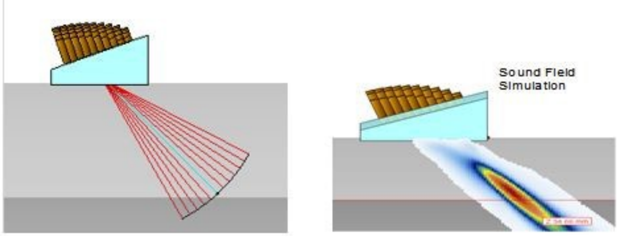

Figure 11.10 Focal law formation (left) and beam simulation (right) for a phased array ultrasonic probe.

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

Research to Support the Review of Subsequent License Renewal Applications

Objective

Continued operation of nuclear power plants (NPPs) for up to 80 years can be achieved if the license applications are in accordance with Title 10 of the *Code of Federal Regulations*, Part 54.31(d). However, the NPPs may need to resolve potential technical challenges from aging effects on passive long-lived systems, structures, and components (SSCs) before the U.S. Nuclear Regulatory Commission (NRC) can approve subsequent license renewal applications (SLRAs).

Accordingly, 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.”*

The objective of this research is to generate independent technical data and confirmatory tools to enable development of regulatory guidance on the aging of SSCs. Moreover, this research is to generate technical data and to enable the development of confirmatory tools to 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, in cooperation with both the Electric Power Research Institute (EPRI) and the U.S. Department of Energy’s (DOE’s) Light-Water Reactor Sustainability Research (LWRS) program, completed research to rank the significance of age-related degradation phenomena that could affect reactor SSCs over 80 years. This research is documented in NUREG/CR-7153, “Expanded Materials Degradation Assessment, Vol. 1-5.” Additional insights regarding NPP performance into the post-40-year PEO can be obtained in the NRC report entitled, “Review of Aging Management Programs: Compendium of Insights from License Renewal Applications and from AMP Effectiveness Audits Conducted to Inform Subsequent License Renewal Guidance Documents," dated June 15, 2016 (ML16167A076). Among its other findings, this study identified tuberculation as an aging mechanism leading to fouling (Figure 11.11) previously unidentified in the license renewal guidance documents.

Figure 11.11 Fouling from tubercles in service water system (NRC presentation at NRC/NEI public meeting, Dec 4, 2014, ML14338A376).

Status

The NRC’s Office of Nuclear Regulatory Research staff continues to interact with the DOE-LWRS Program and EPRI to monitor developments relevant to SLR and, where appropriate, engage in joint research activities. The NRC will continue to conduct longer-term confirmatory research to augment the technical basis for updating regulatory guidance in the future, as necessary, and inform staff reviews of future SLR applications.

For More Information: Contact Amy B. Hull, RES/DE, at Amy.Hull@nrc.gov.

Degradation of Neutron Absorbers in Spent Fuel Pools

Objective

In nuclear 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 requirements in the pool. Therefore, in many SFPs, neutron-absorbing materials (NAMs) containing boron-10 are placed on the rack walls, and are credited for precluding subcriticality.

In the past 15 years, NAMs, especially Boral® and Boraflex®, have shown various types of degradation such as blistering (shown in Figure 11.12) or matrix degradation. Information Notice 09-26, “Degradation of Neutron-Absorbing Materials in the Spent Fuel Pool,” dated October 28, 2009, summarizes specific incidents of excessive degradation. Degradation of credited NAM panels may affect criticality calculations and challenge the subcriticality requirement of $k_{eff} < 0.95$ in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.68, “Criticality Accident Requirements.” Currently, plants detect and manage NAM aging and degradation through surveillance programs such as sample coupons, in situ BADGER[1] testing, and RACKLIFE modeling.

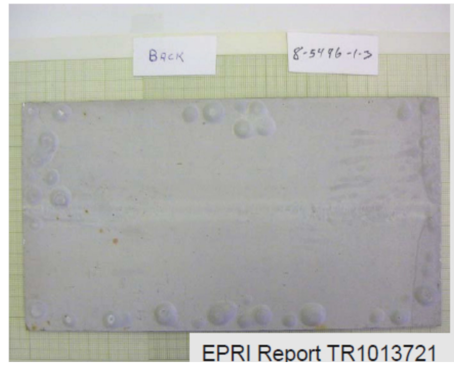

Figure 11.12 Blistering on the aluminum cladding of a Boral© neutron absorber. (Used with permission.)

Current research focuses on evaluating NAM performance in the SFP environment and in-situ neutron attenuation measurements using the BADGER system to characterize degradation mechanisms and measurement uncertainties associated with BADGER results. The results of this research will be used to evaluate the adequacy of SFP surveillance programs and to support the bases for nuclear criticality safety analyses.

Research Approach

The U.S. Nuclear Regulatory Commission (NRC) and the Electric Power Research Institute (EPRI) are conducting cooperative research under a Memorandum of Understanding (MOU) to perform in-situ testing of the NAMs at the Zion SFP using BADGER and to evaluate the performance of Boral® NAM panels harvested from the racks. The NRC’s goal is to correlate the BADGER results with the condition of the panels and to estimate the future performance of the material. The research will help characterize uncertainties associated with BADGER, identify degradation mechanisms, and estimate degradation rates. Also, surveillance methods employed for NAMs in SFPs will be evaluated to determine if the prescribed surveillance frequency is adequate or if some other surveillance frequency should be undertaken for the life of the plant.

Status

BADGER testing at the Zion SFP was performed in December 2015, and the Boral© NAM panels were removed from the Zion SFP in November 2016. The panel evaluations were 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.

Aging Management for Dry 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 from the U.S. Nuclear Regulatory Commission (NRC). Continued 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 renewal of ISFSI licenses for extended operation.

Research Approach

The current research is focused on building on prior knowledge gained through the Extended Storage and Transportation (EST) program to address ISFSI license renewal guidance development. This effort involves an approach similar to that used for reactor license renewal in the Generic Aging Lessons Learned (GALL) report. The process involves identifying safety-related structures, systems, and components (SSCs) in dry cask storage systems (DCSS) as seen in Figure 11.13 and assessing whether their safety function may be impacted by expected degradation based on the degradation scenario (material-environment combination). For degradation scenarios that expect significant aging degradation, the NRC guidance identifies aging management programs (AMPs) that may be developed by licensees to effectively manage the aging effects, often through periodic inspection of the affected SSC. Further research is ongoing to determine the effectiveness of inspection methods for DCSS and to better understand the progression of chloride-induced stress corrosion cracking (CISCC) of stainless steel canisters in marine environments.

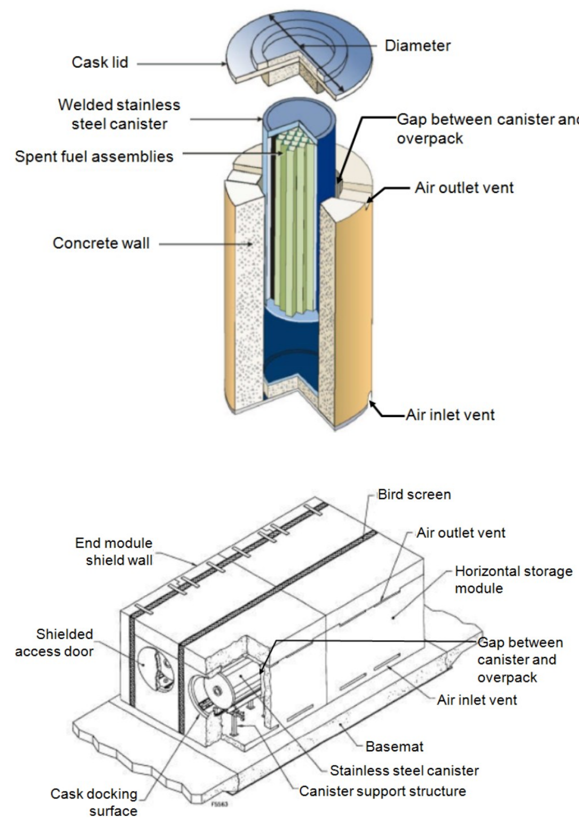

Figure 11.13 Schematics of vertical (top) and horizontal (bottom) DCSS.

