NUREG/CP-0188

Transactions of the 2004 Nuclear Safety Research Conference

To be held at
Marriott Hotel at Metro Center
Washington, DC
October 25-27, 2004

U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research

Transactions prepared by
Brookhaven National Laboratory

AVAILABILITY OF REFERENCE MATERIALS IN NRC PUBLICATIONS

NRC Reference Material

As of November 1999, you may electronically access NUREG-series publications and other NRC records at NRC's Public Electronic Reading Room at http://www.nrc.gov/reading-rm.html. Publicly released records include, to name a few, NUREG-series publications; *Federal Register* notices; applicant, licensee, and vendor documents and correspondence; NRC correspondence and internal memoranda; bulletins and information notices; inspection and investigative reports; licensee event reports; and Commission papers and their attachments.

NRC publications in the NUREG series, NRC regulations, and *Title 10, Energy*, in the Code of *Federal Regulations* may also be purchased from one of these two sources.

1. The Superintendent of Documents
   U.S. Government Printing Office
   Mail Stop SSOP
   Washington, DC 20402–0001
   Internet: bookstore.gpo.gov
   Telephone: 202-512-1800
   Fax: 202-512-2250
2. The National Technical Information Service
   Springfield, VA 22161–0002
   www.ntis.gov
   1–800–553–6847 or, locally, 703–605–6000

A single copy of each NRC draft report for comment is available free, to the extent of supply, upon written request as follows:

Address: Office of the Chief Information Officer,
Reproduction and Distribution
Services Section
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
E-mail: DISTRIBUTION@nrc.gov
Facsimile: 301–415–2289

Some publications in the NUREG series that are posted at NRC's Web site address http://www.nrc.gov/reading-rm/doc-collections/nuregs are updated periodically and may differ from the last printed version. Although references to material found on a Web site bear the date the material was accessed, the material available on the date cited may subsequently be removed from the site.

Non-NRC Reference Material

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, and transactions, *Federal Register* notices, Federal and State legislation, and congressional reports. Such documents as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings may be purchased from their sponsoring organization.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at—

The NRC Technical Library
Two White Flint North
11545 Rockville Pike
Rockville, MD 20852–2738

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

American National Standards Institute
11 West 42nd Street
New York, NY 10036–8002
www.ansi.org
212–642–4900

Legally binding regulatory requirements are stated only in laws; NRC regulations; licenses, including technical specifications; or orders, not in NUREG-series publications. The views expressed in contractor-prepared publications in this series are not necessarily those of the NRC.

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

NUREG/CP-0188

Transactions of the 2004 Nuclear Safety Research Conference

To be held at Marriott Hotel at Metro Center Washington, DC October 25-27, 2004

Manuscript Completed: October 2004
Date Published: October 2004

Compiled by: S. Monteleone, Meeting Coordinator

S. Nesmith, NRC Project Manager

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

Transactions prepared by
Brookhaven National Laboratory

ABSTRACT

This report contains summaries of papers on reactor safety research to be presented at the 2004 Nuclear Safety Research Conference (at the Marriott Hotel at Metro Center in Washington, DC, October 25-27, 2004.

The summaries briefly describe the programs and results of nuclear safety research sponsored by the U.S. Nuclear Regulatory Commission's Office of Nuclear Regulatory Research. Also included are summaries of invited papers concerning nuclear safety issues from U.S. government laboratories, the electric utilities, the nuclear industry, and from foreign governments and industry.

The summaries have been compiled here to provide a basis for meaningful discussion of information exchanged during the course of the meeting, and are in the order of their presentation on each day of the meeting.

2004 Nuclear Safety Research Conference
October 25-27, 2004
Washington, DC, USA

CONTENTS

SESSION 2 : NEW REACTORS

Chaired by: M. Gamberoni (NRC)

Objective: To discuss ongoing recent or planned NRC and industry safety research activities affecting new and advanced reactors and to discuss infrastructure to support next-generation reactor technology.

Tuesday, October 26

SESSION 3A : SPENT FUEL RESEARCH

Chaired by: R. Meyer, M. Hodges (NRC)

Objective: Dry storage and transportation casks impose unique temperature and pressure conditions on spent fuel rods, and the consequences of these conditions are considered in cask licensing. Research results under these conditions and their implications will be presented in this session.

SESSION 3B : HIGH BURNUP FUEL UNDER LOCA CONDITIONS

Chaired by: R. Meyer, F. Akstulewicz (NRC)

Objective: Risk-informed, performance-based rulemaking is planned to revise 10 CFR 50.46 and Appendix K for LOCA analysis. Results of recent research will be presented, and these results will provide the technical basis for changing the embrittlement criteria in the rule.

SESSION 4 : DEVELOPMENT OF A PRA INFRASTRUCTURE

Chaired by: M. Drouin (NRC)

Objective: This session will consider insights on PRA Quality from SPAR and ASP, phased approaches (e.g., Rg 1.200), and international activities. A panel discussion on PRA methods & tools will debate our limitations, priorities, and whether future risk applications might require additional PRA tools, methods and guidance. Additionally, modeling organizational factors, integrating equipment aging into the PRA model, and quantifying the risk of low-power shutdown will be topics covered.

Wednesday, October 27

SESSION 5 : RADIATION PROTECTION

Chaired by: C. Trottier (NRC)

Objective: To discuss current initiatives in radiation protection including new recommendations from the International Council on Radiation Protection (ICRP), the challenges of homeland security, and establishing a more robust materials program.

SESSION 6 : CODES

Chaired by: S. Bajorek (NRC)

Objective: To explore issues in developing and applying computer codes to analyze reactor systems. Topics will include the analysis, applications, and maintenance of codes, the status of the TRACE project, the analysis of source terms, and the assessment and modernization of the MELCOR code.

SESSION 7 : OPERATING EXPERIENCE

Chaired by: J. Ibarra (NRC)

Objective: To present the efforts to enhance the NRC Operating Experience Program utilizing the analysis conducted by RES. Prime concern is the impact on this program from the recommendations of the Davis Besse Lessons Learned Task Force and the August 14, 2003 Northeast Blackout.

Development of Technical Basis to Support Risk-Informed Revision of the Pressurized Thermal Shock (PTS) Rule (10 CFR 50.61)

Mark EricksonKirk, David Bessette, Mike Junge, Shah Malik, Todd Mintz, and Roy Woods
United States Nuclear Regulatory Commission, Office of Nuclear Regulatory Research

Over the past 5 years, the Office of Nuclear Regulatory Research of the United States Nuclear Regulatory Commission has conducted a project aimed at development of the technical basis for a risk-informed revision of the pressurized thermal shock rule (10 CFR 50.61). Pressurized thermal shock (PTS) includes any transient wherein severe cooling of the reactor pressure vessel (RPV) occurs together with, or followed by, pressurization. A number of postulated accidents can thermally shock the vessel. The tensile stresses produced in the vessel wall by the rapid cool-down coupled with pressure induced stresses may, at high levels of neutron embrittlement, be sufficient to drive a crack all the way through the reactor pressure vessel wall. To ensure that the likelihood of vessel failure resulting from PTS is kept extremely low, 10 CFR 50.61 places statutory limits on how embrittled the RPV wall may become before licensees are required to either submit to the NRC a plant-specific rationale supporting the safety of continued operations beyond the statutory embrittlement limit, or are obligated to take steps to keep embrittlement from exceeding the statutory limit during the licensed life of the vessel.

As PWRs approach the end of their original 40-year operating licenses, and licensees consider requesting a 20 year license renewal, compliance with 10 CFR 50.61 can become a factor that limits the operational life of the plant. Addressing this issue on a plant-specific basis has consumed considerable resources within both the NRC and industry. Additionally, it is now widely recognized that state of knowledge and data limitations in the early 1980's necessitated a conservative treatment of several key parameters and models that were used to establish the current statutory embrittlement limit. These conservatisms suggest that re-evaluation of the technical basis for 10 CFR 50.61 based on a more up to date technological approach could justify reduction of unnecessary regulatory burden while maintaining safety.

The re-evaluation includes probabilistic calculations to estimate the risk of vessel failure due to PTS. This approach, which uses experimentally-benchmarked models, considers all of the factors known to influence the risk of vessel failure during a PTS event while accounting for uncertainties in these factors in a consistent manner across a breadth of technical disciplines. Two central features of the approach are a focus on the use of realistic input values and models (wherever possible), and an *explicit* treatment of uncertainties. This approach improves significantly upon that employed to establish the 10 CFR 50.61 embrittlement limits, wherein intentional and un-quantified conservatisms were included in many aspects of the analysis, and where uncertainties were treated *implicitly* by incorporating them into the models. Nevertheless, the current probabilistic model does include a limited number of conservative factors. These conservatisms remain within an overall model focused on producing *realistic* estimates of vessel failure risk due to PTS for one of two reasons: (1) to help ensure the applicability of our findings to PWRs *in general*, or (2) due to the lack of any more realistic models or parameter inputs.

The results of this research, which are summarized in this paper, demonstrate that the likelihood of vessel failure due to PTS is extremely low (~10^{-8}/year) even through the period of license extension. These results provide a basis for potential relaxation of the 10 CFR 50.61 screening limit should rulemaking be undertaken.

LOCA Frequency Evaluation Using Expert Elicitation

Robert L. Tregoning
Lee R. Abramson
United States Nuclear Regulatory Commission
Washington, DC 20555 USA

Paul M. Scott
Battelle Memorial Institute

The emergency core cooling system (ECCS) requirements are contained in 10 CFR 50.46, Appendix K to part 50, and GDC 35. Consideration of an instantaneous break with a flow rate equivalent to a double-ended guillotine break (DEGB) of the largest primary system in the plant generally provides the limiting condition in the required ECCS analysis. However, the DEGB is widely recognized as an extremely unlikely event. Therefore, the NRC is developing a risk-informed revision of the design-basis break size requirements for operating commercial nuclear power plants. A central consideration in selecting a risk-informed design basis break size is an understanding of the LOCA frequency as a function of break size.

LOCA frequency estimates have been developed using an expert elicitation process to consolidate service history data and insights from PFM studies with knowledge of plant design, operation, and material performance. This process is well-recognized for quantifying phenomenological knowledge when data or modeling approaches are insufficient. Separate BWR and PWR piping and non-piping passive system LOCA frequency estimates have been developed as a function of effective break size and operating time through the end of license extension. The elicitation focused solely on determining event frequencies that initiate by unisolable primary system side failures that can be exacerbated by material degradation with age. The expert elicitation process employed in this study is an adaptation of the formal expert judgment process used in NUREG-1150. This process included the decomposition of the complex technical issues which impact LOCA frequencies into fundamental elements in order to more easily assess these important contributing factors. The elicitation process required each member of the elicitation panel to qualitatively and quantitatively assess these LOCA contributing factors and also indicate their uncertainty in this assessment. This information was collected from each panelist in an individual elicitation session.

The qualitative insights provided by the panel members are reasonably consistent. Most panelists agreed that a complete break of a smaller pipe, or non-piping component, is more likely than an equivalent size opening in a larger pipe, or component, because of the increased severity of fabrication or service cracking. Many panelists thought that aging may have the greatest effect on intermediate diameter (6 to 14-inch diameter) piping systems due to the large number of components within this size range and the fact that this piping may receive less attention than smaller or larger diameter piping. Frequency estimates are not expected to change dramatically over the next fifteen years, or even the next thirty-five years. While aging will continue, the consensus is that mitigation procedures are in place, or will be implemented in a timely manner, to alleviate possible LOCA frequency increases.

