# Chapter 6: Human Factors and Reliability Research



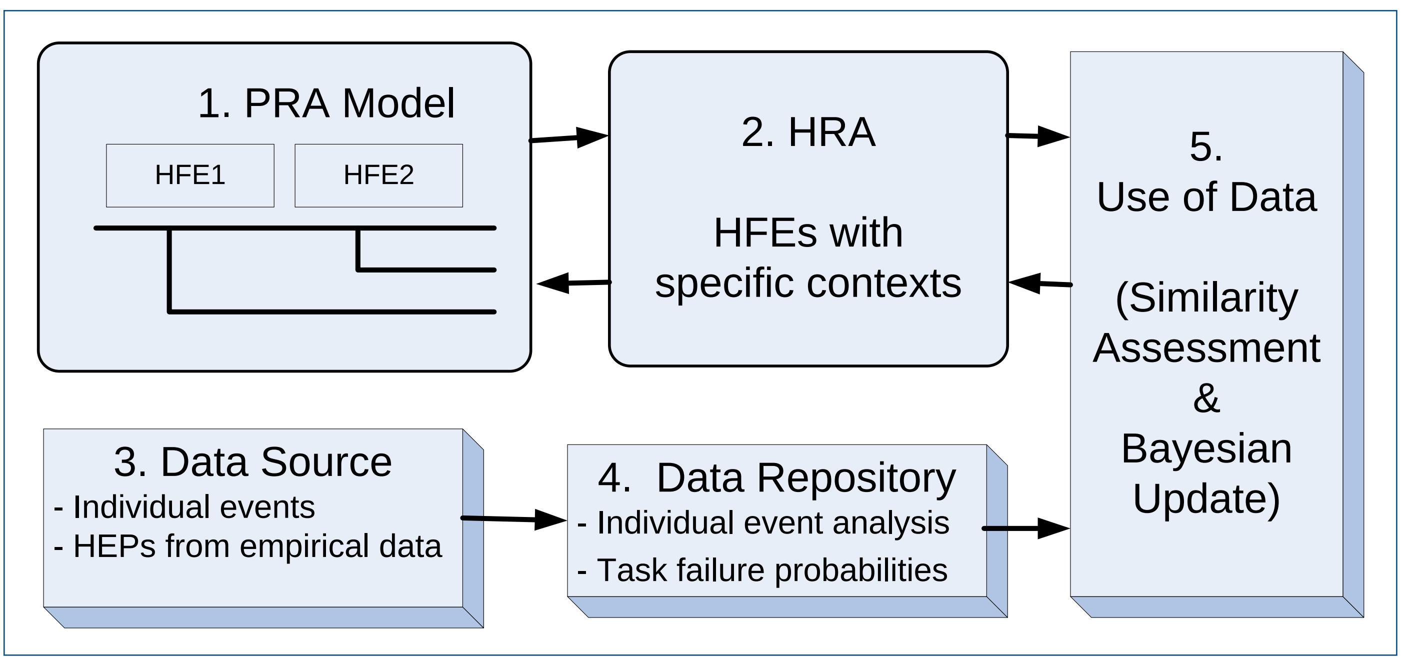



**Diagram illustrating the concept of using empirical data to inform the human error probabilities of human failure events**

# Human Reliability Analysis Data Repository

## Background

Consistent with the NRC's policy statements on the use of probabilistic risk assessment (PRA) and for achieving an appropriate PRA quality for NRC risk-informed regulatory decisionmaking, the NRC has established a phased approach to probabilistic risk assessment quality (see SECY-04-0118, "Plan for the Implementation of the Commission's Phased Approach to Probabilistic Risk Assessment Quality," issued July 2004, and SECY-07-0042, "Status of the Plan for the Implementation of the Commission's Phased Approach to Probabilistic Risk Assessment Quality," issued March 2007). The phased approach to PRA quality includes an action plan for stabilizing the PRA quality expectation and requirements to address PRA technical issues. Human reliability analysis (HRA) is an important PRA element. Data are key to HRA quality. The Commission identified the need for HRA data in staff requirements memorandum (SRM)-M061020, "HRA Model Differences," dated November 8, 2006, and SRM-M090204B, dated February 18, 2009.

Currently, the Office of Nuclear Regulatory Research (RES) maintains the Human Event Repository and Analysis (HERA) system to provide HRA data. The HERA data source relies on analyzing past events and simulator exercises. In order to more effectively support human error probability (HEP) estimates, enhancements to the current HERA are necessary in such areas as data collection methodology, data sources, and the use of collected data to inform HRA. A data framework providing enhancements to these areas has been proposed and is under discussion for development.

## Objective

This project seeks to develop a method to effectively use empirical data to support HRA, with emphasis on HEP estimates.

## Approach

The NRC 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 (HFEs) of interest.

Unlike hardware reliability studies and because of the variability of the HFEs, it is not practical to collect the total number of successes and failures when calculating HEPs. One solution involves grouping together empirical data on tasks with similar human performance characteristics to inform the HEPs. This approach could significantly increase the usability of the empirical data collected.

The staff's approach is to use the six functional elements in most HRA methods for calculating HEPs to form a human performance profile (HPP). The six elements are task analysis (or task decomposition), generic tasks, error modes or error mechanisms, performance-shaping factors, task dependency, and recovery from human failure. The HPP will be used to characterize empirical data and the HFE of interest and to measure the similarity between the HFE of interest and the empirical data. Figure 6.1 illustrates this concept.

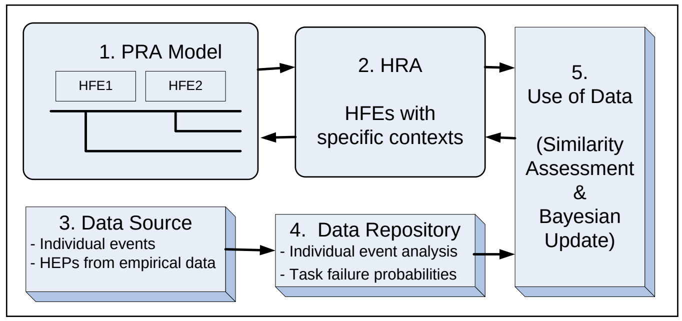



**Figure 6.1 Diagram illustrating the concept of using empirical data to inform the human error probabilities of human failure events**

In the upper portion of Figure 6.1, the HFEs are specified in PRA or HRA (Blocks 1 and 2). Based on the contextual information provided in PRA or HRA, the HPP of the HFE can be specified. The lower portion of Figure 6.1 shows the likely data types (Block 3). These include analyses of individual events (by identifying the key tasks and corresponding human performance in the events) and task failure probabilities. Each instance of success or failure in performing key tasks in past events and each task failure probability are considered as a data point. Each data point is characterized by the HPP and stored in a data repository (Block 4). The data points with similar HPPs to HFEs of interest can be identified from the repository to inform the HFE's HEP.

Key technical challenges to the approach include developing the HPP, similarity measurements, and use of imperfect data.

**For More Information**

Contact Y. James Chang, RES/DRA, at 301-251-7589 or James.Chang@nrc.gov

# Human Reliability Analysis Model Differences

## Background

The NRC's Office of Nuclear Regulatory Research (RES) is supporting the Advisory Committee on Reactor Safeguards (ACRS) to address the November 8, 2006, staff requirements memorandum (SRM-M061020) in which the Commission directed the ACRS to "work with the staff and other stakeholders to evaluate different human reliability models in an effort to propose a single model for the agency to use or guidance on which model(s) should be used in specific circumstances." RES is addressing this issue through collaborative work with EPRI (Electric Power Research Institute), initiated under the RES memorandum of understanding with EPRI on PRA.

## Approach

To address the issue, the project is pursuing a formalization approach and a quantification tool capable of performing HRA in a consistent and efficient manner. The formalization approach aims to build a foundation for HRA that uses the current understanding of human performance and is consistent with the overall PRA framework from the perspective of both failure modeling and estimation of failure probabilities. This approach introduces the crew response tree (CRT) concept, which depicts human failure events in a manner parallel to the PRA event tree process. CRTs provide a structure for identifying the context associated with the human failure events under analysis and use a human information processing model as a platform to identify potential failures.

This approach incorporates behavioral science knowledge by providing the decompositions of human failures, failure mechanisms, and failure factors from both a top-down and bottom-up perspective. The bottom-up approach reflects findings from scientific papers documenting theories, models, and data of interest. The CRT structure and associated lower level models provide a roadmap for incorporating the phenomena with which crews would be dealing, the plant characteristics (e.g., design, indications, procedures, training), and the plant's human performance capabilities (understanding, decision, action). The work aims to create rules, and potentially template-based guidance, for a consistent, efficient, and effective analysis.

For quantification, the formalization approach uses the typical PRA conditional probability expression, delineated to a level adequate for associating the probability of a human failure event with conditional probabilities of the associated contexts, failure mechanisms, and underlying factors (e.g., performance-shaping factors). This mathematical formulation can be used to directly estimate HEPs using various data sources (e.g., expert estimations, anchor values, simulator data, historical data) or can be modified to interface with existing quantification approaches. However, the quantification approach is still under exploration.

The staff anticipates that the methodology will be developed and available for public review and comment by November 2011.

The NRC's costs represent only a fraction of the actual costs for both the international empirical study and the collaborative work with EPRI for addressing SRM-M061020 on HRA model differences. Through these collaborative efforts, the NRC is also able to take advantage of extensive domestic and international PRA and HRA expertise from recognized academics and practitioners.

**For More Information**
Contact Erasmia Lois, RES/DRA at 301-251-7573 or Erasmia.Lois@nrc.gov

# Improving Human Reliability Analysis Methods by Using Simulator Runs

## Background

As part of its efforts to improve human reliability analysis (HRA), the Office of Nuclear Regulatory Research (RES) participates in and supports the International HRA Empirical Study to benchmark HRA models by comparing HRA results to empirical data generated through crew simulator runs. Although the documentation of this study is not yet complete, its findings indicate areas for improvement in HRA methods and practices. But because the study is based on the results of simulator runs using European crews at the Halden Reactor Project (HRP) simulator, the issue of the applicability of the study results to U.S. nuclear power plant crews has been raised.

In its February 2009 staff requirements memorandum (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. RES's benchmarking work is responsive to SRM-M090204B.

The NRC established a memorandum of understanding with a U.S. utility that volunteered to participate in this study and offered simulator facilities, crews, and expertise to support the design and execution of the experimental runs. As a result, a new study was initiated that the HRP staff supports with expertise in the design and execution of simulator runs, as well as the collection and interpretation of crew performance data.

The objective of this new study is to evaluate a specific set of HRA methods used in regulatory applications by comparing HRA predictions to crew performance in simulator experiments performed at a U.S. nuclear power plant. The results will be used to accomplish the following:

- Determine the potential limitations of data collected in non-U.S. simulators when used to evaluate U.S. applications.
- Improve the insights developed from the International HRA Empirical Study.

## Approach

The study approach consists of the following four steps:

### 1. Experimental Design and Performance of Simulated Scenarios

The experimental design is focused on collecting information on the predictive power and consistency of HRA methods—A Technique for Human Even Analysis (ATHEANA), Standardized Plant Analysis Risk—Human Reliability Analysis Method (SPAR-H), Technique for Human Error Rate Prediction/Accident Sequence Evaluation Program (THERP/ASEP), and Cause-Based Decision Tree (CBDT) in particular—through analysis of crew performance in simulated nuclear power plant accident scenarios. It stipulates the collection of information to be used by HRA analysts to evaluate the human failure events (HFEs) involved in the scenarios and to estimate the human error probabilities (HEPs).

The design includes three accident scenarios. The design addresses the plant status before the initiating event, the initiating event, and the associated plant design capabilities and operational characteristics to deal with the event, including procedural guidance; the predetermination and definition of the HFEs to be analyzed for each scenario and associated success criteria; the identification of human performance metrics; the development of crew performance collection protocols and questionnaires to support documentation of observed crew performance; and the development of an information package containing basic probabilistic risk assessment (PRA) and HRA information to be provided to the HRA teams.

The actual experiment consists of the running of the scenarios and the collection and documentation of observations about plant behavior and crew performance by experts (typically plant trainers and PRA/HRA experts). In addition to live observations, crew performance observations are collected through videotapes and debriefings of both the crews and the plant experts who observed the runs.

The experimenters evaluate crew performance by analyzing the information collected during the experiment according to predefined protocols and performance metrics. This part of the study is supported by the staff of the HRP.

## 2. Information Collection and Evaluation of HEPs by HRA teams

Each HRA method is applied by two or three HRA teams composed of NRC and contractor staff. The HRA teams visit the plant to interview plant personnel, view simulator runs (other than the study simulations), and collect relevant plant information. On the basis of the information collected, the teams use their selected HRA methods to perform predictive analysis and to estimate HEPs for the HFEs involved in the simulated scenarios, document the results, and submit them for review and evaluation.

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

## 3. Evaluation of the HRA submittals

An independent group of experts reviews the submitted analyses and compares them to the observed simulator data. These experts perform method-to-method and HRA team-to-team comparisons to determine if and how method differences and analyst differences influence the HRA results. Their analysis includes both qualitative and quantitative comparisons.

Qualitative comparisons examine the extent to which HRA analysts, using their methods, were able to identify key drivers (such as misdiagnosis of equipment failures or lack of adequate procedural guidance for performing the required actions) that could influence the crew's capability to accomplish the required actions. Through such comparisons, the experts identify (1) method limitations with regard to guiding analysts to identify important drivers of human performance, and (2) method limitations with regard to ensuring a consistent use of the method by different analysts (intra-analyst consistency).

Quantitative comparisons involve (1) the ranking of the estimated HEPs, (2) the ranking of the human actions in terms of the level of difficulty that crews appear to have experienced during the simulation, and (3) comparison of the resulting ranking in (1) and (2). These comparisons allow the experts to examine whether or not inconsistencies in ranking stem from the following causes:

- the extent to which the quantification tool can incorporate the important drivers of human performance identified through the qualitative analysis (e.g., the tool allows the use of only a few performance shaping factors in the estimation of HEPs)
- the extent to which the quantification tool can provide a consistent and traceable process to estimate HEPs
- the analysts' capability to correctly apply the tool.

## 4. Documentation of the Results

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

RES expects this study to be completed by September 2011.

**For More Information**
Contact Erasmia Lois, RES/DRA at 301-251-7573 or Erasmia.Lois@nrc.gov

# Pilot Testing of Human Reliability Analysis-Informed Training and Job Aid for NRC Staff Involved with Medical Applications of Byproduct Materials

## Background

In 2003, the Office of Nuclear Material Safety and Safeguards (NMSS) provided the Office of Nuclear Regulatory Research (RES) with a user need for developing human reliability analysis (HRA) capability specific to materials and waste applications (NMSS-2003-003). In this memorandum, NMSS requested two phases of work. Both phases were completed in December 2008.

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

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

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

Follow-up work to pilot the HRA-informed job aid and training materials began in spring 2010.

## Approach

The overall objective is to develop HRA-informed job aids and associated training for NRC staff involved with medical applications of byproduct materials. Although prototypes of the HRA-informed job aid and training materials have been developed, instructions on how to use these tools for specific NRC tasks (e.g., inspections, license reviews) were not developed. Consequently, interaction with NRC staff from the regions, as well as the continued involvement of staff at NRC Headquarters is required in this pilot testing phase of development.

RES is currently making plans for pilot testing of both products at NRC Region I.

### PILOT TESTING TASKS

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

- initial updates to HRA-informed training and job aid (with respect to recent events and new gamma knife technology)
- initial interactions with NRC Region I staff
- onsite HRA-informed training
- onsite demonstration of HRA-informed job aid
- selection of candidates for trial use of HRA-informed job aid
- trial use of HRA-informed job aid
- feedback on trial use
- updates to HRA-informed job aid and associated training (based on feedback)

By the end of calendar year 2010, the first two tasks should be complete and preparations started for Task 3.

**For More Information**
Contact Susan Cooper, RES/DRA at 301-251-7604 or Susan.Cooper@nrc.gov

# Qualitative Human Reliability Analysis for Spent Fuel Handling

## Background

In 2003, the Office of Nuclear Material Safety and Safeguards (NMSS) provided the Office of Nuclear Regulatory Research (RES) with a user need for developing human reliability analysis (HRA) capability specific to materials and waste applications (NMSS-2003-003). In this memorandum, NMSS requested two phases of work, the first of which is completed.

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

- development of HRA methods specific to NMSS needs
- guidance for evaluating the effectiveness of administrative controls
- guidance on good practices for implementing HRA
- guidance for reviewing HRAs
- assistance in incident significance assessments

Initial Phase 2 work on this project began investigating development of HRA methods specific to NMSS needs and guidance on good practices for implementing HRA.

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

## Approach

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

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

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

The result of this work was a July 2006 Sandia National Laboratories (SNL) letter report describing potential vulnerabilities and possible scenarios that could lead to misloads and cask drops.

Currently, the project is developing further HRA-informed insights on cask drops. It is expected that this work will provide useful input to future NRC inspections and reviews.

The current schedule for deliverables for both efforts is the following:

- draft NUREG/CR on initial efforts for misloads and cask drops—September 2010
- draft NUREG/CR on recent expanded efforts on cask drops—September 2010
- preparation of both final NUREG/CRs—December 2010

Continued interactions between NMSS and RES staff are planned as these deliverables are completed.

**For More Information**
Contact Susan Cooper, RES/DRA at 301-251-7604 or Susan.Cooper@nrc.gov

# Human Performance for Advanced Control Room Design

## Background

With the renewed interest in nuclear energy, there are plans to begin constructing new plants within the next several years. The new generation of plants will differ from the existing fleet in several important ways, including the design of the reactors, the instrumentation and control (I&C), and the human-system interface (HSI). Figure 6.2 illustrates one conceptualization of an advanced control room (CR) design. Taken together, these technological advances will lead to concepts of operation that are different from those found in currently operating nuclear power plants. The potential benefits of the new technologies should result in more efficient operations and maintenance. However, if the technologies are poorly designed and implemented, there is the potential they will reduce human reliability, increase errors, and negatively impact human performance—resulting in a detrimental effect on safety. To address these concerns, the NRC sponsored a study to identify human performance research that may be needed to support the review of licensee's implementation of new technology in new and advanced nuclear power plants.

## Approach

To identify the research issues, current industry trends and developments were evaluated in the areas of reactor technology, I&C technology, HSI integration technology, and human factors engineering (HFE) methods and tools. These four research issues were then organized into seven HFE topic areas: (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 (e.g., HFE, I&C) and backgrounds (e.g., vendors, utilities, research organizations) prioritized the issues. Sixty-four issues were distributed among four categories, with 20 research issues placed into the top priority category.

NUREG/CR-6947, "Human Factors Considerations with Respect to Emerging Technology in Nuclear Power Plants," issued October 2008, documents the results of the study. The report contains a summary of the high-level topic areas, the research issues in each topic area, the priorities for each issue, and a human performance rationale that describes the reason why each research issue is relevant. The findings from this study are being used to develop a long-term research plan addressing human performance within these technology areas for the purpose of establishing a technical basis from which regulatory review guidance can be generated.

Of the 20 research projects identified as having a priority 1 research need, four have been completed, five are currently underway, and an additional three projects are scheduled to begin this year. Descriptions of the five projects that are underway are provided below.

**Figure 6.2 One conceptualization of an advanced control room design**

### Advances in Human Factors Engineering Methods and Tools

The methods and tools used to design, analyze, and evaluate the HFE aspects of nuclear power plants are rapidly changing. A previous study identified the current trends in the use of HFE methodologies and tools, identified their applicability to nuclear power plant design and evaluation, and determined their role in safety reviews conducted by the NRC. The study identified seven categories of methods and tools for which additional review guidance may be needed, including (1) application of human performance models, (2) use of virtual environments and visualizations, (3) analysis of cognitive tasks, (4) rapid development engineering, (5) integration of HFE methods and tools, (6) computer-aided design, and (7) computer applications for performing traditional analyses. One outcome of this project to date has been the development of detailed guidance for applying human performance models to the evaluation of nuclear power plant designs. The next phase of the study will provide human factors (HF) guidance for an additional two methods and tool categories.

### Roles of Automation and Complexity in Control Rooms

The overall level of automation in advanced nuclear power plants is expected to be much higher than in plants currently operating in the United States. It is important that the staff be cognizant of current practices and trends in the use of automation in nuclear power plant CRs and understand the influences of automation on CR design, human performance, and conduct of operations. A previous study, "Human-System Interfaces (HSIs) to Automatic Systems," developed a general framework for characterizing automation systems and developed HFE criteria for evaluating automation designs. The present study will further the state of the art by examining the impact of automation on CR design, specifically the impact of automation on (1) operator performance during normal, abnormal, and emergency operations; (2) the reliability of operator's use of automation systems, including existing methods for assessing impacts; and (3) operator performance when the automation fails or is in a degraded state.

### Human Factors Guidance for the Assessment of Computerized Procedures

Applicants for new and advanced reactor design certifications are proposing to incorporate computer-based procedure capabilities as part of their main CR designs. The potential forms of implementation can range from basic applications that are limited to displaying static representations of procedures to those that provide dynamic displays of procedures in conjunction with relevant plant status and process data, context-dependent decision aids, soft controls, and the capability to implement automated sequences of procedure steps. Although the challenges and human factors considerations increase with the level of functionality of these applications, even the most basic application requires consideration of how it will be integrated with other elements of the CR design, how the implementation might affect the roles and responsibilities of the operating crew and standards for conduct of operations, how the operators will transition to backup procedures upon loss of a computer-based procedure system, what the potential failure modes of the application will be, and how those failure modes will be addressed to ensure that acceptable levels of human performance will be maintained. This project will review applicable research literature and operating experience and develop a technical basis document for the development of review guidance that addresses the key issues associated with the use of CR computerized procedures.

### Human Factors Aspects in Concepts of Operations for Modular Designs

Advances in nuclear power plant technology have set the stage for changes to traditional concepts of operations (CONOPS). The CONOPS of new reactor designs introduce such safety-critical performance considerations as the operation of multiple reactors by a reduced crew. The objective of this project is to examine the human factors aspects associated with the monitoring and control of multimodular plants and to provide a technical basis for evaluating the impacts of evolving CONOPS on human performance. The regulatory documents for reviewing modular designs will also be assessed to identify areas that need additional technical basis or guidance to facilitate the staff review of CONOPS for modular reactor designs.

### Update Existing Human Factors Engineering Regulatory Guidance

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

**For More Information**

Contact Stephen Fleger, RES/DRA at 301-251-7905 or Stephen.Fleger@nrc.gov

# Support For Implementation of 10 CFR Part 26 Fitness-for-Duty Programs

## Background

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

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

Most recently, the NRC, with extensive stakeholder input, published an amended, reorganized, and updated rule. The amended 10 CFR Part 26 was published in the *Federal Register* on March 31, 2008. It is organized into 12 subparts that group together related requirements. The NRC permitted licensees and other entities to defer implementation of the majority of the rule's requirements until March 31, 2009, and granted an additional 6 months to implement the rule's new fatigue management requirements.

## Approach

The Office of Nuclear Regulatory Research (RES) participates in a multidisciplinary team of NRC staff that is supporting a myriad of agency initiatives and efforts to facilitate education about the rule and its implementation.

### Fatigue Regulatory Guide

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

### Training Development

To ensure that implementation efforts among the regions and various offices are coordinated and consistent, RES staff and its contractors have developed training materials for inspectors and other NRC staff involved in implementing 10 CFR Part 26. To date, the training has been developed, pilot-tested, and supplemented with computer-based training specifically focused on the fatigue management requirements.

### FFD Web Site Update

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

### Inspection Procedures

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

### Technical Bases for Alternate Specimens and Fatigue Technologies

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

## Future Updates to 10 CFR Part 26

The Commission directed the NRC staff to initiate a new 10 CFR Part 26 rulemaking after publication of the March 31, 2008, amended and revised rule. The Commission asked the NRC staff to review specific elements of the rule related to the technical basis and to evaluate including licensee quality control, quality verification, and quality assurance personnel in the fatigue provisions of 10 CFR Part 26. The RES staff is continuing to provide its technical expertise to staff engaged in the new rulemaking.

**For More Information**

Contact Valerie Barnes, RES/DRA at 301-251-7585 or Valerie.Barnes@nrc.gov

# Safety Culture

## Background

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

### Goal of Safety Culture Activities

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

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

### Industry Safety Culture Assessment Initiative

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

**For More Information**
Contact Valerie Barnes, RES/DRA at 301-251-7585 or Valerie.Barnes@nrc.gov

# Chapter 7: Fire Safety Research

Fire Probabilistic Risk Assessment Methodology for Nuclear Power Facilities

Fire Human Reliability Analysis Methods Development

Fire Modeling Activities

Cable Heat Release, Ignition, and Spread in Tray Installations During Fire (CHRISTIFIRE)

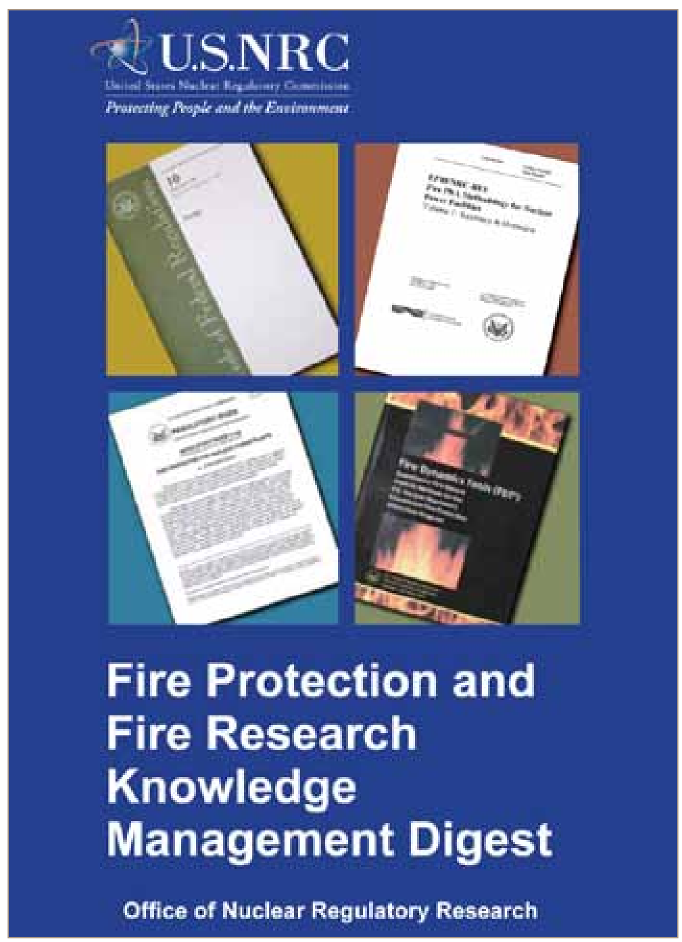
Direct Current Electrical Shorting in Response to Exposure Fire (DESIREEFIRE)

Fire Effects on Electrical Cables and Impact on Nuclear Power Plant System Performance: Phenomena Identification and Ranking Table (PIRT) and Expert Elicitation Programs

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

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

Fire Research and Regulation Knowledge Management




**NUREG/BR-0465**

# Fire Probabilistic Risk Assessment Methodology for Nuclear Power Facilities

## Background

The results of the individual plant examination of external events (IPEEE) program conducted in the 1990s and actual fire events indicate that fire can be a significant contributor to nuclear power plant (NPP) risk, depending on design and operational conditions. In particular, these studies show that failures of fire protection defense in depth (i.e., failure to prevent fires, failure to rapidly suppress fires, or failure to protect plant systems to provide stable, safe shutdown) can lead to risk-significant conditions. Fire probabilistic risk assessment (PRA) provides a structured, integrated approach to evaluate the impact of failures in the fire protection defense-in-depth strategy on safety. Figure 7.1 illustrates a simplified fire PRA event tree representing different sets of fire damage and plant response. The fire PRA directly addresses technical issues such as fire ignition frequency, detection and suppression, fire damage to diverse and redundant trains of core cooling equipment, circuits (i.e., spurious actuations), and plant response.

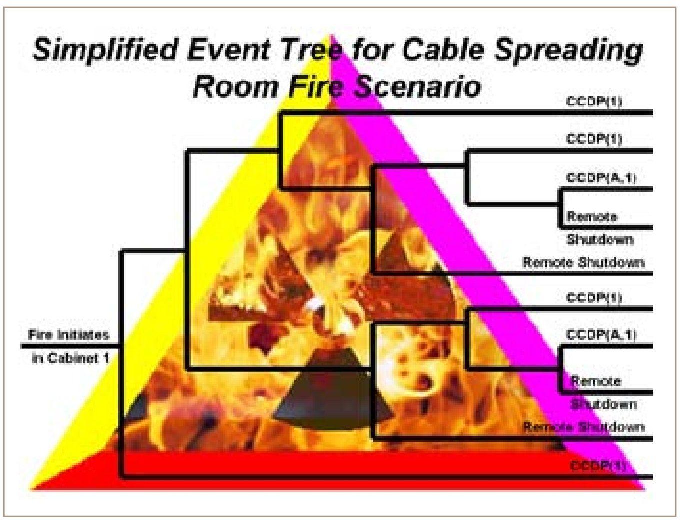



**Figure 7.1 Simplified fire PRA event tree representing different sets of fire damage and plant response**

In 1995, the U.S. Nuclear Regulatory Commission (NRC) adopted a policy statement on PRA with the intent to increase the use of this technology in all regulatory matters, to the extent supported by the state of the art in PRA methods and data. Through the use of PRA, safety is enhanced by gaining insights that supplement the NRC's traditional approach of maintaining defense in depth and safety margin, as well as its overall engineering judgment. In 2004, the NRC amended its fire protection requirements to allow existing reactor licensees to voluntarily adopt the risk-informed, performance-based rule in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.48(c), which endorses National Fire Protection Association (NFPA) 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. Licensees will need a fire PRA to realize the full benefits of making the transition to the risk-informed, performance-based standard.

## Objective

The primary objective of this research is to advance the state of the art in fire PRA methods as directed by the NRC.

## 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 1011989), "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," issued September 2005, which addresses NPP fire risk for at-power operations.

Pilot plants making the transition to the rule, 10 CFR 50.48(c), rely upon NUREG/CR-6850 (EPRI 1011989) to develop their fire PRAs, while the NRC uses it to support reviews. The NRC, with participation by EPRI, has produced interim solutions to all 15 fire PRA issues raised by the pilot plants and EPRI related to NUREG/CR-6850 (EPRI 1011989) in the NFPA 805 frequently-asked-questions (FAQ) program and issued it as Supplement 1.

Additionally, RES and EPRI are working jointly to update and improve the fire events database used for NUREG/CR-6850 (EPRI 1011989). Initially, fire ignition frequencies will be updated; however, other applications are also envisioned. RES is also developing fire PRA methods for low power and shutdown, with EPRI as peer reviewers. Overall, this joint work is producing a significant convergence of technical approaches.

The conditional core damage probability (CCDP) shown in Figure 7.1 is a combination of the following: (1) fire-induced failure only of the cabinet PLUS random failures of trains A and B, (2) fire-induced failures of the cabinet AND train A PLUS random failure of train B, (3) fire-induced failures of all three; all of the above, coupled with failures of any remaining mitigative measures that may still be available, thereby leading to core damage.

## Future Work

A revision to the joint report is in the planning stages as the methodology continues to mature.

**For More Information**
Contact J.S. Hyslop, RES/DRA at 301-251-7611 or JS.Hyslop@nrc.gov

# Fire Human Reliability Analysis Methods Development

## Background

The Individual Plant Examination of External Events (IPEEE) program and the experience from actual fire events found that, depending on design and operational conditions, fire can be a significant or dominant contributor to nuclear power plant (NPP) risk. Human errors have been shown to be a significant contributor to overall plant risk (including the risk from fires) because of the significant role that operators play in the fire protection strategy on reactor safety. Figure 7.2 illustrates operators in an NPP control room. Human reliability analysis (HRA) is the tool used to assess the implications of various aspects of human performance on risk. Currently, the NRC is expanding existing HRA methods to evaluate the impact of human failures in the fire protection defense-in-depth safety strategy.

In 2004, the NRC amended its fire protection requirements to allow existing reactor licensees to voluntarily adopt the risk-informed, performance-based rule, 10 CFR 50.48(c). This rule endorses National Fire Protection Association (NFPA) 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. To realize the full benefits of making the transition to the risk-informed, performance-based standard, plants will need to have a fire probabilistic risk assessment (PRA) that includes quantitative HRA for post-fire mitigative human actions modeled in a fire PRA.

The Electric Power Research Institute (EPRI) and 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 document, NUREG/CR-6850 (EPRI 1011989), "Fire PRA Methodology for Nuclear Power Facilities," that addresses fire risk for at-power operations. This report provides high-level qualitative guidance and quantitative screening guidance for conducting a fire HRA. However, this document does not provide a detailed quantitative methodology to develop best-estimate human error probabilities (HEPs) for human failure events under fire-generated conditions.

## Objective

The overall objective of the effort is to develop fire HRA methods beyond what is currently in NUREG/CR-6850 (EPRI 1011989) and develop an HRA methodology and approach suitable for use in a fire PRA.

The fire HRA guidance developed through this effort is intended to support plants making the transition to 10 CFR 50.48(c), as well as NRC reviewers evaluating the adequacy of submittals from licensees making that transition. It may also be employed as a general fire PRA tool for HRA.