Status

Recently completed research efforts include:

- An assessment of non-destructive examination (NDE) methods for managing aging of DCSS (ML16270A535).
- Aging Management Tables (AMTs), which have been incorporated into the draft Managing Aging Processes for Storage (MAPS) report (ML16235A124) published by the Office of Nuclear Material Safety and Safeguards (NMSS).

Ongoing and planned research efforts are focused in the following key areas:

- Inspection of canisters to address issues such as accessibility, detection, and sizing of likely flaws.
- Assessment of available CISCC growth data and consideration of options for improved understanding of CISCC growth rates.
- Evaluation of potential mitigation and repair methods.

For More Information: Contact Matthew Hiser, RES/DE, at Matthew.Hiser@nrc.gov, or Bruce Lin, RES/DE, at Bruce.Lin@nrc.gov.

Component Integrity Cooperative Research

Objective

The objective of this research is to identify and develop computer software packages, experimental data, numerical procedures, and other analytical methodologies that 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 in several materials research areas allow for leveraging of resources and minimizing duplication of effort.

Research Approach and Status

For the areas described below, the U.S. Nuclear Regulatory Commission (NRC) has separate Memoranda of Understanding (MOUs) with the Electric Power Research Institute (EPRI) and the U.S. Department of Energy (DOE) that promote general information sharing and describe the parameters for conducting cooperative research programs between the two organizations.

Probabilistic Fracture Mechanics Tools and Analyses

The objective of this research is to develop robust analysis tools and methodologies for evaluating reactor coolant system piping rupture probabilities. Such tools and methodologies use realistic input data and models and appropriately treat epistemic and aleatory uncertainties. Cooperative research is ongoing through an MOU with EPRI in the development of Extremely Low Probability of Rupture (xLPR) code. The tool has been verified and validated and is being 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 also ongoing through the PARTRIDGE (Probabilistic Analysis as a Regulatory Tool for Risk-Informed Decision GuidancE) program that is focused on probabilistic fracture mechanics methodologies and has participants from the United States, Canada, Sweden, Korea, Japan, Switzerland, and Taiwan. In addition, the NRC participates internationally through the Nuclear Energy Agency as part of the Committee for the Safety of Nuclear Installations on projects that evaluate the fidelity of analysis modules that are integral to the NRC's probabilistic analysis tools.

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, assess the reliability of NDE methods for selected examinations, and evaluate aspects of inspector qualifications. The NRC will use the results 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 to Assess the Reliability of Emerging Nondestructive Techniques) program, which is focused on the international inspection techniques for dissimilar metal welds and has participation from the United States, Sweden, Japan, Finland, Switzerland, and Korea. The results on blind testing and open testing are documented in NUREG/CR-7235 and NUREG/CR-7236, respectively.

For More Information: Contact Raj Iyengar, RES/DE, at Raj.Iyengar@nrc.gov.

Environmental Degradation Cooperative Research

Objective

The objective of the research is to develop data and methodologies and impacts of environmental degradation, including radiation effects, on the integrity of nuclear-grade materials. The research will provide the technical bases for inspection requirements and aid in regulatory decisions regarding the integrity of components during the extended period of operations.

Research Approach and Status

For the areas described below, the U.S. Nuclear Regulatory Commission (NRC) has separate Memoranda of Understanding (MOUs) with the Electric Power Research Institute (EPRI) and the U.S. Department of Energy (DOE) that promote general information sharing and describe the parameters for conducting cooperative research programs between the two organizations.

Subsequent License Renewal

NRC research supporting subsequent license renewal (SLR) is coordinated with the DOE Light Water Reactor Sustainability (LWRS) research program and with EPRI. These cooperative agreements ensure the timely exchange of information from ongoing and planned aging management research activities. Research continues on such topics as reactor pressure vessel embrittlement, irradiation-assisted degradation of reactor internals, concrete integrity and degradation mechanisms, and electrical cable condition assessment. The goals of this work are to reduce uncertainties associated with long-term materials performance and to enhance the technical bases for SLR guidance documents.

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 to support the reviews of relief requests and to confirm the industry data. The NRC will also use the data to help verify crack initiation and growth models and to evaluate industry inspection programs for dissimilar metal welds in components.

Neutron-Absorbing Materials (NAM) in Spent Fuel Pools (SFP)

The objective of this work 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 study the effects of high-fluence neutron irradiation using harvested ex-plant materials and to provide confirmatory technical basis for the performance of reactor vessel internal materials during potential extended operation up to 80 years. Current research focuses on assessing the significance of void swelling on the structural and functional integrity of pressurized-water reactor internal components and also conducting research on welds and the heat-affected zone at the weld metal interface.

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.

For More Information: Istvan (Steve) Frankl, RES/DE at Istvan.Frankl@nrc.gov.

Chapter 12: Structural Performance Research

Structural performance in nuclear installations is an essential aspect of their safety and security. Structural material degradation and aging effects such as alkali-silica reaction (ASR), containment liner degradation, and loss of prestress have been observed in U.S nuclear power plants. U.S. Nuclear Regulatory Commission research evaluates the significance of material degradation and aging for structural performance and safety to support license renewal of nuclear power plants (NPPs) for up to 80 years. Therefore, this research addresses the following structural performance issues:

Irradiation Effects on Structural Performance of Concrete – For long-term operations, the radiation fluence/dose experienced by the concrete in structures in the proximity of the reactor vesses (primary and biological shield walls and reactor vesses support structures) may approach levels that degrade the concrete. Therefore, research is performed to assess the structural safety significance of concrete irradiation.

Chemical Degradation of Concrete – Concrete at NPPs can deteriorate over time due to the effects of several chemical and physical processes including ASR. This research assesses the structural performance of ASR-affected structures under static and dynamic loading and load combinations throughout the service life of the structure.

Aging of Prestressed Concrete Containment Vessels (PCCVs) – This research evaluates the effects of aging and modifications of PCCVs on the intended safety functions. It examines conditions under which the complex combined stresses from vertical and hoop tendons and operational conditions may lead to creep-induced split cracking (NUREG/CR-7153).

Structural Analysis – This research maintains state-of-the-art structural analysis capabilities on nonlinear structural analysis. It involves benchmarking existing analysis tools, performing sensititivity studies (Figure 12.1) to inform best practices and agencywide studies, and developing new modeling capabilities.

Steel Plate and Concrete Composite Modular Construction – Some designs in the new generation of nuclear power plants have incorporated the use of steel plate composite (SC) modular construction in safety-related structures such as structures that support the reactor coolant system. This research develops the technical bases for evaluating SC construction.

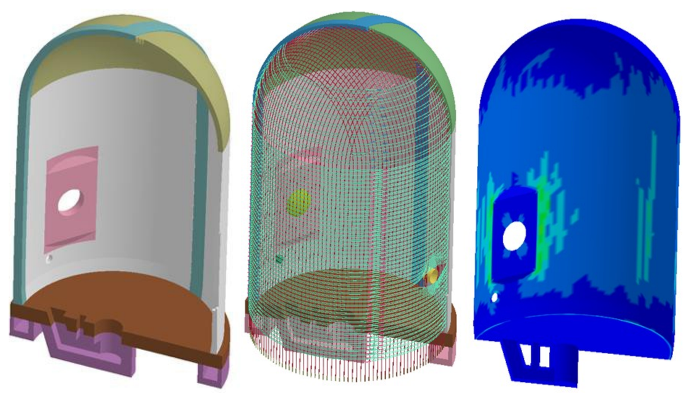

Figure 12.1 Finite element model of a prestressed concrete reactor containment and contours of maximum principal strain in the liner under accident conditions.