The quantitative responses were analyzed separately for each panel member to develop individual BWR and PWR total LOCA frequency estimates of the mean, median, 5th and 95th percentiles. The LOCA frequencies for the individual panelists were then aggregated to obtain group LOCA frequency estimates, along with measures of panel diversity. While there was general qualitative agreement among the panelists about important technical issues and LOCA contributing factors, the individual quantitative estimates are much more variable. Additionally, as the LOCA size increased, the panel members generally expressed greater uncertainty in their predictions, and the variability among individual panelists' estimates increased. Both trends are expected given the underlying scientific uncertainty.

The elicitation LOCA frequency estimates are generally much less than the prior WASH-1400 estimates and more consistent with the NUREG/CR-5750 estimates. The small break (SB) LOCA frequency estimates are similar once the steam generator tube rupture frequencies are added to the NUREG/CR-5750 PWR results. The elicitation medium break LOCA estimates are higher than the NUREG/CR-5750 estimates by factors of 2.5 and 10 for BWR and PWR plant types, respectively. The NUREG/CR-5750 LB LOCA frequency estimates, most comparable to the elicitation LOCA Category 4, tend to be slightly higher (less than a factor of 3) than the current results.

Sensitivity analyses were conducted to examine the robustness of the quantitative results to the underlying analysis procedure. Sensitivity analyses investigated the aggregation method, the effect of panelist overconfidence, percentile calculation, and panel diversity measurement. The bulk of the sensitivity analyses resulted in insignificant changes to the LOCA frequency estimates. However, the method used to aggregate the individual panelist estimates can result in large differences compared to the baseline LOCA frequency estimates. Overconfidence adjustments can also result in large, unsupportable increases in the frequency estimates and need to be done with care given the extreme skewness of many of the individual distributions.

Review of the Reactor Coolant Pressure Boundary Leakage Data and Requirements for U.S. Nuclear Power Plants

M. Srinivasan[1], D. S. Kupperman, W. J. Shack[2], P. Krishnaswamy[3] and G. M. Wilkowski[3]

[1]Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington D.C. 20555-0001, U.S.A; [2]Argonne National Laboratory, Argonne, IL 60439, [3]Engineering Mechanics Corporation, Columbus, OH 4322

In 2002, the discovery of Davis Besse (DB) reactor vessel head corrosion cavity led the NRC to form a Lessons Learned Task Force (LLTF). The LLTF conducted an independent evaluation of the NRC's regulatory processes to ensure reactor vessel head integrity and identified and recommended areas of improvement. One of the recommendations of the task force was to assess the reactor coolant system (RCS) leakage requirements for potential improvements that could provide enhanced safety. Specifically, the requirements needed to be evaluated for sufficiency to (a) provide the ability to discriminate between the RCS unidentified leakage and the reactor coolant pressure boundary (RCPB) leakage; and (b) provide reasonable assurance that plants are not operated at power with RCPB leakage. The NRC Regulatory Guide 1.45 describes the RCPB leakage detection systems and states the regulatory position that the sensitivity and response time of each leakage detection system employed for unidentified leakage should be adequate to detect a leakage rate, or its equivalent, of one gallon per minute (gpm) in less than one hour.

The NRC staff recognizes that simply improving leakage detection and lowering allowable leakage may not be sufficient to provide increased assurance of RCPB integrity. Leakage monitoring assumes that the pressure boundary will fail only under a leak-before-break (LBB) scenario. The leak rates associated with tight stress corrosion (SC) cracks or cracks which may be partially plugged in stress corrosion-susceptible RCPB components could be too low to be detected reliably by existing systems under normal operating conditions. Therefore, the scope of this review also included methods which may be capable of detecting crack initiation and monitoring crack growth before a through-wall crack develops and leakage occurs. To evaluate the current requirements, the NRC's Office of Nuclear Regulatory Research (RES) conducted barrier integrity research at Argonne National Laboratory (ANL) as a follow-up and an update of similar research conducted by the NRC in the late1980s.

This research resulted in a database that identifies the source of leakage, the leak rate, and the resulting actions to remedy RCS leaks in U.S. LWRs. For each leakage event, the database provides information on what equipment detected the leakage, how it was determined that the leakage was through the pressure boundary, and the cause of leakage. Many reported leakages were of very small leakage rate (< 0.01 gpm), were detected visually and were reported as drips, weeping, seepage, "very small," boric acid deposits, etc. Cracks were associated with leaks about 40% of the time with a wide range of leak rates (< 0.01 gpm to > 100 gpm).

This research examined the capabilities of different types of leakage detection systems, which were assessed for their sensitivity, reliability, response time and accuracy. Significant improvements in leak detection capability will require implementation of new systems. Such systems are not only sensitive and accurate, but often can be used to determine leak locations and thus reduce the likelihood of unnecessary shutdowns. Newer commercially available systems include: acoustic emission monitoring, humidity sensors, and air particulate detectors.

This research also determined that acoustic emission (AE) crack monitoring systems can also detect crack initiation and growth before a leak occurs. The same systems used for leak detection can also be used to detect crack initiation and growth. The AE from crack growth and leaks can be separated from noise by monitoring at a high enough frequency. Such technology has been successfully used to monitor selected locations at Watts Bar and Limerick power reactors.

Correlation studies between leak rates and crack size were conducted using computer models. For a given leak rate, it was found that the corresponding crack size could vary by factor of 5 to 10 depending on the applied load on the cracked cross section. More refinements in the model and mechanistic understanding are needed for more effective predictions.

Based on the results of this research, the NRC staff is considering how the regulatory requirements for RCS leak detection and monitoring could be improved as per the DB LLTF recommendation. However, many challenges need to be addressed in recommending improvements, such as the need for backfit analysis or a cost-benefit analysis, that would potentially enhance public safety while reducing unnecessary regulatory burden on utilities. Implementation challenges also exist such as changes in technical specifications, regulatory orders, or potential rulemaking.

The contents of this presentation are based on the authors' knowledge and research sponsored by the NRC Office of Nuclear Regulatory Research. This presentation is an independent product of the authors and does not necessarily reflect the views or regulatory position of the US NRC.

Status and Results of NRC's Proactive Materials Degradation Assessment Program

Joseph Muscara

Materials Engineering Branch, Division of Engineering Technology'
Office of Nuclear Regulatory Research, USNRC

There is a growing recognition both in industry and at the US Nuclear Regulatory Commission (NRC) that proactive approaches to the management of material degradation are needed particularly as plants age and continue to operate into their 20-year license renewal period. Therefore, the NRC Office of Nuclear Regulatory Research (RES) has initiated work to develop a foundation for regulatory actions to prevent materials degradation from adversely impacting safety. A two-step approach will be used for this work. The first step is to identify susceptible materials and locations where degradation can reasonably be expected in LWRs in the future. The second step is to cooperatively develop and implement an international research program for the components and degradation of interest. This research program would address materials and degradation mechanisms, mitigation, repair and replacement, and nondestructive evaluation.

As a starting point for the first step, identification of susceptible materials and locations, RES is conducting a review using the Generic Aging Lessons Learned (GALL) report to identify components that have historically experienced degradation. The effectiveness of in-service inspection and leakage monitoring requirements and techniques for these components is then evaluated with respect to timely detection of known degradation mechanisms. Recommendations for improvements in requirements and techniques will be made as necessary. The review will be conducted by a Nondestructive Examination and Leak Monitoring (NDELM) task group comprised of NRC Office of Nuclear Reactor Regulation (NRR), RES, and national laboratory experts in inspection and leak monitoring techniques and requirements.

In addition to using the GALL report, a concurrent effort is underway to identify LWR components that may be susceptible to material degradation in the future. To accomplish this task, RES is using a modified Phenomena Identification and Ranking Table (PIRT) process to perform the Proactive Materials Degradation Assessment (PMDA). The PIRT process has several qualities that make it ideal for the PMDA: a) provides a structured approach to expert elicitation where background information is provided to an Expert Panel who identify material degradation of interest, b) requires Panel Experts to document the bases for their selections, c) contains a provision for documenting knowledge levels associated with decisions, and d) allows for ongoing documentation over the course of the process.

RES is currently sponsoring a program at Brookhaven National Laboratory (BNL) to collect operational experience and stressor information for the PWR and BWR components to be examined by the PIRT Expert Panel and to manage and document the PIRT Expert Panel and process. The Panel Experts will use the operational experience and stressor information to identify potential degradation mechanisms, if any, for each component. The Expert Panel is comprised of eight world-class materials experts from industry, research laboratories, regulatory agencies, and universities from the US, Canada, Japan, France, and Sweden.

The Expert Panel will meet eight times, with each meeting lasting one week, to examine all of the LWR components of interest. Four of the meetings will be used to examine PWR plant components, and the remaining four will be used to examine BWR plant components. The first Expert Panel meeting to examine the first set of PWR components was held in August 2004 and there will be approximately six weeks between meetings. After each meeting, the components identified by the Expert Panel as being susceptible to future degradation will be provided to the NDELM task group who will assess NDE and leak monitoring techniques and requirements and make recommendations for improvements.

A final report detailing the results of the Expert Panel deliberations for PWRs will be internationally peer reviewed and ready for publication in June 2005. The final peer reviewed report for BWRs will be available in December 2005.

For each of the components susceptible to degradation identified through the GALL review and PMDA PIRT effort, the Conditional Core Damage Probabilities (CCDPs) will be determined. The probabilities of failure and associated uncertainty estimates for the components identified in the above efforts will also be determined. Probability of failure information will be collected where it already exists from other programs and sources such as aging programs, the LBLOCA redefinition effort, and in licensee degradation assessments to support Risk Informed In-service Inspection (RI-ISI) programs. This information will be provided by June 2005. The information collected from these sources is unlikely to contain probability of failure estimates for all of the components identified by the NDELM task group or by the PIRT Expert Panel, nor will it generally contain uncertainty estimates. A comprehensive effort to determine these probability of failure and uncertainty estimates will be completed in FY06.

A Knowledge-Based Approach to Management of Materials Degradation

Robin Dyle, Southern Nuclear Company
Alex Marion, NEI

Members of the Materials Technical Advisory Group

As a result of several recent industry issues involving materials degradation, the NEI Nuclear Strategic Issues Advisory Committee (NSIAC), in May 2003, unanimously approved the "Industry Initiative on Management of Materials Issues" and its associated "Guideline for the Management of Materials Issues," NEI 03-08. NEI 03-08 creates a Materials Executive Oversight Group (MEOG) and a Materials Technical Advisory Group (MTAG) to provide policy direction, oversight and support to the industry's materials programs. An important element of the oversight and support role is the creation and maintenance of a high-level, industry-wide materials management strategic plan that identifies the highest priority challenges/activities. Additionally, an Integrated Materials Management Work Plan has been developed in support of the Strategic Plan. The Work Plan will provide details on each strategic issue, identify which Issue Program(s) is working on particular topics and identify knowledge/technology gaps.

The Initiative's policy-objective is to have a more forward-looking, coordinated industry-wide program, through the effective coordinated operation of the individual Issue Programs, for managing materials aging and degradation. The intent of this strategic approach is to have a comprehensive identification of issues and gaps that can be used by the Issue Programs in developing their specific strategic and annual work plans.

A systematic approach to managing materials issues is necessary to effectively implement this Strategic Plan as well as the elements of an effective materials issues management program. The approach must focus on communicating with stakeholders, and on providing appropriate direction and guidance to the industry. The systematic approach will identify, assess, mitigate and/or repair known and future degradation mechanisms. It will incorporate operating experience, impact of operational changes (e.g. power uprate), relevant laboratory work and input from stakeholders. The first step in that process is to develop fundamental understanding of degradation phenomena/mechanisms, and determine materials (and locations) that are known, or can logically be assumed, to be, susceptible to aging/degradation phenomena when exposed to the operating environment.