**Figure 7.2 Operators in a NPP control room**

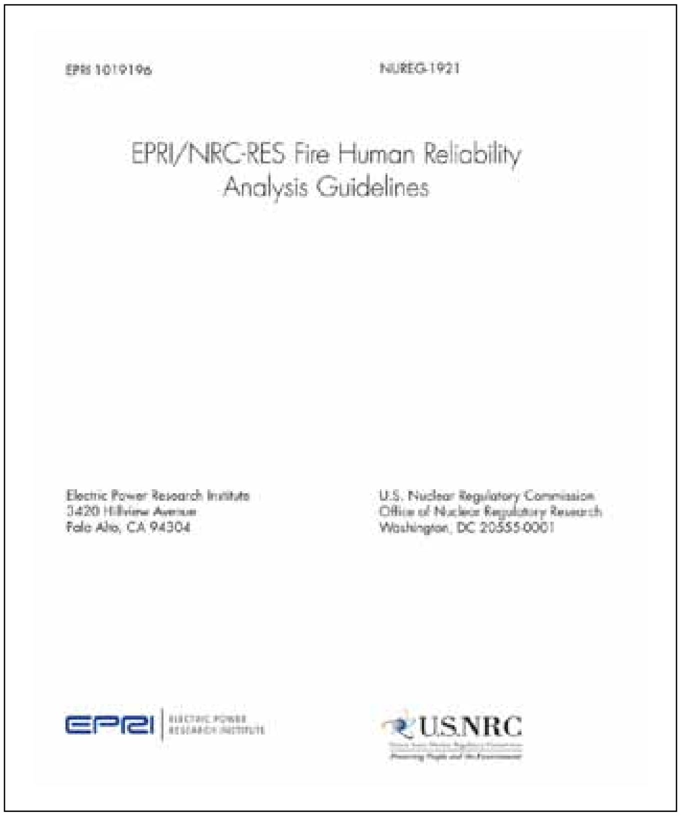
## Approach

RES has worked collaboratively with EPRI to develop a methodology and associated guidance for performing quantitative HRAs for post-fire mitigative human actions modeled in a fire PRA. The NRC issued NUREG-1921 (EPRI 1019196), "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines" (see Figure 7.3), as a draft for public comment in December 2009. It provides three approaches to quantification: screening, scoping, and detailed HRA. Screening is based on the guidance in NUREG/CR-6850 (EPRI 1011989), with some additional guidance for scenarios with long time windows. Scoping is a new approach to quantification developed specifically to support the iterative nature of fire PRA quantification. Scoping is intended to provide less conservative HEPs than screening but requires fewer resources than a detailed HRA. For detailed HRA quantification, the NRC has developed guidance on how to apply existing methods to assess post-fire HEPs.

The NRC plans to release NUREG-1921 (EPRI 1019196) as a final report in spring 2011.

## Future Work

The NRC has added a new HRA module to the NRC-RES/EPRI Fire PRA Workshop to provide training on the use of this methodology. The joint fire HRA methodology development team is scheduled to deliver the fire HRA training at the 2010 workshops, as well as at future fire PRA workshops.



EPRI 1019196 NUREG-1921

EPRI/NRC-RES Fire Human Reliability Analysis Guidelines

Electric Power Research Institute
3420 Hillview Avenue
Palo Alto, CA 94304

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

Figure 7.3 NUREG-1921 cover page

**For More Information**
Contact Kendra Hill, RES/DRA at 301-251-3300 or Kendra.Hill@nrc.gov

# Fire Modeling Activities

## Background

The results of the Individual Plant Examination of External Events (IPEEE) program and actual fire events indicate that fire can be a significant contributor to nuclear power plant (NPP) risk, depending on design and operational conditions. Fire models can evaluate fire scenarios in risk assessments, determine damage to cables and other systems and components important to safety, and characterize the progression of fire beyond initial targets. Used in these ways, fire models are important tools in determining the contribution of fire to the overall risk in NPPs.

## Objective

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

## Approach

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

The NRC and its licensees use the results in NUREG-1824 to provide confidence in the predictive capabilities of the various models evaluated. For example, although engineering calculations have limited capabilities, they provide reasonable estimates of certain phenomena when used within limitations (see Figure 7.4). These insights are valuable to fire model users who are developing analyses to support a transition to NFPA 805, to justify exemptions from existing prescriptive regulatory requirements, and to conduct significance determination process (SDP) reviews under the Reactor Oversight Process (ROP).

The NRC completed a Phenomena Identification and Ranking Table (PIRT) 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, which identified important fire-modeling capabilities needed to improve the NRC's confidence in the results. This study helps define future research priorities in fire modeling.

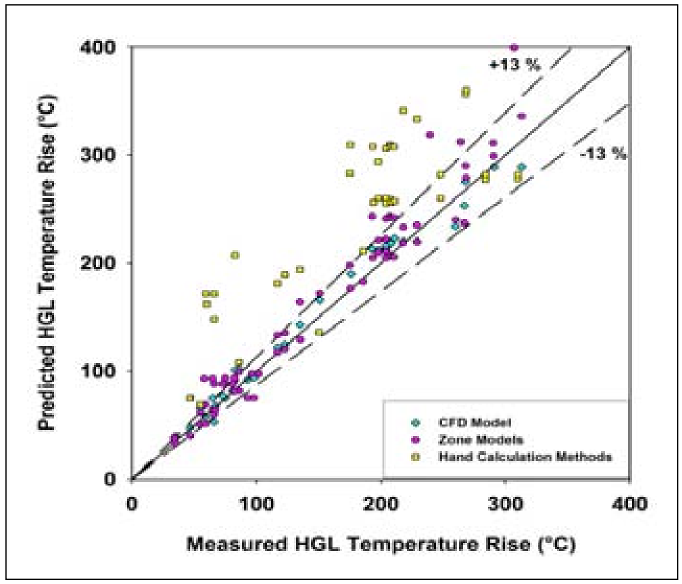



**Figure 7.4 Measured vs. predicted hot gas layer temperature rise**
**The models evaluated provide reasonable estimates of actual temperature rise.**

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

NIST used the THIEF model in both two-zone and computational fluid dynamics (CFD) models. In addition, the NRC incorporated the THIEF model in its fire dynamics tools spreadsheets (NUREG-1805, "Fire Dynamics Tools (FDTs) Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program," issued December 2004). The THIEF spreadsheet is a useful tool for inspectors and licensees to quickly determine the likelihood of cable damage, given a fire, or to indicate the need for further analysis.

Currently, the NRC is again working with EPRI and NIST to develop technical guidance to assist those who conduct fire-modeling analyses of NPPs. This guidance will continue to expand on NUREG-1824 by providing users with best practices

from experts in fire modeling and NPP fire safety.
This application guide contains five commonly available fire modeling tools (FDTs, Fire-Induced Vulnerability Evaluation (FIVE)-Rev1, Consolidated Fire Growth and Smoke Transport Model (CFAST), MAGIC, and Fire Dynamics Simulator (FDS)) that were developed by nuclear power stakeholders or that were applied to NPP fire scenarios. Previously, RES, EPRI, and NIST used these same models in the V&V study documented in NUREG-1824. Figure 7.5 illustrates a isometric view of a room in an NPP showing the temperature profile above an electrical cabinet fire in a fire dynamics simulation.

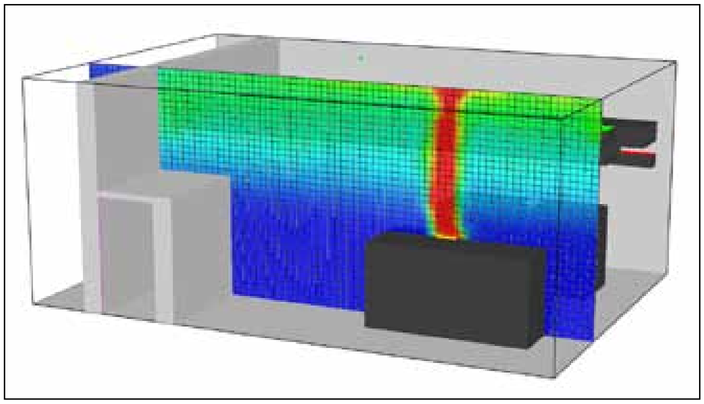


**Figure 7.5 Graphical output from FDS/Smokeview fire model**

The NRC released draft NUREG-1934, "Nuclear Power Plant Fire Modeling Application Guide (NPP FIRE MAG)," for public comment in early 2010. It received numerous comments and suggestions during the public comment period on ways to expand and improve it to better support the model users and reviewers. The NRC is currently working with EPRI and NIST on revising the draft and expects to publish it in early 2011. This report will assist both the user performing the calculation and the reviewers; it includes guidance on selecting appropriate models for a given fire scenario and on understanding the levels of confidence that can be attributed to the model results. The report will also form the foundation for future fire model training being developed by RES and EPRI.

## Future Work

The NRC is continuing to update the fire modeling tools, expand the V & V effort, and develop additional model input data.

**For More Information**
Contact David Stroup, RES/DRA at 301-251-7609 or David.Stroup@nrc.gov

# Cable Heat Release, Ignition, and Spread in Tray Installations During Fire (CHRISTIFIRE)

## Background

Fire can be a significant contributor to nuclear power plant (NPP) risk. In 1975, a serious fire involving electrical cables occurred at the Browns Ferry Nuclear Power Plant (BFN) operated by the Tennessee Valley Authority (TVA). NPPs typically contain hundreds of miles of electrical cables. The burning behavior of cables in a fire depends on a number of factors, including their constituent materials and construction, as well as their location and installation geometry. Burning cables can propagate flames from one area to another, or they can add to the amount of fuel available for combustion. Burning cables also produce smoke containing toxic and corrosive gases. The lower the heat exposure required to ignite the electrical cables, the greater the fire hazard in terms of ignition and flame spread. Electrical cables exposed to fire can lose physical integrity (i.e., melting of the insulation) and insulation resistance, leading to electrical breakdown or short-circuiting or the spread of fire to other cables or combustibles.

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

## Objective

The CHRISTIFIRE (Cable Heat Release, Ignition, and Spread in Tray Installations during Fire) experimental program is an effort to quantify the mass and energy released from burning electrical cables. The program includes fire tests on grouped electrical cables to enable better understanding of the fire hazard characteristics, including heat release rate (HRR) and flame spread. The NRC will use this type of quantitative information to develop more realistic models of cable fires for use in fire probabilistic risk assessment (PRA) analyses, such as those performed using the methods of NUREG/CR-6850 "Fire PRA Methodology for Nuclear Power Facilities" in NFPA 805 applications.

## Approach

Phase 1 of CHRISTIFIRE included experiments ranging from microscale 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 (see Figure 7.6). Meter-long cable segments were slowly fed through a small tube furnace and a variety of spectrometric techniques measured the composition of the effluent (see Figure 7.7). The standard cone calorimeter test (see Figure 7.8) measured the heat release rate per unit area for a variety of cable types at several external heat fluxes.

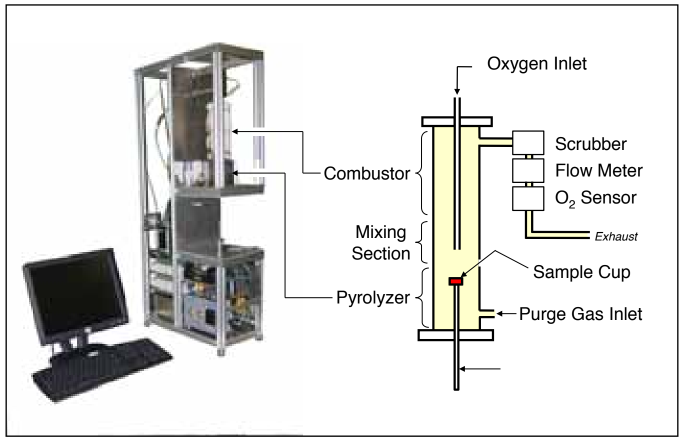



**Figure 7.6 Pyrolysis combustion flow calorimeter (Photograph and diagram from ASTM D 7390)**

A large radiant panel apparatus (see Figure 7.9), specially designed for this test program, measured the burning rate of cables when installed in ladder-back 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 (see Figures 7.10 and 7.11).

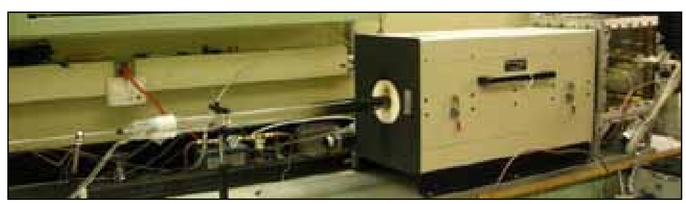


**Figure 7.7 ISO/TS 1970 tube furnace (Photograph of test apparatus used for one set of microscale tests)**

In addition, a simple model of flame spread in horizontal tray configurations, referred to as FLASH-CAT (Flame Spread over Horizontal Cable Trays), 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.

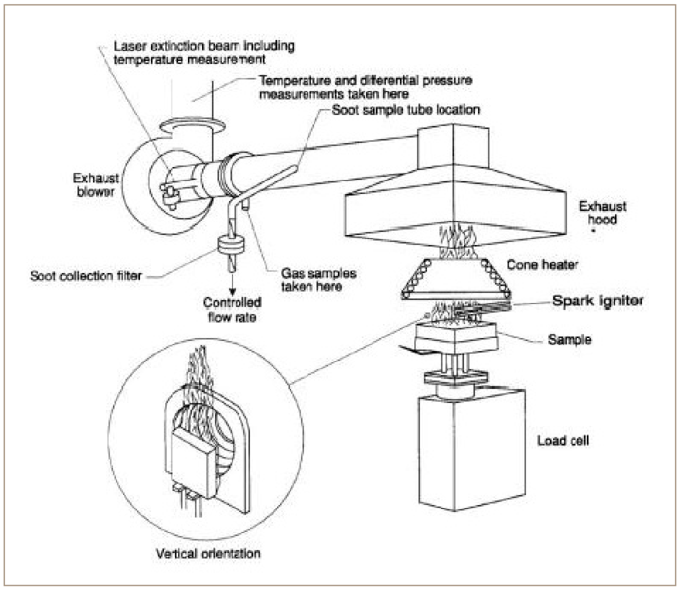



**Figure 7.8 Cone calorimeter**
**(From ASTM D611303; diagram of the cone calorimeter test apparatus)**

**Figure 7.9 Radiant panel cable tray fire test (Side view of burning cables in a tray exposed to a radiant heat source)**

**Figure 7.10 Burning cables during cable tray fire test**
**(Side view of burning cables in trays during a multi-tray test after ignition using a small gas burner)**

**Figure 7.11 Cables in tray**
**(Cables placed in trays before fire test)**

## Future Work

CHRISTIFIRE was the first attempt at developing a more realistic understanding of the burning behavior of grouped cables. Based on its success, future phases of the project will examine the burning behavior of cables installed in vertical trays and the effectiveness of various methods of protection. The FLASH-CAT model will be validated and extended to other configurations.

**For More Information**
Contact David Stroup, RES/DRA at 301-251-7609 or David.Stroup@nrc.gov

# Direct Current Electrical Shorting in Response to Exposure Fire (DESIREEFIRE)

## Background

The Individual Plant Examination of External Events (IPEEE) program results and actual fire events indicate that fire can be a significant contributor to nuclear power plant (NPP) risk. The question of how to determine risk resulting from fire damage to electrical cables in NPPs has been of concern since the Browns Ferry NPP (BFN) fire in 1975. In earlier years, it was generally believed that any system that depended on electric cables passing through a compartment damaged by fire would be unavailable for its intended safety function. The BFN fire and recent testing have prompted wider understanding that short circuits involving an energized conductor can pose considerably greater risk. The resultant "hot shorts" (see Figure 7.12) can cause systems to malfunction so as to inadvertently reposition motor-operated valves and start or stop plant equipment. Plant safety analyses should account for this risk.

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

Some recent tests performed by industry indicate that the results for ac circuits may not be fully representative of what might occur from fire-induced damage to dc circuits. Because of the differences in the operating voltages and circuit design between ac and dc, the previous data gathered for ac circuits may not be applicable to dc circuits.

## Objective

The Direct Current Electrical Shorting in Response to Exposure Fire (DESIREEFIRE) testing (see Figures 7.12 and 7.13 for examples of tests) of risk-significant dc circuits will allow the fire protection community to better understand dc circuit-failure characteristics.

## Approach

The NRC staff elected to perform fire testing of dc circuits using configurations that are representative of safety-significant circuits and components used in NPPs to better understand the probability of spurious actuations and the duration of those actuations in dc circuits.

The DESIREEFIRE testing program used small- and intermediate-scale tests to evaluate the response of dc circuits to fire conditions. Tests include several different circuits:

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

The DESIREEFIRE project is another RES Fire Research Branch working under collaborative research agreement with EPRI. This agreement has provided various components and cabling to the DESIREEFIRE testing program at little or no cost to the NRC.

**Figure 7.12 Direct current electrical cable hot short**

It also provided expert advice on the various aspects of the dc power system and circuit design. Testing is complete, and the NRC plans to issue the report in near future.

**Figure 7.13  Intermediate-scale dc fire tests**

**Figure 7.14  Battery bank for dc fire tests**

## Future Work

The determination for future cable testing programs will be based upon the outcome of the Fire Effects on Electrical Cables and impact on Nuclear Power Plant System Performance Phenomena Identification and Ranking Table (PIRT) and Expert Elicitations.

**For More Information**
Contact Gabriel Taylor, RES/DRA at 301-251-7576 or Gabriel.Taylor@nrc.gov

# Fire Effects on Electrical Cables and Impact on Nuclear Power Plant System Performance: Phenomena Identification and Ranking Table (PIRT) and Expert Elicitation Programs

## Background

Beginning in 1997, the NRC staff noticed a series of Licensee Event Reports (LERs) related to potential plant-specific problems involving fire-induced electrical cable circuit failures. The staff issued Information Notice 99-17, "Problems Associated with Post-Fire Safe-Shutdown Circuit Analysis," in June 1999, to alert the industry. The industry, under the leadership of the Nuclear Energy Institute (NEI), performed a joint series of fire tests with the Electric Power and Research Institute (EPRI) to better understand the issue. The industry used an expert elicitation to review the results and provide recommendations with regard to their use in probabilistic risk assessments (PRAs). EPRI 1003326, "Characterization of Fire-Induced Circuit Faults—Results of Cable Fire Testing," issued December 2002, documented the testing and expert panel results.

On February 19, 2003, the NRC sponsored a facilitated public workshop to discuss the results of the NEI/EPRI tests. Following the workshop, the NRC issued Regulatory Issue Summary (RIS) 2004-03, "Risk-Informed Approach for Post-Fire Safe-Shutdown Circuit Inspections," in December 2004. In that document, the staff identified a number of areas requiring additional testing. The NRC Office of Nuclear Regulatory Research (RES) initiated the Cable Response to Live Fire (CAROLFIRE) test program to address these concerns and documented the results in the three volumes of NUREG/CR-6931 "CAROLFIRE" report, which was published in April 2008. In 2006, a licensee performed independent testing of ungrounded direct current (dc) circuits and obtained unexpected results.

In 2009–2010, the NRC, along with EPRI, initiated the "Direct Current Electrical Shorting in Response to Exposure Fire" (DESIREEFIRE) testing program to better understand the performance of dc circuits. This testing program used small- and intermediate-scale tests to evaluate the response of dc electric cables and circuits to fire conditions. Several different circuits were tested, including dc motor starters, pilot solenoid-operated valve coils, medium-voltage circuit-breaker controls, and instrumentation circuits.

## Objective

Following the development in 2005 of circuit failure probabilities in NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," the NRC added two additional major fire testing programs regarding cable hot shorting: CAROLFIRE in 2008 and DESIREEFIRE in 2011. The objective of these Phenomena Identification and Ranking Table (PIRT) and expert elicitation programs is to improve the state of the art related to understanding and predicting hot shorting when cables are exposed to fire conditions.

## Approach

The NRC plans to convene two separate expert panels. The first will be comprised of electrical engineering experts to review all currently available testing data. This panel will follow the NRC's PIRT process to determine the state of the art in predicting hot shorting when cables are exposed to fire conditions.

The second expert panel will be comprised of fire PRA experts to explore and advance the state of the art in determining realistic probabilities of hot shorting when cables are exposed to fire conditions.

Figure 7.15 below illustrates a typical PIRT panel discussion in progress.

**Figure 7.15 A typical PIRT panel discussion**

## Future Work

The determination for future cable testing will be based upon these PIRT and Expert Elicitation.

**For More Information**
Contact Gabriel Taylor, RES/DRA at 301-251-7576, or Gabriel.Taylor@nrc.gov

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

## Background

The NRC needs data to determine the performance of seals in spent fuel transportation packages during beyond-design-basis fires, similar to the Baltimore Tunnel Fire in 2001. The performance of the package seals is important for determining the potential release of radioactive material from a package during a beyond-design-basis accident. The seals have lower temperature limits than other package components and are a vital part of the containment barrier between the environment and the cask contents.

NUREG/CR-6886, "Spent Fuel Transportation Package Response to the Baltimore Tunnel Fire Scenario," describes in detail an evaluation of the potential release of radioactive materials from three different spent fuel transportation packages. This evaluation used estimates of temperatures resulting from the 2001 Baltimore Tunnel Fire as boundary conditions for finite element models to determine the temperature of various components of the packages, including the seals. For two of the packages, the model-estimated temperatures of the seals exceeded their continuous-use rated service temperature, meaning the release of radioactive material could not be ruled out with available information. However, for both of those packages, the analysis determined, by a bounding calculation, that the maximum expected release would be well below the regulatory safety requirements given in 10 CFR Part 71, "Packaging and Transportation of Radioactive Material," for a release from a spent fuel package during hypothetical accident conditions.

In 2008, a National Institute of Standards and Technology (NIST) study, "Possible Methods for Determination of the Performance of a Transportation Cask in a Beyond-Design-Basis Fire," determined different testing approaches for evaluating package seal performance for containing Chalk River Unidentified Deposit (CRUD) released from the surface of fuel assemblies being transported.

## Objective

The objective of the package seal test is to determine its performance in beyond-design-basis fire scenarios and to provide the physical data needed to better understand the likelihood of a radioactive material release.

## Approach

The Office of Nuclear Regulatory Research (RES) has contracted with NIST to conduct small-scale thermal testing to obtain experimental data regarding the performance of seals during beyond-design-basis fires.

The experimental testing consists of a fabricated small-scale pressure vessel with an American Society of Mechanical Engineers (ASME) flange design (see Figure 7.16), using metallic seals from a selected manufacturer similar to those that might be used on an actual spent nuclear fuel transportation package. The vessel will be heated in an electrical oven to temperatures as high as 800 degrees Celsius (C), which far exceeds the rated temperature of the seals in question. NIST will measure the temperature at different points in the test sample and will also monitor the internal pressure of the vessel to determine if any leaks from the test sample occur.

## Future Work

Future Testing will be determined based upon the outcome of this test series.

**Figure 7.16 Pictures of the small-scale test vessel after 800° C exposure for 9 hours (small-scale test vessel (top left), vessel head after disassembly (top right), and vessel body and metallic seal after disassembly (bottom left and bottom right))**

**For More Information**
Contact Felix E. Gonzalez, RES/DRA at 301-251-7596 or Felix.Gonzalez@nrc.gov

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

## Background

In 1995, the NRC adopted a policy statement on probabilistic risk assessment (PRA) that was intended to increase the use of PRA technology in all regulatory matters to the extent supported by the technical merit of the PRA methods and data. In 2004, the NRC amended its fire protection requirements to allow existing reactor licensees to voluntarily adopt the risk-informed, performance-based 10 CFR 50.48(c) rule, which endorses National Fire Protection Association (NFPA) Standard 805, "Performance Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants", as an alternative to current prescriptive fire protection requirements. Approximately one-half of the current licensed nuclear power plants (NPPs) plan to make the transition to this new rule. In order to realize the full benefits of making the transition to the risk-informed, performance-based standard, plants will need to perform a fire PRA. The fire protection inspection program also uses fire PRAs to perform other regulatory activities, such as the Significance Determination Process (SDP) for inspection findings. Many NPPs use the joint Electric Power Research Institute (EPRI) and NRC document NUREG/CR-6850 (EPRI 1011989), "Fire PRA Methodology for Nuclear Power Facilities," to create fire PRAs for at-power operations. The NRC staff uses it to support reviews of the licensee amendment request (LAR) that a licensee submits when transitioning their fire protection program to NFPA 805. As part of the pilot plants' transition to 10 CFR 50.48(c), the NRC and EPRI have jointly produced interim solutions to fire PRA issues that have been raised concerning the implementation of NUREG/CR-6850 in NFPA 805's frequently-asked-questions (FAQ) program.

The staff is also publishing NUREG-1921 (EPRI 1019196), "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines,“ which it anticipates will be used to develop human reliability analysis (HRA) components of fire PRAs. At the present time, RES, in partnership with EPRI, has drafted NUREG-1934 (EPRI 1019195), "Nuclear Power Plant Fire Model Application Guide" (NPP FIRE MAG). When the NRC issues the final report, it will provide the basis for future fire model training.

## Objective

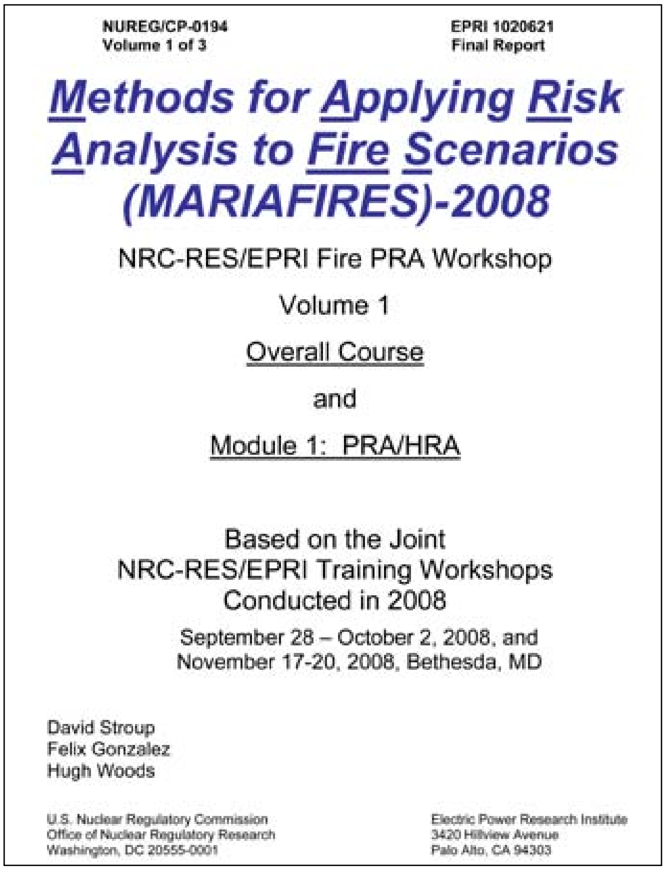
This program supports the NRC's policy to increase the use of PRA technology by providing training for 10 CFR 50.48(c) and other fire protection programs in fire PRA, circuit analysis, HRA, and fire modeling.

## Approach

Since 2005, the NRC and EPRI have jointly conducted training sessions in fire PRA. These sessions, hosted in alternate years by RES and EPRI, are available at no charge to all interested stakeholders. In 2005 and 2006, 3 days of general training covered fire PRA topical areas: PRA, fire models, and fire circuit analysis. Training in 2007 was expanded to 2 weeks per year. The courses offered detailed discussions and hands-on examples for each topical area in parallel for 4 days per week. The 2008 training sessions (Figure 7.17) were video recorded and documented along with their training materials in the three-volume NUREG/CP-0194, "Methods for Applying Risk Analysis to Fire Scenarios (MARIAFIRES)" (Figure 7.18), thus enabling self-study for persons unable to attend the course. This detailed instruction continued through 2009, when it was expanded in 2010 to provide an introduction to fire HRA in NUREG-1921.

In 2009, the NRC endorsed the American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA standard in Regulatory Guide (RG) 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities." Therefore, the 2010 training has also been updated to include the relationship between NUREG/CR-6850 and the fire PRA standard. In addition, the 2010 training includes HRA as a separate topical area to complement existing areas. Overall, this joint work is producing a higher level of understanding of fire PRA methods, which is expected to enhance the efficiency of NRC and industry efforts in fire PRA.

**Figure 7.17 Photo from the 2008 NRC-RES/EPRI fire PRA workshop**



NUREG/CP-0194
Volume 1 of 3

EPRI 1020621
Final Report

**Methods for Applying Risk Analysis to Fire Scenarios (MARIAFIRES)-2008**

NRC-RES/EPRI Fire PRA Workshop

Volume 1

Overall Course

and

Module 1: PRA/HRA

Based on the Joint
NRC-RES/EPRI Training Workshops
Conducted in 2008

September 28 – October 2, 2008, and
November 17-20, 2008, Bethesda, MD

David Stroup
Felix Gonzalez
Hugh Woods

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

Electric Power Research Institute
3420 Hillview Avenue
Palo Alto, CA 94303

**Figure 7.18 NUREG/CP-0194, Volume 1 of 3, cover page (Video recordings of the training sessions covered in each volume are included on a DVD in that volume)**

## Future Work

The Fire PRA, HRA, and Fire Modeling Programs are scheduled to continue into the future. A MARIAFIRES-2010 is also in the planning stages. A Fire Modeling Training Program is expected to be jointly developed by NRC and EPRI after the completion of NUREG-1934, "Nuclear Power Plant Fire Modeling Application Guide (NPP FIRE-MAG)"

**For More Information**
**Contact:**

Fire PRA:
J.S. Hyslop, RES/DRA at 301-251-7611 or
JS.Hyslop@nrc.gov

Fire HRA:
Kendra Hill, RES/DRA at 301-251-3300 or
Kendra.Hill@nrc.gov

Fire Modeling:
David Stroup, RES/DRA at 301-251-7609 or
David.Stroup@nrc.gov

# Fire Research and Regulation Knowledge Management

## Background

The results of the Individual Plant Examination of External Events (IPEEE) program and actual fire events indicate that fire can be a significant contributor to nuclear power plant (NPP) risk, depending on design and operational conditions. During the last 30 years, the NRC has undertaken many studies to better understand fire hazards, fire events, and fire risk in NPPs. The Fire Research Branch (FRB) in the Office of Nuclear Regulatory Research (RES) initiated the Fire Research and Regulation Knowledge Base Project to assemble the collection of NRC fire-related publications issued over the past 30 years. FRB has also undertaken a similar project to document and preserve the history of the influential Browns Ferry NPP (BFN) fire of 1975, and has assembled a Short History of Fire Safety Research to document the agency's research activities.

## Objective

The objective of this research is to support the NRC's knowledge management initiative in the fire protection area by identifying relevant information to be documented.

## Approach

### NUREG/BR-0465: Fire Protection And Fire Research Knowledge Management Digest

The Fire Research and Regulation Knowledge Base is a user-friendly database that provides information needed during such activities as inspections and reviews. The database includes publicly available documents, such as 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities"; guidelines for fire protection in NPPs; fire inspection manuals; fire inspection procedures; generic letters; bulletins; information notices; circulars; administrative letters; regulatory issue summaries; and regulatory guides. The technical knowledge includes NRC 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/BR-0361: The Browns Ferry Nuclear (BFN) Plant Fire of 1975 and the History of NRC Fire Regulations

In 1975, a fire occurred at BFN that challenged the operators' ability to safely shut the plant down. The fire prompted a new series of fire protection regulations and is a formative event in the history of fire protection regulations for NPPs. The brochure and DVD on the BFN plant fire of 1975 (see Figure 7.19) contain all major public documents, publications, regulations, and presentations pertaining to the BFN fire in a one-stop information resource with a user-friendly format, to provide a well-informed perspective about the BFN fire. Combined, these sources create a well-rounded picture of the event for varied types and levels of users; individually, they paint a detailed picture of specific aspects of the event.


**Figure 7.19 Screenshot of "The Browns Ferry Nuclear Plant (BFN) Fire of 1975 and the History of NRC Fire Regulations" (DVD main menu)**

### NUREG/BR-0364: A Short History of Fire Safety Research

The knowledge management program includes "A Short History of Fire Safety Research Sponsored by the U.S. NRC, 1975-2008," which covers its four phases:

- 1975–1987—the Fire Protection Research Program (FPRP) investigated the effectiveness of changes made to NRC's fire protection regulations after the 1975 Browns Ferry NPP fire
- 1987–1993—early fire PRAs were conducted (e.g., the LaSalle Risk Methods Integration and Evaluation Program (RMIEP))
- 1993–1998—incremental improvements were made to the RMIEP methods
- 1998–present—methods were developed to better apply the Commission's PRA technology policy to fire risk technology (to be used where practical in all regulatory matters).

## Future Work

An update to NUREG/BR-0361, "The Browns Ferry Nuclear (BFN) Plant Fire of 1975 and the History of NRC Fire Regulations", and NUREG/BR-0465, "Fire Protection and Fire Research Knowledge Management Digest," are in the planning stages.