Risk-Informed Performance-Based Apporach to Seismic Safety – This research advances possibilities in the use of risk-informed performance-based approaches to seismic safety. Initial research addresses the technical bases for the update to Regulatory Guide 1.208 as well as review of technical bases for the development of staff positions on the recent and forthcoming updates of standards for performance-design and analysis of seismic design of structures and components in nuclear installations. Longer-term efforts would pursue possibilities as those outlined in NUREG/CR-7214.

For More Information: Contact Dogan Seber, RES/DE/SGSEB, at Dogan.Seber@nrc.gov.

Concrete Irradiation Effects on Structural Performance

Objective

The primary objective of this research is to review, evaluate, and enhance the capability to perform confirmatory analyses and testing of the effects of irradiation of concrete on the integrity of structures in the proximity of the reactor pressure vessel over extended periods of operation. The goal is to provide a technical basis to review plant conditions under aging management programs. Another goal is to inform development of guidance updates for the review of aging management programs.

Research Approach

Over extended periods of operation, concrete structures in the proximity of the reactor pressure vessel (RPV) (e.g., the primary and biological shield walls and the RPV support structures) can be subjected to high levels of neutron and gamma radiation together with sustained operating temperature up to about 150° F. Long-term neutron and gamma irradiation on concrete of the reactor supports and shielding structures can affect dimensional change (radiation-induced volumetric expansion), micro-cracking of the cement paste, physical and structural properties of concrete (see Figure 12.2) (e.g., reduction of compressive strength, tensile strength, modulus of elasticity, bond strength) that may affect structural performance and shielding capacity. This research assesses the structural and safety significance of concrete irradiation for long-term operations and addresses it via the following steps:

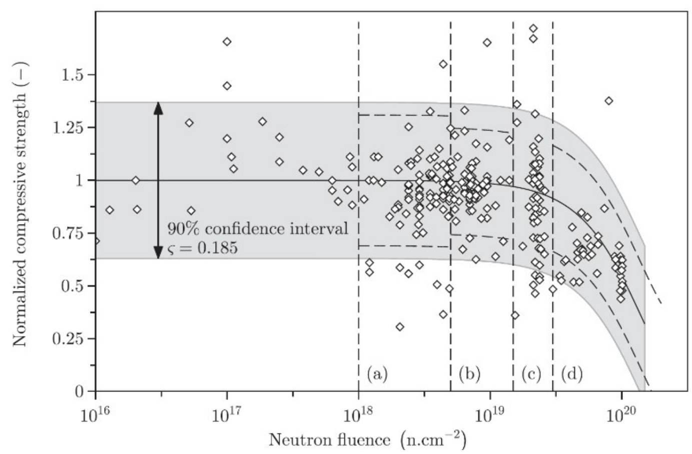

Figure 12.2 Degradation on concrete compressive strength with neutron fluence (source: Yann Le Pape et al.).

- Determine 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 implications of degradation.
- Develop the technical basis for aging management and monitoring methods.

These steps are further enhanced by Office of Nuclear Regulatory Research participation in international collaborative research activities including technical exchanges with the Nuclear Regulatory Authority of Japan and the International Committee for the Irradiation of Concrete.

Status

Although the Electric Power Research Institute (EPRI) and the U.S. Department of Energy (DOE) have jointly published a report on estimated radiation levels for operation up to 80 years, DOE is developing multi-scale mechanical models to simulate damage and EPRI has started developing structural assessments. The U.S. Nuclear Regulatory Commission (NRC) is reviewing the research performed by EPRI and DOE. The NRC is also developing and implementing research plans to (1) study radiation effects on mechanical concrete-steel bonds, which include pursuing testing options under the Halden Research Program; (2) evaluate concrete damage modeling; and (3) assess plant configurations and designs to determine the range of plant conditions that would indicate a need for aging management.

For More Information: Contact Madhumita Sircar, RES/DE, at [Madhumita.Sircar@nrc.gov.](mailto:Madhumita.Sircar@nrc.gov)

Chemical Degradation of Concrete and Structural Effects

Objective

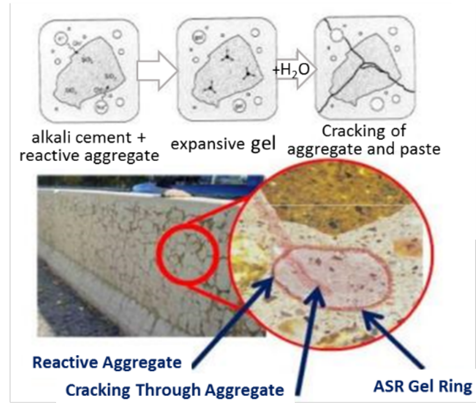

Figure 12.3 Evolution of ASR.

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) (Figure 12.3). 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 objective of the research program is to develop a methodology to determine for an ASR-affected structure (1) the in-situ structural capacity to resist design-basis static and dynamic loads, (2) its future structural capacity, and (3) a recommended aging management program consistent with the guidance in NUREG-1800 for managing the aging effects of ASR in existing structures.

Research Approach

The U.S. Nuclear Regulatory Commission is performing experimental research at the National Institute of Standards and Technology (NIST) that involves a combination of testing and modeling to study ASR effects on nuclear concrete structures. This program involves assessing in-situ mechanical properties of ASR-affected concrete, bond and lap splice lengths of reinforcing bars, and the seismic response characteristics of structural members.

Figure 12.4 ASR concrete block sample.

The research involves the construction of large concrete block specimens (Figure 12.4), each composed of low-, moderate-, and high-reactivity aggregates, as well as a control concrete block specimen of non-reactive aggregate. Each block has two regions with varying confinement provided by different amounts of reinforcement (stirrups and ties), and one region without reinforcement (unconfined). The research involves monitoring the expansion of the concrete for each aggregate type and confinement condition as well as additional testing of concrete samples extracted from these blocks. The blocks are heavily instrumented with triaxial strain gages, humidity sensors, and thermocouples to measure internal concrete expansion, humidity, and temperature, respectively.

Status

NIST is drilling cores from the block specimens and performing compressive and tensile testing on the cores to determine the mechanical properties of the concrete (compressive strength, tensile strength, and modulus of elasticity) throughout the testing duration. Non-destructive testing (NDE) to detect and map the course of ASR degradation in the specimens has also begun. Testing continues for material characterization including reactive aggregate mineralogy and texture effects on ASR for all aggregates. Also continuing is an evaluation of the degree and rate of ASR reaction based on petrographic/image analyses of microstructural features, physical methods and chemical characterization of the evolution of ASR gel in concrete. In addition, this research is also performing an evaluation of the petrographic/ microstructural properties of concrete and their correlations with expansion, surface cracking, and changes in concrete mechanical properties resulting from ASR.

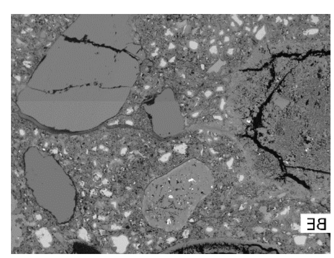

Figure 12.5 ASR gel in crecks.

For More Information: Contact Jacob Philip, RES/DE, at Jacob.Philip@nrc.gov.