In using the systematic approach to develop this Plan, a Degradation Matrix (DM) was developed that addressed material degradation in a top-down fashion. All materials within the scope of the initiative were considered and listed in the DM. Expert elicitation, laboratory studies and field experience were then utilized to identify potential mechanisms by which materials within this broad population can degrade. The material-mechanism combinations were sorted into three large groups. The first group is the set of degradation mechanisms/issues that have been adequately addressed and, where necessary, have ongoing management programs in place. The second group is for those mechanisms/issues that are determined to have work in place for tools/methodologies needed for effective management. The third group is for those mechanisms that are identified as potential challenges for which there is no program or effective methodology for understanding or managing the mechanism at a phenomenological level.

The DM will be maintained as a living document and updated annually. It is a key resource document that will be used with other reference material for the effective management of materials aging and degradation.

Regulatory Structure for New Plant Licensing: Technology-Neutral Framework

Mary Drouin, Tom King, NRC
John Lehner, Trevor Pratt, Vinod Mubayi, BNL
Dennis Bley, Buttonwood Consulting

A risk-informed regulatory structure that can be applied to license and regulate advanced (new) reactors, regardless of their technology, could enhance the effectiveness, efficiency, and predictability (i.e., stability) of new plant licensing. As such, this new process, if implemented, could be available for use later in the decade. The need for development of a risk-informed regulatory structure for new reactors is based on a number of considerations. While the NRC has over 30 years experience of licensing and regulating nuclear power plants, this experience (e.g., regulations, regulatory guidance, policies and practices) has been focused on current light water-cooled reactors (LWRs) and may have limited applicability to new reactors. There will be design and operational issues associated with the new reactors that may be distinctly different from current LWR issues.

The regulatory structure for current LWRs has evolved over five decades, and the bulk of this evolution occurred without the benefit of insights from probabilistic risk assessments (PRAs) and severe accident research. It is expected that future applicants will rely on PRA and PRA insights as an integral part of their license applications. In addition, it is further expected that the regulations licensing these new reactors will be risk-informed. Both deterministic and probabilistic results and insights will be used in the development of the regulations governing these reactors. Consequently, a structured approach for a regulatory structure for new reactors that provides guidance about how to use PRA results and insights will help ensure the safety of these reactors by focusing the regulations on where the risk is most likely while maintaining basic safety principles, such as defense-in-depth and safety margin.

The objective of the regulatory structure for new plant licensing is to provide a technology-neutral risk-informed approach that would enhance the effectiveness and efficiency of new plant licensing in the longer term (beyond the advanced designs currently in the preapplication stage). This regulatory structure is comprised of four major parts:

(1) a technology-neutral risk-informed framework that will provide guidance and criteria to the staff for the development of technology-neutral requirements,

(2) the content for a proposed set of technology-neutral risk-informed requirements that will be based on the guidance and criteria established in the technology-neutral framework,

(3) a technology-specific framework that will provide guidance and criteria for the staff on how to apply the technology-neutral framework and requirements on a technology-specific basis, and

(4) technology-specific regulatory guides that will be derived from the implementation of the technology-specific framework and will provide guidance to licensees on how to apply the technology-neutral regulations on a technology-specific basis.

The key features of the framework are as follows:

- a hierarchal approach based on a safety philosophy that is consistent with the Commission's expectations for new reactors

- protective strategies that provide the basis for the technical requirements needed to ensure that the safety philosophy has been met

- a defense-in-depth approach to address uncertainties that is based on previous Commission guidance, on recommendations from the ACRS and on cornerstones developed as part of the Reactor Oversight Program

- a probabilistic (risk-informed) approach that identifies and selects design basis accidents and that assigns safety classification

- a performance-based approach that will establish performance standards and acceptance criteria for results in development of the requirements

- incorporation of Commission direction on security related matters for new plants

To determine that the overall objectives of the regulatory structure have been met (e.g., enhanced effectiveness and efficiency), the staff identified key characteristics for the regulatory structure. Examples include:

- Flexible—The technology-neutral and technology-specific frameworks are developed in such manner that they allow for changes and modifications to occur, in an efficient and effective manner, that are based on new information, knowledge, etc., and can be adapted to any technology-specific reactor design.

- Risk-informed—Risk information and risk insights are integrated into the decision-making process such that there is a blended approach using both probabilistic and deterministic information.

- Performance-based—The guidance and criteria, when implemented, will produce a set of safety requirements that will minimize prescriptive means for achieving its goals, and therefore, will be performance oriented to the extent practical.

- Uncertainty—The guidance and criteria will include treatment of the different types of uncertainties.

- Defense-in-depth—Defense-in-depth is maintained and is an integral part of the framework.

Testing in the APEX Facility to Investigate Advanced Plant Thermal-Hydraulics

Stephen M. Bajorek
Office of Nuclear Regulatory Research

INTRODUCTION

The Advanced Plant Experiment (APEX) is a unique thermal-hydraulic integral system test facility used to assess the performance of passive safety systems for the Westinghouse AP600 and AP1000 designs. Data from the APEX facility was used extensively as part of AP600 Design Certification, providing approximately 75 tests for use in code assessment and qualification of AP600 safety margin. To address performance specific to the AP1000 design, the APEX facility underwent significant modifications in 2002 in order that it more accurately represent AP1000. Included in the facility modification were an increase in the maximum core power, a new pressurizer and surge line, a larger core makeup tank (CMT), larger diameter fourth stage automatic depressurization (ADS-4) system piping, and decreased line resistances for the CMT, ADS-4, and PRHR heat exchanger.

Testing in the "APEX-AP1000" facility began in 2003, with several integral experiments sponsored by the U. S. Department of Energy (DOE) to investigate performance of AP1000 passive safety systems at design basis accident conditions. The NRC Office of Nuclear Regulatory Research (RES) also conducted several integral tests to explore beyond design basis performance and to provide confirmatory information on thermal-hydraulic processes for which data from the APEX-AP600 series of tests may not have been adequate. In particular, entrainment in the hot leg and upper plenum is of significant interest because of the higher core power in AP1000 than in the AP600. These data are also of interest because they provide unique information on upper plenum thermal-hydraulics that may be useful in understanding carryover from an upper plenum of a conventional pressurized water reactor during long term cooling.

TESTS EVALUATED

The table below lists the NRC APEX-AP1000 tests conducted for which an evaluation was completed. Most of the 2003 tests investigated the double-ended guillotine break of a direct vessel injection (DVI) line.

Test Identifier	Description
NRC-AP1000-01	Double-ended guillotine break of DVI line #1 with complete failure of ADS-1/2/3 system.
NRC-AP1000-03	Double-ended guillotine break of DVI line #1 with 2/4 ADS-4 valves available. Both ADS-4 valves on pressurizer side failed closed.
NRC-AP1000-04	One-inch break at the bottom of cold leg CL-4 with degraded sump.
NRC-AP1000-05	Double-ended guillotine break of DVI line #1 with 2/4 ADS-4 valves available. Both ADS-4 valves on non-pressurizer side failed closed.
NRC-AP1000-06	Two-inch break at the bottom of cold leg CL-4 with 2/4 ADS- 4 valve failure. Both ADS-4 valves on pressurizer side failed closed.

CONCLUSIONS BASED ON NRC's APEX-AP1000 TESTS

1. The APEX-AP1000 tests confirm significant entrainment and carryover of water to the ADS-4 system during and after ADS-4 actuation. As expected, flow quality in the ADS-4 is low when the water level in the vessel is above the bottom of the hot leg. Carryover to the ADS-4 can also be significant, however, after the vessel level decreases below the bottom of the hot leg. In addition, even when the upper plenum collapsed level is low, a water level in the hot legs can still persist. This suggests that the processes of entrainment and flow dynamics within the hot leg remain important to carryover to the ADS-4 even when the water level in the vessel decreases well below the bottom of the hot leg.

2. Design basis tests from the APEX-AP1000 facility did not result in core uncovery and heat up. This significant finding indicates margin to uncovery in the AP1000 design for small break loss of coolant accidents.

3. Both design basis tests by DOE and beyond design basis tests by the NRC (NRC-AP1000-03 versus NRC-AP1000-05) in APEX-AP1000 show that failure of ADS-4 valves on the non-pressurizer side of the system results in a greater delay in IRWST injection than failure of ADS-4 valves on the pressurizer side of the plant. The likely cause is higher two-phase resistance in the pressurizer side ADS vent path by water draining from the pressurizer.

4. Failure of the two of four ADS-4 valves were found to produce a core uncovery (NRC-AP1000-05 and NRC-AP1000-06). While this is a beyond design basis accident scenario, it identifies a sensitivity ADS-4 valve performance.

In general, the NRC tests provided new data for code assessment and model development. The tests helped to confirm robustness in the passive safety system design and operation, since no core uncovery was found to occur unless multiple failures were imposed on the transient.

NRC Nuclear Analysis Research for Reviewing the Advanced CANDU Reactor, ACR-700

Donald E. Carlson
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

Since September 2002, the NRC staff has been conducting a preapplication review of the Advanced CANDU (CANada Deuterium Uranium) Reactor, ACR-700, in response to plans by Atomic Energy of Canada Limited (AECL) to submit an ACR-700 design certification application in March 2005. The NRC Office of Nuclear Regulatory Research (RES) has collaborated with the NRC Office of Nuclear Reactor Regulation (NRR) on the preapplication review activities and is preparing to support the design certification review by adapting and validating NRC's audit analysis codes for use in predicting how the ACR-700 system will respond under operating, transient, and accident conditions. Included in the audit analysis code work for ACR-700 are RES efforts on phenomena identification and ranking, code adaptation and modeling, experiment and test data evaluation, and code bias and uncertainty assessment (i.e., code validation) involving the technical areas of nuclear analysis, thermal hydraulic analysis, and severe accident analysis. This paper describes these RES efforts in the area of nuclear analysis.

In its nuclear design and operating characteristics, the ACR-700 differs significantly both from NRC-licensed light water reactors (LWRs) and from existing CANDU reactors. Important nuclear design changes in relation to existing CANDUs include the following: (1) Light water replaces heavy water as the fuel coolant within the pressure tubes. (2) The core diameter is reduced by decreasing the spacing between the pressure-tube fuel channels. (3) The volume of heavy water moderator between fuel channels is further reduced by enlarging the diameter of the gas-filled calandria tube that encloses each pressure tube. (4) A 43-pin CANFLEX™ fuel bundle design replaces the previous CANDU-standard 37-pin bundle while maintaining essentially the same overall bundle dimensions. (5) The natural uranium oxide (UO_2) pellet composition used in conventional CANDU fuel pins is replaced in ACR-700 by pellet compositions of slightly enriched UO_2 in all but the central CANFLEX™ fuel pin whose pellets are composed of natural UO_2 doped with dysprosium burnable poison. (6) The average fuel bundle burnup at discharge increases from typically 7.5 GWd/t in existing CANDUs to 21 GWd/t in the preapplication reference design of ACR-700.

To enable the simulation of such ACR-700 design features as on-line fueling, pressure-tube core geometry, separate moderator and coolant systems, and control rods that pass vertically between horizontal fuel channels, new capabilities are being added to NRC's nuclear analysis codes that have never been needed for analyzing LWRs. New capabilities being added to the Triton lattice physics modules of the NRC's Standardized Computer Analysis for Licensing Evaluation (SCALE) code system include employing the existing Keno Monte Carlo module to provide the detailed 3-dimensional (3-D) neutron flux solutions needed for the lattice physics treatments of transverse control rods and fuel bundle end effects in ACR-700. The fuel-depletion and control-rod lattice results from SCALE/Triton provide the collapsed and smeared nodal data needed by the NRC's Purdue Advanced Reactor Core Simulator (PARCS) code, which, when coupled to a system thermal hydraulics code model of ACR-700, will provide 3-D static and dynamic solutions of in-core neutron flux and power distributions during reactor operations, transients, and accidents. Further code and modeling developments in SCALE/Triton and PARCS are needed to address the channel-to-channel neutron spectral interactions that govern coolant void reactivity (CVR) effects during the initial "checkerboard" voiding of alternate fuel channels in ACR-700 large-break loss-of-coolant accidents.