**For More Information**

Contact Hugh ("Roy") Woods, RES/DRA at 301-251-7577 or Hugh.Woods@nrc.gov

David Stroup, RES/DRA at 301-251-7609 or David.Stroup@nrc.gov

Felix Gonzalez, RES/DRA at 301-251-7596 or Felix.Gonzalez@nrc.gov

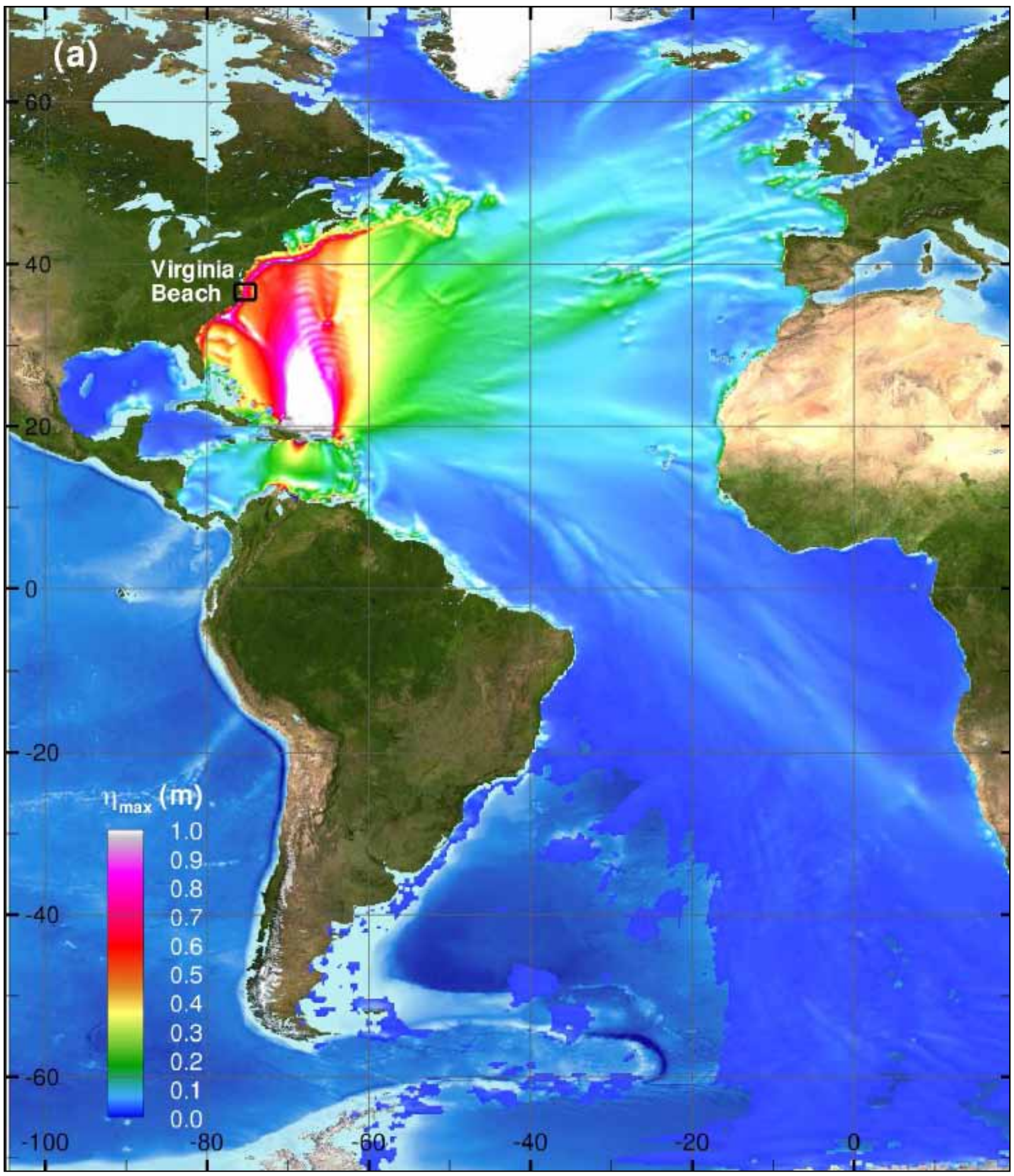
# Chapter 8: Seismic and Structural Research

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

Tsunami Research Program

Seismic Isolation Technology Regulatory Research

Risk-Informed Assessment of Containment Degradation




**Computed maximum tsunami wave amplitude in the Atlantic Basin generated by a Mw 8.8 earthquake in the Caribbean source zone**

# Advances in Seismic Hazard Assessment for the Central and Eastern United States

## Background

Seismic safety in the design and operation of nuclear facilities has been evolving since the development of the first rules and guidance for seismic design by the Atomic Energy Commission. In 1998, the U.S. Nuclear Regulatory Commission (NRC) issued a policy decision to move towards a risk-informed and performance-based regulatory framework. Risk-informed frameworks use probabilistic methods to assess not only what can go wrong, but also how likely it is to go wrong. Over the last decades, significant advances have been made in the ability to assess seismic hazard. The NRC is currently sponsoring several projects in support of both an updated assessment of seismic hazard in the Central and Eastern United States (CEUS) and an enhancement of the overall framework under which the hazard characterizations are developed. Figure 8.1 outlines three of the projects supporting the assessment of seismic hazard. The products of these projects will be used in the determination of seismic hazard design levels.

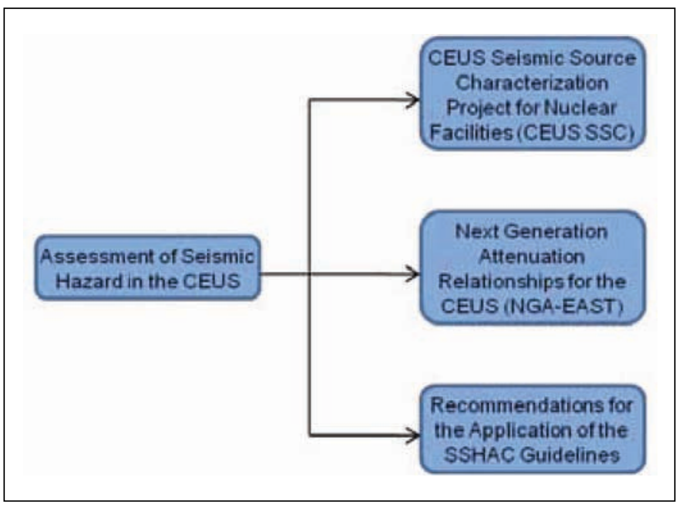



**Figure 8.1 Current projects supporting seismic hazard assessment**

## Research Projects

### The Ceus Seismic Source Characterization for Nuclear Facilities

The objective of the CEUS seismic source characterization (SSC) project is to develop an up-to-date seismic source characterization for the CEUS (see Figure 8.2) that includes (1) full assessment and incorporation of uncertainties, (2) a range of diverse technical interpretations from the informed scientific community, (3) an up-to-date earthquake database, (4) proper and appropriate documentation, and (5) a peer review. Accordingly, the project is being conducted using a process described as a Level 3 project in the Senior Seismic Hazard Analysis Committee (SSHAC) guidance (NUREG/CR-6372, "Senior Seismic Hazard Analysis Committee (SSHAC) Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts"). The NRC, along with the U.S. Department of Energy (DOE), and the Electric Power Research Institute (EPRI), cooperatively sponsor this project, which is scheduled to be completed in 2010.

### Next Generation Attenuation Relationship Development for the CEUS

A probabilistic seismic hazard analysis (PSHA) requires the prediction of ground motions for an earthquake with a given magnitude and distance. This research program will develop new state-of-the art ground motion prediction equations for the CEUS by following up on the successful multiinvestigator project, known as the Next Generation Attenuation (NGA) Relationship project, which focused on the western United States and which the Pacific Earthquake Engineering Research (PEER) Center coordinated. The NRC, DOE, EPRI, and the U.S. Geological Survey (USGS) have cooperatively undertaken this project, which is expected to end in 2014.

### Practical Procedures for Implementing the SSHAC Guidelines and Updating Existing PSHAS

In an effort to standardize PSHAs, the NRC sponsored the development of NUREG/CR-6372. While the SSHAC guidelines provide a robust framework for undertaking PSHAs of different levels of complexity, they do not provide detailed guidance on how to implement PSHAs within the framework. This project will result in a NUREG-series report to complement the SSHAC guidelines by providing practical guidelines for implementing the SSHAC framework, by capturing lessons learned during recent SSHAC Level 3 and 4 projects, and by providing practical guidelines for updating SSHAC-based PSHAs when new information becomes available. This project is scheduled to be completed in early 2011.

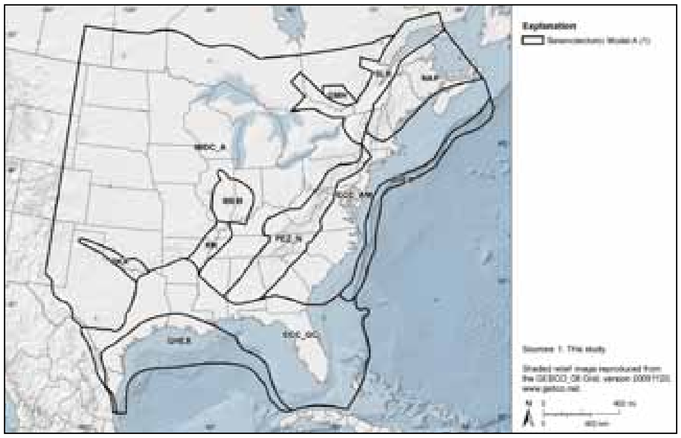



**Figure 8.2  Example source zones from the CEUS SSC for nuclear facilities project**

## For More Information

Contact Annie Kammerer. RES/DE at 301-251-7695 or Annie.Kammerer@nrc.gov

# Tsunami Research Program

## Background

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

This program, which includes cooperative work with the United States Geological Survey (USGS) and the National Oceanic and Atmospheric Administration (NOAA), has already resulted in several important publications on tsunami hazard assessments on the Atlantic Coast of the United States.

## Approach

### Tsunamigenic Source Characterization

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

### Tsunami Generation and Propagation Modeling

The USGS database is now used both for reviews of individual plant applications and as input for tsunami generation and propagation modeling being conducted by the experts at USGS and Texas A&M University. The goal of this modeling is to better understand the possible impacts that the identified sources could have on the coasts.

To undertake modeling of the impact of a flank failure landslide of the La Palma volcano in the Canary Islands, NOAA's Method of Splitting Tsunami (MOST) tsunami generation and propagation model has been coupled with the impact Simplified Arbitrary Lagragean Eulerian (iSALE) code, which can be used for modeling landslide-based tsunamigenic mechanisms. MOST is also being used to investigate the impact of the seismic tsunamigenic sources identified and characterized by the USGS (see Figure 8.3).

The final phases of the program will also explore acceptable probabilistic tsunami hazard assessment methods.

### New Regulatory Guidance

Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants," issued in 1977, briefly discussed tsunami as a source of flooding. The NRC is currently updating this regulatory guide. However, the update of this guide will not include tsunami-induced flooding. The NRC staff is currently preparing a new regulatory guide focused on tsunami hazard assessment and risk. The staff also contributed to tsunami information in draft International Atomic Energy Agency (IAEA) Safety Standard DS-417, "Meteorological and Hydrological Hazards in Site Evaluation for Nuclear Installations."

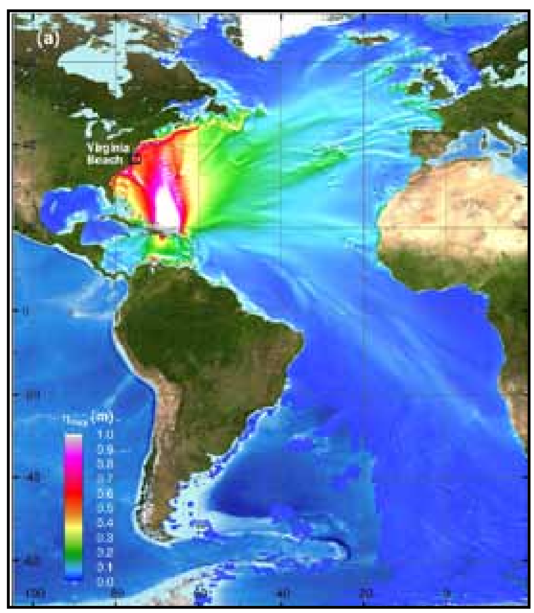



**Figure 8.3 Computed maximum tsunami wave amplitude in the Atlantic Basin generated by a $M_w$ 8.8 scenario earthquake in the Caribbean source zone**

**For More Information**

Contact Annie Kammerer, RES/DE at 301-251-7695 or Annie.Kammerer@nrc.gov

# Seismic Isolation Technology Regulatory Research

## Background

Seismic isolation technologies (also called base isolation technologies) are components and systems that isolate a structure from the motion of the ground during an earthquake. Modern seismic isolation devices and components were principally developed in the 1970s and 1980s, and thousands of conventional buildings, industrial structures, and bridges have been seismically isolated in the United States and abroad (see examples in Figure 8.4). Seismic isolation has been used to design and construct nuclear facility structures in France and South Africa. The renaissance of nuclear energy is leading to an exploration of the use of seismic isolation technologies in U.S. nuclear facilities. Several new advanced reactor designs are expected to include seismic isolation systems. To prepare for the possible use of these technologies in nuclear plant design, the NRC has initiated a program to identify and investigate these technical areas.

## Approach

### Development of NUREG/CR on the Use of Seismic Isolation Systems in Nuclear Power Plants

The NRC, working with Lawrence Berkeley National Laboratory, is addressing a range of technical considerations for analysis and design of safety-related nuclear facility structures using seismic isolation. An associated NUREG/CR under development is intended to serve as a reference for engineers engaged in the design of structures using seismic isolation systems, as well as NRC staff charged with reviewing applications utilizing these technologies. Typically, the seismic isolation components are treated as a civil or structural subsystem of a nuclear power plant whose risk-informed design is governed by specific performance objectives. The treatment of seismic isolation in existing building codes and regulations is being explored as a starting point. The NUREG/CR will discuss the behavior, mechanical properties, modeling, structural response analysis, and design issues for seismic isolation design using the most commonly used seismic isolation devices.

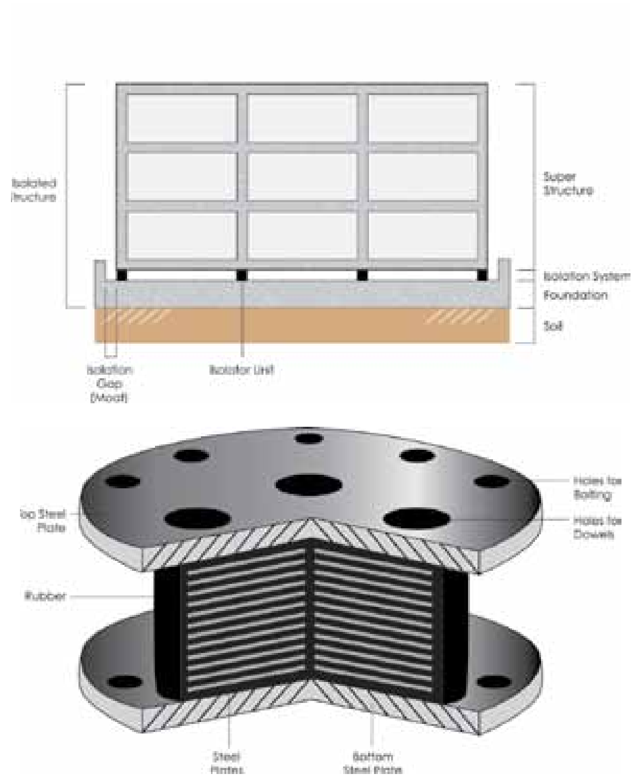



**Figure 8.4 Schematic figures showing typical design of a seismically isolated structural system (above) and a typical rubber-bearing style isolator (below)**

### Upcoming Investigation of Nuclear-Plant-Specific Issues

Further research is required in a number of technical areas. Some of these issues, such as the response to vertical excitation and soil-structure interaction, are already considered for non-isolated nuclear power plant designs such that current guidance could be applicable. Other issues, such as an evaluation of the consequences of impact of the structure against sidewalls during horizontal motion or impact from isolator uplift are new issues for the NRC. An important conclusion from the ongoing work is that base isolation is a viable technology for use in nuclear power plants. Additional research to investigate these critical areas will soon begin to identify acceptable means and methods of analysis and to establish a regulatory basis for review.

### Additional Plant Specific Issues

Additional topics of interest include the following:

- evaluation of isolator displacement capacity and beyond-design-basis events
- evaluation of the effect of differences among the mechanical properties of base isolation devices
- evaluation of the likelihood and possible consequences of rocking of the isolated superstructure on the base isolation devices;

- investigation of the beyond-design-basis aircraft impact load on the base isolation devices
- development, verification and validation of computer simulation models of base isolation devices under multidirectional excitation;
- aging and testing of the base isolation devices
- investigation of the interaction between the base isolation layer, the foundation, and any underlying soil

**For More Information**

Contact Annie Kammerer, RES/DE at 301-251-7695 or Annie.Kammerer@nrc.gov

# Risk-Informed Assessment of Containment Degradation

## Background

Over time, degradation has been observed in the containment vessels of a number of operating nuclear power plants in the United States. Forms of degradation include corrosion of the steel shell or liner, corrosion of reinforcing bars, loss of prestressing, and corrosion of bellows. The containment vessel serves as the ultimate barrier against the release of radioactive material into the environment. Because of this role, compromising the containment could increase the risk of a large release in the unlikely event of an accident. Previous work in this area assessed the effects of degradation on the pressure-retaining capacity of the containment vessel through structural analyses that account for degradation. These analyses provided useful information about the effects of the degradation on the structural capacity of the containment in both deterministic and probabilistic fashions. However, additional studies are still required to identify adequate metrics and related methods that can be used to examine the effects of degradation in specific cases.

### *Approach*

The NRC is sponsoring research at Sandia National Laboratories (SNL) to assess the effects of containment vessel degradation in containment vessels in a risk-informed manner. Goals for the research include supporting license renewal reviews and inspections by providing methods to examine, on a case-by-case basis, potential degradation effects from aging and repairs. Initially, the study evaluated the effects of degradation on several types of containments with respect to the guidelines given in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." The study integrated fragility curves developed for nondegraded and postulated degraded conditions using structural analysis with preexisting probabilistic risk assessment models used in NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants." That phase of the study concluded that several cases of postulated degradation involving corrosion of the liner (see Figure 8.5) or shell showed small increases, no increases, or even decreases in the large early release frequency (LERF). Rather than leading to a containment rupture, the postulated liner degradation causes the containment to fail by leakage, with an increase in small early release frequency (SERF).

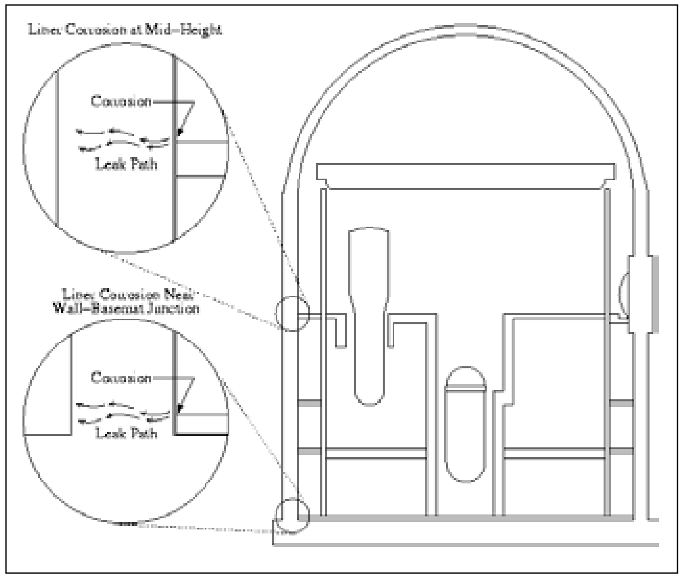



**Figure 8.5 Example of reinforced concrete containment leak paths for postulated corrosion degradation (NUREG/CR-6920)**

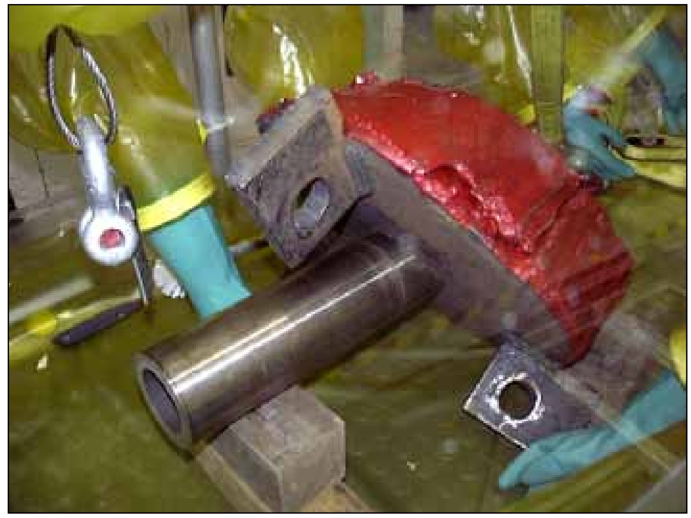
Since Regulatory Guide 1.174 does not provide guidance on the limits of SERF, additional deterministic analyses were performed to assess the effects of degradation on consequences to evaluate the feasibility of using metrics other than LERF. The study is continuing to assess the extent of corrosion, other containment types, and other degradation modes. Because most U.S. power plants have unique designs, a research goal is to develop results, approaches, and metrics that can be used for case-by-case examination of degradation effects.

**For More Information**

Contact Dr. Jose Pires, RES/DE at 301-251-7696 or Jose.Pires@nrc.gov

# Chapter 9: Materials Performance Research





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

# Extremely Low Probability of Rupture

## Background

The staff of the U.S. Nuclear Regulatory Commission (NRC) describes in Standard Review Plan (SRP) Section 3.6.3, “Leak-Before-Break (LBB) Evaluation Procedures,” acceptable analysis and assessment methodologies. Specifically, the SRP outlines a deterministic assessment procedure that can be used to demonstrate compliance with the requirement of General Design Criterion (GDC) 4, “Environmental and Dynamic Effects Design Bases,” in Appendix A, “General Design Criteria for Nuclear Power Plants,” to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities,” for primary system pressure piping to exhibit an extremely low probability of rupture. SRP Section 3.6.3 does not allow for assessment of piping systems with active degradation mechanisms. However, it is known that primary water stress-corrosion cracking (PWSCC) is occurring in systems that have been granted LBB exemptions to remove pipe-whip restraints and jet impingement shields.

To address this issue, the NRC has determined through a qualitative approach that these LBB-approved systems remain in compliance (see NRC Regulatory Issue Summary 10-07, “Regulatory Requirements for Application of Weld Overlays and Other Migration techniques in Piping Systems Approved for Leak-Before-Break,” dated June 8, 2010). This approach includes the following:

- As a qualitative rationale, the great majority of observed cracking is of limited extent and of shallow depth. These factors tend to mitigate the risk of piping rupture.
- PWSCC mitigation activities have been implemented (e.g., stress improvement and material replacement with overlays, mechanical stress improvement, inlays, onlays).

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

## Approach

As part of the effort for quantitatively ensuring the long-term extremely low probability of rupture, in accordance with GDC 4, the Office of Nuclear Regulatory Research (RES) is embarking on an effort to develop a modular-based computer code for the determination of the probability of failure for reactor coolant system (RCS) components. In doing so, RES has sought the support of national laboratories and commercial contractors and communicates with the domestic nuclear industry under the auspice of the Electric Power Research Institute (EPRI). This computer code will be capable of considering all degradation mechanisms that may contribute to low probability failure events while properly handling the uncertainty in the failure process. The code will be structured in a modular fashion so that, as additional operational experiences arise, additions or modifications can be easily incorporated without code restructuring. The first arm of the modular code to be developed deals directly with primary piping integrity and is coined xLPR for “extremely low probability of rupture.”

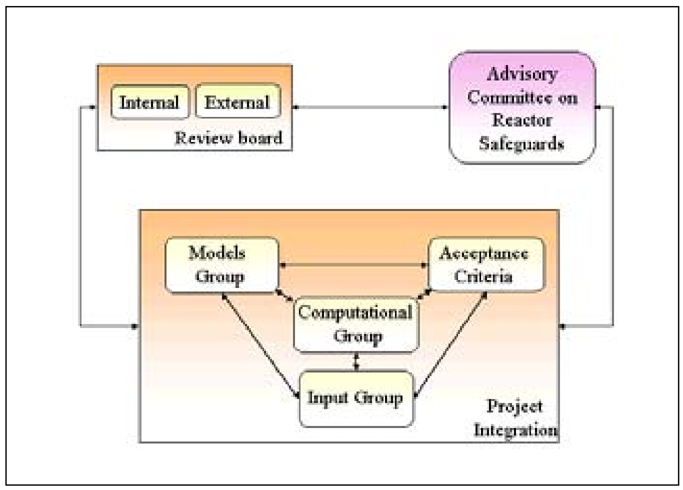



**Figure 9.1 xLPR organizational development structure**

As part of the ongoing 2-year pilot study effort, RES developed a group of teams as shown in Figure 9.1, each with specific long-term and short-term technical objectives. These teams will develop the quantification of extremely low probability of rupture. As part of the pilot study, the team effort will be focused on a particular problem. (i.e., the failure of a pressurizer surge nozzle dissimilar metal weld as seen in Figure 9.2 with a circumferential crack.)

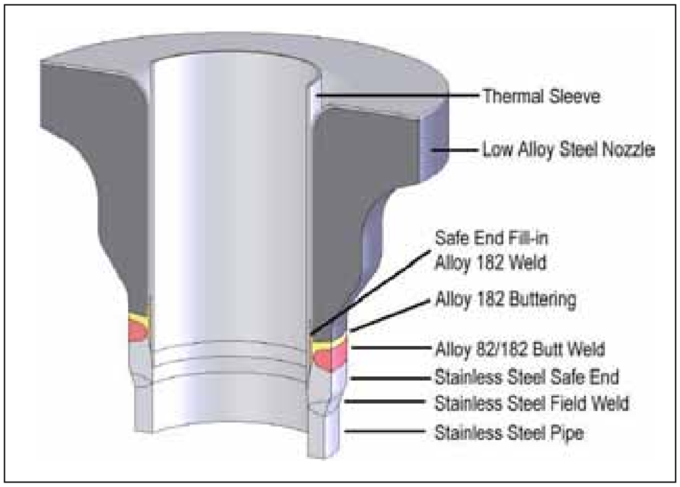



**Figure 9.2 Pressurizer surge nozzle illustration**

As the pilot study draws to a close, an initial version of the xLPR code (Version 1.0) is complete and will be used to demonstrate the feasibility of conducting these calculations using a fully verified, vetted, document controlled code. The pilot study outcome will be a demonstration of the feasibility of this process, both computationally and organizationally, to develop a complex fracture mechanics based code to calculate the probability of rupture for primary piping systems. In addition, an understanding of the limitations associated with these codes, and a firm basis for developing a more robust modular-type code will be developed. In the long term, focus shifts to the more generic problems associated with RCS integrity. The long-term outcome will be a modular computer code based with verified and validated methodologies for predicting low probability of failure events.

## Schedule

The planned schedule for the xLPR program is as follows:

xLPR pilot study complete – December 2010
xLPR modular code – 1st Quarter 2013
Long Term – Generic modular code – 1st Quarter 2015

**For More Information**
Contact David L. Rudland, RES/DE at 301-251-7622 or David.Rudland@nrc.gov

# Research to Support Regulatory Decisions Related to Second and Subsequent License Renewal Applications

## Background

Materials degradation phenomena, if not appropriately managed, have the potential to adversely impact the functionality and safety margins of nuclear power plant (NPP) systems, structures, and components (SSCs), especially as they continue to operate for longer periods. the Office of Nuclear Regulatory Research (RES) has initiated a multiyear research program to develop an improved understanding of materials degradation failure mechanisms to better predict potential impacts on the long-term operability of NPP SSCs, to provide necessary technical data to support regulatory decisions, and to inform the development of aging management programs (AMPs) to ensure continued safe plant operation.

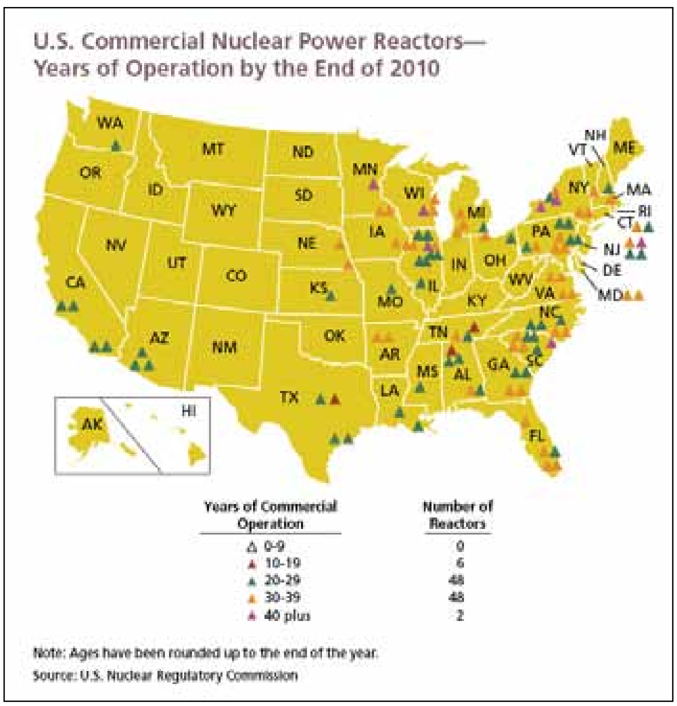



**Figure 9.3 Years of commercial operation (2010)**

As shown in Figure 9.3, several NPPs have entered into the first period of extended operation, and to date over half (59) have been granted an initial license extension of 40–60 years.

The U.S. commercial nuclear power industry has publicly informed the NRC staff of its intentions to submit, in the 2015–2019 timeframe, license renewal applications (LRAs) for a subsequent license renewal, which will cover a potential 80-year operating period. The agency permits requests for a subsequent license renewal under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants." However, potential technical challenges from aging effects on passive SSCs may need to be resolved before the licensee enters into an operating period beyond 60 years, including aging effects on the reactor pressure vessel (RPV), the RPV internals, primary piping, safety-related secondary piping, buried and submerged structures, electric cable insulation, and concrete exposed to high temperature and radiation.

To ensure that the NRC is prepared for a timely review of possible LRAs for a subsequent renewal, research to support the agency's regulatory decisionmaking on such LRAs is needed to ensure the availability of the necessary technical information.

## Objective

The objective of this research is to provide a sound technical basis to support timely reviews of potential subsequent LRAs.

## Approach

The NRC and industry have already expended considerable resources over the last several decades to better understand the safety implications and risk associated with aging of SSCs. Key activities have included an assessment of the technical basis for an alternate pressurized thermal shock (PTS) rule (10 CFR 50.61a, "Alternate Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events"), aging of electrical cables, and environmentally assisted cracking (EAC) of materials. Further, in February 2008, the NRC and the U.S. Department of Energy (DOE) cosponsored a "Workshop on U.S. Nuclear Power Plant Life Extension Research and Development," which requested stakeholder input into aging management research areas for "Life Beyond 60." (A summary of the workshop proceedings is provided in the Agencywide Documents Access and Management System (ADAMS) at Accession No. ML080570419.) Based on the results of this workshop, and the staff's long-term research plan, potential additional areas of focus for a subsequent license renewal include aging management of reactor vessel and internal materials, cable insulation, buried and submerged structures, and concrete exposed to high temperature and radiation.

The NRC staff is presently expanding the original NUREG/CR-6923, "Expert Panel Report on Proactive Materials Degradation Assessment," issued February 2007, to include longer timeframes (i.e., 80 or more years) and passive long-lived SSCs beyond the primary piping and core internals, such as the concrete containment building and cable insulation. This will allow the staff to (1) identify significant knowledge gaps and any new forms of degradation that may have arisen since the original proactive materials degradation assessment report was developed; (2) capture the current knowledge base on materials degradation mechanisms; and, (3) prioritize materials degradation research needs and directions for future efforts. This effort is being accomplished through a collaborative effort with a complementary DOE program—the LWR Sustainability (LWRS) program.

In recent years, there have been a variety of related research initiatives, such as the creation of the Materials Aging Institute by the Electric Power Research Institute (EPRI), Électricité de France (EDF), Tokyo Electric Power Company (TEPCO), and others, as well as the development of networks and technical meetings focused on some elements of proactive management of materials degradation (PMMD). However, no forum currently exists to bring together these diverse activities and provide coordinated information exchange and prioritization of PMMD topics. The NRC is working with other national regulators and nongovernmental organizations (NGOs) to implement an International Forum for Reactor Aging Management (IFRAM) that would create a network of international experts who would exchange information on operating experience, best practices, and emerging knowledge. These experts would work jointly to leverage the separate efforts of existing national programs into a coordinated research activity. This coordination enables a pooling of technical expertise and avoids unnecessary redundant efforts. The participants would share responsibility, accountability, resources, and rewards from this coordinated activity.

The staff will also be holding recurrent NRC/industry workshops on the status of operating experience from the initial renewal term and industry research activities to address aging management of technical issues for a subsequent license renewal term. This is a followup to the initial February 2008 workshop; the next workshop is planned for the first quarter of 2011.

In a related activity, RES is initiating an effort to collect the results from implementation of AMPs committed to by licensees for the initial license renewal period, along with any information from other licensee activities that will provide greater insights to materials aging phenomena in the renewed license operating period. This information and improved understanding will be used to identify any need for enhancements to AMPs for plant operation out to 80 years.

**For More Information**
Contact C.E. ("Gene") Carpenter, Jr., RES/DE at 301-251-7632 or Gene.Carpenter@nrc.gov

# Steam Generator Tube Integrity

## Background

Steam generator (SG) tubes (see Figure 9.4) are an integral part of the reactor coolant system (RCS) pressure boundary. They serve as a barrier to isolate the radiological fission products in the primary coolant from the secondary coolant and the environment. The understanding of SG tube degradation phenomena is continually evolving to keep pace with advances in SG designs and materials. To date, many modes of degradation have been observed in SG tubes, including bulk corrosion and wastage, crevice corrosion, pitting, denting, stress-corrosion cracking, and intergranular corrosion attack. Flaws have developed on both the primary and the secondary side of SG tubes. If such flaws go undetected or unmitigated, they can lead to tube rupture and possible radiological release to the environment.

**Figure 9.4 Recirculating steam generator tube bundle**

## Overview

The main objective of this research program is to develop a technical basis for SG tube integrity evaluations. This basis is needed to ensure that SG tubes continue to be inspected appropriately, flaw evaluations continue to be conducted correctly, and repair or plugging criteria are implemented appropriately. To aid in regulatory decisions and to assess code applications, as depicted in Figure 9.5, this research program addresses the following areas:

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

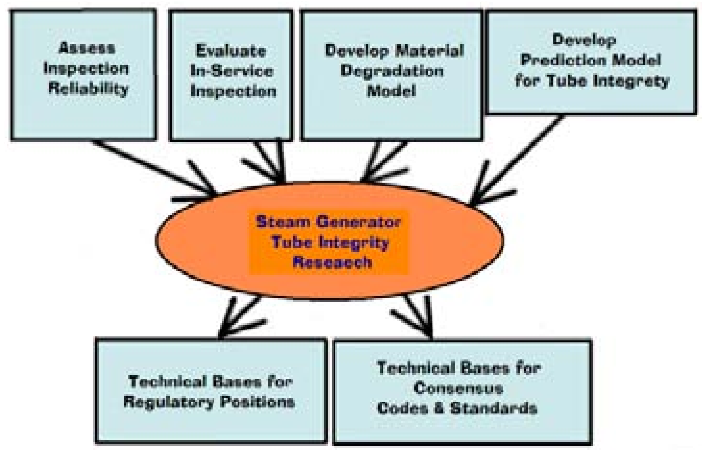



**Figure 9.5 Tube integrity research schematic**

## Approach

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

### Assessing Inspection Reliability

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

### Inservice Inspection Technology

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

### Research on Tube Integrity And Performance Modeling

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

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

Another ongoing study examines the consequences of exposing RCS materials to high temperatures during severe accident scenarios. Such accidents may challenge the integrity of SG tubes, so analyses are being conducted to determine whether certain RCS components may fail before SG tubes. Such a scenario would be preferable to an initial release through SG tubes, because RCS leaks would leak into containment, while SG tube leaks could lead to a radiation release to the outside environment.

### Research On Degradation Modes

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

### International Cooperation

The NRC is currently administering the fourth, 5-year term of the International Steam Generator Tube Integrity Program (ISG-TIP-4). In this program, regulators and researchers from member countries conduct and share research on tube integrity and inspection technologies. Current participants include organizations from Canada, France, Japan, Korea, and the United States.