Aging of Prestressed Concrete Containment Vessels

Objective

The objective of this project is to study the effects of aging and modifications of Prestressed Concrete Containment Vessel (PCCV) on their intended safety functions. Volume 4, "*Aging of Concrete and Civil Structures*," of NUREG/CR-7153, known as the Expanded Materials Degradation Assessment (EMDA) report, identified creep and potential for creep-related fracture as a topic for further study in relation to the extended period of operation of nuclear power plants. This topic is applicable to PCCVs because of the sustained, multi-axial loading from the prestressing. This is relevant for aged PCCVs where reinforcement in the radial direction is not present and when prestressing systems require adjustments such as modifications for steam generator replacement, other repairs, and loss of prestress. As with other concrete aging mechanisms, creep and potential for creep-fracture also may interact with other degradation mechanisms such as alkali silica reaction, corrosion, and freeze-thaw.

The Electric Power Research Institute (EPRI), U.S. Department of Energy, and international organizations, such as the Committee for the Safety of Nuclear Installations of the Organization for Economic Development and Cooperation (OECD/CSNI) maintain or plan research activities on this topic. An OECD/CSNI activity identifies shrinkage, creep, drying, and moisture transport as some of the several significant aging topics for study. Specifically, this activity focuses on benchmarking aspects of VERCORS (*In French: Vérification réaliste du confinement des réacteurs, In English: Realistic verification of the reactor containment*). VERCORS is a comprehensive multiyear study of a 1/3 scale PCCV that focuses on aging effects, computational modeling, and use of non-destructive evaluation techniques and sensors for structural monitoring. Electricité de France, which started, funds and executes VERCORS, is making information and data from this study available to the OECD/CSNI activity.

Research Approach

The research examines conditions, if any, in which creep under the complex combined stresses from vertical and hoop tendons may lead to creep-induced split cracking. It also examines, for example, the potential for creep re-activation by the re-tensioning of tendons to contribute to damage. The research reviews operating experience on the performance of PCCVs (Figure 12.6), examines potential degradation modes, and improves understanding of computational modeling of structural behavior for degraded PCCVs under accident scenarios involving thermo-mechanical loading. It also includes evaluating the efficacy of modern techniques for inspection and monitoring of prestressing systems inclusive of NDE and sensors. The research tasks are:

- Study of the operating experience.
- Review of research results on creep and potential creep-fracture.
- Participation in the VERCORS mock-up program.

Figure 12.6 Delamination of Crystal River Unit 3 PCCV.

Status

The Office of Nuclear Regulatory Research (RES), in coordination with Sandia National Laboratories, is reviewing EPRI reports and operating experiences. In addition, RES is receiving data from the VERCORS program and developing a modeling approach for benchmarking. The date for completion of pre-testing benchmarking analysis is March 30, 2018. Intermediate pressurization test results for the VERCORS testing will be available in April 2018, with a workshop planned for August of 2018 to compare the results from the various research teams.

For More Information: Contact Madhumita Sircar, RES/DE, at Madhumita.Sircar@nrc.gov.

Structural Analysis

Objective

The research in this topic maintains a state-of-the-art capability in nonlinear structural analysis to support regulatory functions of the U.S. Nuclear Regulatory Commission. Uses of this capability include confirmatory analysis, assessments of design margins, and obtaining insights on modeling and analysis practices. These assessments and insights inform staff reviews and the development and update of design and review guidance. This page illustrates the research in this topic with benchmarking analysis of full-scale and scaled regulatory drop tests of spent nuclear fuel transportation (SNFT) casks.

Progress of design optimization enabled in part by improvements of three-dimensional (3D) non-linear finite element (FE) software capabilities tends to produce complex and optimized cask designs that may be subject to high stresses or strains during transportation accidents. Typically, safety assessments of SNFT casks subjected to transportation accident scenarios involve drop test data and numerical analysis using a range of three-dimensional (3D) FE calculations with varying complexity. The analyses provide additional understanding of the stresses and strains in critical structural elements of the cask beyond those directly derived from the limited sensor data from the tests. The objective of the benchmarking studies using full-scale and scaled test data coupled with sensitivity analyses is to provide useful technical data to inform and enhance confidence on staff and industry practices for the design and review of SNFT cask designs.

Research Approach

Under a cooperative research agreement with the Federal Institute for Materials Research and Testing (BAM) of Germany, the staff gained access to information and data from regulatory drop tests for SNFT transportation accidents performed by BAM. This study involves comparing simulations with LSDYNA to results from a full-scale test with a German cask (Figure 12.7) as well as to results from full-scale and scaled tests with Japanese casks (Figure 12.8). The study involves performing sensitivity analysis to assess effects of mesh refinement, material properties and models, and finite elements on accelerations and strains. It also assesses the ability of the finite element analysis to bound response quantities of interest and related uncertainties, scale effects, and secondary effects as permitted by the test data.

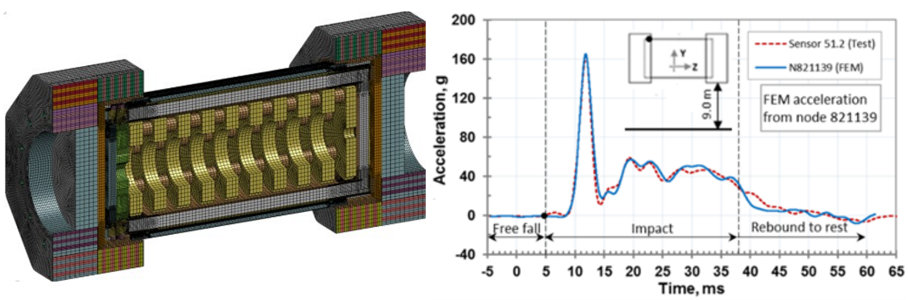

Figure 12.7 Full-scale finite element model of a SNFT cask and simulation acceleration versus test data.

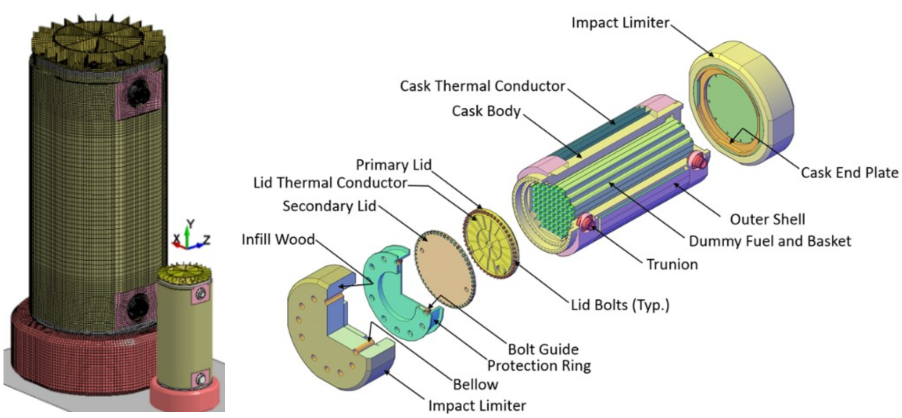

Figure 12.8 Full- and 40-percent scale finite element model of a SNFT cask and internal components modeled.

Status

This research was completed in 2017.

For More Information: Contact Hernando Candra, RES/DE, at Hernando.Candra@nrc.gov.

Steel Plate and Concrete Composite Modular Construction

Objective

New and advanced reactor designs have adopted a modular construction approach called Steel-Plate Composite (SC) structures as one of the major features for some of their structures. Structural SC walls consist of plain concrete with steel plates on both faces attached to the concrete by shear connectors and steel tie bars connecting steel plates on opposite faces of the wall (Figure 12.9). Floor and roof slabs of SC construction consist of a concrete slab with a steel plate attached to the bottom face of the concrete by shear connectors and conventional steel reinforcing bars in the concrete.