Evaluating calculation bias and uncertainty is especially important in this context because confirmatory measurements of CVR in an operating conventional CANDU or ACR-700 core are inherently difficult and have never been performed in existing CANDUs. Associated ACR-700 CVR validation activities in RES therefore include adapting the existing sensitivity and uncertainty analysis methods in SCALE's Tsunami code modules for use in assessing the applicability and adequacy of AECL's existing and planned sets of semi-prototypic critical experiments and fuel irradiations. The modified SCALE/Tsunami modules will also be used to estimate the bias and uncertainty in code predictions of CVR and other safety-significant nuclear effects in the ACR-700 core based on the emerging sets of code-to-data benchmark results. While motivated mainly by specific needs and analysis issues associated with the staff's upcoming ACR-700 design certification review, these improvements to NRC's nuclear analysis code infrastructure will provide the staff with enhanced audit analysis capabilities that will prove broadly applicable to understanding the safety-related behavior of other nuclear systems.

Pebble Bed Modular Reactor
Research and Development for an Innovative Small Reactor

Edward G. Wallace
Senior General Manager - US Programs
PBMR Pty, LTD., Centurion, South Africa

The pebble bed modular reactor (PBMR) being developed in the Republic of South Africa is a development derived from German technology first deployed in the mid-1960s at the AVR plant in Jülich, Germany. The AVR was a 30 MWth, indirect cycle research reactor that demonstrated the basic characteristics of pebble fuel reactors. This demonstration was followed by the 300 MWe THTR indirect cycle reactor which operated in the mid-1980s for over two years, before being shut down for non-technical reasons. The PBMR design is founded on the development work and insights gained from the research, construction and operations of these two predecessors. The PBMR Demonstration Power Plant is a full scale, single module demonstration of the follow-on commercial multi-module plant design.

The design of the PBMR is different than its predecessors in that it employs a direct cycle (i.e., Brayton cycle) arrangement and operates at slightly higher core exit temperatures of 900°C. The PBMR project began in 1993 in South Africa. This presentation will cover the research and development, past, present and future, in four phases; basic R&D for concept finalization; initial development for component design; verification programs and advanced research and development.

Basic R&D Programs

Initial test programs were developed for core design, primary loop simulation, high temperature reactor support components and power conversion unit components. PBMR and its key suppliers performed a series of tests on part-scale and full-scale equipment to confirm the conceptual design approach. Significant among the testing were the development and operation of a high temperature nitrogen gas, three-shaft, Brayton cycle representation (PBMR Micro-Model or PBMM) of the PBMR thermo-dynamic cycle. This facility demonstrated the basic controllability of the three-shaft Brayton cycle power conversion unit (PCU) and provided valuable benchmark data for PBMR thermodynamic codes. With the completion of the PBMM testing, the initial concept design, system requirements and arrangements were completed.

Component Development Testing Programs

In 1999-2001, PBMR began development of individual test programs for high temperature helium components and other reactor auxiliary items, and laboratory-scale fuel equipment and associated quality measurement process development that were critical to the conceptual design or fuel manufacturing process. This included a wide variety of fuel handling system components and development of critical PCU components. The results of these development tests confirmed the adequacy of a large number of individual components for use in the plant, but were not under full environmental conditions. In the case of the PCU, test results coupled with basic design analysis made way for a rearrangement of the PCU from a vertical, three-shaft cycle to a single, horizontal PCU.

Verification Testing Programs

Beginning in 2002, PBMR has identified a number of additional test facilities that would be useful to verify and validate design performance codes and to demonstrate full reactor and auxiliary component operability in the design high temperature PBMR environment. PBMR will be constructing a High Temperature Facility (HTF) that will demonstrate system and component operability and reliability under full design temperature and pressure conditions. The HTF will also be full height for reactor and auxiliary system testing, where operating system performance could be influenced by the physical scale of the system. The HTF will be constructed in 2005-6 and begin testing in mid-2006. PBMR will be constructing a Heat Transfer Test Facility to validate all heat transfer mechanisms and coefficients in a pebble core configuration. In 2004-5 PBMR is conducting testing on the NACOK facility in Germany to further explore natural circulation conditions and air ingress phenomena. Finally, PBMR is constructing fuel production facilities at the Pelindaba site for the Koeberg Demonstration Power Plant. This facility will demonstrate production fuel manufacturing capability and also provide early fuel for the PBMR fuel irradiation testing program (FITP). The FITP will confirm the quality of the PBMR manufactured fuel for the PBMR design envelop conditions and add additional confirmation data to the established German fuel data base.

Advanced R&D Programs

PBMR has identified several different testing needs for the future to extend the PBMR design beyond the technology used in the basic PBMR commercial design. These programs are focused on three primary objectives: to improve fuel performance, utilization and disposal options for existing PBMR plants; to develop component, materials, and fuel for VHTR conditions approaching 1200°C; and to use PBMR-based plants in other applications such as hydrogen co-generation, desalination, or other process heat applications.

ACR Research and Development

A.J. White and D.J. Wren
Atomic Energy of Canada Limited

A new advanced reactor design, the ACR-700, is currently being developed to meet the needs of the electrical utilities for safe, reliable and economic power generation technologies. The ACR-700 is an evolutionary design that is based on the design of the CANDU reactors in operation around the world. A few key design features differentiate the ACR-700 design from the reference CANDU design, notably the use of slightly-enriched uranium fuel and the use of light water as a coolant. The ACR-700 design is currently being reviewed by the US-NRC under a pre-application review program.

The development of the ACR-700 is supported by a research and development program that addresses the safety analysis of the design and the performance of the safety systems. In addition, the ACR-700 development program includes activities to improve the economic reliability and performance of the plant. In both cases, the research programs are incremental and the technologies involved are based on the results of decades of research and operational experience.

A key element of the ACR-700 development program is the extension of the safety analysis code suite used to analyze the safety performance of CANDU reactors for application to the ACR-700 design. A comprehensive program is in place to extend the validation of the computer codes used to model the core neutronics and the transient thermalhydraulics of the reactor coolant system under accident conditions. The former includes experiments in the critical lattice facility (ZED-2) at the Chalk River Laboratories, analyses of data from other relevant core configurations, and comparison with other code calculations (including MCNP). The thermalhydraulics studies extend the validation of the CATHENA code for the ACR-700 design using results from the large-scale RD-14M test loop at the Whiteshell Laboratories.

The ACR-700 design, like the reference CANDU design, has a number of inherent, passive features that slow the progression of potential severe accidents, mitigate the consequences, and provide an extended time period for operator action to further mitigate event consequences. The ACR-700 research program in this area is strongly coupled to the research program supporting the operating CANDU plants. Research is in progress to support understanding and modelling of the core degradation processes that are unique to the ACR-700/CANDU design.

THIS PAGE INTENTIONALLY BLANK

Advanced Nuclear Energy Research in the United States

Carl Sink
U.S. Department of Energy

Advanced reactor and fuel cycle efforts in the United States sponsored by the Department of Energy are carried out under three programs: the Generation IV Nuclear Energy Systems Initiative, the Nuclear Hydrogen Initiative, and the Advanced Fuel Cycle Initiative. These programs all support an overall strategy to create a new generation of nuclear energy systems which will be deployable no later than 2030 in both developed and developing countries, both for generation of electricity and to support a future hydrogen economy.

A technology roadmap for Generation IV nuclear energy systems was developed to identify the most promising nuclear energy systems (consisting of both a reactor and associated fuel cycle) for meeting the challenges of safety, economics, waste, and proliferation resistance and physical protection. More than 100 experts from 10 countries evaluated more than 100 systems proposed by the worldwide R&D community leading to the selection of six Generation IV systems: the Gas-Cooled Fast Reactor (GFR), the Lead-Cooled Fast Reactor (LFR), the Molten Salt Reactor (MSR), the Sodium-Cooled Fast Reactor (SFR), the Supercritical-Water-Cooled Reactor System (SCWR), and the Very-High-Temperature Reactor (VHTR). Significant R&D is necessary to overcome technical challenges to develop each of the six concepts. Crosscutting R&D is also being conducted in the areas of fuel cycle, fuels and materials, energy products, risk and safety, economics, proliferation resistance and physical protection.

The Nuclear Hydrogen Initiative has been established to develop the production technologies that can be most effectively coupled to next generation nuclear reactors for hydrogen production. Efficient processes that can utilize the high-energy output of a nuclear reactor include thermo chemical splitting of water and high-temperature electrolysis. The Advanced Fuel Cycle Initiative is working to provide advanced fuel cycle technologies ready to be deployed with light water- and gas-cooled reactors that have the ability to achieve a significant reduction in the amount of high-level spent fuel requiring geologic disposal, to reduce significantly the amount of accumulated plutonium in civilian spent fuel, and to extract more useful energy from spent nuclear fuel components.

THIS PAGE INTENTIONALLY BLANK

High-Burnup Cladding Mechanical Performance during Cask Storage and Post-Storage Handling and Transportation

Robert S. Daum, Hanchung Tsai, Yung Liu, and Michael C. Billone

Argonne National Laboratory (ANL)
Argonne, Illinois, USA

The assessment of cladding performance during spent-nuclear-fuel (SNF) pre-storage operations, dry-cask storage, and post-storage handling and transport, including hypothetical accidents, is important in ensuring that sub-criticality is maintained, that radioactivity is contained, that cask external dose rates are limited, and that SNF assemblies can be safely retrieved, handled and transported at the end of dry-cask storage. Although cladding failure is not prohibited by federal regulations (CFR 71 and 72), the failure mode and extent of cladding failure have a significant effect on the possible reconfiguration of SNF within storage and transport casks. High-burnup, SNF Zircaloy-4 (Zry-4) cladding from pressurized-water reactors is more susceptible to failure and possible fuel dispersal than low-burnup Zry-4 due to higher hydrogen pickup, higher internal pressure and higher corrosion level. Hydride precipitation, corrosion (i.e., wall thinning), irradiation-induced defects, and possibly even post-reactor thermal creep, will reduce cladding ductility and impact resistance. Also, transfer-and-drying operations conducted from the SNF pool to the storage cask may result in thermo-mechanical conditions that promote radial-hydride-induced degradation of cladding ductility and impact resistance during severe loadings associated with hypothetical accidents. Such cladding would be more susceptible to brittle failure at the end of dry storage when temperatures are expected to be 150-250°C – below the ductile-to-brittle transition temperature for zirconium hydrides. A testing program has been developed to investigate these effects in high-burnup Zry-4 cladding, with particular emphasis on the conditions that promote radial-hydride precipitation and the effects of these radial hydrides on cladding integrity. Preliminary experimental results are presented from axial tensile and thermal creep tests, as well as conditions for radial-hydride precipitation. The test plan, which includes ring-compression-ductility and ring-crush-impact screening tests, is also described

The current regulations for storage and transportation of SNF are designed primarily to maintain sub-criticality and to ensure that doses are less than regulatory limits, that the cask provides adequate fuel confinement and containment, and that the fuel is retrievable. Interim Staff Guidance No. 11, Revision 3 limits high-burnup cladding temperatures to ≤400°C during short-term operations and normal storage conditions. Although this temperature limit is intended to minimize radial-hydride precipitation in high-burnup cladding, license applications for high-burnup storage, storage-and-transport, and transport casks are evaluated on a case-by-case basis due to the lack of pertinent data to assess high-burnup cladding behavior during the complete cycle of pre-storage operations, storage and post-storage handling and transport. The data generated within the ANL test program are to be used both by applicants for high-burnup cask licenses (e.g., nuclear industry and DOE-RW) and by cask-license evaluators (NRC-SFPO).