**For More Information**

Contact Charles Harris, RES/DE at 301-251-7637 or Charles.Harris@nrc.gov

# Consequential Steam Generator Tube Rupture Program

## Background

The NRC and the nuclear power industry have expended considerable resources over the last two decades to better understand the safety implications and risk associated with consequential steam generator tube rupture (C-SGTR) events (i.e., events in which steam generator (SG) tubes leak or fail as a consequence of the high differential pressures or SG tube temperatures, or both, predicted to occur in certain accident sequences). Key activities included an assessment of temperature-induced creep-rupture of the reactor coolant system (RCS) components in the NUREG-1150 study entitled, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, issued December 1990; a representative analysis of the potential for induced containment bypass by an ad hoc NRC staff working group in NUREG-1570, "Risk Assessment of Severe Accident-Induced Steam Generator Tube Rupture" issued 1998, and recent thermal-hydraulic (T/H) analyses and risk analyses as part of the steam generator action plan (SGAP). Severe accident analyses performed as part of the state-of-the-art reactor consequence analyses (SOARCA) project provide additional insights into the likelihood and impact of subsequent failure of the reactor hot leg shortly following a C-SGTR event.

Prior investigations of a Westinghouse plant concluded that the contribution of C-SGTR events to the overall containment bypass frequency is at best at the same order of magnitude, if not lower than, the containment bypass fraction associated with other internal events for most pressurized-water reactors (PWRs). Thus, plant risk assessments should consider and monitor the risk associated with C-SGTR in a manner commensurate with its expected importance at each plant. Although important conclusions were made, these investigations identified certain limitations of scope, as well as a lack of thorough RCS component modeling with advanced simulation tools. It is important to address these limitations to advance our understanding of associated risks and to develop an enhanced risk assessment tool for C-SGTR events.

## Objectives

To close the technical gaps and to develop an enhanced risk assessment procedure for C-SGTR, the current RES program will attempt to fulfill the following objectives:

- Update computational fluid dynamics (CFD) and system code models for Combustion Engineering (CE) plants.
- Evaluate the impact of in-core instrument tube failures on natural circulation.
- Update SG flaw distributions.
- Complete structural analyses of CE and Westinghouse RCS components.
- Develop a user-friendly methodology for assessing the risk associated with consequential tube rupture and leakage in design-basis accidents and severe accident events.
- Conduct a reassessment of the conditional probabilities of C-SGTR based on updated flaw distributions and updated T/H analyses.

Compile and summarize key research, building upon NUREG-1570 (work performed as part of SGAP activities.

## Approach

### CE Thermal-Hydraulic and Severe Accident Analysis

The updated modeling approach and lessons learned from these most recent Westinghouse plant predictions will be applied to a CE plant model in order to improve the T/H predictions. This effort will update the hot-leg flow and mixing model, as well as hot-leg thermal radiation modeling.

The CE CFD model will be updated to include a simplified upper plenum, hot leg, surge line, and the SG primary side. This model will be used to predict hot leg and inlet plenum mixing rates, as well as the variations in temperature of the flow entering the hottest tubes in the SG.

The system code modeling effort will include the development of a MELCOR CE plant model which incorporates all of the lessons learned from the recent Westinghouse predictions completed in support of the SGAP. The modeling will also incorporate the updated CE CFD model predictions.

### Assess Impact of In-Core Instrument Tube Failures

In December 2009, RES completed a study on the impact of the consequences of instrumentation tube failure during severe accidents, which is detailed in ERI/NRC-09-206, "Analysis of the Impact of Instrumentation Tube Failure on Natural Circulation During Severe Accidents" (ADAMS Accession No. ML100130402). This work assesses the impact of instrumentation tube failures for Three Mile Island, Unit 2 (TMI-2) (a Babcock & Wilcox (B&W) design with a once through SG), and Zion (a 4-loop Westinghouse design with a U-tube SG). After a thorough review of this work, a detailed assessment of in-core instrument failures will be prepared.

### Updated Steam Generator Flaw Distributions

To assess the probability of an induced SGTR, detailed knowledge of the characteristics of SG tube flaws is needed with the tube temperature and stress profile during postulated accidents. For statistical analysis, flaw density distribution data as a function of size, shape, orientation, location, and type are needed. The potential for failure depends primarily on the upper tail of the size distribution (i.e., the most severe flaws) for a given flaw type and location. A verification process will also be used to confirm that the flaw distributions are consistent with operating experience for observed leakage rates.

By means of an existing memorandum of understanding addendum between the NRC and EPRI, RES will work with the industry to update flaw distributions originally developed in the mid-1990s. This update will include (1) evaluating the effect of improved inspection techniques on flaw density distributions; (2) developing distributions for both crack-like and wear-like defects; (3) accounting for flaws in the SG tube within the tubesheet regions; and (4) identifying any changes in flaw distribution caused by new tube materials, new SG designs, or new inspection techniques.

### Structural Analysis of CE And Westinghouse RCS Components for Prediction of RCS Piping Failure

RES structural analyses will build upon the latest T/H and severe accident analyses to include specific RCS components for Westinghouse and CE plants (e.g., hot-leg nozzle and hot leg-to-surge line nozzle). The failure analysis will consider uncertainty resulting from the shape, size, and location of potential flaws in the RCS components.

RES plans to identify, characterize, and model relevant RCS nozzles to assess their potential for failure during severe accidents for Westinghouse and CE plants. Two-dimensional axisymmetric and three-dimensional models will be developed, addressing variables such as nozzle geometries and configurations, boundary conditions, loading conditions, fabrication effects, stress-corrosion cracking mitigations, and degraded conditions. These models will be used to determine the time to failure for each analyzed component and the associated sensitivity to loadings and flaw geometry. Because of the importance of incorporating uncertainty, RES will develop a semiempirical methodology, based on numerical experiments, to predict failure of critical RCS components. The resulting methodology is expected to be more conducive to the procedure adopted in the C-SGTR risk assessment method developed as part of the program.

### Simplified Method for Assessing the Risk Associated With C-SGTR

In March 2009, RES provided the NRC's Office of Nuclear Reactor Regulation (NRR) with a report describing a method for assessing C-SGTR risk (ADAMS Accession No. ML083540412). RES intends to extend the methods described in this previous report to incorporate a number of enhancements. These enhancements will include consideration of the updated T/H conditions, SG flaw distribution, and RCS component analyses. Additionally, C-SGTR risk assessment methods described in previous NRC, Electric Power Research Institute (EPRI), and industry reports will be reviewed to identify useful insights and modeling approaches for use with the new simplified method. RES anticipates that the level of analysis in the new approach will be comparable to that of the previous RES C-SGTR risk report and the earlier NUREG-1570 study. Consistent with previous C-SGTR risk assessment work, the new simplified method will consider both pressure-induced and thermally induced SG tube failures.

### Reevaluation of C-SGTR Conditional Probabilities

In support of SGAP, RES previously developed an SG tube failure probability calculator tool. RES plans to extend the framework and modeling approaches used in this tool, including pressure- and temperature-induced challenges. Consequently, this program will focus on further validation of the detailed modeling used in the calculator, extension of calculator capabilities, updates to basic data and parameters (including provisions for future data updates), improvements in calculator usability, and development of supporting documentation.

## Deliverables

The following deliverables are anticipated at the completion of the C-SGTR program:

- probabilistic risk assessment report
- risk assessment tool
- draft regulatory guidance on risk-informed decision making regarding C-SGTR
- draft Risk Assessment Standardization Project (RASP) handbook section on assessment of CSGTR suitable to support revisions to the Inspection Manual Chapter 0609 appendices supporting the Significance Determination Process (SDP).
- summary report compiling key research results

**For More Information**

Contact Raj Mohan Iyengar, RES/DE at 301-251-7907 or Raj.Iyengar@nrc.gov

# Reactor Pressure Vessel Integrity

## Background

One aspect to the safe operation of a nuclear power plant is maintaining the structural integrity of the reactor pressure vessel (RPV) during both routine operations (i.e., heat up, cool down, and hydro test) and during postulated accident scenarios (e.g., pressurized thermal shock (PTS)). To do this, procedures are needed to estimate and compare the driving force for structural failure to the resistance of the structure to this driving force (and the effect of radiation on this resistance). Current statutory procedures for these estimates are found in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.61, "Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events (i.e., the PTS rule); Appendix G, "Fracture Toughness Requirements," to 10 CFR Part 50; Appendix H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR Part 50; Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials;" and Regulatory Guide 1.161, "Evaluation of Reactor Pressure Vessels with Charpy Upper Shelf Energy Less Than 50 ft-lb." While these methods generally depend on empirically based engineering methods, they are known to incorporate large implicit conservatisms adopted to address state-of-knowledge deficiencies that existed at the time of their promulgation. When coupled with the deterministic basis of current regulations, these conservatisms may unnecessarily reduce the possibility for continued operation and potential license renewals.

## Objectives

1. Integration of the advances in the state of knowledge, empirical data, and computational power that has occurred in the 20+ years since the adoption of the current regulatory requirements to develop the technical bases for state-of-the-science and risk-informed revisions to the statutory procedures that regulate the structural integrity of the current operational boiling and pressurized reactor fleets.

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

An additional objective is to apply insights from probabilistic structural integrity assessment gained from the first objective and the predictive material models developed in the second objective to develop and validate a modular probabilistic computer code that can be used to assess the structural integrity assessment of any pressurized structure in a nuclear power plant.

## Approach

RES has recently completed a multiyear study conducted in cooperation with the Oak Ridge National Laboratory (ORNL), other national laboratories and Government contractors, and the domestic nuclear power industry under the auspices of the Electric Power Research Institute (EPRI) Materials Reliability Project (MRP) to develop the technical basis for a risk-informed revision to the PTS rule. The Office of Nuclear Reactor Regulation (NRR) has used this technical basis to develop a voluntary alternative to the PTS rule which relaxes many of the conservatisms in the current rule without impacting the public health and safety. The NRC completed this voluntary alternative rule in 2010.

Also in the coming years, RES will publish and make available for public comment a revised version of Regulatory Guide 1.99, along with its technical basis. This revision is based on over five times the quantity of empirical data used to develop the current regulatory guide. The insights gained from these activities provide a large part of the work needed as the technical bases to support revisions to Appendices G and H to 10 CFR Part 50, which are both scheduled for completion in 2011.

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

- Development and validation of a method capable of identifying embrittlement mechanisms in reactor materials before they occur in commercial reactor service.
- Development and validation of a modular computational tool to perform probabilistic structural integrity assessments of passive primary reactor pressure boundary components.

**For More Information**
Contact Mark Kirk, RES/DE at 301-251-7631 or Mark.Kirk@nrc.gov

# Environmentally Assisted Fatigue of Components Exposed to the Reactor Water Environment

## Background

Environmentally assisted fatigue (EAF) deals with the effects that reactor coolant environments have on the fatigue life of components exposed to those environments. The American Society of Mechanical Engineers (ASME) Code, Section III, design fatigue curves, which were developed based on air testing of laboratory specimens, do not explicitly address EAF, and test data indicate that the ASME Code fatigue curves may not always be adequate for coolant environments (see Figure 9.6).

EAF of components exposed to the reactor water environment was first identified as a part of the NRC's Fatigue Action Plan, which was completed in 1995 (SECY-95-245, "Completion of the Fatigue Action Plan," dated September 25, 1995). By memorandum dated December 26, 1999, the NRC identified the need to evaluate environmental fatigue for nuclear power plants pursuing license renewal as a part of the close out of Generic Safety Issue (GSI) 190, "Fatigue Evaluation of Metal Components for 60-Year Plant Life." NUREG-1801, Revision 1, "Generic Aging Lessons Learned (GALL) Report," provides guidance for licensees. Specifically Chapter X.M1, "Metal Fatigue of Reactor Coolant Pressure Boundary," recommends the use of the methodology contained in NUREG/CR-6583, "Effects of Light-Water Reactor (LWR) Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," for carbon and low-alloy steels and NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," for austenitic stainless steels.

The NRC also identified the need to evaluate environmental fatigue for new reactors in Regulatory Guide 1.207, "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light Water Reactor Environment for New Reactors," with associated methodology documented in NUREG/CR-6909, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials." This methodology is also allowed for use by license renewal applicants.

The NRC must make additional effort in this area to facilitate the future review of licensees' environmental fatigue evaluations submitted to the agency for review and approval by both license renewal and new reactor applicants.

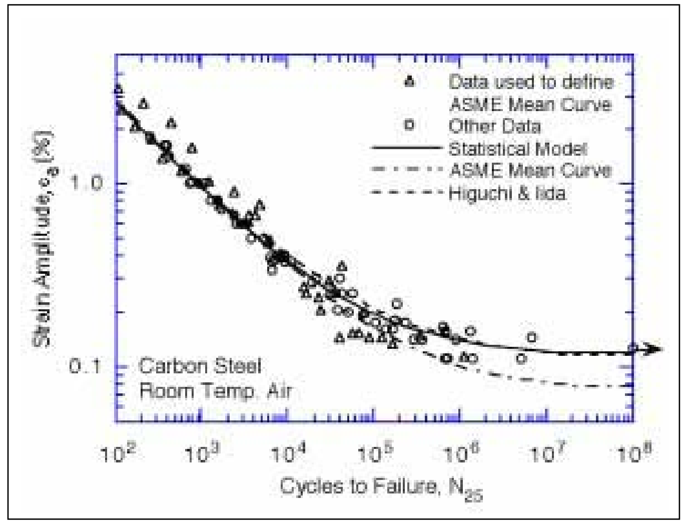



**Figure 9.6 Fatigue S–N behavior for carbon steel**

## Objective

Discussions among the Office of Nuclear Reactor Regulation (NRR), RES, and the Office of New Reactors (NRO) identified several further efforts necessary to address EAF:

- a transient stress evaluation software tool for rapidly determining thermal transient stresses in reactor components
- an ASME Code, Section III, fatigue calculation software tool for estimating fatigue usage factors in reactor components
- revised fatigue cumulative usage factor (CUF) evaluation criteria for postulated high-energy line break (HELB) locations
- technical support from Argonne National Laboratory (ANL) to update existing environmental fatigue methodology and develop application techniques

RES has planned activities for the time period of August 2009 through December 2011 to address these needs.

## Planned Tasks

The specifics of the RES tasks planned for this topic are as follows:

1. Transient Stress Evaluation Software Tool

   In this task, a software "mathematical integrator" tool will be developed and benchmarked that performs Duhamel integration for any user-specified input thermal transient using a unit stress response to develop thermal transient stress histories. The concept behind this approach is to utilize established mathematical integration techniques to simplify the stress evaluation of thermal transients. Guidance will also be provided regarding the finite element evaluation that is needed to develop the unit stress response necessary for use of this software. A technical/user report will also be developed for this software.

2. Section III Fatigue Calculation Software Tool

   In this task, a software tool will be developed and benchmarked that performs fatigue evaluation in accordance with ASME Code, Section III, as follows:

   - user-inputs consisting of a six stress-component tensor time-history for a point on a RPV component for thermal transients (including pressure and mechanical loadings).
   - scaling and combination of the above stresses, as appropriate, into a total stress time-history tensor for the location being evaluated
   - stress-range pairing of the time-history tensor and conversion to principal stress ranges
   - cycle counting of the stress ranges
   - calculation of alternating stress intensities
   - calculation of CUF
   - calculation of environmental CUF in accordance with Regulatory Guide 1.207

   A technical/user report will also be developed for this software.

3. Develop a Technical Basis for Postulated HELB Locations

   The outcome of this task will be a technical report which documents the available background for this issue and any available information that was used to establish the previous CUF limit, the essential elements of refined fatigue analyses that could be performed for plants operating beyond 40 years to satisfy a revised CUF limit, the essential elements of a flaw tolerance approach that could be used in lieu of the current CUF limit for the operating reactor fleet, and a technical basis for a CUF limit to be used in HELB evaluations for affected piping systems in new reactors and plants operating to 60 years.

4. Technical Consulting from ANL

   This task includes technical and consulting support from ANL (with whom the NRC contracted for much of its earlier work on this subject) in the following areas:

   - reviewing proposed ASME Code Cases on environmental fatigue
   - reviewing data to determine the impact of strain threshold and holdtime effects
   - reviewing additional available laboratory data collected over the past decade to determine whether revision of Regulatory Guide 1.207 is necessary
   - investigating several practical issues associated with application of the Regulatory Guide 1.207 methodology
   - providing technical support to NRC staff in addressing environmental fatigue issues related to license renewal and new reactor design and providing NRC staff with knowledge transfer and subject matter turnover.

**For More Information**
Contact Gary L. Stevens, RES/DE at 301-251-7569 or Gary.Stevens@nrc.gov

# Degradation of Reactor Vessel Internals from Neutron Irradiation

## Background

The internal components of light-water reactor (LWR) pressure vessels are fabricated primarily with austenitic stainless steels because of their relatively high strength, ductility, and fracture toughness in their unirradiated state. During normal reactor operational conditions, the internal components 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 and increases their strength and their susceptibility to irradiation-assisted stress-corrosion cracking (IASCC). Exposure also decreases their ductility and fracture toughness. Cracks caused by IASCC have been found in a number of internal components in LWRs, including control rod blades, core shrouds, and bolts (see Figure 9.7).

As nuclear power plants age and neutron irradiation dose increases, the degradation of the vessel internals becomes more likely and potentially more severe. Preliminary data suggest that the significance of LWR vessel internals degradation could increase during both the license extension period (i.e., 40 to 60 years) and during even longer term operation of nuclear power plants.

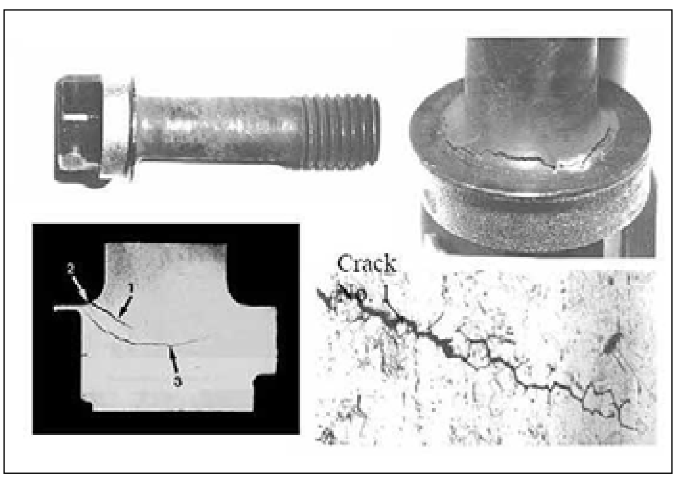



**Figure 9.7 Cracking of a baffle bolt in a pressure water reactor (PWR).**

## Objective

The NRC has developed a broad research plan to address the degradation of reactor vessel internals from neutron radiation. The results of the research will be used to provide insights into the causes and mechanisms of IASCC in boiling-water reactors (BWRs) and PWRs and to inform regulatory decisions regarding the reliability of reactor vessel internals during long-term operation.

## Approach

The NRC has been conducting research to characterize and evaluate irradiation-induced degradation by doing the following:

defining a threshold neutron dose above which irradiation begins to affect material properties

evaluating the adequacy of data used to estimate cyclic fatigue and IASCC growth rates for both the BWR and PWR vessel internal materials

assessing the significance of void swelling and irradiation stress relaxation/creep on the structural and functional integrity of PWR internal components.

Test specimens have been and will be irradiated over a broad range of prototypical exposure levels to evaluate the expected performance of plant materials. In addition, the research plan includes the harvesting of internal structural materials from decommissioned nuclear reactors, such as the Zorita reactor in Spain. Materials from the Zorita reactor have higher levels of radiation exposure than experimental samples and would provide information on the expected behavior of domestic BWR and PWR components during long-term operation. The plan also provides for participation in other collaborative research efforts that will leverage resources, extend knowledge acquired from previous research, and utilize unique testing facilities within the international community.

Presently, a systematic research effort is underway to determine the causes of IASCC, establish a fracture toughness degradation threshold, and investigate saturation effects in BWR and PWR internals. Representative reactor internal materials are being irradiated at the Halden Nuclear Reactor facility in Norway, and experimental testing is being carried out at Argonne National Laboratory (ANL). Specifically, within BWR environments, the effects on IASCC and fracture toughness from the hydrogen concentration in the reactor coolant and the concentration of light elements, such as sulfur and oxygen, within the steels are being evaluated. This portion of the work is nearing completion. In the next phase, the effects of neutron dose on IASCC and fracture toughness and the synergistic effects of neutron and thermal embrittlement on fracture toughness are being investigated for PWR environments. Longer term research will focus on effects expected during plant operation beyond 60 years.

As previously indicated, the NRC staff is completing a multiyear study of the effect of BWR environments on IASCC of austenitic stainless steel vessel internals. The products of this program have been used to evaluate licensee submittals related to managing degradation of these components and to inform other aspects of the regulatory process, such as inspection requirements and responses to relief requests. This program's results have led to the resolution of regulatory issues, as well as the development, validation, and improvement of regulations and regulatory guidelines.

**For more information**

Contact Dr. Appajosula S. Rao, RES/DE at 301-251-7636 or Appajosula.Rao@nrc.gov

# Primary Water Stress-Corrosion Cracking

## Background

Primary water stress corrosion cracking (PWSCC) in primary pressure boundary components composed of nickel-based alloy is a degradation mechanism that can affect the operational safety of pressurized water reactors (PWRs). PWSCC cracks found in control rod drive mechanism (CRDM) nozzle J-groove welds at North Anna Unit 2 are shown in Figure 9.8. The narrow cracks are often located in complex structures either within or adjacent to welds and are difficult to detect and characterize. Undetected PWSCC has led to reactor pressure boundary leaks and subsequent boric acid corrosion of the low-alloy steel reactor pressure vessel head at Davis-Besse in 2002 (see Figure 9.9).

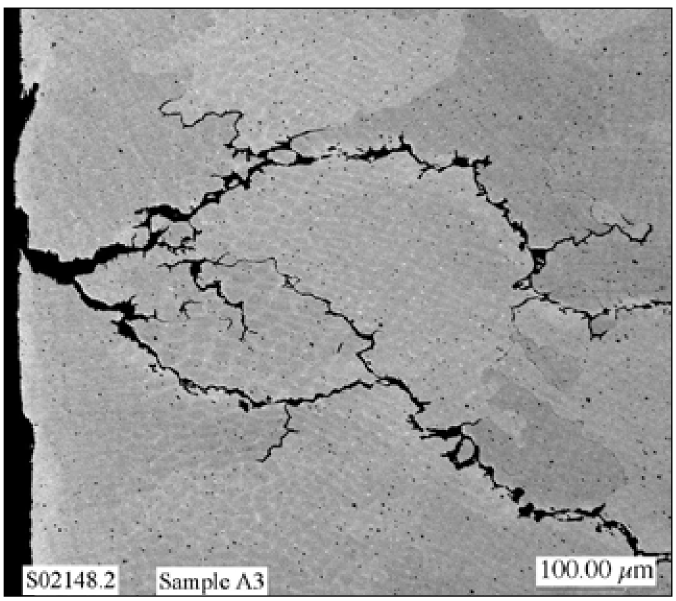



**Figure 9.8 PWSCC cracks in the Alloy 182 Jgroove weld in North Anna-2 Nozzle 31**

Alloy 690 and associated weld metals, Alloy 52 and 152, which have nominal chromium concentrations of 30 percent, have been used in replacement components, including steam generators, PWR replacement heads, reactor coolant system piping, nozzles, and instrument penetrations. PWSCC mitigation of the more susceptible alloys has been conducted using Alloy 52 and 152 weld overlays. To be successful, an improved understanding of the complex interrelations between stresses in the affected components, material microstructure, and the aggressive nature of the PWR environment is necessary.

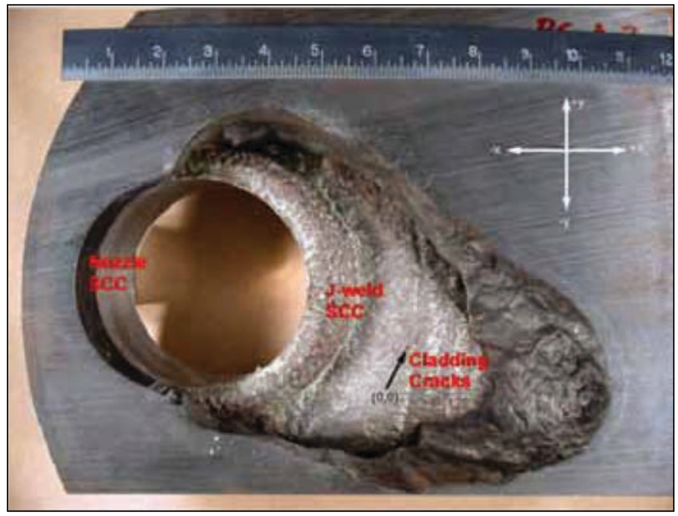



**Figure 9.9 Photograph showing extensive boric acid corrosion in the low-alloy steel Davis-Besse reactor pressure vessel head. Reactor coolant leaked from PWSCC cracks in the Alloy 600 control rod drive mechanism nozzle and the nozzle J-groove weld.**

## Objective

The objectives of this program are to evaluate the PWSCC susceptibility of high-chromium Alloy 690 and its weld metals, Alloys 152/52 and their variations, and to determine the relationship between PWSCC susceptibility and metallurgical characteristics of the chromium-containing nickel-based alloys used in replacement and new construction components. The work will also provide valuable information to assess potential mitigation methods for the lower chromium nickel-based alloys (600/182/82) originally used in PWRs and known to be susceptible to PWSCC.

Information obtained will be used to develop regulatory guidance and establish inservice inspection requirements necessary to ensure continued safe operation of PWRs.

## Approach

The NRC is sponsoring confirmatory research consisting of crack growth rate measurements on nickel-based alloys in simulated PWR environments, as well as microstructural and fracture surface analyses of test materials. The NRC is also participating in an international cooperative effort to evaluate factors that influence the PWSCC susceptibility of nickel-based alloys.

### NRC-Sponsored Research

The NRC has ongoing research activities on the PWSCC susceptibility of nickel-based alloys. Specific tests are being conducted to evaluate the importance of the following:

- fabrication processes and thermal treatments on Alloy 690
- shielded metal arc welding (SMAW) and gas tungsten arc welding (GTAW) processes
- heat-affected zones adjacent to SMAW and GTAW welds
- weld defects, including hot cracking and ductility dip cracking
- dilution zones in dissimilar metal welds

Examination of test specimen fracture morphology, along with metallurgical analyses and crack tip characterizations of test specimens and actual plant components which have been removed from service, will provide data to determine how the microstructural features affect PWSCC growth rates.

Results obtained from the NRC-sponsored research have shown that possible combinations of cold work and thermal treatments can significantly affect the PWSCC susceptibility of Alloy 690. High-chromium weld filler alloys are generally more resistant to PWSCC; however, higher susceptibility of some welds is still being investigated.

## PWSCC International Cooperation

The NRC is also participating in an international cooperative effort that includes representatives from the Electric Power Research Institute (EPRI), industry, and licensees. This cooperative effort has led to the development of PWSCC testing protocols and analysis methods, evaluation of representative plant materials, and testing of newly developed weld alloys. The cooperative effort provides a forum for the dissemination and discussion of research results which benefits all participants.

**For More Information**

Contact Darrell S. Dunn, RES/DE at 301-251-7621 or Darrell.Dunn@nrc.gov

# Primary Water Stress-Corrosion Cracking Mitigation Evaluations and Weld Residual Stress Validation Programs

## Background

In pressurized-water reactor (PWR) coolant systems, nickel-based dissimilar metal (DM) welds are typically used to join carbon steel components, including the reactor pressure vessel (RPV), steam generators (SGs), and the pressurizer, to stainless steel piping. Figures 9.10 and 9.11 show a representative nozzle to piping connection cross-section, including the dissimilar metal (DM) weld. The DM weld is fabricated by sequentially depositing weld beads as high-temperature molten metal that cools, solidifies, and contracts, retaining stresses that approach or, potentially, exceed the material's yield strength.

These DM welds are susceptible to primary water stress-corrosion cracking (PWSCC) as an active degradation mechanism that has led to RCS pressure boundary leakage. PWSCC is driven by tensile weld residual stresses (WRS) and other applied loads within the susceptible DM weld material. Hence, proper assessment of these stresses is essential to accurately predict PWSCC flaw growth and ensuring component integrity.

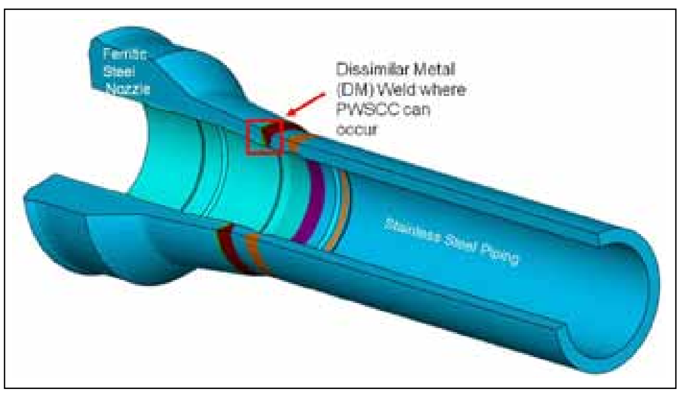



**Figure 9.10 Cutaway view of a carbon steel nozzle DM weld and stainless steel piping typical in a light-water cooled nuclear power plant**

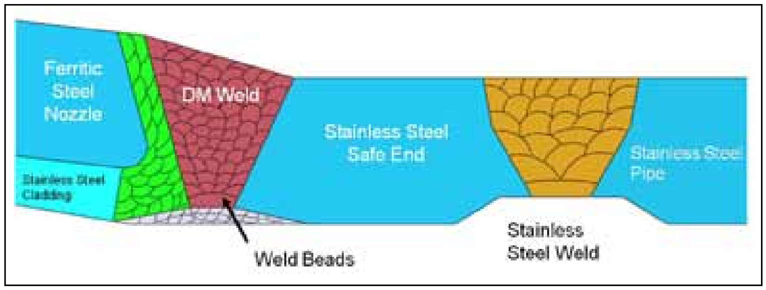



**Figure 9.11 Cross-section of nozzle to pipe weld highlighting weld bead pattern**

The nuclear power industry has developed several PWSCC mitigation techniques for DM welds that are currently being implemented in the PWR fleet. Examples include the following:

- full structural and optimized weld overlays, in which replacement material less susceptible to PWSCC is welded onto the outer diameter of the affected joint that also imparts a stress improvement to the susceptible joint
- weld inlays, in which a layer of replacement material less susceptible to PWSCC is welded to the inner diameter to act as a barrier between the corrosive reactor coolant and the DM weld material (e.g., similar to cladding)
- Mechanical Stress Improvement Processes (MSIP) in which the pipe is squeezed using a large hydraulically driven clamp that imparts a stress improvement to the susceptible joint

Weld overlays and MSIP reduce, and in some cases reverse, tensile residual stresses in DM welds, decreasing the driving force for crack growth. However, weld inlays have been shown to increase tensile WRSs, potentially increasing PWSCC initiation and growth, but, of the less susceptible replacement material.

## Validation Program

Recent improvements in computational capabilities have facilitated advances in WRS predictions using finite element analysis (FEA). Although no universally accepted methodology exists to model WRS using FEA, Electric Power Research Institute (EPRI) has developed draft guidelines for streamlining these procedures. The assumptions and estimation techniques vary from analyst to analyst, causing variability in the predicted WRS profiles.

RES is supporting the Office of Nuclear Reactor Regulation (NRR) in developing appropriate regulatory requirements to address PWSCC in reactor coolant piping systems. A portion of this effort includes the Weld Residual Stress Validation Program aimed at refining and validating the FEA procedures for modeling WRS and characterizing the uncertainties in the resulting predictions. The WRS Validation Program is being conducted cooperatively with EPRI under a memorandum of understanding addendum.

Figure 9.12 shows a typical WRS FEA performed using the ABAQUS software for a RPV-to-pipe nozzle DM weld. The distribution in stresses shows where a flaw may initiate (typically on the inner diameter of the DM weld), propagate, and cause leakage or structural instability. The results of this analysis are being validated through comparison of predicted temperature, thermal strain, and residual stress fields with a variety of physical measurements performed on actual and representative plant components and mockups.

The Weld Residual Stress Validation Program has enjoyed a number of successes thus far, including the following:

- evaluations of various PWSCC mitigation techniques (full structural weld overlays, optimized weld overlays, MSIP, and inlays)
- safety evaluation report technical basis development provided to NRR for approving several PWSCC mitigation techniques for use by the PWR fleet
- input to ASME Code Case reviews
- multiple plant-specific PWSCC flaw evaluations for NRR review

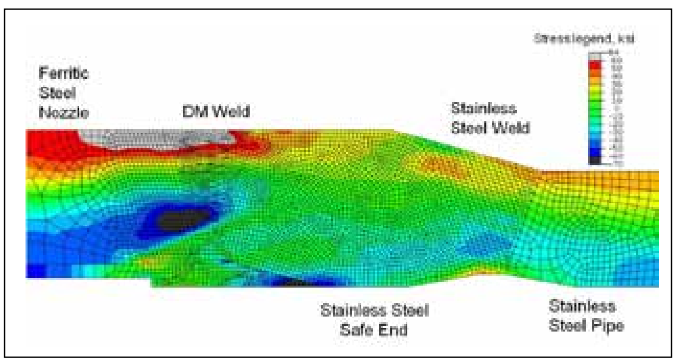



**Figure 9.12 Stress magnitude distribution in a nozzle to pipe weld configuration**

## Remaining Work

RES, RES contractors, and in cooperation with the nuclear power industry through an NRC/EPRI memorandum of understanding addendum, are currently completing a multiphase program to validate predictions of WRSs based on FEA. A major element of this program involves the International Weld Residual Stress Round Robin, in which 15 organizations are blindly and independently analyzing the WRSs in a representative pressurizer surge nozzle DM weld mockup, as seen in Figure 9.13. RES is conducting a blind validation of this mockup by measuring WRS and comparing the measurements to blindly conducted FEA predictions.

**Figure 9.13 Pressurizer surge nozzle DM weld mockup being measured for WRS**

Once completed, the WRS Validation Program will facilitate improvements in the following:

- WRS FEA predictive methodologies
- PWSCC flaw evaluation procedures and NRR staff review of licensee submittals
- determining estimates for the uncertainty and distribution of WRS, which are needed in probabilistic analyses (e.g., xLPR Code).