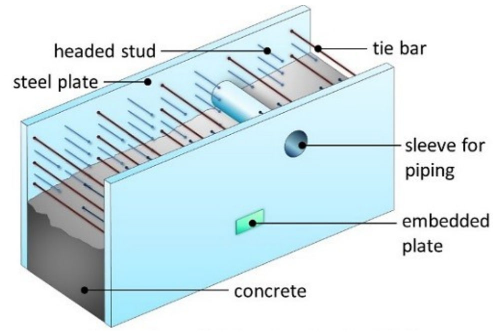

Figure 12.9 Illustration of SC wall construction.

Although a considerable body of international and U.S. research on SC structures exists and two international design standards have been available for a few years, current reviews by the U.S. Nuclear Regulatory Commission (NRC) for new reactor applications and license amendments that incorporate the use of SC construction are done on a case-by-case by basis. The American Institute of Steel Construction (AISC), through a multiyear effort, issued the first U.S. consensus standard for the design fabrication and erection of safety-related SC structures as Appendix N9 to Supplement 1 to the 2012 edition of the AISC standard N690, N690s1-15, *Specification for Safety-Related Steel Structures for Nuclear Facilities*.

The objective of this research is to review the technical bases for the American Institute of Steel Construction (AISC) standard N690s1-15 to inform the development of generic, up-to-date, and stable regulatory guidance for safety-related steel and SC structures other than containments. The expectation is that the industry and staff will use this guidance as needed for new reactor designs including small modular reactors and non-light water reactor reactors. The guidance also will be available for the review of license amendments for operating plants that invoke the new guidance and involve SC or steel structures.

Research Approach

The Office of Nuclear Regulatory Research (RES) sponsors and conducts information-gathering public meetings on the technical bases of the provisions of the N690s1-15. During these meetings, the staff and its contractors engage AISC experts to get clarifications in the technical bases and justification of certain provisions in the standard to inform resolution of their possible acceptance with or without exceptions. RES staff also reviews domestic and international experimental data (using international collaborative research opportunities, when needed) and conducts its own independent confirmatory research. RES contracted Brookhaven National Laboratory to assist the staff with the review of the technical basis for the N690s1-15 standard and awarded a research grant to Purdue University that assists confirmation of provisions for design to protect against impact loads such as those from tornado missiles.

Status

RES is reviewing information gathered following the most recent public meeting held in November of 2015 and is planning an additional public meeting. The staff also will review the AISC design guide for SC structures when published to obtain additional clarification on the implementation of the standard provisions. The projected date for completion of a draft regulatory guide for public review is in the last quarter fiscal year 2018.

For More Information: Contact Marcos Rolon Acevedo, RES/DE, at [Marcos.RolonAcevedo@nrc.gov.](mailto:Marcos.RolonAcevedo@nrc.gov)

Risk-Informed, Performance-Based Approach to Seismic Safety

Objective

This resarch advances possibilities in the use of risk-informed, performance-based approaches to seismic safety. In addition of obtaining possible new safety insights and regulatory clarity for efficient reviews, an anticipation is that these approaches would enable, for example, focusing resources promptly where they matter the most for safety so that regulation could be met with overall fewer resources and could lead to designs that are overall less costly to design, construct, operate, and maintain.

Research Approach

Design approaches, analysis methods, regulations, and other regulatory positions that support the seismic safety of the nuclear power plant fleet have been evolving towards approaches that increasing rely on performance-based and risk-informed ideas. This is in agreement with broader U.S. Nuclear Regulatory Commission staff efforts to advance risk-informed performance-based approaches in various areas of safety reviews including seismic safety. Figure 12.10 schematically illustrates the evolution towards a performance-based and risk-informed approach to the analysis and design of structures, systems, and components (SSCs) important to safety up to the present. The figure also illustrates the next steps in this process, which also respond to proposals by the industry, the design community, and developments by the U.S. Department of Energy. Initially, the research addresses:

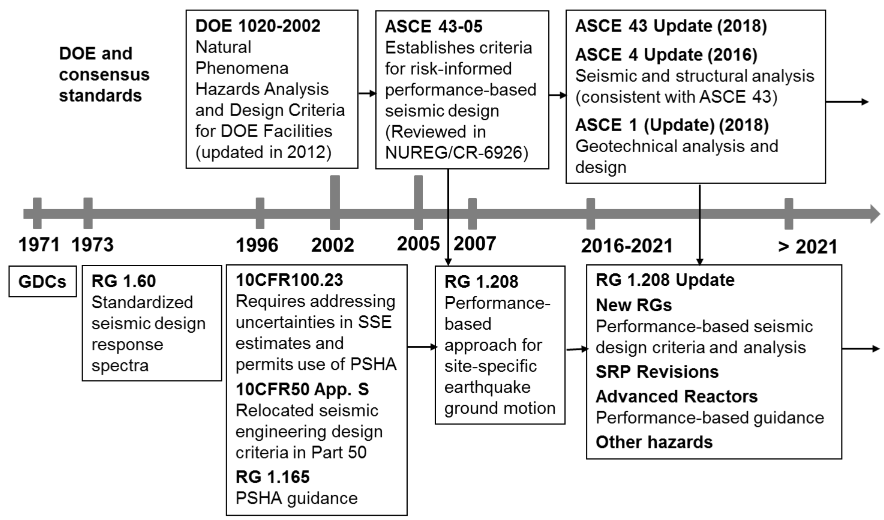

Figure 12.10 Evolution of risk-informed performance-based approach to seismic safety.

- Continuing the development of the technical bases for the update of the performance-based, site-specific, ground motion response spectra (update of Regulatory Guide [RG] 1.208).
- Bridging the gap between RG 1.208 and an integrated, performance-based approach to seismic/ structural analysis and design of SSCs important to safety.
- Completing the technical basis for the development of staff positions on recent and forthcoming updates of standard for performance-based seismic analysis and design of structures and components in nuclear installations. Standards of interest include the American Society of Civil Engineering (ASCE) standards for seismic analysis and design of nuclear installations, namely ASCE 4-16 and the forthcoming updates of ASCE 43 and ASCE 1.

Longer-term efforts would pursue possibilities as those in NUREG/CR-7214, “Toward a More Risk-Informed and Performance-Based Framework for the Regulation of the Seismic Safety of Nuclear Power Plants.”

Status

This research started in 2017 with the review of ASCE 4-16. The research coordinates this topic with topics in Chapter 10 of this report.

For More Information: Contact Jose Pires, RES/DE, at Jose.Pires@nrc.gov.

Structural Performance Cooperative Research

Objective

The objective of these activities is to ensure that the research in the area of structural safety accounts for and leverages ongoing relevant research at other U.S. and international institutions working on the safety of nuclear installations. The cooperative research activities that the U.S. Nuclear Regulatory Commission's (NRC's) Office of Nuclear Regulatory Research (RES) maintains with those organizations involve exchange of technical information, round robin analyses for benchmarking of modeling approaches, research workshops, and joint testing programs, as needed.

Aging Effects on Concrete Structures and Aging Management

Domestic Cooperative Research – This cooperative research reviews, assesses, and where needed augments the technical basis for review guidance related to concrete aging and degradation effects on the safety of nuclear power plants. The U.S. Department of Energy (DOE) and the Electric Power Research Institute (EPRI) conduct research to support aging management of concrete structures in nuclear power plants. RES maintains separate collaborative research agreements with each of these organizations to exchange technical information related to the safety of long-term operations. RES technical interchange meetings with DOE and EPRI staff have been concentrated on irradiation effects on concrete, alkali-silica reactions (ASR) effects, aging management, and supporting technologies like non-destructive examination.