Experiments have concentrated on the microstructural characterization and mechanical-property testing of stress-relieved Zry-4 (15x15 design) irradiated in H.B. Robinson Unit No. 2 to a rod-average burnup of 67 GWd/MTU and a fast neutron fluence (E > 1 MeV) of 14×10^{26} n/m^2. In Grid Spans 3 and 4 (fuel midplane to 0.7-m above the fuel midplane), destructive examinations of the as-irradiated cladding have been performed. Circumferentially averaged (8 locations) corrosion layers of 70±10 to 100±10 μm, with no spalling, have been measured, along with cross-section-averaged (90° segments) hydrogen contents of 550±80 to 750±90 wppm. Precipitation of circumferential hydrides shows varying distribution, density, and particle size along the axial, azimuthal, and radial directions of the cladding. In a few cladding locations, transverse metallography in an etched condition shows a significantly high hydride density localized in roughly a 90° arc directly under the outer-surface corrosion layer and to a radial depth of ≈100 μm, suggesting the presence of a hydride "lens." Such hydride microstructures are known to reduce cladding ductility under tensile loading, but little is known about the evolution of these microstructures under thermo-mechanical conditions associated with SNF drying, transfer and storage.

As compared to non-irradiated Zry-4 (15x15 design), room-temperature axial tensile properties of the high-burnup (690±40 wppm H) Zry-4 at a strain rate of 0.1%/s show an increase in yield (600 → 770 MPa) and ultimate tensile (765 → 950 MPa) strengths and a decrease in uniform (6 → 3%) and total (14 → 4%) elongations. The strength increase appears to be due mainly to radiation-induced hardening, while the ductility decrease appears to be due to both radiation- and hydride-induced embrittlement. Thermal annealing tests with high-burnup Zry-4 samples show that strength properties (based on microhardness data) appear to recover by ≈75% and circumferential hydrides tend to homogenize across the cladding radius after 72 hours at 420°C. These results suggest that SNF drying operations may partially anneal radiation-induced hardening. The degree of ductility recovery with annealing remains to be demonstrated.

Two thermal creep tests (C14 and C15) have been completed using defueled high-burnup Zry-4 cladding specimens, which are top-welded to active internal-gas-pressurization systems in order to maintain constant gas pressure inside the creep specimens. These tests were conducted at 400°C for 101 days at a pressure of 29.5 MPa and an initial hoop stress of 190 MPa. The specimens were depressurized periodically prior to cooling to room-temperature for diameter measurements. Both specimens remained intact, no localized bulging (precursor to rupture) was observed, the average hoop creep strains were ≈3.6% and the peak hoop creep strains were ≈5%. In order to induce radial-hydride precipitation, the C15 specimen was cooled at ≈2.4°C/s from 400°C under full pressure during the final shut-down. The sample depressurized at 205°C and a midplane true hoop stress of ≈205 MPa due to failure in the upper weld region. Post-test metallography at three axial locations showed significant radial-hydride and negligible circumferential-hydride precipitation. However, post-test hydrogen measurements indicated substantial loss of hydrogen from the C15 specimen (≈670 → 320 wppm at the midplane) to the Zircadyne-702 end-fittings (12 → 210 wppm at the bottom end-plug). Redesign of the thermal creep test train and furnace is in progress to minimize the hydrogen loss and the axial temperature gradient from the specimen midplane (400°C) to 30 mm above the midplane (≈390°C). However, it is interesting to note that the hydrogen solubility of non-irradiated Zry-4 is ≈210 wppm at 400°C. The absence of visible circumferential hydrides at the specimen midplane (with 320 wppm) suggests that high-burnup Zry-4 is capable of trapping about 100 wppm of hydrogen, which most likely precipitates during rapid cooling as sub-micron-size hydrides. It will be interesting to determine if this excess hydrogen precipitates as visible radial hydrides under the slow cooling rates (<4°C/day) typical of drying-transfer-storage.

The preliminary axial-tensile, thermal-creep, and hydride-reorientation results have been used to develop a test plan to better understand the mechanical behavior of high-burnup Zry-4 cladding under drying-transfer, storage, and post-storage handling-transport conditions. In additional to tensile and creep tests of pool-stored high-burnup Zry-4, sealed specimens will be annealed for ≈3 days at 380-420°C and at hoop stresses of 0, 60, 90, 120, and 150 MPa and slow-cooled at ≈3°C/day under decreasing pressure. Rings cut from these 100-mm-long samples will be subjected to ductility (diametral compression at 0.1%/s and 100%/s) and crush-impact failure-energy screening tests. These tests will be conducted at room-temperature and 150°C. The decreases in ductility and failure-impact energy will be correlated to the extent of radial hydride formation to map out cooling conditions – especially stress at 400°C – that are detrimental to high-burnup Zry-4 cladding integrity. Additional tests (e.g., fracture toughness) may be conducted on cladding subjected to these detrimental cooling conditions.

THIS PAGE INTENTIONALLY BLANK

Data Needs for the Transportation and Storage of High Burnup Fuel

RE Einziger, CL Brown, CG Interrante and GP Hornseth

US Nuclear Regulatory Commission
Spent Fuel Program Office

The storage and transportation of low to medium burnup spent nuclear fuel (<45 GWd/MTU) are mature, ongoing operations with a strong safety record. In order to improve nuclear reactor utilization, utilities are operating plants with fuel that is licensed for extended burnup. Fuel with burnups up to 62 GWd/MTU and possibly beyond will need to be stored and transported. In addition, events of recent years have raised the question of the safety of transport and storage casks that might come under terrorist attacks.

The NRC has been recently focusing on the storage and transportation of the higher burnup fuel because it's cladding may have degraded mechanical properties relative to lower burnup fuel. The increased fluence and time-in-reactor have caused changes in the fuel pellet and cladding characteristics, In addition, newer cladding alloys are in use that have been developed to meet in-reactor performance standards.

During extended operation, the fuel is in the reactor for a longer time, hence more cladding oxidation occurs. Approximately 15-20% of the hydrogen generated during the oxidation process diffuses into the cladding, and for the most part accumulates at the outer cooler edge. Depending on the alloy, the hydrogen concentration in this outer rim of the cladding can reach above 600 wppm compared to 300 wppm or less in lower burnup fuel. As the temperature of the fuel is raised in the drying process, after cask loading, much of this hydrogen will go back into solution. Later cooling of the fuel will re-precipitate the hydrogen as hydrides as the solubility limit decreases with decreasing temperature. If the applied hoop stress due to the internal rod gas pressure is sufficient, radial hydrides will form. Radial hydrides may degrade the mechanical properties of the cladding and possibly lead to rod breach during transportation and storage conditions. At the current time, this critical stress is not well defined.

To meet the performance needs in reactor, the vendors have developed a number of new alloys that have more corrosion resistance than Zircaloy at comparable burnups. The corrosion of these alloys produces less hydrogen thus making the potential for hydride generation and reorientation potentially less detrimental. A comparative database on the creep behavior, fracture toughness, and other mechanical properties must be established by the licensee to determine if these alloys fall under the same guidelines for storage and transportation as the Zircaloy alloys.

At higher burnup, the fuel pellet forms a rim region, representing about 4-8% by volume of the fuel. This rim retains fission gas under high pressure, restructures to a submicron grain size, and has higher plutonium content than the body of the pellet. Very little is known about the behavior of this rim region under impacts that might be characteristic of a severe drop or terrorist attack. Since the fuel grains are already in the respirable size range, it is important to know the relative fracture and dispersal behavior of this fine-grained material compared to behavior of grains from lower burnup fuel that are 100 times larger.

To address the effects of these changes on the safe transportation and storage of spent fuel, expansion of the current databases by the vendors and utilities is expected. The NRC has instituted a confirmatory data gathering activity to allow the staff to evaluate the adequacy of these databases when license applications for the storage and transportation of high burnup fuels are submitted.

The NRC continues to seek data and analysis methods from the nuclear industry to support the safe storage and transportation of high burnup fuel. This paper will discuss data that are needed to evaluate the storage and transportation of high burnup spent nuclear fuel, and the NRC confirmatory programs to obtain data. Furthermore, the implications of the uncertainties on pending changes for Interim Staff Guidance ISG-11 Revision 3 will also be discussed.

Perspective on Requirements for Spent Fuel Storage and Transportation

Albert Machiels
Electric Power Research Institute, Inc.

A significant regulatory milestone was achieved when the Spent Fuel Project Office published Interim Staff Guidance (ISG) 11, Revision 2 entitled: *Cladding Considerations for the Transportation and Storage of Spent Fuel.* The acceptance criteria specified in Rev. 2 were supplemented by an additional set of acceptance criteria for low burnup fuel in Rev. 3 published in November 2003.

In both revisions, the acceptance criteria address dry storage only, but not transportation. This can be considered unusual for the following two reasons:

1. Most systems to be used for high-burnup spent fuel are intended to be dual-purpose. Clearly, the acceptance criteria for loading spent fuel in such dual-purpose systems should envelop both storage and transportation applications.
2. The most limiting considerations that led to the acceptance criteria for dry storage were actually based on transportation, i.e., on limiting the potential formation of radial hydrides in the spent-fuel cladding during dry storage in order to minimize potential degradation of the cladding mechanical properties used in analyzing transportation accidents.

Presently, the U.S. regulations (Part 71) do not have specific criteria with regard to performance of cladding under hypothetical accident conditions.[1] However, the configuration of the spent fuel in the transportation package after an accident is an input to the shielding and criticality evaluation, as well as possibly to the confinement evaluation. Clearly, if it can de demonstrated that no significant damage occurs either to the spent-fuel itself (no re-configuration), or to the package (no potential for moderator ingress), the criticality analysis, generally considered as a key driver from a regulatory perspective, would be greatly simplified.

Confirmatory and new experimental work is being conducted at ANL with the participation and funding of several organizations, including the US NRC, US DOE, EPRI, and the fuel vendors. This work is expected to demonstrate the conservative, but appropriately realistic, technical basis, which resulted in the acceptance criteria for storage contained in Rev. 2/3 of ISG-11, and its applicability to transportation applications. Concurrently, modeling of spent-fuel performance under impact loading conditions is a necessary activity for both guiding the experimental work and getting the most value from its results.

These activities are, or will be, supplemented by other generic efforts:

1. Risk assessment of criticality event during transportation
2. Implementation of full-burnup credit

[1] However, Part 71 contains cladding performance criteria for normal conditions of transport.

This three-prong approach (risk assessment, fuel cladding performance, and burnup credit) is expected to support the contention that transportation risk minimization is a direct function of the reduction in the number of shipments. If this is indeed the case, the use of high-capacity packages should generally be the preferred implementation path for dual-purpose systems.

LOCA Integral Test Results for High-Burnup BWR Fuel

Yong Yan, Tatiana A. Burtseva, and Michael C. Billone

Argonne National Laboratory (ANL), Argonne, IL 60439

LOCA integral tests with high-burnup BWR fuel have been conducted at ANL to provide NRC and industry with data for assessing the applicability of the LOCA embrittlement criteria in 10 CFR50.46 to high-burnup fuel rods. Similar tests are planned for high-burnup PWR fuel. These criteria limit peak cladding temperature to 2200°F (1204°C) and maximum oxidation (equivalent cladding reacted, ECR) to 17% during high temperature steam oxidation to ensure residual ductility during ECCS quench ($T \geq 135°C$) and during possible post-LOCA seismic events ($T \approx 100°C$). Appendix K specifies the use of the Baker-Just (BJ) correlation for calculating reaction rate and ECR. NRC Regulatory Guide 1.157 (1989) allows for the use of best-estimate correlations (e.g. Cathcart-Pawel [CP]) for calculating reaction rate and ECR. At 1204°C, 17% BJ-ECR = 13% CP-ECR. In anticipation of the degrading effects of high-burnup operation on the cladding, NRC Information Notice 98-29 (1998) specifies that ECR should be based on total oxidation (corrosion plus transient steam oxidation). For PWR Zircaloy-4 (Zry-4) cladding, high-burnup operation results in peak corrosion layers of ≈100 μm, corresponding to 8-10% ECR, and peak hydrogen concentrations of 600-800 wppm. The ANL LOCA integral tests, which are conducted with fueled specimens, are designed to improve our understanding of the behavior of high-burnup fuel exposed to a LOCA transient, as well as to provide data for the assessment of the LOCA embrittlement criteria.