Figures 9.10–9.12 courtesy of Dr. Lee Fredette of Battelle Memorial Institute, Columbus, OH.

**For more information**
Contact Howard J. Rathbun, RES/DE at 301-251-7647 or Howard.Rathbun@nrc.gov

# Nondestructive Examination

## Background

As required by 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 their safety function is performed and that the requirements of the ASME Code are met. Research on nondestructive examination (NDE) of light-water reactor (LWR) components and structures provides the technical basis for regulatory decision-making related to these requirements. Results from the NDE of these components and structures are also used to assess models developed to predict the effects of materials degradation mechanisms and as initial conditions for component-specific fracture mechanics calculations. The Pacific Northwest National Laboratory (PNNL) is conducting this work.

## Regulatory Needs

Areas of concern addressed by NDE research include the following:

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

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

## Approach

### Evaluation of NDE Reliability and ISI Techniques

Research activities include NDE of fabrication flaws and destructive verification. The research objectives are to (1) determine the relationships among preservice inspection methods, inservice degradation (cracking, aging), and ISI practice and results; and (2) evaluate the effectiveness, accuracy, and reliability of new techniques expected to be applied by licensees in current, new, and advanced reactors. Certain materials, degradation mechanisms, and locations are difficult to inspect in the current fleet of reactors and will most likely remain challenging for new reactors. The NRC is using fabricated mockups and components removed from reactors, including some canceled plants and some operating reactors, to determine the effectiveness of existing and emerging NDE techniques (see Figures 9.14 and 9.15).

**Figure 9.14 Sectioning of reactor vessel head penetrations from WNP-1, a cancelled plant**

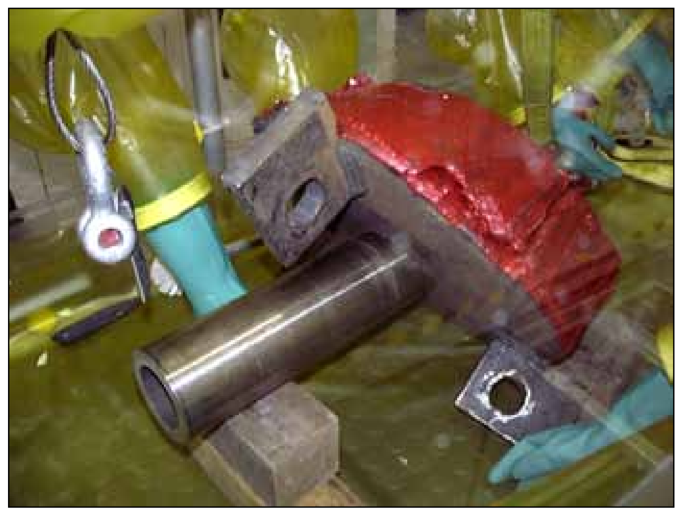


**Figure 9.15 Nondestructive and destructive examination of salvaged CRDM penetrations and J-groove welds from North Anna, Unit 2**

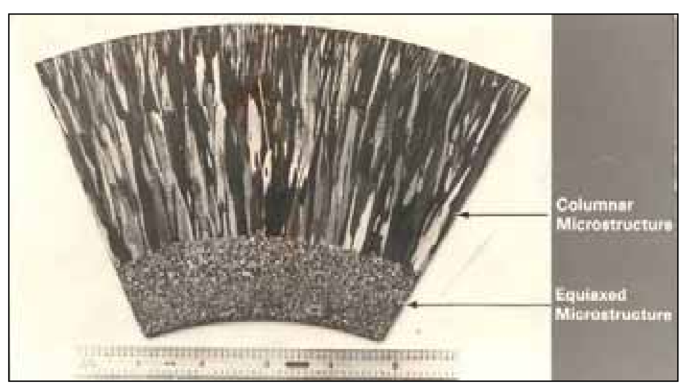



**Figure 9.16 Sample illustrating the coarse grain microstructure of centrifugally cast stainless steel**

The NRC performs some of this work under cooperative agreements to help defray costs and to gain access to the expertise of other organizations.  For example, the ability to detect and characterize PWSCC in LWR components is being evaluated under an NRC-initiated international project known as the Program for the Inspection of Nickel-Alloy Components (PINC).  This program is linking NDE performance to crack morphology and is developing NDE reliability data of advanced ultrasonics and other new NDE methods.  Eight organizations participate in PINC and exchange information and test results from their related research.

The NRC is directing, under its current program at PNNL, research on the inspection of coarse-grained austenitic alloys and welds (see Figures 9.16 and 9.17).  NDE of these components is difficult because of signal attenuation and reflections.  In these materials, grain boundaries and other microstructural features appear similar to cracks.  Research findings will support appropriate inspection requirements for these components so as to ensure safety.

### Enhanced Signal Processing and Analysis Systems

Modern NDE systems (Figure 9.17) produce a significant amount of data that must be examined during ISIs.  Automated data analysis algorithms reduce the processing time for large amounts of NDE data and thus improve ISI reliability by allowing more extensive inspections.  Computer-aided data analysis methods may further improve NDE reliability by reducing or eliminating operator-related errors.  Advanced processing techniques also support the use of alternative NDE techniques (e.g., high-resolution eddy current and phased array inspections).  The research is focused on determining the accuracy and reliability of advanced NDE for complicated defects in comparison with conventional techniques, confirmed by destructive examination.

### Advanced Inspection for Fabricated Components

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

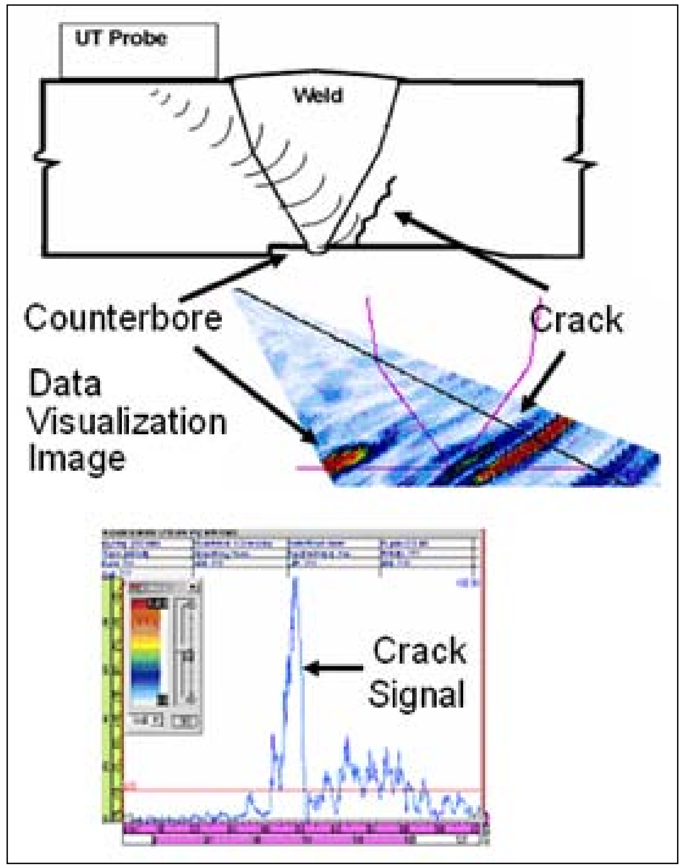



**Figure 9.17  Schematic view of flaw detection at the far side of a weld, using a phased array ultrasonic (PA-UT) technique.  PA-UT improves flaw detection in coarse-grained metals and welds**

### In Situ Material and Stress-State Characterization

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

## Summary

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

**For More Information**

Contact Aladar Csontos, RES/DE at 301-251-7640 or Aladar.Csontos@nrc.gov

# International Nondestructive Examination Round Robin Testing

## Background

Between November 2000 and March 2001, leaks were discovered in Alloy 600 control rod drive mechanism (CRDM) nozzles and associated Alloy 182 J-groove attachment welds in several pressurized water reactors (PWRs). Destructive examination of several CRDMs showed that the leaks resulted from primary water stress corrosion cracking (PWSCC). By mid-2002, over 30 leaking CRDM nozzles had been reported in the United States. Moreover, during this same time, a circumferential hairline crack was detected in the first weld between the reactor vessel nozzle and the A-loop hot-leg piping at another PWR that was subsequently determined to be PWSCC. Such events, both domestic and international, made it apparent that additional research was necessary to address PWSCC in dissimilar metal welds.

The NRC executed agreements with organizations from Japan, Sweden, South Korea, Finland, and the United States to establish the Program for the Inspection of Nickel-Alloy Components (PINC). Pacific Northwest National Laboratory (PNNL) assisted the NRC with the coordination of this program. Figure 9.18 depicts the organization of PINC.

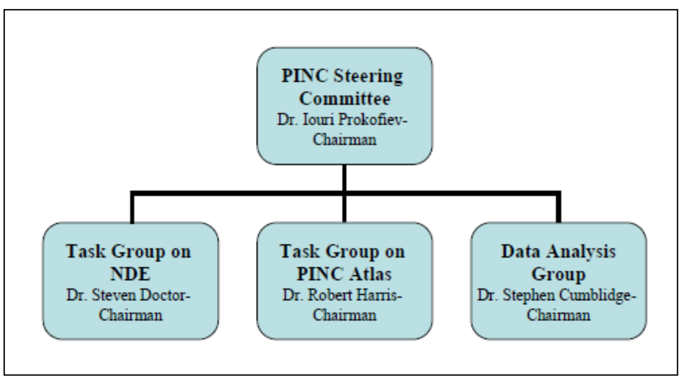



**Figure 9.18 Organization of PINC**

## Objective

The purpose of this program is to compile a knowledge base (archived through the information technology tool, PINC Atlas) on cracking in Alloy 600 and similar nickel-based alloys in nuclear power plants, including the crack morphology and nondestructive examination (NDE) responses. In addition, the program will identify and quantitatively assess the capabilities of current NDE techniques to detect, size, and characterize tight defects using NDE mockups with simulated PWSCC-like cracks.

## Approach

As part of their international collaboration, PINC participants identified, ranked, and determined which component configurations should be considered for the study.

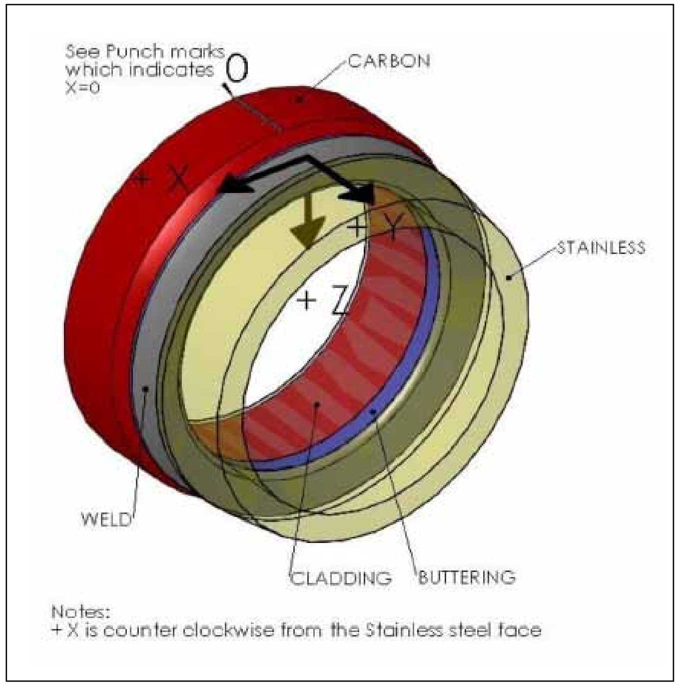



**Figure 9.19 Test Block 2.10 from Swedish Radiation Safety Authority Qualification Center**

A series of test blocks with cracks were then designed and fabricated by different contributors (such as shown in Figure 9.19) to simulate the selected component configurations. NUREG/CR-7019, "Results of the Program for the Inspection of Nickel Allow Components," issued August 2010, describes the results of the round robin tests that were performed to assess the NDE effectiveness and reliability.

The primary objective of the study was to produce an electronic resource on PWSCC in nickel-based alloys. This included documenting the material generated in support of an improved understanding of (1) PWSCC morphology, (2) NDE responses to PWSCC, and (3) the capability of NDE to reliably detect and accurately size PWSCC.

With regard to the second objective (i.e., investigate the capability of various NDE methods to detect and size the through-wall extent of PWSCC), NUREG/CR-7019 describes the efforts of the PINC participants to detect and measure the lengths of cracks. The surface conditions, access to both sides of the weld, and inspection conditions for the PINC specimens provided the inspectors with less challenging conditions than would be expected in field inspections of PWR components. Although the inspection conditions were less challenging, team performance was highly variable. This finding supports continuation of performance demonstration efforts in the nuclear industry to ensure adequate qualification of inspectors. The variability in team performance should be factored into the decisionmaking process when applying the results of this study.

Other key insights from the report include the following:

- Eddy current inspection from the cracked surface demonstrated the highest probability of detection for the examination of the dissimilar metal weld specimens.
- None of the NDE techniques in this round robin study demonstrated the capability to accurately measure the depths of flaws in dissimilar metal welds to ASME Code, Section XI, requirements.
- The study suggests that, in certain situations, examinations would be improved through the use of several NDE techniques to ensure adequate flaw detection and sizing.

## Program to Assess Reliability of Emerging Nondestructive Techniques

The results from PINC helped substantiate the fact that current NDE technology is sufficient to detect damage during only its final stages. Thus, the need for follow-on confirmatory research on emerging techniques for earlier detection of damage to plant components was clear.

In 2010, the NRC began an additional international cooperative project with a two-fold objective of early detection and prediction. Early detection will involve subjecting samples to temperature and stress and studying the damage in situ using, for example, acoustic emission. Predictions will involve developing NDE techniques to detect susceptibility of materials to damage.

The new Program to Assess Reliability of Emerging Nondestructive Techniques (PARENT) for dissimilar metal welds will focus on tight cracks, including PWSCC and hot cracks, in welds in piping and in other nuclear power plant components. It will assess the reliability of emerging NDE techniques to detect and characterize PWSCC in nickel-based primary reactor coolant system components.

The result of inspections on NDE test specimens containing representative simulated and fabrication flaws using more advanced, emerging techniques will be relevant to weld inlay and overlay repairs for existing reactors and to fabrication welds in new reactors. The Atlas information tool with PWSCC crack morphology and corresponding NDE results, developed under the PINC program, will be reviewed, applied, and extended to support inservice inspectors (see Figure 9.20).


**Figure 9.20 Information technology tool—PINC Atlas**

**For More Information**
Contact Dr. Iouri Prokofiev RES/DE at 301-251-7655 or Iouri.Prokofiev@nrc.gov

# Containment Liner Corrosion

## Background

Commercial nuclear power plant containment buildings are designed to act as a barrier to prevent radioactive release under severe accident conditions. Many nuclear power plants have containment buildings constructed with either reinforced or posttensioned concrete in contact with a thin steel containment liner, which serves as a leaktight membrane. Although the integrity of containment liners is assessed by leak rate tests and NRC-required inspections, three instances of through-wall corrosion of the liner have occurred since 1999. In all cases, liner corrosion was associated with foreign material embedded in the concrete during original construction (see Figure 9.21). Prior leak tests or inspections detected neither the foreign material nor the corrosion-related material loss. Active corrosion was identified after penetration of the liner had occurred.

**Figure 9.21 Photograph of the through-wall corrosion detected at Beaver Valley, 2009. A piece of wood embedded in the concrete during original construction was found behind the corroded area of the steel containment liner. The area was identified by a large paint blister which was filled with steel corrosion products.**

## Objective

The objectives of this program are to evaluate historical information about liner corrosion events, determine the mechanisms for through-wall corrosion, and determine whether plant designs and construction practices influence the susceptibility to liner corrosion. The results of the program will be used to assess the current methods of inspecting the liner and possible methods to mitigate liner corrosion. Knowledge gained will also be applied to the effects of plant aging on the integrity of the containment structure and the steel liner.

## Approach

Historical information on incidents of liner corrosion was gathered from several sources, including the following:

- In-service Inspection (ISI) reports and leak rate test results
- NRC inspection reports
- Licensee event reports
- International operating experience

Information is being analyzed to determine the relationships between liner corrosion incidents and plant design, operational parameters, and the presence of construction defects. The analysis conducted will be used to identify whether additional research or regulatory action is needed.

## Results

Review of historical information showed that containment liner corrosion initiating on the inside surface of containment liners as a result of degraded or damaged coatings and water collection behind moisture barriers occurs more frequently than corrosion at the liner-concrete interface. Although damage to moisture barriers and coating are more frequent, NRC-required inspections have resulted in early detection and mitigation of these incidents.

Operating experience indicates construction defects, such as fragments of wood present from the time of original construction, are a major contributor to liner corrosion at the concrete-liner interface. For containment structures designed so that the liner is in contact with the concrete (Figure 9.22), a foreign material in contact with the steel may retain moisture, promote crevice corrosion, and be the source of acidic decomposition products.

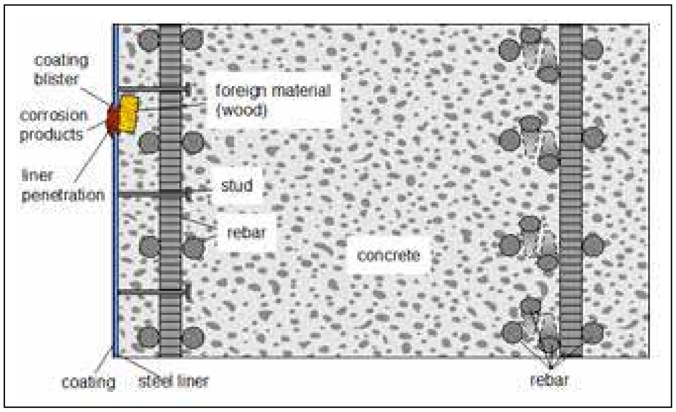



**Figure 9.22 Schematic showing a cross-section of a reinforced concrete containment structure with embedded foreign material from original construction and corrosion penetration of the steel containment liner**

## Future Efforts

The efficacy of current inspection methods and the value gained from augmented inspections will be assessed. Testing and modeling efforts may be beneficial to understand the effects of construction defects on corrosion of the steel containment liner and the potential benefits of coatings, sealants, concrete overlays, inhibitors, and cathodic protection systems as mitigation methods for concrete degradation and liner corrosion. Evaluation of the aging and degradation of these passive components will be necessary as nuclear power plants age and enter extended operation beyond 40 and 60 years of service.

**For More Information**
Contact Darrell S. Dunn, RES/DE at 301-251-7621 or Darrell.Dunn@nrc.gov

# Atmospheric Stress Corrosion Cracking of Dry Cask Storage Systems

## Background

Commercial nuclear power plants refuel every 18 to 24 months. Fuel removed from the core is placed in spent fuel pools for a minimum of 5 years. Independent spent fuel storage installations (ISFSIs), licensed under Title 10 of the Code of Federal Regulations Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste," are used when spent fuel pools have reached capacity (see Figure 9.23 for map of ISFSI locations). ISFSIs are initially licensed for 20 years, and license renewals for 40 years were recently completed for three ISFSI sites.

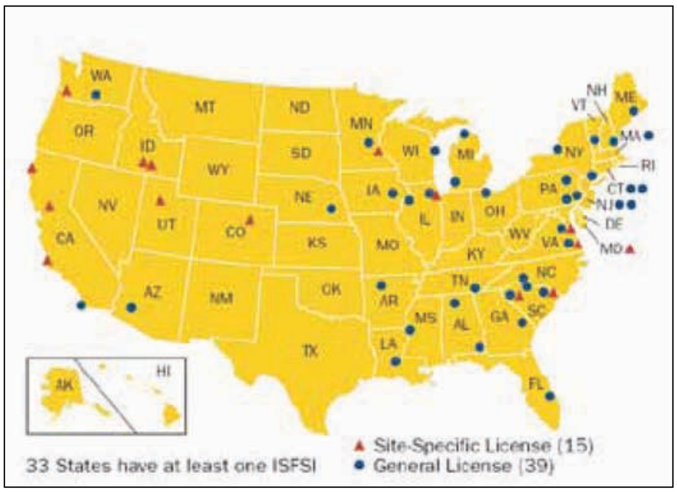



**Figure 9.23 ISFSI locations**

Dry storage systems at operating ISFSIs consist of canisters constructed using austenitic Type 304/304L/316/316L stainless steels (see Figure 9.24). Some of the current and possibly future ISFSI sites are located in coastal atmospheres where chloride containing salt as an airborne aerosol may deposit on the canister surfaces. A review of previous research provided little insight on the possible effects of salt accumulation over the expected range of operating temperatures for dry storage system canisters. Understanding the environmental conditions and material factors that influence atmospheric chloride stress-corrosion cracking (SCC) of austenitic stainless steel is necessary to evaluate the long-term operation of dry cask storage systems.

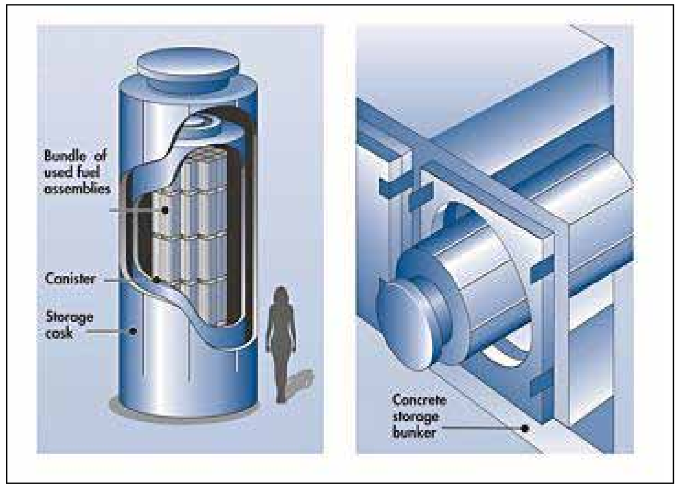



**Figure 9.24 Dry storage system designs**

## Objective

The objective of this research was to evaluate the potential for canister degradation at ISFSIs, including the deposition and accumulation of sea salts that may induce SCC. Evaluation of SCC susceptibility must consider the time-dependent changes in the environmental conditions on the surfaces of the stainless steel canisters, canister construction materials, and fabrication effects. Information obtained will help identify potential issues and regulatory requirements for long-term ISFSI operation in coastal atmospheres.

## Approach

The NRC sponsored research to evaluate the chloride SCC susceptibility of austenitic stainless steel dry storage systems exposed to coastal atmospheres. Accelerated laboratory tests were conducted using standardized U-bend test specimens produced from stainless steel Types 304, 304L, and 316L base metals, as well as 304/308, 304L/308L, and 316L/316L gas tungsten arc welded (GTAW) alloys. Accelerated atmospheric testing was conducted by placing the test specimens in an atmospheric chamber and heating the specimens to 40, 85, and 120 degrees Celsius (C) (104, 185, and 248 degrees Fahrenheit (F)). Dry sea salt deposited on specimens over a 2-week period was equivalent to an 18-month exposure in a coastal atmosphere. Environmental conditions inside the test chamber were controlled with alternating high and low relative humidity intervals to simulate daily fluctuations. Test specimens were examined after exposure periods of 4, 16, 32, and 52 weeks.

## Results

As illustrated in Figure 9.25, the high relative humidity led to the formation of chloride-containing solutions and chloride SCC on all specimens tested at 40 degrees C (104 degrees F).

Type 316L stainless steel was found to be slightly more resistant to chloride SCC as compared to Types 304 and 304L. SCC was observed on the Type 304 and 304L specimens after only 4 weeks of accelerated testing, whereas SCC on the Type 316L specimens occurred after 32 weeks. SCC was observed on U-bend specimens with and without welds. No SCC occurred on specimens tested at 85 and 120 degrees C (185 and 248 degrees F). Lower relative humidity at the higher temperatures precluded the formation of chloride-containing solutions.

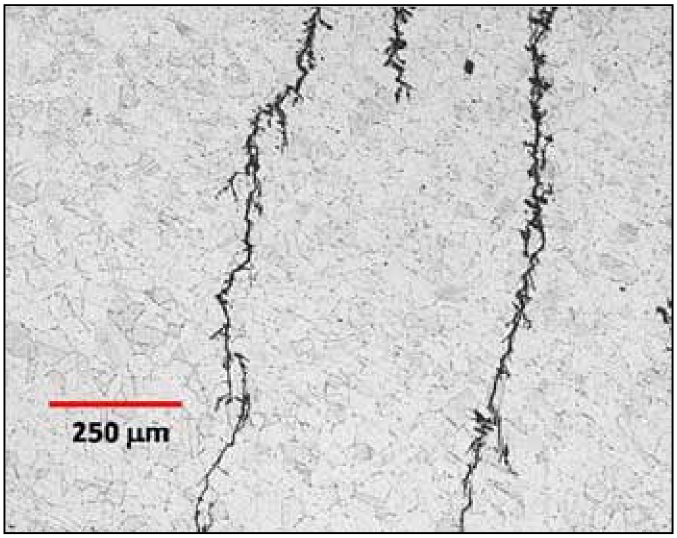



**Figure 9.25 Cross-section of a Type 304 U-bend specimen showing intergranular cracks after testing for 16 weeks at 40 °C (104 °F)**

## Summary and Future Work

- Results of accelerated testing under conservative conditions indicate that deposited sea salts can form chloride-containing solutions at high relative humidity values and promote SCC in austenitic stainless steels.
- Higher temperatures and lower relative humidity prevent the formation of chloride-containing solutions that can promote SCC.
- The implications of this research suggest that the SCC of ISFSI storage casks appears to be limited to a narrow range of conditions but may be more likely during extended operation as the storage canister surface temperatures decrease.
- The NRC will share the results of this research with industry as part of the ongoing cooperative efforts to address the safe long-term storage of spent fuel.

**For More Information**
Contact Darrell S. Dunn, RES/DE at 301-251-7621 or Darrell.Dunn@nrc.gov

# High-Density Polyethylene Piping Research Program

## Background

As seen in Figure 9.26, carbon steel piping used for nuclear power plant Class 3 safety-related service water systems have experienced general corrosion, microbiologically induced corrosion (MIC), and biofouling resulting in leakage and flow restriction.

**Figure 9.26 Corrosion and biofouling of carbon steel service water system piping**

As a result, the nuclear power industry has proposed to replace buried carbon steel piping service water systems with high-density polyethylene (HDPE). HDPE piping is typically immune to general corrosion, MIC, and biofouling; is less costly to install; and has a potential service life exceeding 50 years. HDPE piping is used extensively in natural gas distribution systems, as well as in municipal water piping systems, with great success. The mining and oil drilling industries offer examples of other applications of HDPE piping.

ASME Code, Section III, governs the design and installation of Class 3 safety-related service water piping systems. However, the ASME Code does not include the design and installation of HDPE piping systems. The ASME Section III/XI Special Working Group—Polyethylene Piping developed Code Case N-755, which provides rules for the design and installation of HDPE piping systems. Code Case N-755 addresses many of the issues related to using HDPE piping in Class 3 safety-related buried piping systems, but the NRC identified several issues related to the allowable service life conditions, pipe fusion, and inspection that need resolution before the agency will allow its general use by licensees. ASME is working to resolve these issues.

Since the NRC has not approved Code Case N-755, licensees have submitted relief requests for the substitution of carbon steel piping with HDPE piping for Class 3 safety-related applications. The agency has granted two such relief requests, which relied heavily on Code Case N-755, but the NRC imposed several additional requirements to help ensure piping and fusion joint integrity (see Figure showing HDPE piping installed at a nuclear power plant).

**Figure 9.27 Installation of underground Class 3 safety-related HDPE piping**

## Regulatory Needs

The objective of this program is to conduct confirmatory research to assess the service life, design, fabrication, and inspection requirements proposed in Code Case N-755. Since HDPE is a new material for safety-related applications at nuclear power plants, data and analyses are needed to independently verify the requirements in Code Case N-755 and its application to existing and new nuclear power plants.

## Approach

RES is performing confirmatory tests and analyses on HDPE piping to evaluate the following:

- allowable service life conditions for pipe and fusion joints
- piping system design requirements
- fusion procedure qualification requirements
- nondestructive testing methods and procedure qualification requirements

RES is active in ASME Code activities related to HDPE piping and coordinates HDPE piping issues with the Office of Nuclear Reactor Regulation (NRR) and the Office of New Reactors (NRO) for eventual ASME resolution.

**For More Information**
Contact Eric Focht, RES/DE at 301-251-7649 or Eric.Focht@nrc.gov

# Neutron Absorbers in Spent Fuel Pools

## Background

After use in a nuclear reactor, fuel bundles are stored in the spent fuel pool in cells formed by a stainless steel rack structure. Subcriticality in the pool is often credited to panels of boron-10 containing neutron absorber materials which are placed within the rack walls. In the past 30 years, neutron absorber materials have shown various types of degradation, such as blistering or matrix degradation (see Figures 9.28 and 9.29). Incidents of excessive degradation are summarized in Information Notice 09-26, "Degradation of Neutron-Absorbing Materials in the Spent Fuel Pool," dated October 28, 2009. Degradation of credited neutron absorber panels may invalidate the geometric and areal density assumptions used for the original criticality calculations of record and challenge the requirement for $k_{eff}$ to be less than 0.95 as specified by Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.68, "Criticality Accident Requirements".

Currently, plants assess the neutron absorption capabilities of these materials by three means: (1) mapping of degradation through software calculation packages, such as RACKLIFE; (2) direct in situ measurement of neutron absorption, such as Boron-10 Areal Density Gauge for Evaluating Racks (BADGER) testing; and (3) analysis of test coupons positioned in the pool. If significant degradation is detected, plants may reduce $k_{eff}$ by reracking existing bundles and replacing or adding neutron absorber panels/inserts. However, as extended plant operations produce more and more spent fuel bundles, which need to be placed into previously unoccupied cells, and as neutron absorber materials age further, reracking and panel/insert addition or replacement have become more difficult.

## Objective

In light of these new concerns, the NRC staff is initiating a research program to catalog plants' current use of neutron absorbers and evaluate the efficacy of current surveillance programs. Results of the program will guide future regulatory decisions pertaining to spent fuel pools.

## Approach

### Compilation of Existing Data

The NRC staff is currently collecting available historical data concerning spent fuel pools and absorbers from public licensing and relicensing documents. The staff is also collaborating with industry groups to obtain additional information. A complete databank of neutron absorber information will allow the staff to identify degradation trends as a function of factors such as panel age, fluence, or pool environment.

### Evaluation of Surveillance Methods and Programs

Over the next 2 years, the staff is planning to conduct research to verify the accuracy of the BADGER testing and the RACKLIFE degradation modeling program. The staff is also studying mechanisms and rates of neutron panel degradation to determine whether these or other surveillance methods and programs can detect loss of neutron absorber capability and if such detection will occur in a timely manner.

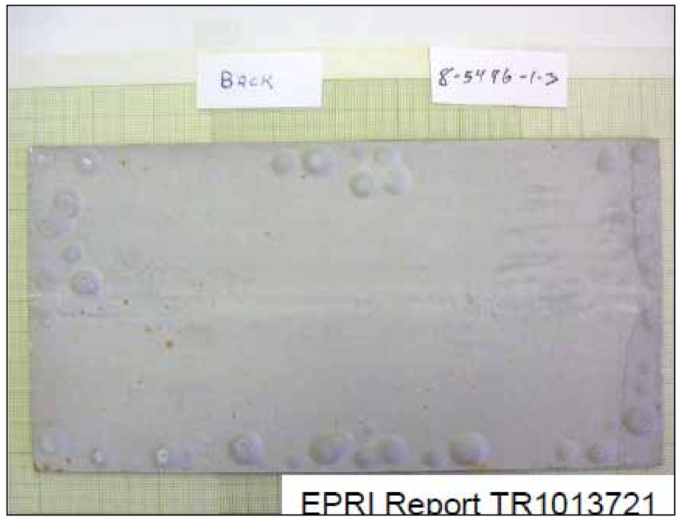



**Figure 9.28 Blistering on the aluminum cladding of Boral neutron absorber**


**Figure 9.29 Degradation of the composite matrix in Boraflex neutron absorber**

**For More Information**

Contact April Pulvirenti, RES/DE at 301-251-7976 or April.Pulvirenti@nrc.gov

# Chapter 10: Digital Instrumentation and Control and Electrical Research



# Digital Instrumentation and Control

## Background

The digital instrumentation and control (I&C) area continues to evolve as the technology changes and the U.S. Nuclear Regulatory Commission (NRC) continues to refine its regulatory approach. Current control rooms are dominated by analog equipment, such as electromechanical switches, annunciators, chart recorders, and panel-mounted meters. However, as operating nuclear power plants (NPPs) upgrade their control rooms, analog equipment is being replaced with modern digital equipment, including flat screen operator interfaces and soft controls. Future plants will have highly integrated control rooms similar to those in Figure 10.1. The NRC has seen a substantial increase in the proposed use of digital systems for new reactors and retrofits in operating reactors. As a result, the NRC continues to update applicable licensing criteria and regulatory guidance and perform research to support licensing these new digital I&C systems.

In the 1990s, the NRC developed guidance to support the review of digital systems in NPPs. Since that time, the NRC has been effectively using the current licensing guidance for review of applications of digital technology in operating reactors and in certification of new reactor designs. In an effort to continually improve the licensing process, the NRC commissioned the National Academy of Sciences' National Research Council to review issues associated with the use of digital systems. The National Research Council issued its report, "Digital Instrumentation and Control Systems in Nuclear Power Plants," and made several recommendations, which included a recommendation to update the NRC research program to balance short-term regulatory needs and long-term anticipatory research needs. The Advisory Committee on Reactor Safeguards (ACRS) has also encouraged research in the digital I&C area to keep pace with the ever-changing technology.

## Overview

In 2005, the Office of Nuclear Regulatory Research (RES) developed a comprehensive 5-year Digital System Research Program Plan, which defined the I&C research programs to support the regulatory needs of the agency. In 2007, the NRC formed a Digital I&C Steering Committee and seven task working groups (TWGs) to work with the nuclear industry in improving regulatory guidance for digital I&C system upgrades in operating reactors, support design certification submittals for new reactors, and support review of digital I&C systems in fuel cycle facilities. The TWGs issued new interim staff guidance to address specific digital I&C regulatory issues. In 2010, the agency developed an updated Digital System Research Plan with input from several sources, including the National Research Council's report on digital I&C systems at nuclear power plants, ACRS, external stakeholders, and the NRC staff. The updated research plan consists of five research program areas: (1) safety aspects of digital systems, (2) security aspects of digital systems, (3) advanced nuclear power concepts, (4) knowledge management, and (5) carryover projects. The products of these research programs include technical review guidance, information to support regulatory-based acceptance criteria, assessment tools and methods, standardization, and knowledge management initiatives.