International Cooperative Research – In relation to concrete irradiation, RES has a bilateral cooperative research agreement with the Nuclear Regulatory Agency of Japan to gather test data critical for validation, verification, and evaluation of irradiation damage models. RES participates in the International Committee on Irradiated Concrete in nuclear power plants to exchange information on characterization, quantification, and simulation of the effects of radiation on concrete structures. RES also is exploring the potential to research irradiation effects on steel-concrete bond via the Halden Research Program.

Regarding ASR, RES participates in a cooperative research activity of the Committee for the Safety of Nuclear Installations of the Nuclear Energy Agency (NEA/CSNI) called ASCET. This activity assesses modeling of the behavior of ASR-affected structures using test data provided by the Canadian Nuclear Safety Commission. RES also is finalizing a bilateral agreement with L'Institut de Radioprotection et de Sûreté Nucléaire (IRSN) to exchange research information on their respective concrete degradation research. The IRSN research program, ODOBA, is a 10- to 15-year project on reinforced concrete structures subject to various degradations.

Figure 12.11 Construction of EDF's 1/3-scale PCCV. testing mock up

Related to prestressed concrete containment vessels (PCCVs), RES participates in a NEA/CSNI benchmarking activity that analyzes the response and behavior of a 1/3-scale model of a PCCV (Figure 12.11) to study effects of shrinkage, creep, drying, and moisture transport on containment performance under accident scenarios. The activity uses information and data provided by Electricité de France from its multi-year VERCORS research program.

Impact Loads on Concrete Walls – RES 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 involves round-robin benchmarking analysis studies within the auspices of the CSNI. Benefits to the NRC include (1) reducing uncertainty in confirmatory assessments of impact loads on nuclear installations, (2) ensuring that assessments performed for U.S. reactors represent the state of the art, and (3) maintaining capability in this area. The IMPACT project is highly cost-effective in that all 10 participants share the testing costs. Both programs are expected to be completed in the third quarter of FY 2018.

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Chapter 13: Digital Instrumentation and Controls Research

Instrumentation and Controls Research Program

The instrumentation and controls (I&C) area continues to evolve, and the U.S. Nuclear Regulatory Commission (NRC) continues to refine its regulatory approach to comport with technological advancements in this area. As operating nuclear power plants (NPPs) upgrade their control rooms, analog equipment is being replaced with modern digital equipment including, but not limited to, flat screen operator interfaces and soft controls. Future plants will have highly integrated control rooms similar to Figure 13.1. The NRC continues to perform research that supports development of licensing criteria to evaluate new digital I&C systems. In an effort to continually improve the licensing process, the NRC has accepted a recommendation to balance short-term regulatory needs and long-term anticipatory research needs from the National Research Council report, "Digital Instrumentation and Control Systems in Nuclear Power Plants."

Figure 13.1 Highly Integrated Control Room.

Overall Program

The research program in this area is focused on supporting updates to the existing regulatory infrastructure that ensure effective and efficient reviews of digital technology proposed by licensees and suppliers.

Modernization of the Instrumentation & Control Regulatory Infrastructure

To meet regulatory challenges associated with digital technologies, the NRC has developed an Integrated Action Plan (IAP) for the modernization of the existing regulatory infrastructure. Regulatory challenges include ensuring that regulations for digital I&C technologies are performance based, technology neutral, unambiguous, and applicable to modification in operating plants and new reactor designs. The Office of Nuclear Regulatory Research (RES) supports these strategic modernization efforts, as needed, by performing confirmatory and anticipatory research to produce technical bases for addressing the regulatory challenges in a safe and secure manner.

Space Weather and Geomagnetic Disturbances

Electronic systems, and especially digital systems, are susceptible to electromagnetic phenomena such as transients induced by solar storms or coronal mass ejection events and electromagnetic interference from various sources. Such phenomena have the ability to affect a nuclear power plant directly and impact the grid. Impacts to the grid can affect both the ability of the grid to absorb the energy produced by the reactor and to provide power for achieving and maintaining a safe shutdown. This project assesses the potential magnitude, frequency, and impact of such phenomena in an effort to better understand how best to mitigate their consequences. This effort includes an ongoing effort to revise Regulatory Guide (RG) 1.180 on electromagnetic radio-frequency interference in safety-related I&C systems.

Cyber Security

RES supports continued review of RG 5.71, "Cyber Security Programs for Nuclear Facilities," to ensure that digital I&C systems can maintain safe operating environments in nuclear facilities. The NRC I&C Research staff participate in governmentwide, academic, and industry working groups that provide the latest information and tools to address cyber threats. Continuous evaluation is required to maintain expertise that can address concerns that arise in this rapidly changing environment.

Digital Instrumentation and Control (DI&C) — Modernization of the Instrumentation & Controls Regulatory Infrastructure

Objective

This research supports agency efforts described in the *Integrated Action Plan to Modernize Instrumentation and Controls Regulatory Infrastructure* (the IAP, ML17102B307) to review and update the regulatory infrastructure for applications of digital technology. The IAP was developed in response to Commission direction provided in SRM-SECY-15-0106 (ML16126A140). The Office of Nuclear Regulatory Research (RES) provides both research and regulatory support under this project, as needed.

Products from this research will provide the technical bases for addressing regulatory challenges associated with digital technologies in a safe and secure manner. Regulatory challenges include ensuring that regulations for digital instrumentation and controls (I&C) technologies are performance based, technology neutral, unambiguous, and applicable to modification in operating plants and new reactor designs.

Research Approach

Modernization Plan 4 (MP4) in the IAP describes the agency effort to assess and modernize the Instrumentation and Controls Regulatory Infrastructure. The effort is multifaceted and requires identifying and addressing technical and regulatory challenges through both confirmatory and anticipatory research projects.

To focus the objectives and desired outcomes of each project under this umbrella effort, the U.S. Nuclear Regulatory Commission staff is evaluating the impact of the current overall I&C regulatory infrastructure to impede safe adoption of digital technologies. Projects will broadly consider important insights from past review experiences, ongoing and past licensing review and research efforts, lessons learned from operating experience, insights from other safety-critical industries, and international perspectives. This research will be coordinated, as appropriate, with national and international strategic partners such as the U.S. Department of Energy national laboratories, private contractors, and international organizations like the Halden Research Project.

Active research to identify and address regulatory and technical gaps is expected to span multiple years. Success in this research will be reflected by a simpler, streamlined, and agile I&C regulatory infrastructure.

Status

The completion of the MP4 tasks is expected to span multiple years.

The RES staff is actively involved in identifying and clarifying the technical and regulatory relationships that impact the ability of the regulatory staff to evaluate digital components and systems for use in nuclear applications. This is a joint effort with the Office of Nuclear Reactor Regulation (NRR) and the Office of New Reactors (NRO).

The RES staff has performed an initial capability assessment of known contractors and strategic partners in anticipation that national and international collaborative efforts will be needed to produce the technical bases for addressing the technical and regulatory challenges.

For More Information: Contact Paul Rebstock, RES/DE, at Paul.Rebstock@nrc.gov.

Infrastructure Protection: Space Weather and Cyber Security

Objective

This research is partly driven by Executive Orders (EO) addressing Critical Infrastructure Protection (CIP). These include “Coordinating Efforts to Prepare the Nation for Space Weather Events” and “Strengthening the Cybersecurity of Federal Networks and Critical Infrastructure.” The U.S. Nuclear Regulatory Commission (NRC) has addressed some of these issues in rulemaking through actions such as petition for rulemaking (PRM) 50-96 from the Foundation for Resilient Societies.