LOCA integral test results are reported for fueled high-burnup BWR specimens. These results are compared to baseline data for zirconia-pellet-filled, nonirradiated Zry-2 cladding specimens exposed to the same tests. Four LOCA integral tests have been conducted with specimens from Limerick BWR fuel rods at 56 GWd/MTU. In the as-discharged condition, the Limerick cladding (Zr-lined Zry-2 GE-11 9×9 design) has a corrosion layer of ≈10 μm and a hydrogen content of ≈70 wppm. The specimens were internally pressurized with helium to a gauge pressure of ≈8.3 MPa at 300°C. During heating in steam at 5°C/s, the internal pressure rose to ≤9 MPa prior to burst at ≈750°C. The full LOCA sequence (Fig. 1) calls for heating in steam at 5°C/s to 1204°C, holding for ≤5 minutes at 1204°C (≤20% CP-ECR), slow-cooling at 3°C/s to 800°C and bottom-flooding to quench the cladding from 800 to 100°C.

The ICL#1 test specimen was ramped to burst in argon and slow cooled. The ICL#2 specimen was exposed to the LOCA test sequence with the exception of quench. The nondestructive results from these tests indicated more similarities than differences between high-burnup specimens and non-irradiated specimens. The ICL#3 specimen achieved partial quench (800°C to 470°C) before failure of the quartz chamber that surrounded the specimen. The full LOCA sequence with quench was demonstrated in the ICL#4 test (see Fig. 1). Nondestructive examinations included photography and profilometry for all 4 specimens and gamma scanning for the ICL#3 and #4 specimens. Destructive examinations were performed on ICL#2 and #3 specimens to determine oxide layer thickness, fuel morphology, and axial profiles of hydrogen and oxygen concentration. Oxide-layer thickness and oxygen-content results indicate two-sided oxidation in the ballooned-and-burst region of both high-burnup and nonirradiated specimens. The axial profile of hydrogen pickup for ICL#2 and #3 specimens is shown in Fig. 2 and compared to the data for nonirradiated Zry-2 cladding. For nonirradiated specimens, the hydrogen pickup was low in the burst region and very high at 70-90 mm above and below the burst mid-plane. For high-burnup-fueled cladding, the hydrogen peak was towards the burst mid-plane. Because of the large secondary hydriding from the cladding inner surface, significant degradation of post-quench ductility (PQD) is expected for the ICL#4 ballooned region, even at 100-135°C. A 5th hot-cell integral test, with a 2-minute hold time at 1204°C, is in progress to determine if the 17% BJ-ECR criterion is sufficient to protect the ballooned region from embrittlement due to steam oxidation and hydrogen pickup at 1204°C.

In the uniform burnup region (within grid spans 2-5), the high-burnup PWR cladding for the next set of LOCA tests differs from BWR cladding in terms of corrosion layer thickness (≈40 to 100 μm) and hydrogen content (≈400 to 800 wppm). For test planning purposes, the separate effects of hydrogen on diametral-compression PQD have been investigated with prehydrided, nonirradiated 15x15 Zry-4 cladding rings after oxidation at 1204°C and quench. For as-received (≈10 wppm H) cladding that was oxidized at 1204°C, the ductile-to-brittle-transition CP-ECR was 8% at room-temperature, 12% at 100°C, and 14% at 135°C. In contrast, cladding with 400-to-800 wppm hydrogen exhibited significant embrittlement, even after moderate oxidation at 1204°C. Samples prehydrided to 400-800 wppm and oxidized at 1204°C to 8% CP-ECR exhibited no ductility. With anticipated secondary hydrogen update from the cladding inner surface, the embrittlement ECR is expected to be <<17% for high-burnup PWR specimens subjected to LOCA integral tests at 1204°C, even if the ECR is determined by the sum of the corrosion layer and the BJ-calculated transient ECR. The baseline data from prehydrided cladding are being used to plan the PWR hold-times at 1204°C such that the embrittlement ECR can be determined effectively in the ballooned and non-ballooned regions of the PWR LOCA integral test specimens.

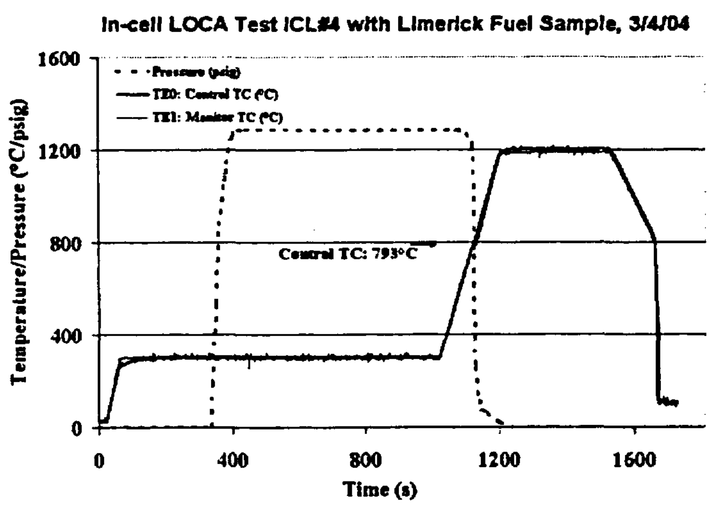

Fig 1. Temperature and pressure histories for LOCA integral test (ICL#4) with high-burnup BWR fuel.

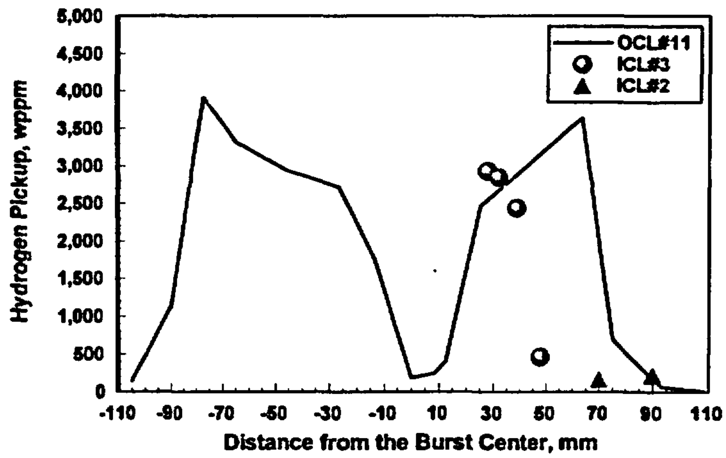

Fig 2. Axial profile of hydrogen pickup in high-burnup BWR cladding used in LOCA integral tests ICL#2 and ICL#3 (5 minutes at 1204°C); OCL#11 results are for nonirradiated Zry-2 cladding.

THIS PAGE INTENTIONALLY BLANK

LOCA Testing at Halden

E. Kolstad, W. Wiesenack, V. Grismanovs
(OECD Halden Reactor Project)

The safety criteria for loss-of-coolant accidents are defined to ensure that the core will remain coolable. Since the time of LOCA experiments in the '70s, which were largely conducted with fresh fuel, changes in fuel design, the introduction of new cladding materials and in particular the move to high burnup have generated a need to re-examine these criteria and to verify their continued validity. Hot cell programs concentrating on embrittlement and mechanical properties of high burnup cladding have been initiated in some countries.

The Halden reactor is suited for integral in-pile tests on fuel behavior under LOCA conditions. It is intended to utilize fuel rods irradiated in commercial reactors to burnup levels >50 MWd/kg with a thorough characterization regarding the state of the cladding and the bonding with the fuel. Participating organizations have supplied both PWR and BWR fuel with desired characteristics. It is the intention to include medium burnup (40-45 MWd/kg) fuel in the test series in order to assess the difference between medium burnup and very high burnup fuel (>60 MWd/kg.

The Halden experiments are single pin tests and will focus on effects that are different from those studied in out-of-reactor tests. A prototypical bounding LOCA transient does not exist, and it was recommended that the test conditions be selected to meet the following primary objectives:

- to maximize the balloon size to promote fuel relocation, and to evaluate its possible effect on cladding temperature and oxidation, and
- to investigate the extent (if any) of - "secondary transient hydriding" - on the inner side of the cladding around the burst region.

Target peak clad temperatures (PCT) for the pre-irradiated rods have been set at 800°C and 1100°C for high and medium burnups.

The first LOCA trial runs were carried out in the Halden reactor in May 2003, using a fresh, unpressurized PWR rod with Zr-4 cladding. The main objective was practicing, to determine how to run the later experiments with pre-irradiated segments. PCTs in the range 800°C - 1100°C for the initial six transients were successfully achieved.

The rig with the fuel rod was located in a high-pressure flask connected to an ex-reactor high-pressure loop incorporating a blow-down system. The cladding temperature transients can be controlled by rod power and an annular heater surrounding the rod. A spray system at the top of the rig is used to supply steam for the oxidation and hydriding processes. The extensive rig/rod instrumentation enables power calibration and neutron flux monitoring, and includes a fuel centre thermocouple (first rod only), three cladding thermocouples (at two elevations), rod pressure sensor, cladding extensometer, heater thermocouples etc. The geometry of the test section is shown in Fig. 1.

The second trial LOCA test run was successfully carried out on 28 May this year. The test was performed with a fresh, pressurized PWR rod and consisted of a blowdown phase, heat-up, hold at target PCT and termination by reactor scram (Fig. 2). The main objective was to achieve ballooning and cladding failure to find out how to run later experiments with pre-irradiated rodlets.

The target cladding temperature of 1050°C was achieved, and rod rupture occurred at 800°C, as evidenced by rod pressure and elongation measurements (Fig. 3) as well as the gamma monitor on the blowdown line to the dump tank. The hold time above 900°C was 390 s and the average temperature increase rate between 600 and 800°C was ~7°C/s. The azimuthal temperature variation was small prior to cladding failure, within ± 2-3°C, and the tensile hoop stress ~55 MPa. The spray was applied intermittently during the high temperature period and the test was terminated by a reactor scram. The rod with its capsule will undergo gamma-scanning at Halden before it is shipped to Kjeller hotcells for detailed PIE.

Pre-test calculations were carried out by VTT using the FRAPTRAN/GENFLO code. The code predicted the maximum cladding temperature with good accuracy. Also the timing and temperature of the rod failure was well predicted. Further calculations will be performed in preparation of the next test (the first with a pre-irradiated PWR segment).