**Figure 10.1 Highly integrated control room**

## Research Program

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

Work in the area of safety aspects of digital systems includes analytical assessment research to support safety evaluations of digital I&C systems. Ongoing research is developing an inventory and classification for NPP digital systems, elicitation of expert opinions on the state of the art in analysis of safety critical systems, failure mode and operational experience analysis, and a safety demonstration framework. This research will improve the understanding of how digital systems may fail and develop the commensurate criteria to ensure that these systems will not compromise their safety functions and not affect NPP safety adversely. Other research projects are investigating fault-tolerant testing techniques and advanced diagnostics and prognostics. The NRC and the industry are interested in risk-informing digital safety system licensing reviews. One of the major challenges to risk-informing digital system reviews is developing an acceptable method for modeling digital system reliability. The staff examined a number of reliability and risk methods that have been developed in other industries, such as aerospace, defense, and telecommunications. Based on its review of these techniques and available failure data, the staff performed benchmark studies of digital system modeling methods, including traditional event-tree/fault-tree and dynamic methods. Internal staff and ACRS reviews of the studies challenged the viability of the methods and availability of data needed. Further research on the failure modes of digital systems and quantitative software reliability is being pursued.

With respect to the security aspects of digital systems, the staff developed a new Regulatory Guide 5.71, "Cyber Security Programs for Nuclear Facilities," in support of Title 10 of the *Code of Federal Regulations* (10 CFR) 73.54, "Protection of Digital Computer and Communication Systems and Networks." The staff is actively engaged in ongoing cyber research to explore cyber vulnerabilities in digital systems and networks, including wireless networks that are expected to be deployed in NPPs. This research will ultimately provide improved regulatory guidance and tools for evaluating digital systems and networks for cyber vulnerabilities, including potential vulnerabilities arising from safety and nonsafety system interconnections.

The staff is also staying abreast of advanced nuclear power concepts in the digital systems area. In support of the U.S. Department of Energy's advanced reactor design programs, Next Generation Nuclear Plant, and the proposed license applications for small modular reactors, research projects to investigate unique regulatory aspects for advanced I&C are underway.

In the knowledge management area, collaborative research efforts in the United States and internationally support sharing regulatory standards and research data for digital systems. There are ongoing efforts to share operational experience data and analysis techniques with industry via the Electric Power Research Institute; with other Government agencies, such as the National Aeronautics and Space Administration, and with research organizations in other countries. Research supports international NPP digital system standards harmonization and NRC knowledge management and regulatory efficiency improvements.

**For More Information**

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

# Digital Instrumentation and Control Probabilistic Risk Assessment

## Background

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

The current licensing process for digital I&C systems is based on deterministic engineering criteria. In its 1995 policy statement on probabilistic risk assessment (PRA), the Commission encouraged the use of PRA technology in all regulatory matters to the extent supported by the state of the art in PRA methods and data (Volume 60, page 42622, of the *Federal Register*). Although many activities have been completed in the area of risk-informed regulation, the risk-informed analysis process for digital I&C systems has not yet been satisfactorily developed. Since, at present, no consensus methods exist for quantifying the reliability of digital I&C systems, one of the programs included in the NRC digital I&C research plan addresses risk assessment methods and data for digital I&C systems. The objective of this research is to identify and develop methods, analytical tools, and regulatory guidance to support (1) nuclear power plant licensing decisions using information on the risks of digital systems and (2) inclusion of models of digital systems in PRAs of nuclear power plants.

## Approach

Previous and current Office of Nuclear Regulatory Research (RES) projects have identified a set of desirable characteristics for reliability models of digital systems and have applied various probabilistic reliability modeling methods to an example digital system (i.e., a digital feedwater control system (DFWCS)). Figure 10.2 provides an illustration of one of these modeling methods. Several NUREG/CR reports, which have received extensive internal and external stakeholder review, document this work. The results of these benchmark studies have been compared to the set of desirable characteristics to identify areas where additional research might improve the capabilities of the methods. One specific area currently being pursued by RES is the quantification of software reliability. To examine the substantial differences in PRA modeling of software (versus conventional nuclear power plant components), in May 2009, RES convened a workshop involving a team of experts with collective knowledge of software reliability and/or nuclear power plant PRA. At the workshop, the experts established a philosophical basis for modeling software failures in a reliability model. RES is now reviewing quantitative software reliability methods and plans to develop one or two technically sound approaches to modeling and quantifying software failures in terms of failure rates and probabilities. Assuming such approaches can be developed, they will then be applied to an example software-based protection system in a proof-of-concept study.

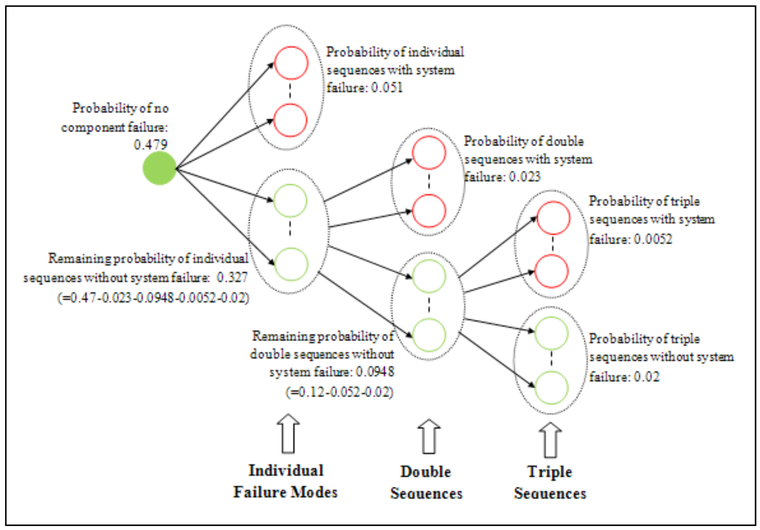



**Figure 10.2 Condensed Markov state transition model for quantifying DFWCS failure frequency from hardware failures**

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

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

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

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

### International Collaboration

In October 2008, RES staff led a technical meeting on digital I&C risk modeling for the working group on risk (WGRisk) of the Organization for Economic Cooperation and Development (OECD), Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI). The objectives of this meeting were to make recommendations regarding current methods and information sources used for quantitative evaluation of the reliability of digital I&C systems for PRAs of nuclear power plants, and identify, where appropriate, the near- and long-term developments necessary to improve modeling and evaluation of the reliability of these systems. While the meeting did not produce specific recommendations of the methods or information sources that should be used for quantitative evaluation of the reliability of digital I&C systems for PRAs of nuclear power plants, it did provide a useful forum for the participants to share and discuss their experience with modeling these systems. The report documenting the meeting is available on the NEA Web site at http://www.nea.fr/nsd/docs/2009/csni-r2009-18.pdf. A follow-on WGRisk activity is now getting underway that will focus on development of a failure mode taxonomy for digital I&C systems for use when incorporating digital I&C systems into PRAs of nuclear power plants.

## References

NUREG/CR-6962, "Traditional Probabilistic Risk Assessment Methods for Digital Systems," October 2008.

NUREG/CR-6997, "Modeling a Digital Feedwater Control System Using Traditional Probabilistic Risk Assessment Methods," September 2009.

Chu, T.L., et al., "Workshop on Philosophical Basis for Incorporating Software Failures into a Probabilistic Risk Assessment," Brookhaven National Laboratory, Technical Report, BNL-90571-2009-IR, November 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML092780607).

**For More Information**

Contact Alan Kuritzky, RES/DRA at 301-251-7587 or Alan.Kuritzky@nrc.gov

# Analytical Assessment of Digital Instrumentation and Control Systems

## Background

New and proposed digital instrumentation and control (I&C) systems in nuclear power plants (NPPs) pervade and affect nearly all plant equipment, with increasing interdependencies (e.g. through interconnections, resource sharing, and data exchanges), complexity is increasing. These interdependencies are becoming increasingly difficult to identify, analyze, and understand. Configurations of these networked systems tend to have plant-specific differences, such that no two systems are identical. The operating history of such systems is relatively short and, by the very nature of the systems and expected changes, is not likely to become statistically significant. In addition, unanticipated failure modes could create very confusing situations that might place the plant, or lead operators to place the plant, in unexpected or unanalyzed configurations. Under these conditions, evaluation for licensing has become very challenging. The Advisory Committee on Reactor Safeguards has also observed these concerns.

## Overview

This research project, which addresses these concerns, is driven by a combination of the Commission's staff requirements memoranda (SRM)-M070607 and SRM-M080605B and the needs expressed by the regulatory offices through the fiscal year (FY) 2010–FY 2014 NRC Digital Systems Research Plan (ADAMS Accession No. ML100541484).

Using existing theoretical knowledge in the fields of software and systems engineering for high-confidence, real-time control systems, this research will develop a framework of knowledge about how and why digital I&C systems may fail. The framework will allow continuous enrichment with new knowledge gained from operating experience and other research inside and outside the NPP application domain.

## Objectives

In support of the safety evaluation of digital I&C systems, this research will improve the understanding of how systems may fail and develop the commensurate criteria to ensure that these systems will not compromise their safety functions and affect NPP safety adversely.

Knowledge in the form of a causality framework will be useful in improving root-cause analysis of operating experience and will serve to inform companion research in PRA. Knowledge about modes of degradation will also inform research in the effects of degraded I&C on human performance.

## Approach

Based on an inventory of current and future digital I&C devices and systems in NPPs, the three preapproved digital I&C platforms, and emerging trends in digital technology, this research will characterize the NPP application domain and, for this bounded domain, it will identify credible failure and fault modes and analyze their effects, including the operating crew, the plant, and other affected systems. Since the limited failure modes of mature technology hardware components is relatively well understood and the practice of their application is relatively mature, the scope of this research focuses on understanding the failure modes of systems and systems of systems, the causes of such failures, and the criteria or conditions to avoid or prevent such failures. Of particular interest are the system failures caused by complex logic, whether implemented in the form of software, a field-programmable gate array, or a complex programmable logic device.

To acquire relevant knowledge outside the NPP industry, the NRC reached out to the world's leading researchers in safety-critical software and systems engineering and pursued an elicitation process culminating in a two-day clinic. The results from this expertise elicitation activity have shaped the direction of some of the research described below.

### Inventory and Classification of Digital Instrumentation and Controls Systems

In cooperation with the nuclear industry, this study will establish an inventory of current and future digital I&C devices and systems in NPPs. The purpose is to understand the scope and nature of the systems on which safety assessment research should be focused. The inventory will include enough information to allow characterization of the domains of applications in NPPs (the digital I&C devices and systems and their relationships to their environments) and clustering the inventoried items into classes of similarities. Example elements of information include: (1) the role or NPP function in which the item is applied, (2) whether the item stands alone or is interconnected, (3) various aspects and indicators of the complexity of the item, (4) the degree of verification or qualification, (5) properties of its architecture, and (6) properties of its development process to the extent that these elements or information are available. The intent of such characterization is to facilitate the understanding of possible adverse behaviors and approaches to ensure freedom from adverse behaviors.

### Digital Instrumentation and Control Failure and Fault Modes Research

This study will establish an analytical framework for organizing knowledge about how and why digital I&C systems may fail. The scope of the study will be limited to the domain of digital I&C devices and systems (e.g., classes of devices and systems represented in the existing inventory, NRC preapproved platforms, and trends in new licensing applications). The scope includes an analysis of systems with tightly coupled integration of traditionally decoupled or loosely coupled functions, applications (e.g., reactor trip system, engineered safety features actuation system), signals, and infrastructural services, as exemplified in new licensing applications.

Knowledge about failures, faults, and their causes will be organized in a reusable manner. Coupled with causal knowledge will be research on criteria or conditions to avoid or prevent such faults (e.g., constraints on the architecture and the development process).

## Knowledge Elicitation from Experts

To acquire relevant knowledge outside the NPP industry, the NRC reached out to the world's leading researchers in safety-critical software and systems engineering and pursued an elicitation process culminating in a two-day clinic held in January 2010 to identify the following:

- current limitations in the assurance of complex logic and areas of uncertainties
- evidence needed for effective assurance
- areas in need of research and development

The pool of experts represented a broad diversity in cultural backgrounds, application domains, and thought processes. Countries of origin included the United Kingdom, Sweden, Germany, Australia, New Zealand, Canada, and the United States. Application domains included defense, space flight, commercial aviation, medical devices, automobiles, telecommunications, and railways.

Through a chain of referrals by the experts, the NRC built a candidate pool of over 75, of which more than 30 experts were available for individual interviews. Based on common patterns emerging from the collective interviews, the NRC drafted a reference position to confirm with the experts the areas of general agreement and to identify areas for deeper discussion. While certain findings confirmed NRC staff positions, other findings revealed opportunities to improve the rigor and depth of NRC reviews. For example, the experts confirmed that the safety assessment for a digital I&C system will continue to require high caliber judgment from a diverse team, commensurate with the complexity of the system and its development process and environment, such as in systems containing complex software or other manifestations of complex logic. To exercise reasonable judgment, the review team will require a variety of complementary types of evidence, integrated with reasoning to demonstrate that the remaining uncertainties will not affect system safety adversely. The experts recommended that, in the absence of such demonstration, there should be diverse defensive measures, independent from digital safety systems using complex software or other implementations of complex logic or products of software-intensive tools.

## Safety Demonstration Framework

In accordance with recommendations from the experts, as mentioned above, the NRC is investigating the application of the evidence-argument-claim structure (variously known as an assurance case or a safety case) to systematize the safety evaluation of a complex digital I&C system (see Figure 10.3). Although the NRC has a comprehensive regulatory guidance framework, certification or licensing applications submitted for review tend to address the various requirements and guidelines separately, rather than in a safety-goal-oriented integrative manner. This research will explore mapping the NRC's regulatory guidance framework into a safety-goal-oriented evidence-argument-claim framework.

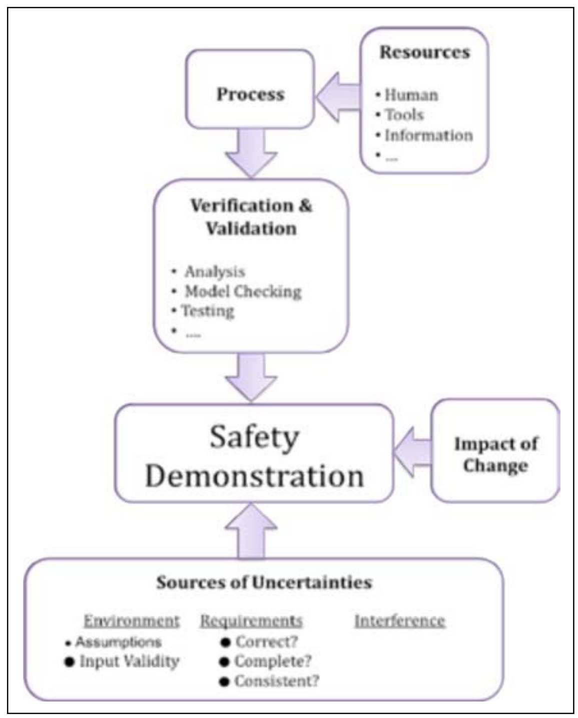



**Figure 10.3 Integrating different types of evidence to demonstrate that a system is safe**

**For More Information**
Contact Luis Betancourt, RES/DE at 301-251-7409 or Luis.Betancourt@nrc.gov

# Cyber Security for Digital Instrumentation and Control Systems

## Background

As the nuclear industry continues to integrate modern digital technology into control and protection systems, new challenges and issues arise with ensuring that the critical functions at NPPs are protected from both malicious and unintentional acts through computer-based resources and communication networks. In 2009, the NRC issued a new cyber security rule in Title 10 of the Code of Federal Regulations (CFR) 73.54, which requires licensees to provide high degree of assurance that digital systems and networks are adequately protected from cyber threats up to and including the design-basis threat. In January 2010, the NRC office of research (RES) issued an accompanying Regulatory Guide 5.71, "Cyber Security Program for Nuclear Facilities," which provides guidance to licensees on how to comply with the recently issued cyber security rule.

## Objective

The Digital Instrumentation and Controls Branch within RES has been actively involved in cyber security research for several years. One objective of this research is to ensure that NRC's regulatory framework is appropriately updated to consider the anticipated increase of digital technology in existing and new reactors. This objective is covered by several cyber security research projects outlined in the Digital System Research Plan for FY 2010–FY 2014 and by establishing new regulatory guidance, such as Regulatory Guide 5.71.

## Issuance of Regulatory Guide 5.71

The cyber security rule (10 CFR 73.54) requires licensees to protect digital computer and communication systems and networks associated with safety-related functions, important to safety functions, security functions, emergency preparedness functions, and relevant support systems and equipment. Regulatory Guide 5.71 provides a performance-based approach that the NRC staff believes is acceptable for complying with 10 CFR 73.54. The approach outlined in Regulatory Guide 5.71 provides guidance to formulate a viable cyber security plan, identify critical digital assets (CDAs), and apply extensive National Institute of Standards and Technology (NIST) cyber security controls tailored specifically for NPPs.

The NIST cyber security standards (Special Publication 800-53, "Recommended Security Controls for Federal Information Systems," and Special Publication 800-82, "Guide to Industrial Control Systems (ICS) Security") were adapted for use in nuclear facilities by considering some of the unique challenges posed by the intersection of industrial control systems and traditional information technology (IT) systems.

Since Regulatory Guide 5.71 provides a comprehensive approach to securing digital systems and networks, ongoing cyber security research seeks to produce tools and processes to better facilitate reviews of the cyber security plans, inspections of the cyber security plans, and technical reviews for future digital safety systems.

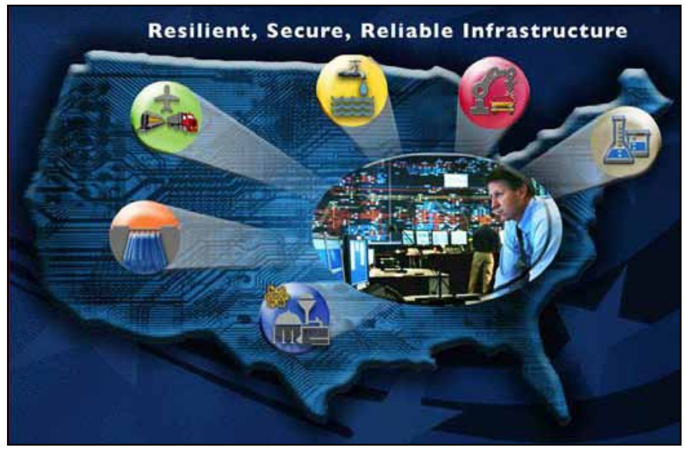



**Figure 10.4 Graphic of United States showing infrastructure of various utilities**

## Digital Safety System Vulnerability Assessments

Potential interactions between safety and security are an identified concern for digital upgrades in existing reactors and all-digital control and protection architectures for new reactors. The purpose of vulnerability assessments is to identify potential platform-based weaknesses. A critical digital safety system, such as a digital reactor protection system, may consist of several CDAs and could have multiple potential access points that might weaken the security posture if exploited by an adversary, such as a disgruntled insider. This project aims to first identify weaknesses, or vulnerabilities, in generic digital safety systems and then identify appropriate controls or practices that could mitigate or eliminate the identified vulnerabilities.

The approach for the digital safety systems vulnerability assessment is based on the Sandia National Laboratories' red teaming process, which is a planned vulnerability assessment performed from the perspective of a postulated adversary. The red teaming process is adapted to model a defined adversary threat level and to observe the test systems' response to attack progressions performed in a safe laboratory environment. This project will support development of regulatory positions on cyber security. The deliverables will also provide the NRC staff with enhanced processes and tools for identifying and assessing security vulnerabilities. Deliverables from the digital safety system vulnerability assessments could also assist NRC staff reviews of cyber security plans, aid in training for regional cyber security inspections, and inform technical reviews for safety requirements.

## Network Security

The network security project supports regulatory priorities discussed with the NRC's Nuclear Security and Incident Response Office by identifying generic protection and mitigation measures appropriate to NPP environments. The project will support the review of digital I&C system upgrades in currently operating nuclear plants and future plants. This project also supports the Advanced Reactor Research Program.

Networking (both wireless and wired) is the interconnection of components (e.g., controllers, actuators, and sensors) with the objective of communicating among the associated subsystems. This networking of subsystems within a larger system framework can present security vulnerabilities in the system as a result of weaknesses in the network design that could be exploited during a cyber attack propagated through a vulnerable subsystem. These vulnerabilities could be inherent in the system features or could be incorporated into the system features during system development or before system installation.

The network security research project addresses secure network design techniques for networks yet to be installed in nuclear facilities. This research will obtain from digital industry security experts information regarding cyber vulnerability mitigation strategies that can be built into or added onto digital system architecture designs during the network design and development phase. The research also will identify strengths and weaknesses of various network architecture designs, including built-in and added-on cyber vulnerability mitigation strategies. The areas to be addressed will include preferred practices that prevent or mitigate insider cyber attack vectors, outsider cyber attack vectors, and developer cyber attack vectors.

## Wireless Network Security

In wireless communications, a signal is transmitted through a shared medium instead of a controlled conductive path, such as wires. Irrespective of the transmission medium, wired or wireless, many established security controls, such as those identified in Regulatory Guide 5.71, apply to any network. If a wireless network is not directly associated with a critical system at an NPP, it could provide a pathway to wired assets, which would qualify it as a wireless CDA according to guidance in Regulatory Guide 5.71.

Because wireless communications travel through air, and not a controllable path, interference from the environment, surrounding equipment, and structures becomes a dominant security and performance issue. The use of a shared transmission medium makes wireless network architecture and security implementation different from that of the wired network. Past research (e.g., NUREG/CR-6882, "Assessment of Wireless Technologies and Their Application at Nuclear Facilities," issued July 2006) has identified and assessed numerous security-related issues associated with implementing wireless systems, such as denial of service, wired equivalent privacy encryption, wireless telephony, and unsecured access points. Examples of the combinations of defensive measures to be explored in this project include password protection, encryption, administrative controls, network diversity and segmentation, firewalls, access point management (roaming), signal/noise/strength level monitoring, effects of wireless sensor usage, signal strength management, and even signal direction management. These security considerations are identified and addressed in a deliverable for the Wireless Network Security project being led by experts from the Oak Ridge National Laboratories.

**For More Information**
Contact Jeanne Dion, RES/DE at 301-251-7482 or Jeanne.Dion@nrc.gov

# Susceptibility of Nuclear Stations to External Faults

## Background

Offsite power is considered to be the most reliable electrical source for safe operation and accident mitigation in Nuclear Power Plants (NPPs). It is also the preferred source of power for normal and emergency NPP shutdown. When offsite power is lost, emergency diesel generators provide onsite power. Consequently, if both power sources are lost, a total loss of alternating current power could occur, resulting in station blackout, which is one of the significant contributors to core damage frequency.

In August 2003, an electrical power disturbance in the northeastern part of the United States caused nine NPPs to experience a loss-of-offsite power (LOOP) event. This event, which was initiated by an overgrown tree touching electrical transmission lines, resulted in cascading outages, caused trips of NPP stations, and disabled offsite power supplies. Thus, the design and maintenance practices for NPP switchyard protection systems can affect the reliability and availability of the plants' offsite power sources.

Since the deregulation of the electric power industry, NPP switchyards may have become more vulnerable to external faults because most of those switchyards are no longer owned, operated, or maintained by companies that have an ownership interest in the NPPs. Instead, the switchyards are maintained by local transmission and distribution companies, which may not fully appreciate the issues associated with NPP safety and security. Maintenance practices may also be inconsistent among these companies. In addition, circuit breaker components (i.e., relays, contacts, and opening/closing mechanisms) have begun to show age-related degradation. Improper maintenance of these components could affect the detection and mitigation of faults, which could, in turn, delay protective actions at NPPs.

At Catawba Nuclear Station on May 20, 2006, both units tripped automatically from 100 percent power following a LOOP event. (See the licensee event report for Event Number 41322006001, "Loss of Offsite Power Event Resulted in Reactor Trip of Both Catawba Units from 100% Power.") That event began when a fault occurred within a current transformer associated with one of the switchyard power circuit breakers. A second current transformer failure, along with the actuation of differential relays associated with both switchyard buses, deenergized both buses and separated the units from the grid.

## Objective

The NRC staff initiated a research project to develop a better understanding of the current power system protection in electrical switchyards and identify the system vulnerabilities that contribute to electrical fault propagation into nuclear facilities.

## Approach

This research project comprises multiple tasks. First, the contractor will review the operation of electrical protection systems associated with events that resulted in plant trips and LOOPs (e.g., Palo Verde, Catawba, and Peach Bottom). The contractor will then identify the root causes of the propagation of external electrical faults into the NPP switchyards, assess the level of protection of current NPP switchyard breaker arrangements and relay schemes used for protection, and coordinate this study with the North American Electric Reliability Corporation (NERC) and Federal Energy Regulatory Commission (FERC) and its assessments of switchyard protection. Lastly, the contractor will illustrate through analysis and modeling how an actual fault outside an NPP switchyard will affect an operating NPP station and will compare existing NPP switchyard designs with modeling and analysis of the settings and identify the desirable level of protection offered for responding to external faults.

## Products

Upon completion of this research project, the NRC may develop a NUREG-series report to provide an assessment of the NPP switchyard protection designs in its response to external electrical faults and will consider publishing a regulatory guide, in coordination with NERC and FERC, to address the desirable level of protection acceptable for NPP switchyards.

**For More Information**

Contact Darrell Murdock, RES/DE at 301-251-7629 or Darrell.Murdock@nrc.gov

# Evaluation of Equipment Qualification Margins to Extend Service Life

## Background

According to Title 10 of the *Code of Federal Regulations* Part 50 Section 49 (10 CFR 50.49), "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," Class 1E electrical equipment located in a harsh environment must be environmentally qualified to perform its safety-related function during and following a design-basis event such as a loss-of-coolant accident.

In particular, 10 CFR 50.49, known as the Environmental Qualification (EQ) Rule, states that "margins must be applied to account for unquantified uncertainty, such as effects of product variations and inaccuracies in test instruments."

The Institute of Electrical and Electronics Engineers (IEEE) Standard (Std.) 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," defines margin as the difference between the most severe specified service conditions of the plant and the conditions used in type testing.

Furthermore, Section 6.3.1.5 of IEEE Std. 323-1974 lists suggested factors for licensees and equipment manufacturers to apply to service conditions for type testing, including temperature, pressure, radiation, voltage, frequency, time, vibration, and environmental transients.

The margins, as indicated in IEEE Std. 323-1974, are utilized in the test profiles to determine the qualified life of equipment. However, the margins are expected to account for the following:

- manufacturing tolerances and measurement uncertainties
- lack of sufficient oxygen in the test chamber
- lack of simultaneous age conditioning (temperature and radiation)
- high dose rate for radiation aging
- license renewal to extend the life of equipment to 60 years
- inconsistencies in activation energy values used in the Arrhenius equation[1] for thermal aging

[1] The Arrhenius equation is a methodology for addressing time-temperature aging effects, where the key assumption is that material thermal degradation is dominated by a single chemical reaction whose rate is determined by the temperature of the material and a material constant called the activation energy.

Since manufacturing tolerances and measurement uncertainties cannot be readily quantified when establishing qualified life, margins are added to ensure that the equipment can perform its safety function. The lack of oxygen in the test chamber during accelerated aging could impact the qualified life since the equipment could have greater degradation because of the oxidation effects. As a result, equipment testing does not consider the effects of oxygen, and the margins account for this phenomena. Furthermore, recent data have shown that simultaneous aging (radiation and thermal) may produce synergistic effects that reduce the qualified life when compared to sequential aging. The same margins are also used to account for any variations between sequential and simultaneous aging. Using a smaller dose rate for the radiation aging of cables would more adequately result in showing radiation degradation effects, but the margin is also credited for the use of a high radiation dose rate. The existing margin is applied to extend the life of equipment to 60 years for license renewal, but when an imprecise activation energy is utilized, the impact on the time needed for thermal aging can be affected. Therefore, to correct for any errors in activation energy, margins are added. As a result, the margins are used to account for a variety of factors, as opposed to only production variations.

The regulatory use for this research will be to establish the technical basis for assessing the qualified life of electrical equipment in light of the uncertainties identified following the initial qualification testing.

## Approach

Through this research, the staff aims to (1) confirm whether EQ requirements for electrical equipment are being met throughout the current and renewed license periods of operating reactors, (2) quantify the margin, and (3) verify its adequacy to address the uncertainties discussed above. This research will assess the existing margins and evaluate its adequacy in light of known problems. The contractor will perform a background literature search and include the review of several key reports on the aging of cables. The NRC will publish a NUREG/CR report at the completion of this project outlining the margin available to address the known uncertainties when qualifying electrical equipment.

**For More Information**
Contact Sheila Ray, RES/DE at 301-251-7644 or Sheila.Ray@nrc.gov

# Charging Current as an Indicator of a Fully Charged Battery

## Background

The NRC is sponsoring confirmatory battery testing research to determine whether charging current is a suitable indicator of a fully charged condition for lead-calcium batteries, to evaluate the impact of overcharging on battery capacity and service life, and to simulate battery aging and monitor its impact on battery life and performance. The research program will determine the level of current monitoring needed to ensure a fully charged condition and maintain operational readiness while in standby mode.

## Approach

Traditionally, the typical plant technical specifications required the measurement of specific gravity to determine if the battery was fully charged. To overcome the uncertainties in specific gravity measurements to assess the state of charge, the industry developed an alternate technique to measure charging current as an indicator. The Institute of Electrical and Electronics Engineers (IEEE) revised IEEE Standard (Std.) 450-1975, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations," to accommodate this new method. IEEE Std. 450-2002, "Recommended Practice for Maintenance, Testing, and Replacement of Vented Leaded-Acid Batteries for Stationary Applications," recommends measuring the battery's charging current in lieu of specific gravity for determining a vented lead-calcium battery's state-of-charge. The standard provides the recommended practices, test schedules, and testing procedures, including recommended methods for determining a battery's state of charge to maintain permanently installed vented lead-acid storage batteries (typically of the lead-calcium type) for their standby power applications. The NRC staff endorsed this new standard and issued Regulatory Guide 1.129, Revision 2, "Maintenance, Testing, and Replacement of Vented Lead-Acid Storage Batteries for Nuclear Power Plants."

To confirm that the battery has the capability to perform its design function, the staff initiated the research and arranged the testing of batteries to be performed in three phases:
(1) evaluation of charging current as a monitoring technique,
(2) evaluation of the use of charging current to monitor battery capacity, and (3) impact of overcharging on batteries.

The approach for this research project will involve testing of lead-acid batteries from three different types of vendors to obtain a good sample of what is currently being used at the Nuclear Power Plants (NPPs) (see Figure 10.5). The batteries will be installed in a configuration similar to that used in the plants and will be subjected to deep discharge/charge cycles to simulate an expected service life for the batteries. All testing will be performed in accordance with IEEE Std. 450-2002, along with a quality assurance plan developed specifically to meet the needs of the project in order to ensure an acceptable level of quality for the test results.

Upon the completion of the testing, the NRC will issue a NUREG-series report to document the assessment of the new test methods involving charging current. In addition, the NRC will consider issuing regulatory guidance to describe the various battery cell metallurgies and the best methods to verify the operational readiness of battery systems in NPPs.

**Figure 10.5 NRC staff reviewing the first set of batteries that the contractor has received before commencing confirmatory battery testing**

**For More Information**
Contact Liliana Ramadan, RES/DE at 301-251-7642 or Liliana.Ramadan@nrc.gov

# Chapter 11: New and Advanced Reactor Research

Thermal-Hydraulic Analyses of New Reactors

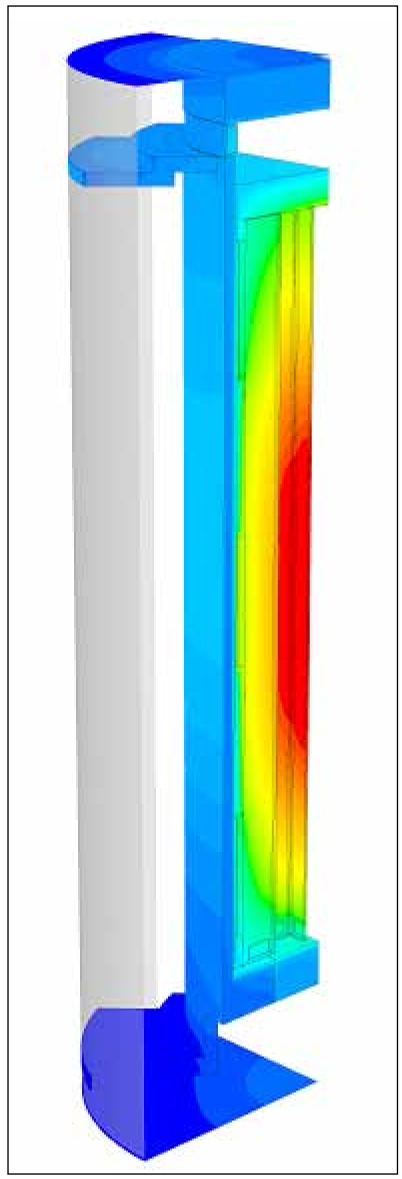
Computational Fluid Dynamics in Regulatory Applications

Advanced Reactor Research Program

Next Generation Nuclear Plant

Materials Issues for High-Temperature Gas-Cooled Reactors

Confirmatory Analysis Tool for Structural Integrity Evaluation of Creep and Creep-Fatigue Crack Growth in Next Generation Nuclear Plant Metallic Components



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

# Thermal-Hydraulic Analyses of New Reactors

## Background

The U.S. Nuclear Regulatory Commission (NRC) uses the Transient Reactor Analysis Code/Reactor Excursion and Leak Analysis Program (TRAC/RELAP) Advanced Computational Engine (TRACE) code to perform confirmatory calculations in support of design certification and combined operating license reviews for all new reactors—the Advanced Passive 1000 Megawatt (AP1000), U.S. Advanced Pressurized-Water Reactor (U.S. APWR), the U.S. Evolutionary Power Reactor (EPR), the Economic Simplified Boiling-Water Reactor (ESBWR), and the Advanced Boiling-Water Reactor (ABWR). The modeling of various integral pressurized-water reactor (IPWR) designs has been undertaken to assess the applicability of NRC codes in anticipation of confirmatory analyses.