Research Approach

The commercial nuclear industry is a vital part of the power generation segment of the Nation’s infrastructure. The NRC is the responsible agency for the regulation of this sector. To regulate this sector, the NRC is primarily focused on the CIP threat from cyber and electromagnetic (EM) attacks and the CIP risk from extreme Space Weather (SW). These events, while vastly different in initiation, are related in their expected mode of impact—potential loss of a facility system’s ability to perform required safety functions. In many cases, these events would result in a long-term loss of offsite power.

When threat/risks to facilities from such events as an EM weapon attack had been evaluated, the finding was that nuclear power plants (NPPs) would maintain a safe state. However, the threat/risk environment has now become very volatile. The NRC takes an informed approach to manage this risk/threat, evaluating the potential for an event along with the impact of the event against the bounding factor of safety/resilience. If the bounding factor no longer encompasses the potential event, then additional actions are required.

To validate the coverage of the bounding factor, a continuing survey of new or modified threat vectors and risk scenarios is required. Coordinated notification to responsible parties in other offices ensures that any new information that will cause a significant impact to the regulatory environment will be addressed. In addition, new and evolving requirements for CIP are reviewed and integrated into the process. The intent is to address CIP concerns through the regulatory framework that is already in place. Targeted regulations will be reserved for extreme scenarios. To identify these scenarios and to maintain confidence in the regulated facilities ability to operate safely, the ever-changing threat/risk environment must be continually reviewed. Although a small effort, this is vital to meeting the CIP requirements.

Status

The NRC officially evaluated the general risk from EM attacks starting in the early 1980s and updated the review in 2010. The NRC determined that projected SW and EM events would not exceed a facility’s ability to operate safely. In 2012, the NRC received PRM 50-96. After review and evaluation, it was decided that the increased risk would be addressed by Fukushima Action Items.

The NRC participates in the North American Electric Reliability Corporation Working Group on EM threats and the implementation of the new Federal Energy Regulatory Commission Reliability Rule. Information from the National Aeronautics and Space Administration, the Space Weather Prediction Center, and the U.S. Geological Survey’s development of 3D ground conductivity models are used in the ongoing evaluation/validation process. The NRC also participates in Infragard (a Federal Bureau of Investigation critical infrastructure threat discussion/alerting community) and the Cyber Security and Information Assurance Working Group providing valuable threat information to be used.

To summarize, this is a small effort with extreme importance in ensuring that events such as EM, cyber, or SW will not impact NRC sector critical infrastructure components’ ability to maintain a safe status. A continuing evaluation is required to maintain confidence in this position due to the changing environment.

For More Information: Contact Roy Hardin, RES/DE, at Leroy.Hardin@nrc.gov.

Digital Instrumentation and Control Cooperative Research

Objective

The U.S. Nuclear Regulatory Commission's (NRC's) ongoing research efforts for digital instrumentation and controls (I&C) include domestic and international cooperative and collaborative initiatives to leverage support for the NRC's I&C research and regulatory programs.

Research Approach

Domestic and international collaborative research efforts support exchanging technical information for digital systems. The products of these collaborations include technical review guidance, information to support regulatory-based acceptance criteria (e.g., through NUREG-series reports and publications and research information letters), assessment tools, methods, and standards.

Status

Figure 13.2 Halden Reactor Project.

Domestically, the NRC has a research Memorandum of Understanding with the Electric Power Research Institute in key technical areas that support collaborative research and sharing of research results. Work in the area of safety and security aspects of digital systems includes analytical assessment research to support safety evaluations of digital I&C systems (e.g., the use of new methods for hazard analysis). The NRC participates in interagency research and development coordination activities such as the Federal Networking and Information Technology Research and Development program to learn about relevant research in other Federal Government agencies such as the National Aeronautics and Space Administration, Food and Drug Administration, Federal Aviation Administration, U.S. Department of Defense, Department of Transportation, and National Institute of Standards and Technology.

Internationally, the NRC provides funding for research conducted by the Halden Reactor Project. Research collaboration for a safety demonstration framework, where a claim is supported with explicit evidence and reasoning, is intended to lead towards a performance-based regulatory infrastructure, substantially reducing the volume of information to be submitted by an applicant and the associated effort in preparing and reviewing a safety analysis report. In addition, Halden's project on safety demonstration framework will be used to assess methodologies that can be applied by vendors and licensees to demonstrate that their design is safe by regulators. The finalized methodology will be used for the development of a performance-based, safety-focused regulatory framework for I&C. The NRC participates in the Regulator Task Force on Safety Critical Software for nuclear reactors that comprises I&C experts from regulatory organizations and their technical support organizations representing the United Kingdom, Sweden, Finland, Belgium, Germany, Spain, South Korea, and Canada, to learn of issues encountered in their licensing experiences and their preventative common positions. The NRC also participates in the Software Certification Consortium, where it learns about relevant research enablers for third party certification in several safety-critical application categories (e.g., medical devices, aircraft, automobiles, and nuclear power plants). Staff work in standard development organizations such as the Institute of Electrical and Electronic Engineers and the International Electro-technical Commission supports international digital system standards harmonization and NRC knowledge management and regulatory efficiency improvements.

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Chapter 14: Electrical Research

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 preserve nuclear safety. The Fukushima event demonstrated the severe safety impacts of loss of offsite and onsite emergency power.

The U.S. Nuclear Regulatory Commission (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 confirmatory research of the critical design and performance aspects of these systems in operating NPPs and new reactors.

Power Source Reliability

The NRC is 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 further assess reliability and ruggedness of onsite normal and emergency power in plant safety systems.

Figure 14.1 Electrical switchgear.

Station Battery System Testing

The NRC has confirmatory research projects on NPP station batteries and DC distribution systems that include 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. Three different NRC-sponsored battery tests were conducted at Brookhaven National Labs for different purposes and outcomes. The test findings and results have been published in three documents: NUREG/CR–7148, "Confirmatory Battery Testing: The Use of Float Current Monitoring to Determine Battery State-of-Charge;" NUREG/CR-7188, "Testing to Evaluate Extended Battery Operation in Nuclear Power Plants;" and NUREG/CR-7229, "Testing Conducted to Determine the Battery and Battery Charger Short-Circuit Current Contributions on a DC Distribution System."

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 refining methods used for simulated aging of electrical equipment as well as condition monitoring to confirm that past equipment qualification practices were adequate and to identify suitable 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 concurrent effects of radiation and temperature aging with humidity to simulate close adherence to operating plant conditions. A number of condition-monitoring techniques will be applied during and following aging protocols to identify techniques that could measure progression of aging effects. Finally, the aged cables will be subjected to a simulated loss-of-coolant accident to determine the functional capability in a harsh environment for an extended period of operation.

Electrical Cable Qualification and Condition Monitoring

Objective

The U.S. Nuclear Regulatory Commission (NRC) confirmed in its review of responses to 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. 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. The data collected from GL 2007-01 showed that a significant number of failures occurred under normal service conditions within the service interval of 20-30 years, much before the renewed license period and before the end of the normally 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 evaluate the effectiveness of commonly used cable condition-monitoring techniques for specific insulation materials; assess qualified life predictions to 50, 60, and 80 years of operation; and enhance the understanding of the condition-based environmental qualification methodology.

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 80 years of operations. For better estimates of cable performance, the aging will be performed concurrently with a low radiation dose and low temperature for 24 months to produce homogeneous degradation in the cable samples closely adhering to actual plant 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 their capability to detect continued aging degradation of cables during and after the accident.

Status

The NRC has contracted with the National Institute of Standards and Technology (NIST) to execute this research. Currently, NIST is acquiring the services of Sandia National Laboratories radiation facilities to perform the thermal and radiation aging of the cable samples. The research is scheduled to be completed in late 2020 with the publication of a NUREG/CR.