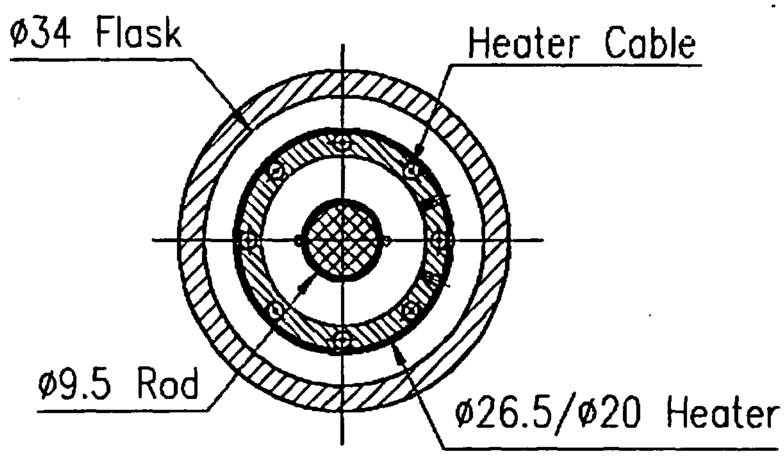

Fig. 1. The geometry of the test section

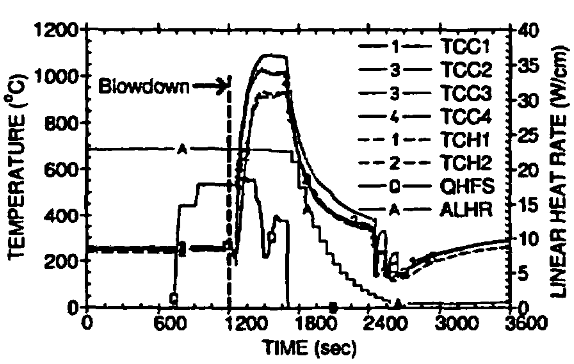

Fig. 2. Cladding, heater and loop inlet and outlet temperatures during the LOCA test. Also fuel and heater power is shown. Nomenclature: A = fuel power, Q = heater power, I = inlet loop temperature, dashed 1-2 = heater temperature

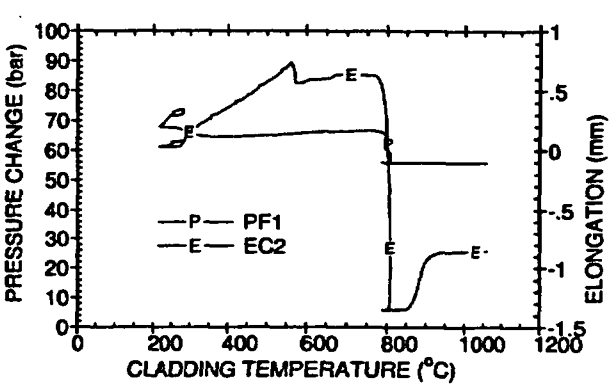

Fig. 3. Measurements of rod pressure and cladding elongation as function cladding temperature during the LOCA test (Rod rupture at 800°C)

THIS PAGE INTENTIONALLY BLANK

Results from Studies on High Burn-Up Fuel Behavior Under LOCA Conditions

Fumihisa NAGASE and Toyoshi FUKETA

Department of Reactor Safety Research
Japan Atomic Energy Research Institute

To promote a better understanding of high burnup fuel behavior under loss-of-coolant accidents (LOCAs), a research program is being conducted at the Japan Atomic Energy Research Institute (JAERI). The program consists of integral thermal shock test and other separate tests for oxidation rate and mechanical property of fuel claddings.

1. Integral thermal shock test

In the test, short rods are heated up, burst, oxidized in steam and quenched by flooding water in order to evaluate fracture-bearing capability of oxidized fuel claddings under the simulated LOCA condition. The Japanese LOCA criteria for cladding embrittlement are based on results from the integral thermal shock tests. Tests were performed with non-irradiated cladding tubes, which were mechanically thinned and pre-hydrided to evaluate the separate effects. Claddings fractured into two pieces during the quench, depending primarily on the oxidation. The threshold of the fracture in terms of the oxidation decreases with the higher concentration of initially absorbed hydrogen and with the larger load in axial constraint.

Two PWR fuel rods, irradiated to 39 and 44GWd/t (rod average) at Takahama unit-3 reactor, are subjected subsequently to the tests. The cladding material is low-Sn (1.3%Sn) Zircaloy-4 and thickness of oxide layer formed in the reactor ranged 18 to 25 μm. Before the tests, fuel pellets were removed from 190mm-long segments, and alumina dummy pellets were loaded in the defueled claddings. Zircaloy end-plugs were welded at the both ends of the claddings and the fabricated test rods were pressurized to about 5MPa with argon gas. Six test rods were quenched after rupture at about 800 deg C and isothermal oxidation at 1030 to 1192 deg C. Two claddings which were oxidized to about 26 to 30%ECR* fractured during quench. The fracture of the irradiated claddings agrees with the failure criteria for non-irradiated claddings containing similar hydrogen concentrations. Four claddings oxidized to about 16 and 25%ECR survived the quench. These indicate that fracture/no-fracture threshold is not reduced so significantly by irradiation to the examined burnup level.

2. Mechanical tests of oxidized and quenched cladding

Besides the integral thermal shock tests, mechanical tests are performed to develop methodology for predicting cladding fracture on quenching as well as to examine mechanism of cladding embrittlement.

Ring-tensile and ring-compression tests were performed on non-irradiated Zircaloy-4 claddings which were pre-hydrided to 400 and 800 ppm, oxidized at 1000 to 1250 deg C, and finally quenched.

- Ductility reduction observed in the ring-tensile tests was not remarkable for the oxidation between 10 and 20%. Uniform tensile stress in the circumferential direction is applied to the cladding in the ring tensile test, and this stress state is quite different from that is applied during quench. This suggests that test methods should be carefully selected in order to estimate cladding embrittlement under LOCA condition.

- The ring-compression tests detected the sudden ductility drop above 15% oxidation for claddings which were oxidized at 1200 deg C without hydriding. The significant ductility reduction occurred at lower oxidation level in the pre-hydride claddings (400 and 800 ppm). This indicates that pre-hydriding enhances cladding embrittlement of oxidized cladding and agrees with the results of the integral thermal shock tests.
- It has been generally considered that slow cooling after high-temperature oxidation enhances oxygen diffusion from oxide layer into metallic prior-β phase, and consequently microstructure and ductility of metallic prior-β phase changes depending on the cooling rate. To confirm that, ring compression tests were performed with claddings which were cooled at different rates after oxidation at 1100 and 1200 deg C. It is shown that the influence of slow cooling differs depending on oxidation temperature and oxidation amount. The influence becomes greater in the cladding oxidized at 1200 deg C and at the lower oxidation level.

Acknowledgment

The integral thermal shock test with irradiated PWR fuel claddings has been performed as a corporative research program between JAERI and Japanese PWR utilities.

* ECR is estimated by the Baker-Just equation, taking account of double sided oxidation and wall thinning by ballooning. The nitial cladding thickness used in the estimation is metallic thickness after corrosion during the reactor operation.

Realistic Assessment of Fuel Rod Behavior Under Large-Break LOCA Conditions

M. E. Nissley, C. Frepoli, K. Ohkawa
Westinghouse Electric Company

Fuel clad swelling and rupture can occur during a loss of coolant accident (LOCA), depending on the core heatup transient and the pressure differential across the cladding. Clad rupture will lead to release of fission products from the fuel, and double-sided metal-water reaction (oxidation) within the ballooned region. In order to simplify the radiological dose calculations, it is typically assumed that 100% of the rods in the core fail. However, it is instructive to consider what a realistic failure fraction might be under more representative conditions. The objective of this study will be to assess the extent of failure and the consequences for the cladding oxidation for the large break LOCA scenario, with and without detailed treatment of uncertainties.

The first assessment will use a deterministic calculation of a large break LOCA under normal operating (baseload) conditions, using the realistic computer program WCOBRA/TRAC. In order to ensure some cladding rupture, a full train of ECCS will be assumed lost (worst single failure) and bounding rod power conditions in the lead fuel assembly will be used. Estimates of the extent of rupture throughout the core will be made by considering peak cladding temperature dependence on rod power, rupture temperature as a function of cladding pressure differential, burnup effects on rod internal pressure, and a core-wide census of rod power and burnup. A comparison of the maximum local oxidation within and away from the ballooned region will also be made.

The second assessment will use the results from a best-estimate plus uncertainties analysis of a large break LOCA, performed using methods consistent with US design basis LOCA regulatory requirements. Uncertainties in thermal-hydraulic models, plant operating conditions, and fuel rod models are accounted for in this method by simultaneously sampling from the uncertainty distributions of each parameter for each transient case. The plant operating conditions considered in the uncertainty analysis include transient power distributions, such that more severe axial shapes and higher linear heat rates are considered than in the first assessment. The extent of rupture within the uncertainty cases will be examined, and conclusions drawn relative to the threshold for rupture. Maximum local oxidation within and away from the ballooned region will be reviewed for the most limiting cases, and those results will be assessed for their dependence on the related fuel rod uncertainty parameters (burst strain, degree of fuel relocation, etc.).

The information presented in these assessments should be interpreted as illustrative and representative. Extent of rupture and degree of oxidation are highly dependent on the transient conditions, which are highly dependent on plant-specific parameters such as core power, nuclear peaking factors, ECCS capacity and other factors.

THIS PAGE INTENTIONALLY BLANK

Post-Quench Ductility of Advance Alloy Cladding

Michael C. Billone, Yong Yan, and Tatiana A. Burtseva

Argonne National Laboratory (ANL), Argonne, IL 60439

Diametral (ring)-compression screening tests have been conducted to assess the ductility of 17×17 Zry-4, ZIRLO and M5 samples oxidized to 0-20% ECR at 1000°C, 1100°C and 1200°C. The 25-mm-long samples were exposed individually to two-sided steam oxidation in the same test apparatus for the same test times, slow cooled to 800°C and water-quenched. Test times were calculated using the Cathcart-Pawel weight-gain correlation and a reference wall thickness of 0.57-mm (Zry-4 and ZIRLO). Based on sample weight increase (normalized to the surface area), weight gain was determined and compared to Cathcart-Pawel (CP) predictions. As expected, good agreement was achieved among Zry-4, ZIRLO and M5 and the CP predictions for the 1100°C- and 1200°C-oxidized samples, while differences in weight gain vs. time were observed for the alloys oxidized at 1000°C. After ≈3400 s at 1000°C, the weight gains of M5 and ZIRLO were ≈36% and ≈20% less than Zry-4, respectively. For lower test times, the M5 weight gain was consistently lower than Zry-4, while the ZIRLO and Zry-4 weight gains were about the same. The experimental weight gains, along with the sample thickness (0.61 mm for M5), were used to determine experimental ECR values.

Similar tests were performed with E110 tubing (0.71-mm) at 1000°C and 1100°C to characterize the onset of breakaway oxidation, subsequent hydrogen pickup, and decrease of post-oxidation ductility. Weight gain for as-received E110 could not be determined accurately because of the early (<300 s at 1000°C) breakaway oxidation resulting in oxide flaking and spalling. However, polished and/or machined-and-polished (0.58-0.69 mm wall) E110 exhibited stable oxide growth for oxidation times up to ≈300 s at 1000°C and >1000 s at 1100°C. For these samples, the E110 weight gains were similar to M5: lower than Zry-4 at 1000°C and about the same as Zry-4 at 1100°C.

Ring-compression samples (8-mm-long) were cut from the oxidized samples and tested initially at room temperature and 0.033 mm/s displacement rate (0.35%/s diametral strain rate). Load-displacement curves were analyzed by the traditional offset-displacement method to determine plastic ductility. It was found that this method over-predicts plastic displacement, determined directly from pre- and post-test diameter measurements along the loading direction, by ≤0.2 mm (2% strain). For rings with offset strains < 3%, direct measurement of post-test diameter after the first through-wall crack proved to be a more reliable measure of ductility. Rings with permanent strains < 1% were classified as brittle.