New reactor designs include evolutionary advances in light-water reactor technology and thus pose unique modeling challenges as a result of novel systems and operating conditions. Many of these modeling challenges are associated with passive systems that rely on phenomena such as gravity, pressure differentials, natural convection, or the inherent response of certain materials to temperature changes. Most developmental assessments conducted for currently operating light-water reactors cover the phenomenology necessary in thermal-hydraulic simulations for new reactor designs. However, the modeling of some of the novel systems and operating conditions of new reactors requires further code development and additional assessments against specific experimental data.

## New Reactor Designs

### AP1000

The AP1000 (see Figure 11.1) relies extensively on passive safety systems. Passive systems are used for core cooling, containment cooling, main control room emergency habitability, and containment isolation. These systems challenge system codes in predicting fluid flow induced by small driving heads. The applicability of TRACE to simulate AP1000 transients was demonstrated through comparisons with data from relevant integral and separate-effects test facilities.

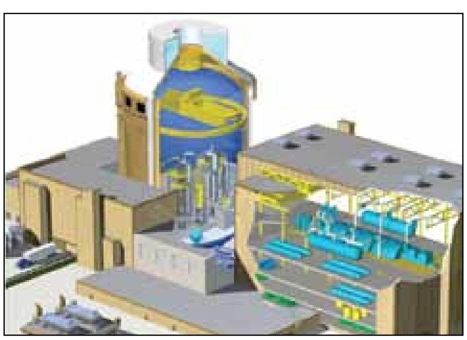


Figure 11.1 AP1000

### U.S. APWR

Most of the major components of the U.S. APWR (see Figure 11.2) are very similar to those of existing pressurized-water reactors (PWRs). The major exception is the advanced accumulator that eliminates the need for pumped low-pressure safety injection. The ability of TRACE to predict the behavior of advanced accumulators has been demonstrated with separate-effects data. Furthermore, detailed three-dimensional phenomena, such as cavitation, nitrogen ingress, and mass flow rate, have been modeled using computational fluid dynamics tools, and the results were coupled as needed with system code simulations.

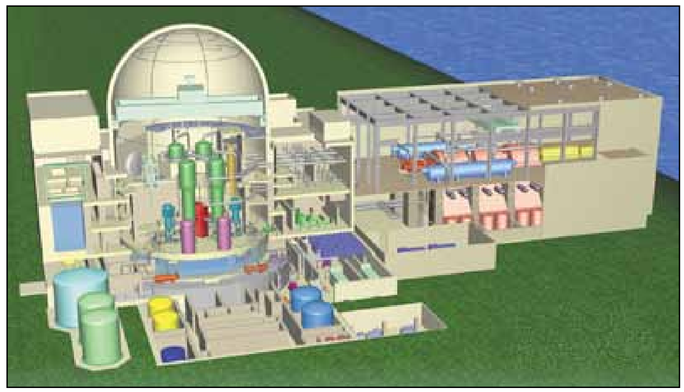


Figure 11.2 U.S. APWR

### EPR

The EPR (see Figure 11.3) is an evolutionary PWR design that uses rapid secondary-side depressurization for mitigation of loss-of-coolant accidents (LOCAs). This increases the emphasis on the ability of TRACE to predict reflux condensation in steam generator tubes. To demonstrate the applicability of TRACE to the 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)-IV, and ROSA-V.

Figure 11.3 U.S. EPR

## ESBWR

The ESBWR (see Figure 11.4) has a passively driven containment cooling system and a gravity-driven cooling system. Both of these systems rely entirely on natural phenomena for the convection of mass and energy. The prediction of void distributions and two-phase natural circulation is very important for the ESBWR. Integral test data from the 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 this application. In addition, proper modeling of film condensation in the presence of noncondensable gases at low power levels posed a significant challenge in the ESBWR analysis. Improved models in TRACE predicted these phenomena very well.

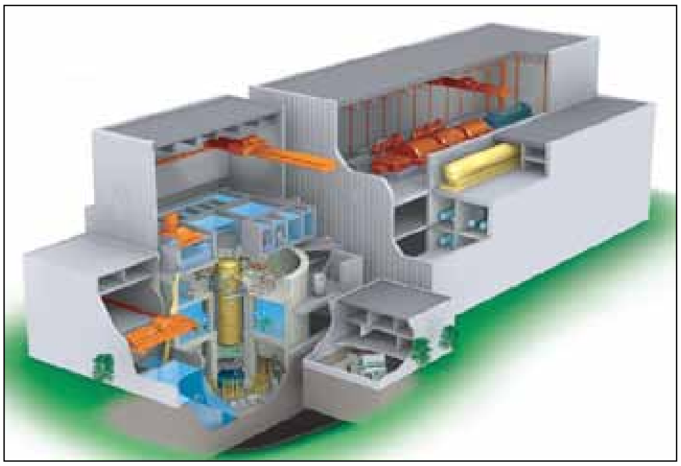


**Figure 11.4 ESBWR**

## ABWR

The ABWR (see Figure 11.5) is an evolutionary boiling-water reactor that includes such design enhancements as recirculation pumps internal to the reactor vessel and digital controls.

**Figure 11.5 ABWR**

TRACE will be used to simulate the plant response to LOCAs, as well as to anticipated operational occurrences and other transients. Modeling internal pumps and incorporating the logic needed for digital controls will pose potential challenges to the code.

## IPWR

The current IPWR designs (see Figure 11.6) eliminate the external reactor coolant piping and integrate the steam generator and pressurizer into the reactor vessel as one integral primary system. The NuScale IPWR design uses helical tube steam generators, and the mPower IPWR design uses once-through tube steam generators to produce superheated steam in the secondary system. Test data from the Multi-Application Light Water Reactor (MASLWR) and integrated system test (IST) facilities are being used to assess the NRC codes for applicability to these designs. Proper modeling of helical tube heat transfer, film condensation, and natural circulation are the main challenges for TRACE simulation.

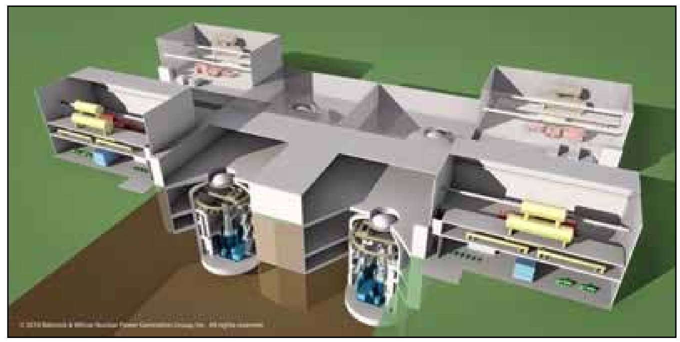


**Figure 11.6 IPWR**

**For more information**
Contact Jaclyn Dorn, RES/DSA 301-251-7565 or Jaclyn.Dorn@nrc.gov

# Computational Fluid Dynamics in Regulatory Applications

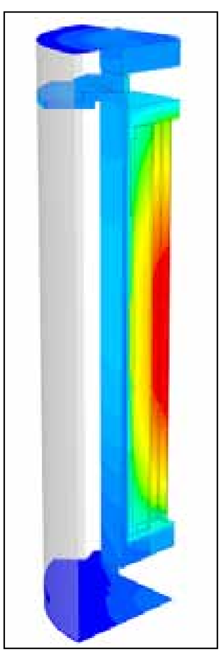
## Background

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

The NRC's Office of Nuclear Regulatory Research (RES) has developed a state-of-the-art CFD capability that supports multiple offices within the agency. RES uses the commercial CFD codes from ANSYS Inc. (FLUENT) and CD-adapco (STAR-CCM+) and has supported the development of multiphase modeling capabilities in research codes. The office maintains a Linux cluster with over 200 processors to provide the capability needed to solve the large-scale problems that are characteristic in the nuclear industry. RES staff is actively involved in national and international CFD programs and maintains a high level of expertise in the field. This state-of-the-art capability provides a robust infrastructure for both confirmatory and exploratory CFD computations.

## Applications

### Spent Fuel Transportation And Storage



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

The CFD approach has been used to study cask designs under a variety of external conditions, such as fires, reduced ventilation, and hotter fuels. This work supports dry cask certification efforts by improving the agency's technical bases for licensing decisions (see Figure 11.7).

**Figure 11.7 Temperature contours of a ventilated dry cask that uses ambient air to passively cool the spent fuel stored inside the canister surrounded by a concrete overpack**

### Operating Reactors

CFD predictions have also aided in understanding detailed fluid behavior for broad-scope analyses, such as pressurized thermal shock, induced steam generator tube failures, boron dilution and transport, and spent fuel pool analyses. In most cases, CFD results are used iteratively with system code predictions, or they provide boundary or initial conditions for other simulations (see Figure 11.8).

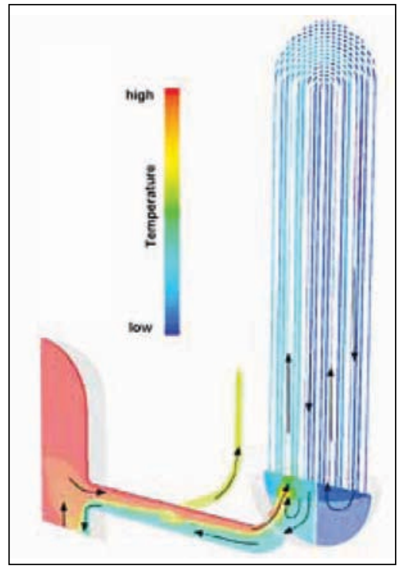



**Figure 11.8 During a particular severe accident scenario, hot gases from the core circulate through the hot legs and steam generator in a counter-current flow pattern. The risk of induced failures is considered.**

### New And Advanced Reactors

The agency has used CFD to confirm the distribution of injected boron in the ESBWR. In the design certification of the U.S. APWR, CFD was used to investigate the performance of an advanced accumulator (see Figure 11.9). The phenomena of interest are cavitation and nitrogen ingress, which exceed typical system code capabilities. The validation of the CFD simulation against experimental data was particularly challenging for this application, especially because CFD results were also used to examine possible scale effects.

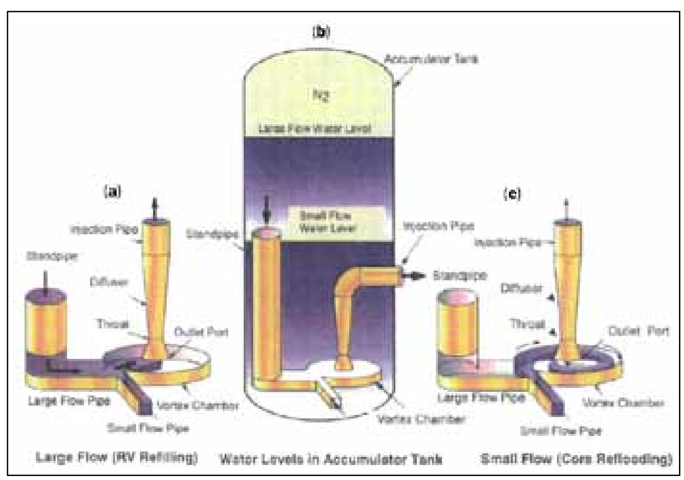



**Figure 11.9 The advanced accumulator (b) is a water storage tank with a flow damper in it that switches the flow rate of cooling water injected into a reactor vessel from a large (a) to small (c) flow rate**

**For More Information**

Contact Kenneth Armstrong, RES/DSA, at 301-251-7551 or Kenneth.Armstrong@nrc.gov

# Advanced Reactor Research Program

## Background

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

## Overview

The revised ARRP documents the NRC's current assessment of its research infrastructure needs and the agency's planned safety research to support its review of HTGR and VHTR licensing applications. These include a combined license (COL) application for a VHTR to be constructed at the Idaho National Laboratory (INL) in connection with the Next Generation Nuclear Plant (NGNP) Project, as directed by the Energy Policy Act of 2005 (EPAct) (Public Law 109-58), and a potential design certification application for the pebble bed modular reactor.

The update also includes a high-level survey of the technical infrastructure development and initial safety research that the NRC would need to conduct to prepare for its review of a potential sodium-cooled fast reactor (SFR) licensing application. Such licensing applications include a near-term application for design approval for the Toshiba Super Safe, Small and Simple (4S) reactor and a longer term licensing application for a commercial advanced fast-burner reactor being developed by the U.S. Department of Energy (DOE) for nuclear fuel recycling. The SFR research infrastructure survey was conducted at a higher level than the HTGR and VHTR reassessment. The survey identifies the key technical, safety, and research issues associated with SFR licensing. The survey provides a framework for a potential follow-on in-depth SFR research infrastructure assessment similar in scope to the HTGR and VHTR assessment. As an example, the NRC HTGR accident analysis evaluation model concept schematic shown in Figure 11.10 demonstrates the applicability of research results to reactor plant systems analysis.

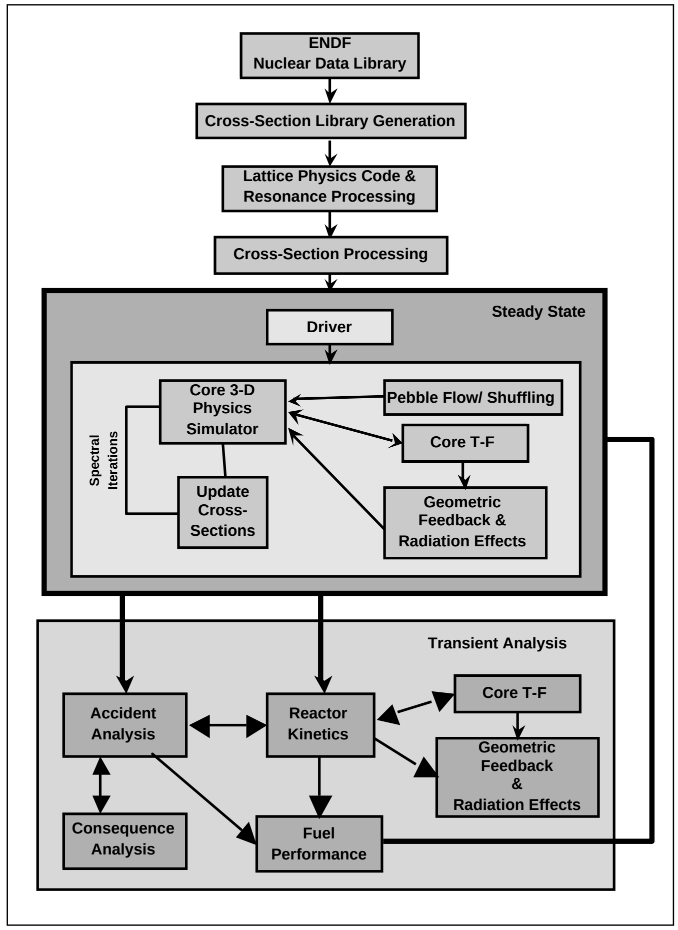



**Figure 11.10 Schematic of NRC HTGR accident analysis evaluation model concept**

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

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

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

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

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

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

## Summary

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

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

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

**For More Information**
Contact Tarek Zaki, RES/DSA at 301-251-7986 or Tarek.Zaki@nrc.gov

# Next Generation Nuclear Plant

## Background

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

## Approach

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

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

DOE has determined that the NGNP nuclear reactor will be a very-high-temperature gas-cooled reactor (VHTR) for the production of electricity, process heat, and hydrogen (see Figure 11.11). The VHTR can provide high-temperature process heat (up to 950 degrees Celsius) that can be used as a substitute for the burning of fossil fuels for a wide range of commercial applications. Since the VHTR is a new and unproven reactor design, the NRC will need to adapt its licensing requirements and processes, which have historically evolved around LWR designs, for licensing the NGNP nuclear reactor.

### NGNP Reactor Technology

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

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

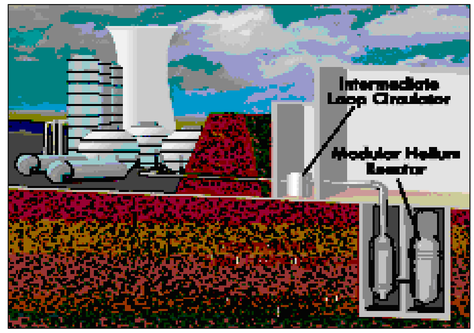



**Figure 11.11 Artist's rendition of an NGNP plant**

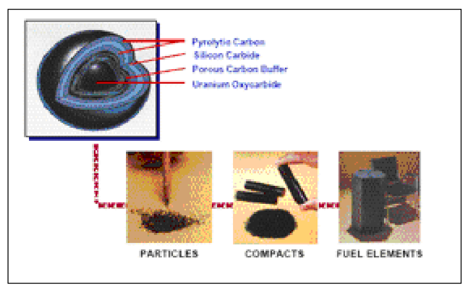



**Figure 11.12 TRISO fuel for NGNP**

### NGNP Licensing Requirements

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

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

### Analytical Tools Development and Other R&D

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

To address regulatory and safety issues for an NGNP design in major technical areas, and, in particular to identify important safety-relevant phenomena associated with these design concepts and to assess the knowledge base, a phenomena identification and ranking table (PIRT) exercise was conducted in 2007. The PIRT process involved assembling groups of experts in each of the identified major areas, facilitating focused discussions among the experts in these areas, annotating expert deliberations and finally, assessing the knowledge gaps in these areas based on expert deliberations.

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

- thermal fluids and accident analysis
- high-temperature materials including graphite
- process heat and hydrogen cogeneration
- fission product transport (FPT) and dose
- tristructural isotropic (TRISO)-coated fuel particles (see Figure 11.12)

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

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

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

## Project Status

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

**For More Information**
Contact Sudhamay Basu, RES/DSA at 301-251-7521 or Sudhamay.Basu@nrc.gov

# Materials Issues for High-Temperature Gas-Cooled Reactors

## Background

High Temperature Gas-Cooled Reactors (HTGRs) operate in an environment quite different from that of Light Water Reactors (LWRs). Challenges to the pressure boundary metallic materials and graphite and other ceramic core components are considerably more severe. The HTGR coolant does not change phase and the graphite core components (GCCs) are exposed to very high temperatures and neutron fluence.

The integrity of metallic and graphite components is important to maintaining safety. Integrity of components is necessary to avoid air, water, or steam ingress into the pressure boundary and to maintain core geometry. The pressure boundary also acts as a barrier to release of radioactivity. A sound technical basis is necessary for evaluating the integrity and failure modes of components.

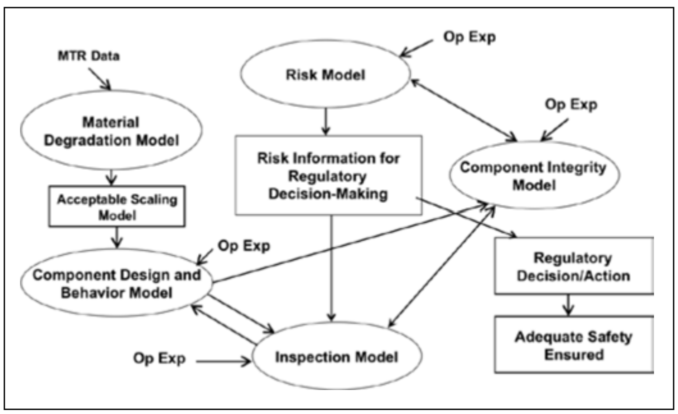



**Figure 11.13 Influence of material behavior, inspection, flaw evaluation, and component integrity assessment on risk assessment**

As illustrated in Figure 11.13, information from various aspects of materials research, such as degradation mechanisms, inspection efficacy, stress analysis, and component integrity assessment, obtained from probability of failure estimate, is needed for conducting probabilistic risk assessments resulting from the failure of these components to perform their intended functions. Note that failure probability data are not available from experience; therefore, large uncertainty information may be developed from research to identify and quantify degradation processes

The EPAct established the NGNP to demonstrate the generation of electricity or hydrogen, or both, with an HTGR. The NRC is responsible for licensing and regulatory oversight of the NGNP. A combined construction and operating license application for the NGNP is presently scheduled for submittal to NRC in fiscal year (FY) 2013.

As of summer 2010, the design of the NGNP HTGR is still conceptual, and final component specification and material selection have yet to be determined. Therefore, the NRC continues to research generic component performance requirements. Candidate materials for specific applications are being evaluated to identify potential qualification and acceptance gaps.

The staff expects the HTGR applicant to provide complete data with the COL or design certification (DC) application. This will include technical bases to support the designed functions of high-temperature materials and GCCs. The staff evaluation of the design will rely on the applicant-provided information. During reactor operation, the licensee will confirm designed performance of GCC via periodic inspections and coupon tests.

The NRC began to develop a research plan during 2003 on materials issues related to HTGR and has updated it on an annual basis. The research plan has been coordinated with the Office of New Reactors (NRO) and has been presented periodically to the Advisory Committee on Reactor Safeguards (ACRS). The research plan identified the lack of consensus codes and standards for high-temperature materials and graphite as a leading hurdle for staff review of an HTGR license application. A number of research projects have been conducted since then.

### High-Temperature Materials Research

For metallic materials, the staff analyzed the limitations involved in extending the known properties at lower temperatures to the HTGR operating temperature. The agency published the results of this analysis, conducted by Argonne National Laboratory (ANL), as NUREG/CR-6816, "Review and Assessment of Codes and Procedures for HTGR Components." Seven codes and procedures were analyzed, including five American Society of Mechanical Engineers (ASME) codes (Section III, Subsection NB, and Subsection NH) and Code Cases (N-499-1, N-201-4, and the draft Code Case for Alloy 617); one French code (RCCMR); and one British procedure. The report concluded the following:

- Most of the materials needed for HTGR were not included in the code cases; therefore, new code cases are needed for these materials.
- Codes and code cases did not provide specific guidelines for environmental effects, especially the effect of impure helium, on the high temperature behavior (e.g., creep and creep-fatigue) of the materials considered.
- Data on environmental effects should be collected or generated, if not available, so that the specific guidelines for these effects can be developed.

ANL also examined in considerable detail high-temperature material properties of relevance to HTGR and published the results in NUREG/CR-6824, "Materials Behavior in HGTR Environments." The report identified the materials used for structural applications (such as pressure vessel and reactor primary circuit components, including internals) and for the power conversion system, with emphasis on gas-turbine-based HTGRs.

The NRC staff has been participating in ASME Code, Section III, Subsection NH, development activities to ensure that code cases consider in sufficient detail the environmental effects on important alloying elements and their distribution and affected properties whose degradation will influence the design margin for maintaining the integrity of the coolant pressure boundary.

The NRC also jointly participated with DOE in sponsoring ASME S&T LLC in developing a roadmap for updating the Subsection NH Code to be applicable to the NGNP HTGR. The following key issues continue to be addressed:

- flaw evaluation for design margin assessment
- component classifications
- development of risk-informed inspection program, including reliability and integrity management (RIM) of passive metallic components

During 2010, a research contract was placed at Pacific Northwest National Laboratory (PNNL) to determine a suite of reliable flaw evaluation techniques for HTGR high-temperature materials, including acoustic emission. In addition, PNNL staff will review the documents currently being generated by the ASME Code Division 5 Section XI Special Working Group for HTGR inservice inspection (ISI) requirements. PNNL will identify additional information needed to evaluate the applicant's design and to review licensee-proposed ISI programs.

During 2007, the NRC conducted a PIRT exercise with a panel of high-temperature materials experts to determine data needs which have high importance to safety and low knowledge. The exercise identified five phenomena in this category. NUREG/CR-6944, "Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs)," Volume 4, "High-Temperature Materials PIRTs," issued March 2008, contains the results of the PIRT exercise. Of these five phenomena, the NRC decided to focus research on the development of a time-dependent creep and creep-fatigue crack growth predictive methodology that will be integrated into the modular probabilistic fracture mechanics (PFM) computer code. This work, being conducted at Oak Ridge National Laboratory (ORNL), is the subject of a separate information sheet.

## Nuclear Graphite—Research

Consistent with the graphite research plan, the NRC, in cooperation with DOE, conducted a PIRT exercise with an international panel of nuclear graphite experts. NUREG/CR-6944, Volume 5, "Graphite PIRTS," issued March 2008, presents the results of this effort. Of the several phenomena identified, five were ranked to be of high importance and low knowledge. During 2009, the NRC conducted a technical information gap analysis international workshop and identified specific technical areas which are not addressed by the HTGR applicants' and other current worldwide research. To conduct effective technical review, the workshop panel recommended that the NRC staff develop a broad knowledge base in nuclear graphite technology and actively participate in the development of irradiation data, behavior modeling and interpretation, and codes and standards development.

In 2010, the NRC initiated independent research in two major areas. The first is exploratory research on the release of stored (Wigner) energy of irradiated graphite when it is heated subsequently to temperatures greater than the irradiation temperature. Such a scenario is possible, for example, in a LOCA, leading to excessive heat generation and loss of graphite with potential release of radionuclides. The second is research to develop a confirmatory finite element stress analysis (FEA) tool, which will provide the staff an independent capability to conduct time (dose)-integrated, nonlinear, three-dimensional FEA for GCC. The input data for model and procedure development will originate from DOE/INL/ORNL and other worldwide research. The model and the procedures will be validated and verified using the ASME Code and DOE and other vendor data and benchmark calculations on idealized core component shapes. The staff can use this FEA tool, projected to be available by 2013, to confirm applicant assumptions, stresses, design factors of safety, and the retention of design margin over the reactor life. The staff will also use this tool to perform confirmatory analyses of applicant designed deformation limits for GCC.

## Ceramic and Carbon-Carbon Composites

The NRC currently has not planned any specific research on these materials because of the paucity of specific information on these materials from NGNP designers, especially with respect to the design envelope, expected material interactions with the environment, and safety classification. The staff expects that safety concerns pertaining to nuclear graphite will generally apply to these materials.

**For More Information:**

Contact Dr. Makuteswara Srinivasan, RES/DE at 301-251-7630 or makuteswara.srinivasan@nrc.gov

# Confirmatory Analysis Tool for Structural Integrity Evaluation of Creep and Creep-Fatigue Crack Growth in Next Generation Nuclear Plant Metallic Components

## Background

The Energy Policy Act of 2005 (EPAct) established the Next Generation Nuclear Plant (NGNP) to demonstrate the generation of electricity or hydrogen, or both, with an HGTR. The NRC is responsible for licensing and regulatory oversight of the NGNP. A combined construction and operating license application for the NGNP is planned for submittal to the NRC in fiscal year (FY) 2014.

Creep and creep-fatigue crack growth of preexisting flaws or flaws that are initiated early in service life of metallic components (e.g., intermediate heat exchanger (IHX), cross vessel (CV)/duct, steam generators (SG), reactor pressure vessel (RPV)) in the NGNP, a very high temperature reactor (VHTR), are of particular concern if they are not detected during in-service inspection (ISI) because of accessibility or other issues. A macroscopic crack might grow to a critical size that triggers other structural failure modes, such as creep rupture due to reduced section thickness or brittle fracture of ferritic steel components during heatup or cooldown. A crack might also grow through the wall thickness, leading to a breach of the pressure boundary or the primary/secondary boundary and causing fission product release and/or air/steam/water ingress. Oak Ridge National Laboratories (ORNL) is conducting this work.

The NRC has identified subcritical crack growth from creep and creep-fatigue loading of NGNP high-temperature metallic components as a phenomenon that has a high importance ranking and a low knowledge level (NUREG/CR-6944, Volume 4). Time-dependent creep and creep-fatigue crack growth evaluation methodologies and analysis tools are necessary to support the independent assessment of the structural integrity of NGNP pressure boundary metallic components under normal operating conditions, design-basis accident and beyond-design-basis conditions, and other conditions that result in significant component degradation and failure.

## Objective

The objective of this research is to develop a confirmatory analysis tool to perform independent structural integrity evaluation of NGNP metallic components operating in high-temperature range where creep or creep-fatigue deformation is significant.

## Approach

The focus is on development of a validated time-dependent creep and creep-fatigue crack growth predictive methodology that will be integrated into a modular probabilistic fracture mechanics (PFM) computer code that the NRC is currently developing. The evaluation model consists of three modules, as shown in Figure 11.14 and described in the text below. This independent confirmatory capability is planned to be completed in FY 2014 to support NRC licensing reviews of the NGNP metallic components.

### Methods Development

The development of time-dependent crack-tip parameters (CTPs) is the main focus of this group of efforts. The NGNP candidate metallic materials exhibit three stages of creep behavior (primary, secondary, and tertiary). The approach to the development of CTPs is to perform crack-tip singularity analysis for each of the three creep deformation regimes. Once the time-dependent fracture mechanics methodology is developed, a data analysis procedure would be available to determine the correlation between the CTPs and the creep crack growth rate data.

### NGNP-Specific Crack Growth Correlations

This module is involved with the development of crack growth correlations specifically for the materials of construction for the NGNP IHX, CV/duct, SG, RPV, and other components. The evaluation model development will include both base metal and weldments. The required NGNP-specific crack growth correlations and material constants will be developed from confirmatory or new crack growth test data.

### Model Implementation into PFM Code

This module is involved with the implementation of the deterministic flaw evaluation procedure in the computer program module. It is anticipated that flaw evaluations using either best-estimate or statistical upper limits for the crack growth rates could be performed by the computer program. After completion of verification and validation, the deterministic flaw evaluation computer program will be incorporated into NRC's modular PFM computer code upon inclusion of various sources of aleatory and epistemic uncertainties in the data and models.

## Data Needs

A set of scoping tests to generate creep and creep-fatigue crack growth data will be needed to develop creep and creep-fatigue crack growth correlations. Judging from the available information in the literature, representative nickel-based Alloy

800H and its associated weldment are good candidates for the scoping tests since Alloy 800H has creep behavior similar to the materials being considered for the NGNP in a temperature range of 750–800 degrees Celsius. A list of data needs and the dates they are needed has been provided to the Department of Energy/Idaho National Laboratories NGNP project. Confirmatory NGNP-specific crack growth data will also be needed to validate creep crack growth correlations and to support NGNP license review. While the environment that these materials will be exposed to during service is impure helium, and possibly steam, the test data will be generated primarily in the air environment. Thus, any potential material degradation mechanisms in impure helium, and possibly in steam, that could accelerate the crack growth rates as compared with those in the air environment will need to be addressed through additional limited number of confirmatory environmental tests.

**For More Information:**
Contact Dr. Shah Malik, RES/DE at 301-251-7657 or Shah.Malik@nrc.gov

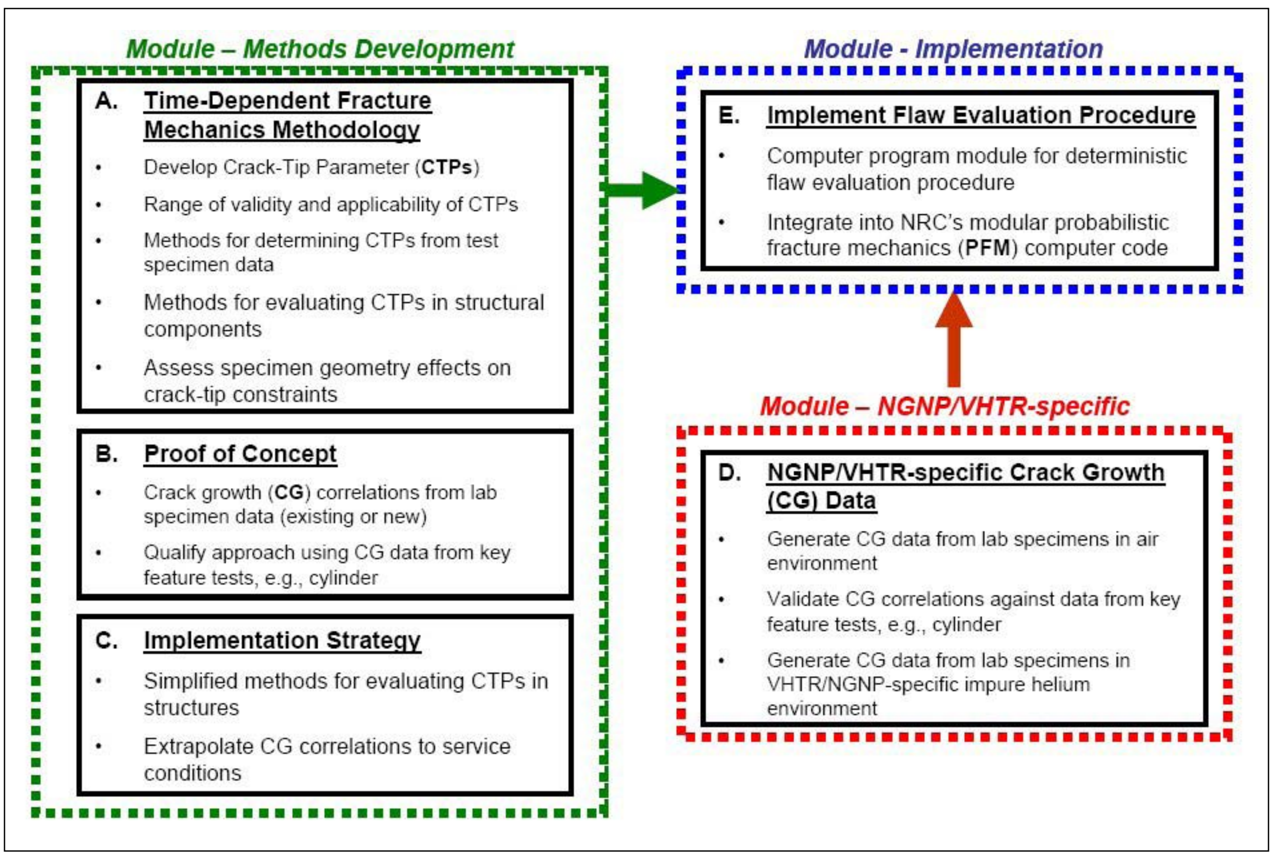



**Figure 11.14 Roadmap for development of a confirmatory analysis tool for creep and creep-fatigue flaw evaluation of the NGNP metallic components**

# Chapter 12: International Cooperative and Long-Term Research

Cooperative International Research Activities and Agreements

The Organization for Economic Cooperation and Development Halden Reactor Project

The Organization for Economic Cooperation and Development/Nuclear Energy Agency PKL2 Project

The Organization for Economic Cooperation and Development ROSA-2 Program

Studsvik Cladding Integrity Project

International Cooperative Research on Impact Testing

Round Robin Analysis of Containment Performance under Severe Accident—Collaboration between the U.S. Nuclear Regulatory Commission and the Atomic Energy Regulatory Board (India)

Agency Forward-Looking and Long-Term Research

**Completed prestressed concrete containment vessel ¼-scale model at Sandia National Laboratories**

# Cooperative International Research Activities and Agreements

## Cooperative Research Agreements

The U.S. Nuclear Regulatory Commission's (NRC's) Office of Nuclear Regulatory Research (RES) has implemented 100+ bilateral or multilateral agreements with 20+ countries and the Organization for Economic Cooperation and Development (OECD). These agreements cover a wide range of activities and technical disciplines, including severe accidents, thermal-hydraulic (T/H) code assessment and application, digital instrumentation and control (I&C), nuclear fuels analysis, seismic safety, fire protection, human reliability, and more.