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Battery-Testing Program

Objective

The U.S. Nuclear Regulatory Commission (NRC) conducted confirmatory nuclear station battery testing at Brookhaven National Laboratory (BNL). This research program was intended to validate if the batteries generally used in the nuclear industry remained in a fully charged condition within the full span of the service life in operational readiness and whether the charging current is a suitable indicator of a fully charged condition for lead-calcium batteries. This research also verified performance capabilities of the batteries for extended loss of alternating power (ELAP) conditions. Additional testing in response to an issue at an operating plant verified 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 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 to ensure fully charged state, (2) evaluation of the use of charging current to monitor battery capacity, and (3) identification of the conditions when a battery can be returned to service to meet its design requirements.

To evaluate the batteries' response to extended loss of alternating current power, the staff tested plant DC profiles under ELAP from four nuclear power plants (three pressurized-water reactors and one boiling-water reactor).

To enhance understanding of short-circuit contributions, a series of tests were conducted with batteries and battery chargers paralleled to a fault that occurs on a DC system. The confirmatory research to implement the approach as outlined was conducted at Brookhaven National Laboratory (BNL).

Status

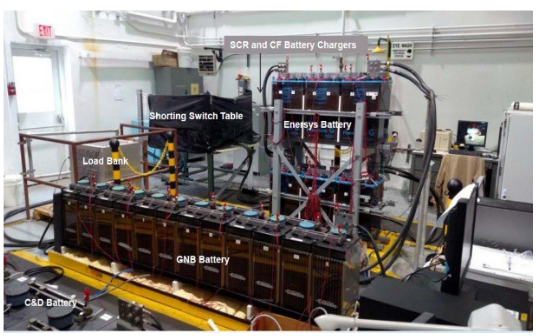

Figure 14.2 BNL battery facility.

Figure 14.2 shows the BNL battery test facility. All tests have been completed, and results for the confirmatory battery testing are documented as follows: NUREG/CR - 7148, "Confirmatory Battery Testing: The Use of Float Current Monitoring to Determine Battery State-of-Charge;" the extended battery testing is documented in NUREG/CR- 7188, "Testing to Evaluate Extended Battery Operation in Nuclear Power Plant;" the short-circuit contributions testing is documented in NUREG/CR-7229, "Testing to Evaluate Battery and Battery Charger Short-Circuit Current Contributions to a Fault on the DC Distribution System."

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Electrical Cooperative Research

Objective

The U.S. Nuclear Regulatory Commission (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. The NRC staff actively participates in both domestic and international research collaborations to advance the agency's mission.

Research Approach

The Office of Nuclear Regulatory Research (RES) developed a comprehensive Electrical Systems Research Program Plan that 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 to support NRC research objectives and to improve the quality of research projects.

Figure 14.3 EPRI headquarters.

Status

Domestically, RES has a research Memorandum of Understanding with the Electric Power Research Institute (EPRI) supporting two of the electrical research programs—Battery Testing research and 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 the 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. The 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 participates in standards development activities with organizations such as the Institute of Electrical and Electronic Engineers and the International Electrotechnical Commission to harmonize international nuclear power plant electrical system standards, facilitate NRC knowledge management, and gain regulatory efficiency and effectiveness.

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Chapter 15: International and Domestic Cooperative Research

The U.S. Nuclear Regulatory Commission's (NRC's) Office of Nuclear Regulatory Research (RES) has been implementing cooperative agreements with international and domestic organizations over the last 25 years. 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 35 countries and the Organization 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 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 th 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.

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, four nuclear safety databases, and joint research projects) and the Committee on Radiation Protection and Public Health. The RES Deputy Director is on the CSNI Bureau, 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 U.S. lead in the International Atomic Energy Agency's (IAEA's) Nuclear Safety Standards Committee, RES coordinates NRC contributions to the many IAEA safety standards requirements and guidance documents.

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, improve analytical methods, enhance assessments of plant risk, and 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 U.S. Nuclear Regulatory Commission (NRC) and its predecessor, the U.S. Atomic Energy Commission, have been participating in the Organization 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 2018–2020. 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.

Figure 15.1 Halden Boiling Water Reactor.

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 is used to review and audit industry analyses. The NRC is particularly interested in loss-of-coolant accident (LOCA) tests that 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. Of particular interest to the NRC, HRP's materials testing program will perform irradiation and testing to investigate the irradiation-assisted stress-corrosion cracking of weld materials that were harvested from the decommissioned Zorita reactor in Spain.

Man-Technology-Organization Research

The HRP research facilities also include several labs for Man-Technology-Organization (MTO) research. Among those is the Halden Man Machine Laboratory (HAMMLAB). 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 effects of computerized procedures on operator performance, integrated system validation and verification, and coordination and decisionmaking in severe accident management. This research will contribute to the technical basis for human factors guidance and human reliability analysis. The MTO laboratory also conducts research on the safety of digital instrumentation and controls (I&C) where the NRC's primary interest is developing a safety demonstration framework that enables a risk-informed, performance-based regulatory infrastructure.

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 about 86 percent 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 U.S. Nuclear Regulatory Commission’s (NRC’s) Office of Nuclear Regulatory Research (RES) 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

The CADAK Project is the continuation of the cable aging part of the Cable Aging Project (SCAP). CADAK aims to establish the technical basis for assessing the qualified life of electrical cables in light of the uncertainties identified following the initial qualification testing. In addition, CADAK will investigate the adequacy of the margins and their ability to address the uncertainties.

The following are the CADAK technical objectives:
1. Develop a database that contains records of cable failures and condition-monitoring test results.
2. Estimate the remaining qualified lifetime of cables used in nuclear power plants.
3. Analyze the information collected on cable failures and condition-monitoring test results to develop topical reports in coordination with the OECD/NEA and CSNI.
4. Promote the sharing of ongoing and future research among the participating country members.

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.
3. Provide users with tools to apply the database for regulatory decisionmaking.
4. Manage a relational event database that 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 3-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 3-year term, from December 2014 to December 2017, included 11 participants: Canada, Chinese Taipei, Czech Republic, France, Germany, Japan, Korea (Republic of), Slovak Republic, Spain, Switzerland, and the United States. The third 3-year term is currently under negotiation but is expected to begin in 2018.

Fire Incidents Records Exchange (FIRE) Project

Fire can be an important contributor to core damage and plant damage states; however, 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 and currently has 13 participating countries: Belgium, 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 by 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, root cause information, extinguishment, 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 United States. The database facilitates the development of qualitative insights into the root causes of fire events that can then be used to derive approaches for their prevention or mitigation.

This project is also facilitating improvements of existing international reporting systems trending and indicators for risk based inspections. The database project also provides a valuable forum for international communication on other fire safety issues and led to the identification of the problem of High Energy Arcing Faults (HEAF) in electrical equipment which matured into a separate NRC led OECD Project on Joint Analysis of ARC Faults (JOAN of ARC).

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, 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.
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. In addition to analysis of the above listed components, the ICDE project has also developed topical reports on important issues related to CCF and risk assessment. These include multi-unit CCFs, cross-component and cross-system CCFs, and CCFs due to external environmental factors.

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The Office of Nuclear Regulatory Research (RES) supports the mission of the U.S. Nuclear Regulatory Commission (NRC) by providing technical advice, tools, and information to identify potential safety and security issues and resolve them as appropriate, assessing risk and other nuclear safety and security issues, and developing and coordinating regulatory 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 RES staff uses its own expertise and collaborates with partner offices at the NRC, commercial entities, national laboratories, other Federal agencies, universities, and international organizations.

This NUREG describes research being conducted by RES 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. RES has organized this collection of information sheets by topical areas that summarize projects currently in progress. Each sheet provides the name(s) of the RES technical staff who can be contacted for additional information.

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