## Bilateral, Multilateral and Code User Groups

Many of the agreements are established bilaterally with a foreign regulator or research institution for participation in one of the two largest nuclear safety computer code sharing programs. The Code Applications and Maintenance Program (CAMP) includes thermal-hydraulic code analysts from 20+ member nations. The Cooperative Severe Accident Research Program (CSARP) includes about 20 member nations who focus on the analysis of severe accidents using the MELCOR code. Both programs include user group meetings at which participants share experience with the NRC codes; identify code errors; perform code assessments; and identify areas for additional improvement, experiments, and model development.

The OECD's Nuclear Energy Agency (NEA) coordinates most of the NRC's multilateral research agreements. A few examples show how diverse the agreements can be. Large-scale experiments include the Halden Reactor Project (HRP) based in Norway and the domestically based Sandia Fuel Pool project. The OECD Piping Failure Data Exchange Project database is a different sort of shared resource for participants. RES applies a set of established criteria when considering cooperative research program proposals it receives. Considerations include cost, benefit, timeliness of expected results for current and expected regulatory uses, and more.

NRC participation in these agreements allows broader sharing of data obtained from physical facilities not available in the United States. As a result, NRC tools, data, and safety knowledge stay current and are state of the art. This enhances the NRC's ability to soundly make realistic regulatory and safety decisions based upon worldwide scientific knowledge and promotes the effective and efficient use of agency resources. Data obtained are used to develop new analytical models; to validate NRC safety codes; to enhance assessments of plant risk, including decisionmaking, fire, and human performance and reliability; and to develop risk-informed approaches to regulation.

### NEA Activities

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 the CSNI's seven working groups and three joint task groups) and the Committee on Radiation Protection and Public Health (CRPPH). The RES Director serves on the CSNI Bureau and on the Halden Reactor Project's Board of Management.

### IAEA Activities

RES also serves as the agency lead on codes and standards. By acting as the agency lead in the International Atomic Energy Agency's (IAEA's) Nuclear Safety Standards Committee, RES coordinates NRC contributions to the many IAEA safety standards guides. RES also participates in two "extra-budgetary programs" within IAEA entitled, "Protection against Tsunamis and Post Earthquake Consideration in the External Zone," and "Seismic Safety of Existing Nuclear Power Plants," which feeds into the IAEA's International Seismic Safety Center.

### Bilateral Information Exchange

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

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

**For more information**
Contact Donna-Marie Sangimino, RES/PMDA at 301-251-7673 or Donna-Marie.Sangimino@nrc.gov

# The Organization for Economic Cooperation and Development Halden Reactor Project

## Background

The NRC and its predecessor, the U.S. Atomic Energy Commission (AEC), have been participating in the Organization for Economic Cooperation and Development (OECD) Nuclear Energy Agency (NEA) Halden Reactor Project (HRP) since its inception in 1958. During this period, the NRC has used numerous research products from this internationally funded cooperative effort, which is located in Halden, Norway, and managed by the Norwegian Institute for Energy Technology (Institutt for Energiteknikk (IFE). For example, Halden tests on high-burnup fuel under loss-of-coolant accident (LOCA) conditions supported an NRC research information letter on cladding embrittlement. As another example, Halden's human factors research has supported regulatory guidance in areas such as alarm systems, hybrid control rooms, display navigation, and guidance for the review of proposed staffing configurations in computer-based control rooms.

## Facilities and Activities

### Fuels and Materials Research

The Halden Boiling-Water Reactor (HBWR) (see Figure 12.1), which currently operates at 18 to 20 megawatts, is fully dedicated to instrumented in-reactor testing of fuel and reactor materials. Since its initial startup, the reactor facility has been progressively updated and is now one of the most versatile test reactors in the world. The HRP fuels and materials program focuses on the performance of fuel and structural materials under normal or accident conditions using the numerous experimental channels in the core that are capable of handling many test rigs simultaneously.

Recent NRC reviews of industry fuel behavior codes have directly employed data from the HRP fuels program. These data are also essential for updating the NRC's fuel codes and materials properties library, which are used to audit industry analyses. Currently, the NRC is particularly interested in the previously mentioned LOCA tests, which are investigating such phemonena as axial gas flow, maintaining or breaking fuel-to-cladding bonding, fuel axial relocation, and fuel fragment spillage through cladding burst opening.

Regarding the HRP's nuclear reactor materials testing program, the HRP has, over the years, provided fundamental technical information to support the understanding of the performance of irradiated reactor pressure vessel (RPV) materials and supplemented results generated under NRC research programs. Recently, the HRP has been an essential partner in evaluating the irradiation-assisted stress-corrosion cracking (IASCC) of light-water reactor (LWR) materials. The HRP has irradiated materials that were later tested under the NRC's research program at Argonne National Laboratory to measure crack initiation, fracture toughness, and crack growth rate under representative LWR conditions. The HRP's ongoing work on IASCC and other areas (e.g., irradiation-induced stress relaxation) supplements NRC-sponsored research and addresses existing knowledge gaps. The NRC staff is using this information to inform reviews of licensee aging management programs.

### Man-Technology-Organization Laboratory

IFE's Halden facility also includes the IFE Man-Technology-Organization (MTO) Laboratory. The Halden Man-Machine Laboratory (HAMMLAB) (see Figure 12.2) is one of the principal experimental facilities in this laboratory. HAMMLAB uses a reconfigurable simulator control room that facilitates research into instrumentation and control (I&C), human factors, and human reliability analysis (HRA). Currently, HAMMLAB has hardware and software enabling it to simulate the Fessenheim pressurized-water reactor (PWR) plant in France, the Forsmark-3 boiling-water reactor (BWR) plant in Sweden, and the Ringhals-3 PWR plant in Sweden.

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

Recently, HRP-designed and executed HAMMLAB experiments provided the foundation for the International Empirical HRA Study, a multinational study aimed at developing an empirically based understanding of the performance, strengths, and weaknesses of HRA methods used in risk-informed regulatory applications. The NRC will be using the study's results to address outstanding HRA technical issues, including those related to HRA model differences identified in a November 8, 2006, staff requirements memorandum (SRM). Currently, ongoing HRP experiments are addressing a number of topics of interest to the NRC, including control room staffing strategies, the role and effects of automation in advanced control room designs, and aids to improve control room teamwork. The NRC expects that this research will contribute to the technical basis for human factors guidance, especially for new reactor designs.

The IFE MTO Laboratory also includes a virtual environment center and an integrated operations laboratory. The former is used to perform research involving mixed reality applications (e.g., training), and the latter is used to address issues associated with remote operations.

Finally, the MTO Laboratory also conducts research on I&C systems. Past efforts include work in the area of instrumentation surveillance and monitoring techniques based on advanced decision algorithms. A number of HRP-developed systems have been evaluated for use by U.S. plants.

The current HRP digital systems research activities contribute to three phases of a system lifecycle:

- Development, assurance and deployment of high integrity software important to nuclear power plant safety,
- Condition monitoring and maintenance support, where engineering and technical support teams are the intended beneficiaries of the research results. This research will improve accuracy and usability of current methods and develop novel techniques to improve diagnostics and condition-based maintenance.
- Development and application of software systems for operational support, where plant operators are the intended beneficiaries of the research results. The research program includes interaction of advanced control systems with human operators and issues related to the implementation and use of operational procedures.

## Summary

The HRP has provided and continues to provide valuable information to the NRC. Much of this information addresses gaps that are otherwise not being addressed by current NRC research activities, and some of this information is foundational to NRC's efforts to improve the technical basis of key models, methods, and tools. Furthermore, because the NRC is one of several contributors to the HRP budget, the HRP enables the NRC staff to significantly leverage its resources.

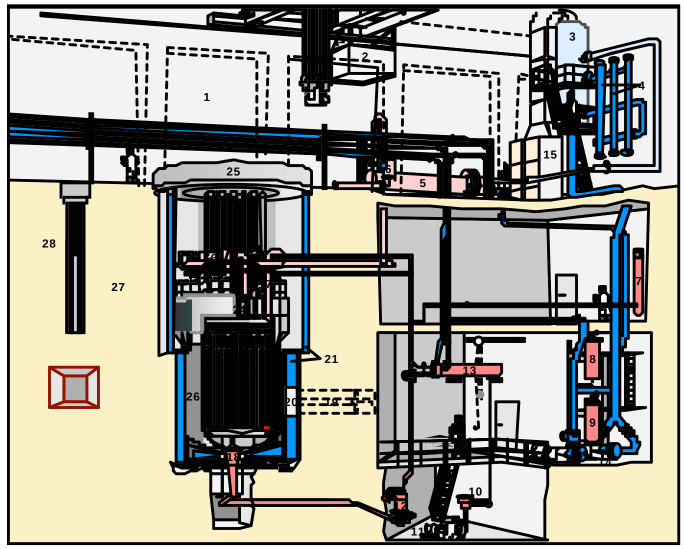


**Figure 12.1 HBWR test reactor**

**Figure 12.2 HAMMLAB control room simulator**

**For More Information**
Contact Alysia Bone, RES/DRA at 301-251-7602 or Alysia.Bone@nrc.gov

# The Organization for Economic Cooperation and Development/ Nuclear Energy Agency PKL2 Project

## Background

Since 2001, the NRC has been involved in a series of Organization for Economic Cooperation and Development (OECD)-fostered programs that use the Primärkreisläufe (PKL) facility to investigate safety-related issues relevant to current and new pressurized-water reactor (PWR) designs. The latest of such programs is the OECD/Nuclear Energy Agency (NEA) PKL2 Project (PKL2), a 3.5-year program that focuses on complex heat transfer mechanisms in steam generators (SGs) and boron precipitation processes under postulated accident conditions. Participation in this program is expected to help reduce known uncertainties in the area of thermal-hydraulics and provide data for use in assessing and enhancing the applicability of the NRC's Transient Reactor Analysis Code/Reactor Excursion and Leak Analysis Program (TRAC/RELAP) Advanced Computational Engine (TRACE) code.

Designed and built in the 1970s by AREVA NP GmbH (formally Siemens/KWU), the PKL facility is a full-height, 1:145 volume-scaled replica of a German PWR. Configured for enhanced realism, the facility has four identical reactor coolant loops arranged symmetrically around a reactor pressure vessel that contains a simulated core. Each of the four loops is equipped with a fully functional SG and a reactor coolant pump, and the core is simulated using 314 electrically heated rods. Each SG contains 30 U-tubes of original size and material, and each reactor coolant pump is equipped with an active speed controller to enable the simulation of different pump characteristics. The bundle of rods representing the core are capable of generating 2.5 megawatts (MW) of core power, which is equivalent to 10 percent of the nominal power rating of the 1,300-MW PWR used as the basis for the facility's design.

## Approach

For PKL2, 19 of the 28 OECD member countries have agreed to the following program of experimentation:

- G1: Systematic investigation of the heat transfer mechanisms in SGs containing nitrogen, steam, and water (2 tests)
- G2: Cooldown procedures with SGs isolated and emptied on the secondary side
- G3: Fast cooldown transients (e.g., main steamline break (MSLB))
- G4: Systematic study of heat transfer in SGs under reflux condenser conditions
- G5: Boron precipitation processes after large-break loss-of-coolant accident (LBLOCA)
- G6/G7: Subjects yet to be determined

Before an experiment is conducted, its scope and configuration are discussed and agreed upon during biannual review meetings. These meetings also allow members to review results from completed tests, exchange information on modeling best practices, and compare computer code results from posttest calculations.

The first TRACE posttest calculation performed during PKL2 was of the Test G3 MSLB. To perform the calculation, a TRACE model representing each of the major components and control systems present in the facility was developed (Figure 12.3). The TRACE results showed that the code was capable of predicting all of the key phenomena reasonably well. The results also made apparent the uniqueness of the four-loop data in illuminating the asymmetric effects of the test, which proved to be a challenge for the code to simulate.

As more tests are completed, similar calculations will be performed and analyzed to assess the applicability of TRACE and provide further insight into safety-related issues. Of particular interest is the boron-precipitation test, Test G5, which will investigate the factors affecting boron precipitation during long-term cooling and help determine the adequacy of modeling techniques employed by licensees to simulate the phenomena.

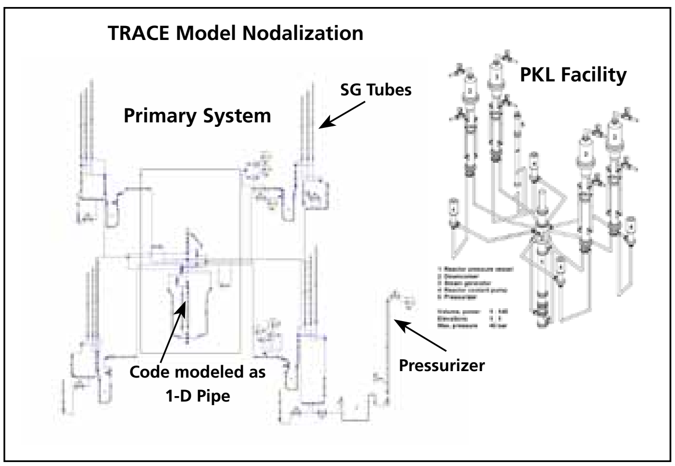



Figure 12.3 TRACE nodalization and schematic of PKL facility

### For More Information

Contact Shawn O. Marshall, RES/DSA at 301-251-7523 or Shawn.Marshall@nrc.gov

# The Organization for Economic Cooperation and Development ROSA-2 Program

## Background

The NRC has been participating in the Rig of Safety Assessment (ROSA) program for many years under the Organization for Economic Cooperation and Development/Nuclear Energy Agency. The ROSA-2 program is the latest phase of the program to conduct thermal-hydraulic (T/H) accident experiments in PWRs. The ROSA-2 program started in 2009 and is scheduled to be completed in 2012.

## Approach

The ROSA programs use the Large Scale Test Facility (LSTF) operated by the Japanese Atomic Energy Agency (JAEA) to conduct T/H accident experiments (see Figures 12.4 and 12.5). The LSTF, which has been in use since 1985, is an instrumented full height, 1/48 volumetrically scaled test facility intended to perform system integral experiments simulating the T/H response at full-pressure conditions of existing and next generation PWR designs during loss-of-coolant accidents (LOCAs) and other operational and abnormal transients.

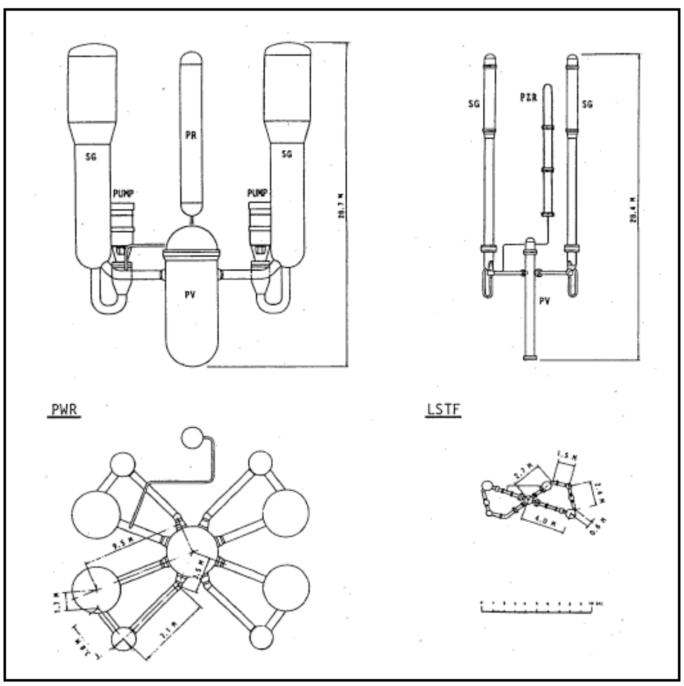


**Figure 12.4 Size comparison of ROSA/LSTF to a four-loop PWR**

Six tests are planned for the ROSA-2 program. As part of the ROSA-2 program, testing at the LSTF facility will specifically investigate the following safety issues:

Three intermediate-break LOCAs, including risk-informed break size definition and verification of safety analysis codes, will be performed.

Improvements and new proposals for accident management mitigation and emergency operation will be investigated. Two tests, focused on the recovery from a steam generator tube rupture (SGTR), one with and the second without a main steam line break (MSLB), will be performed.

A counterpart test with the *Primärkreislauf-Versuchsanlage* (primary coolant loop test facility) PKL test facility is being developed. The PKL facility in Erlengen, Germany, is operated by AREVA NP. Counterpart testing at the ROSA-2/LSTF and PKL facilities will provide test data that reflect the design scaling of the two facilities and produce two sets of test data for computer code validation. Program participants will finalize the description of the counterpart ROSA-2/LSTF-PKL test in the near future.

The NRC staff members participating in this international project investigate unresolved safety issues relevant to current PWRs and new PWR designs. The ROSA-2 test data will be used to validate the TRACE computer code and expand the usefulness of the code as an audit tool.

The ROSA-2 test program has already completed testing of a 17-percent intermediate hot-leg break and a 17-percent intermediate cold-leg break; however, only preliminary test data are currently available. The NRC staff has developed a TRACE model of the primary and secondary sides of the LSTF test facility to analyze these two tests. Preliminary test data for these two tests have been compared to TRACE blind and posttest predictions.

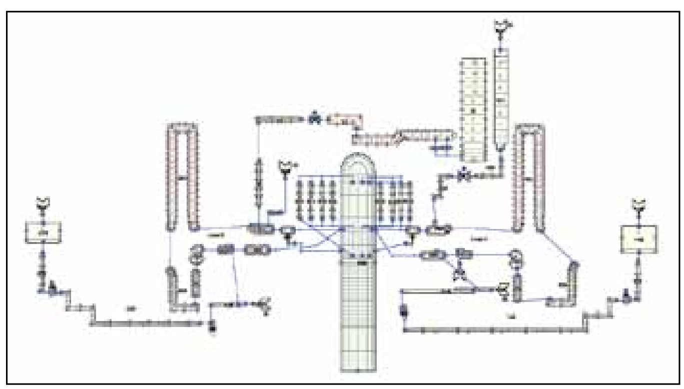


**Figure 12.5 LSTF primary system TRACE model used for 17% intermediate hot-leg break**

**For More Information**

Contact William Krotiuk, RES/DSA at 301-251-7541 or William.Krotiuk@nrc.gov

# Studsvik Cladding Integrity Project

## Background

The Studsvik Cladding Integrity Project (SCIP) is an Organization for Economic Cooperation and Development/Nuclear Energy Agency supported international program launched in 2004 and now extended to 2014, with participants from Europe, Japan, the United States, Russia, and Korea. The participants represent four categories: those who supply and manufacture the fuel, the power companies themselves, regulators, and laboratories with similar assignments to Studsvik's.

## Objective

SCIP is focused on improving the ability to predict mechanisms that can cause damage to cladding under normal operation and during transients. The program is conducted in the form of experiments, studies of fundamental mechanisms, development of suitable testing methods, and knowledge transfer.

The SCIP experiments and studies of fundamental mechanisms enable the understanding and quantification of key parameters important to hydrogen-induced failures, stress-corrosion cracking failures, and pellet-cladding mechanical interaction failures. This work provides valuable information for the development of operating restrictions.

The development of testing methods includes in-cell and out-of-cell mechanical testing techniques, as well as postirradiation analysis methods. This work enables the characterization of changes in cladding and pellets that take place with irradiation and provides valuable and unique characterization of advanced cladding and fuel pellet designs.

## Approach

Multiple laboratories are performing the technical work in SCIP II. Power transient testing is conducted in the Halden Boiling-Water Reactor (HBWR). Studies of the irradiated rods are then made at the Studsvik Hot Cell Laboratory, leading to a series of mechanical tests in other laboratories at Studsvik.

### Use Of Scip Data In The Integral Assessment Of Fuel Rod Computer Codes

As part of the NRC's fuel performance code development effort, new code versions are exercised to assess the integral code predictions to measured data for various performance parameters. The documentation of the integral assessment is publicly available and serves to demonstrate the code's ability to accurately predict integral fuel response under normal and off-normal conditions. As new data are generated, new assessment cases are added to the integral assessment suite.

The latest integral assessment added 10 SCIP ramps to the assessment suite. The ramps were modeled to assess the ability of FRAPCON 3.4 to predict cladding hoop strain during power ramps. Peak node plastic strain values from SCIP ramp data were compared to predicted values. Measured versus predicted values of plastic strain were compared as a function of burnup and ramp terminal level. These ramp tests were the first ramp tests that FRAPCON 3.4 was compared to with burnup greater than 45 gigawatt day per metric ton of uranium (GWd/MTU).

The comparison of predicted to the measured values in these ramp tests provided valuable insight into FRAPCON's ability to predict fuel and cladding response during power ramps. In this comparison effort, it was noted that FRAPCON 3.4 underpredicted the measured hoop strain in high burnup rods. The underprediction was most severe for those ramp tests with long hold times, as can be seen in Figure 12.6. The NRC is now revisiting the FRAPCON 3.4 strain model to investigate the source of this underprediction, and, if possible, to improve the modeling capabilities of FRAPCON.

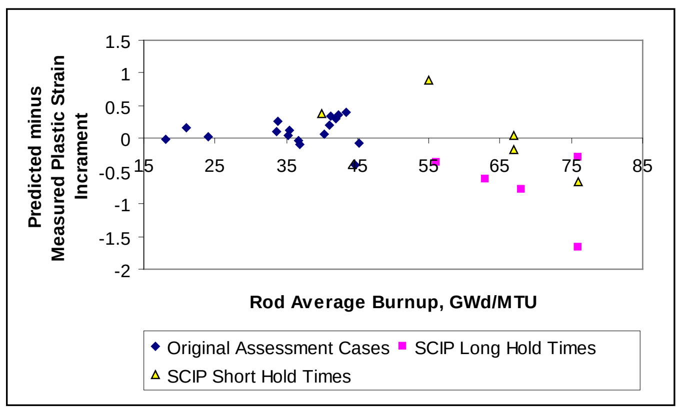



**Figure 12.6 FRAPCON 3.4 predicted minus measured permanent hoop strain as a function of burnup, indicating an underprediction at high burnups**

**For More Information**

Contact Michelle Flanagan, RES/DSA at 301-251-7547 or Michelle.Flanagan@nrc.gov

# International Cooperative Research on Impact Testing

## Background and Objectives

The NRC believes that it is prudent for nuclear power plant designers to take into account the potential effects of the impact of a large, commercial aircraft on nuclear facilities. RES has been conducting research in the area of impact loads on nuclear power plant structures that contributes to maintaining and developing critical skills needed to carry out the agency's mission of ensuring the safety of nuclear installations. Currently, the NRC participates in two international collaborative research programs in this area--one with theTechnical Research Center of Finland (VTT) and one with the Nuclear Energy Agency (NEA) Committee on the Safety of Nuclear Installations (CSNI's) Working Group on Integrity and Aging of Components and Structures (IAGE-WG) Concrete Subgroup. The expected benefits of these programs are (1) to benchmark the various computer codes that the NRC staff and its contractors utilize in impact assessments against experiments and (2) to synthesize the results of benchmarking into recommendations for good practices. These collaborative programs also provide opportunities to interact and exchange information with nuclear regulators abroad and with international nuclear safety organizations, ensuring NRC cognizance of ongoing impact research in various countries.

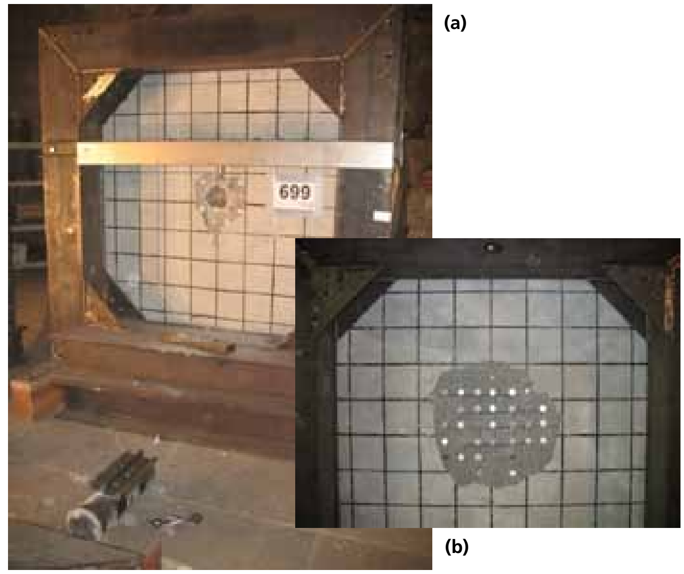
Anticipated benefits to the NRC from its participation in these programs include (1) reducing uncertainty associated with assessments of impact loads on nuclear installations and (2) ensuring that the assessments performed for U.S. reactors represent the state of the art in ensuring the safety of the public and protection of the environment.

## Approach

### Impact Test Agreement with the Technical Research Center of Finland

The NRC, the VTT, and nuclear regulators and nuclear safety research organizations in other countries participate in a multiyear international experimental research program, called IMPACT, to collect and analyze new data on the performance of reinforced and prestressed concrete walls subject to impact loads. All testing data under this program are provided by VTT using unique testing facilities not readily available elsewhere in the world, while the technical work of the NRC and the other participants focuses on analytical efforts.

Specific aims of the project include (1) obtaining new data on the time-varying hydrodynamic shock pressures from the impact on rigid structures of empty tanks, tanks filled with concrete (i.e., hard missiles; see Figure 12.7), and tanks filled with liquids (i.e., soft missiles; see Figure 12.8); (2) collecting new data on the response of reinforced concrete walls (e.g., displacements, strains) to these impact loads; (3) use of the new data to develop insights on the behavior of structures under impact conditions; and (4) use of the new data to benchmark computer simulation codes.


(a)

(b)

**Figure 12.7 Hard missile impact on reinforced concrete slab: (a) impact face and (b) back face (VTT)**

**Figure 12.8 Soft missile impact on reinforced concrete slab (VTT)**

VTT tests for the IMPACT program assess various reinforcement conditions, including prestressing, support conditions, slab thickness, impact speeds, and missile hardness. The first phase of the program tested over 20 impacts on concrete slabs, and a similar number of tests is planned for the second phase of the program already underway.

The IMPACT program includes regular workshops in which the participants exchange information on benchmarking, including benchmarking being done by RES staff (see Figure 12.9).

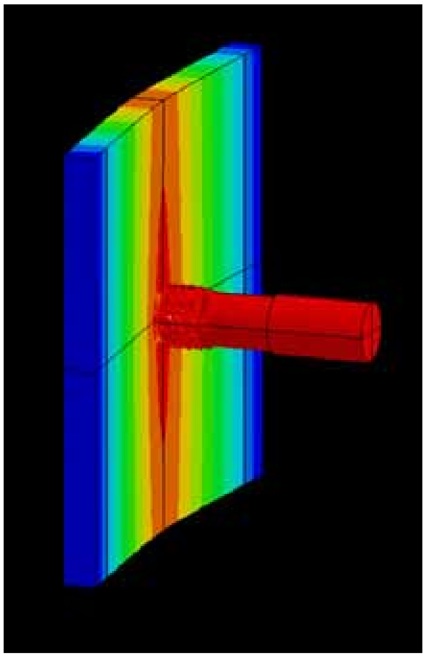


**Figure 12.9 Displacement contours for simulated missile impact on vertical wall**

## Research to Support the CSNI Project on Impact Assessment

The CSNI IAGE-WG Concrete Subgroup, developed a round robin benchmark exercise entitled, "Improving Robustness Assessment Methodologies for Structures Impacted by Missiles." The purpose of this project is to develop guidance that outlines effective methods of evaluating the integrity of structures impacted by missiles and to compare various methods in a round robin study of impact data. The project will use publicly available data from simple, reduced-scale tests and will reinterpret previous tests with newly available data, modeling capabilities, and results. The exercise will consider several types of structures ranging from structural components and box-shaped structures of reduced size to reactor building-like structures of reduced size. The project is expected to produce a state-of-the-art report collecting the contributions and proposing synthesis and recommendations for good practices.

To support its participation in this program, the NRC contracted Sandia National Laboratories (SNL) to benchmark different types of numerical simulation tools and to develop improved insights on modeling and damage criteria aimed at increasing confidence in numerical simulations for the assessment of existing and planned facilities.

**For More Information**
Contact Dr. Syed Ali, RES/DE 301-251-7658 or Syed.Ali@nrc.gov

# Round Robin Analysis of Containment Performance Under Severe Accidents—Collaboration Between the U.S. Nuclear Regulatory Commission and the Atomic Energy Regulatory Board of India

## Background

As part of the Indo-U.S. Civilian Nuclear Agreement, the NRC and the Atomic Energy Regulatory Board (AERB) of India are working together through the USNRC-AERB Nuclear Safety Co-Operation Program. As a result of this program, the two agencies met and discussed areas of mutually beneficial areas of study. The two agencies agreed to cooperate in the following areas: (1) new reactor designs, (2) probabilistic risk assessment (PRA) methods and applications and severe accident analysis and management, (3) proactive material degradation program, (4) digital systems reliability and qualification, and (5) operating experience feedback in India and the United States.

Also through this program, the NRC and AERB agreed to organize and participate in the Standard Problem Exercise #3 (SPE #3) round robin analyses. The SPE #3 will build on the previous round robin analysis of the NRC and the Nuclear Power Engineering Corporation of Japan (NUPEC) 1:4-Scale Prestressed Concrete Containment Vessel (PCCV) model tests conducted at the Sandia National Laboratories (SNL). The aim of SPE #3 is to undertake an analytical exercise on concrete containment structural performance. This will be accomplished by the benchmarking of the SNL PCCV model test to develop a consensus on modeling approach to assess pressure versus leakage behavior and to determine ultimate load behavior (see Figure 12.10).

Research into the integrity of containment structures for nuclear power plants has been conducted in both national and international Round Robin analyses. While the contributions of each of these efforts to the understanding of the role of containment in ensuring the safe operation of nuclear power plants is important, the most comprehensive experimental effort has been conducted at SNL, primarily under the sponsorship of the NRC. NUREG/CR-6906, "Containment Integrity Research at Sandia National Laboratories: An Overview," summarizes the major results of the experimental efforts and the observations and insights gained from the analytical efforts of more than 25 years of containment integrity research at SNL. Before pressure testing the scale models, a number of regulatory and research organizations were invited to participate in a pretest round robin analysis to perform predictive modeling of the response of scale models to overpressurization. Seventeen organizations responded and agreed to participate in the pretest round robin analysis activities. The purpose of the SNL containment integrity research was to provide a forum for researchers in the area to apply current state-of-the-art analysis methodologies to predict the capacity of steel, reinforced, and prestressed concrete containment vessels. The SPE #3 organized by the NRC and AERB progresses from these past efforts. In addition to the NRC and AERB, other international organizations from France, Finland, Korea, Sweden, Germany, and the United Kingdom are participating in SPE #3. The exercise is expected to produce a joint report describing the exercise and summarizing the results of the analyses performed.

**Figure 12.10 Completed prestressed concrete containment vessel ¼-scale model at SNL**

AERB SANDIA USNRC

**For More Information**
Contact Herman Graves, RES/DE at 301-251-7625 or Herman.Graves@nrc.gov

# Agency Forward-Looking and Long-Term Research

## Background

### Forward-Looking Research

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

### Long-Term Research

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

## Approach

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

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

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

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

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

The committee forwards the results of the review to the RES Office Director and posts the results on an internal Web site. In this way, the review committee's ratings are available to the staff as feedback on the input suggestions. Since 2010, during the planning, budgeting, and performance management process (PBPM), the RES Office Director, along with the directors of the agency's regulatory offices, agree on those long-term research projects that should receive a "high" priority and should be actively supported through those phases of the PBPM process under their control.

**For More Information**

Contact Patricia Santiago, RES/DSA at 301-251-7982 or Patricia.Santiago@nrc.gov

| NRC FORM 335 (9-2004) NRCMD 3.7 — U.S. NUCLEAR REGULATORY COMMISSION | 1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.) |
|---|---|
| **BIBLIOGRAPHIC DATA SHEET** *(See instructions on the reverse)* | NUREG-1925, Rev. 1 |
| 2. TITLE AND SUBTITLE<br>Research Activities FY2010 - FY2011<br>U.S. Nuclear Regulatory Commission | 3. DATE REPORT PUBLISHED<br>MONTH: 12 YEAR: 2010<br>4. FIN OR GRANT NUMBER |
| 5. AUTHOR(S)<br>Patricia A. Santiago, Brett Rini, Ken Armstrong, Dan Hudson | 6. TYPE OF REPORT<br>NUREG<br>7. PERIOD COVERED (Inclusive Dates)<br>2010-2011 |

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

Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Mail Stop C6D20M
Washington, DC 20555-0001

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

same as above

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

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

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

This NUREG provides a collection of information sheets, organized by topical ar eas and specific projects, that summarize programs currently in progress.

| 12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)* | 13. AVAILABILITY STATEMENT |
|---|---|
| Digital Instrumentation and Control Research, Fire Safety Research, Human Facto rs and Reliability Research, Materials Perfomance Research, New and Advanced Reactors, Radiation Protection, Reactor Safety Codes and Analysis, Generic Issues, Risk a nd Consequence Analysis, Seismic and Structural Research, Severe Accident Research, MELCOR, Wi nMACCS, Phebus-Fission Products, Zirconium Fire Research, SPAR Model, TRACE, SOARCA, VA RSKIN,PRA, Vessel Integrity, Tube Integrity, Materials Degradation | unlimited<br>14. SECURITY CLASSIFICATION<br>(This Page) unclassified<br>(This Report) unclassified<br>15. NUMBER OF PAGES<br>16. PRICE |

www.nrc.gov
NUREG-1925, Rev. 1
December 2010