Zry-4, ZIRLO and M5 exhibited ductile behavior (offset strains ≥ 3% and/or permanent strains ≥ 1%) after oxidation at 1000°C and 1100°C for CP-calculated ECR ≥ 17%. The 1000°C results are interesting in that the all three alloys exhibit ≈3% offset-strain ductility after oxidation for the same test time (≈3400 s) at 1000°C, even though the measured ECR values were 22.4%, 18.0% and 13.3% for Zry-4, ZIRLO and M5, respectively. These results suggest that oxidation time at 1000°C and CP-calculated ECR correlate better with ductility than ECR based on actual weight gain, especially for M5. For as-received E110, embrittlement occurs after oxidation at 1000°C for ≈625 s, corresponding to an average hydrogen pickup of ≈300 wppm. The hydrogen concentration in this oxidized sample was highly non-uniform (25-560 wppm) in the circumferential and axial directions with the high hydrogen concentrations occurring under local areas of breakaway oxidation. The results suggest that hydrogen entering E110 through cracks in the oxide layer is essentially "frozen" in position during the course of the test. For polished and machined-and-polished E110 oxidized at 1100°C for ≤1011 s, the material was ductile up to hydrogen concentrations occurring under local areas of breakaway oxidation. The results suggest that hydrogen entering E110 through cracks in the oxide layer is essentially "frozen" in position during the course of the test. For polished and machined-and-polished E110 oxidized at 1100°C for ≤1011 s, the material was ductile up to 19% CP-ECR based on the machined-and-polished wall thickness of 0.58 mm. While surface polishing was found to stabilize oxide growth on E110 surfaces, pre-etching with solutions containing HF tended to de-stabilize oxide growth at earlier test times than observed for as-received E110 tubing.

For Zry-4, ZIRLO, M5 and E110 (RRC-KI/RIAR data) samples oxidized at 1200°C, the room-temperature offset strains decreased rather abruptly from 5 to 10% ECR. Based on interpolation of the permanent strain data, the embrittlement ECR values at room-temperature were: ≈10% for Zry-4 and ≈12% for ZIRLO and M5. Based on the RRC-KI/RIAR offset strain data, the embrittlement ECR for E110 oxidized at 1200°C was ≈8%.

Zry-4, ZIRLO and M5 alloys oxidized at 1200°C were retested at 135°C and 0.35%/s, as well as 3.5%/s for Zry-4. The enhancement in post-quench ductility with test temperature was remarkable. As shown in Figs. 1 and 2, ZIRLO and M5 (extrapolated) retained significant ductility for measured and CP-calculated ECR values >17%. Zry-4 also maintained post-quench ductility for ECR > 17% under these conditions. At the higher strain rate of 3.5%/s, Zry-4 was also ductile for ECR > 17%. The implication of these results is that as-fabricated Zry-4, ZIRLO and M5 satisfy the LOCA embrittlement criteria during and shortly after quench. Although testing of the 17x17 cladding alloys at 100°C has not yet been conducted, the results from testing of 15x15 Zry-4 at RT, 100°C and 135°C (Y. Yan, this meeting) suggest that these alloys will retain ductility at the longer-term post-quench temperatures of ≈100°C.

However, in-reactor corrosion results in hydrogen pickup. This hydrogen can enhance embrittlement directly, as well as indirectly by increasing the oxygen solubility and embrittlement in the prior-beta layer of the post-quench cladding alloy. Preliminary test results at 135°C with prehydrided 17x17 and 15x15 (Y. Yan, this meeting) Zry-4 samples oxidized at 1200°C indicate that ≈5%-ECR Zry-4 embrittles at ≈600 wppm H, that ≈8-9%-ECR Zry-4 embrittles at ≈350 wppm H and ≈11%-ECR Zry-4 embrittles at ≈300 wppm H. Future work will focus on the post-quench ductility at 100-135°C of prehydrided-nonirradiated ZIRLO and M5 and high-burnup Zry-4, ZIRLO and M5.

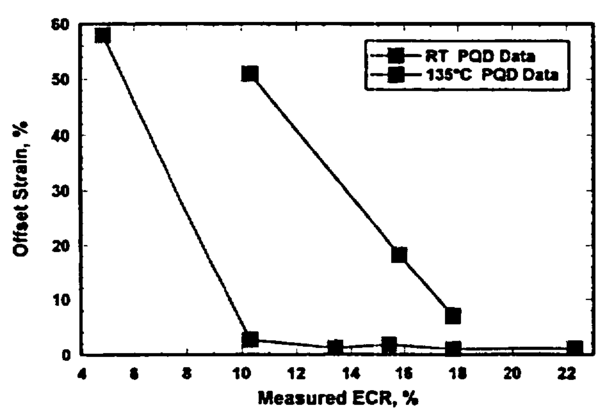

Fig. 1. Post-quench ductility of ZIRLO oxidized at 1200°C and ring-compressed at RT and 135°C.

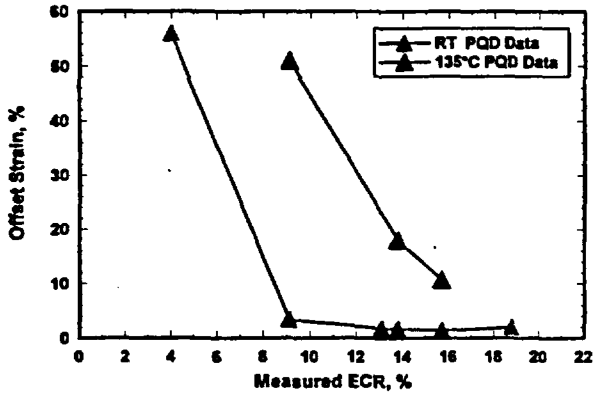

Fig. 2. Post-quench ductility of M5 oxidized at 1200°C and ring-compressed at RT and 135°C.

THIS PAGE INTENTIONALLY BLANK

Overview of the CEA Data on the Influence of Hydrogen on the Metallurgical and Thermal-Mechanical Behavior of Zircaloy-4 and M5™ Alloys Under LOCA Conditions

J.C. Brachet(*)(1), L.. Portier(1), V. Maillot(1), T. Forgeron(2), J.P. Mardon (3), P. Jacques(4), A. Lesbros(4)

(1) CEA-DEN, DMN/SRMA, CEA/Saclay, 91191 Gif-sur-Yvette Cedex, France
(2) CEA-DSNI-Comb., CEA/Saclay, 91191 Gif-sur-Yvette Cedex, France
(3) FRAMATOME ANP Nuclear Fuel, 10 rue Juliette Récamier, 69456 Lyon Cedex 06, France
(4) EDF, SEPTEN, 12-14 avenue Dutrievoz, 69628 Villeurbanne Cedex, France
(*) corresponding author : jean-christophe.brachet@cea.fr

A few years ago, within the framework of the CEA/EDF/Framatome-ANP R&D cooperative program, we made the assumption that the burn-up influence on the thermal-mechanical behavior of the fuel cladding tubes under LOCA conditions should be strongly linked to the hydrogen up-take due to the in-service oxidation (see for example discussion pp. 276-277 in [1]). Thus, since that time, an extensive experimental program has been conducted in CEA labs on as-received and pre-hydrided Zy-4 and M5™ advanced alloys of Framatome-ANP to get a better insight into the influence of the hydrogen on the thermal-mechanical cladding behavior during the first phase of the LOCA transient (ballooning and rupture) and for post-quenched conditions (residual ductility/toughness...) [2] [3] [4].

On the one hand, one of the main assumptions here was that the microstructural defects, and the resultant hardening produced under heavy neutron irradiation within the zirconium matrix, are annealed early upon the first phase of the LOCA transient (i.e. first thermal ramp) and thus, that the main effects of high burn-up should come from the hydrogen uptake. To assess this hypothesis, specific thermal-mechanical tests have been performed on virgin, pre-hydrided and irradiated cladding tubes. This confirmed that the effect of hydrogen uptake dominates over that of irradiation on the thermal-mechanical response of the materials.
So, in a first part of the presentation, we will briefly summarize the main results obtained here and, from the metallurgical point of view, we will illustrate the strong influence of hydrogen on the decrease of the alpha-to-beta phase transformation temperatures of the zirconium alloys studied.

On the other hand, studies have been performed on the post-quench mechanical behavior of as-received and pre-hydrided cladding tubes after single-face oxidation at 1000-1200°C and quenching. In parallel with these mechanical tests, in-depth metallurgical investigations have been developed [5], to be able to quantify the resultant phase thickness (that is, ZrO_2, Alpha(O) and Ex-Beta phase layers) and their specific chemical composition - especially their oxygen content which is known to influence strongly the residual mechanical properties. Also, fractograph analysis has been applied on failed samples to get a better knowledge of the failure

mechanism as a function of the materials and of the hydrogen concentration, for different oxidation conditions.
So, in the second and major part of the presentation, we will focus on LOCA post-quenched behavior of as-received and prehydrided Zy-4 and M5™ cladding tubes for typical hydrogen contents ranging from ~200 up to ~600 wt-ppm depending on the alloy. Ring compression, impact, and bending tests at Room Temperature have been performed for different oxidation conditions. The mechanical results will be presented and briefly discussed, taking into account the metallurgical analysis (resultant phase morphology and thickness, chemical composition – oxygen contents, failure mode,...).

References

[1] T. FORGERON, J.C. BRACHET, et al., « Experiment and modelling of advanced fuel rod behaviour under LOCA conditions : $\alpha \Leftrightarrow \beta$ phase transformation kinetics and EDGAR methodology », *Zirconium in the Nuclear Industry: 12th. Int. Symposium, June 1998, Toronto, Canada*, ASTM STP 1354, (2000), pp. 256-278

[2] J.C. BRACHET, L. PORTIER, et al., "Influence of hydrogen content on the $\alpha \Leftrightarrow \beta$ phase transformation temperatures and on the thermal-mechanical behavior of Zy-4, M4 (ZrSnFeV) and M5™ (ZrNbO) alloys during the first phase of LOCA transient," *Zirconium in the Nuclear Industry: 13th. Int. Symposium, June 10-14 2001, Annecy, France,* ASTM STP 1423, (2002), pp. 673-701

[3] L. PORTIER, T. BREDEL, J.C. BRACHET, V. MAILLOT, J.P. MARDON, A. LESBROS, "Influence of Long Service Exposures on the Thermal-Mechanical Behavior of Zy-4 and M5™ Alloys in LOCA Conditions", *Zirconium in the Nuclear Industry: 14th. Int. Symposium, June 13-17 2004, Stockholm, Sweden,*, to be published in ASTM-STP

[4] J.-C. BRACHET, J. PELCHAT, D. HAMON, R. MAURY, P. JACQUES, J.-P. MARDON, « Mechanical behavior at Room Temperature and Metallurgical study of Low-Tin Zy-4 and M5™ (Zr-NbO) alloys after oxidation at 1100°C and quenching", *Proceeding of TCM on "Fuel behavior under transient and LOCA conditions", Sept. 10-14, 2001, organised by IAEA, Halden, Norway*

[5] J.C. BRACHET et al., "Quantification of the α(O) and Prior-β phase fractions and their oxygen contents in high temperature (HT) oxidised Zr alloys (Zy-4, M5™)", *Proceeding of "SEGFSM Topical Meeting on LOCA", 25-27th. of May 2004, ANL, Chicago, USA*

Insights on PRA Quality from the SPAR Model Development Program

Patrick O'Reilly, Michael Cheok
U.S. Nuclear Regulatory Commission

The purpose of the NRC's Standardized Plant Analysis Risk (SPAR) model development program is to provide NRC staff with readily available and easy-to-use analytical tools for performing plant-specific risk assessments. The Office of Research has completed SPAR models for all 72 plant sites for internal initiating events at full power operation. In addition, models for the low power/shutdown modes of operation, for external initiating events, and for calculating large early release frequencies are being developed. SPAR models are used to evaluate risk significance of inspection findings as part of the Reactor Oversight Process, to evaluate risk associated with operating events as part of the Accident Sequence Precursor program, to perform analyses in support of generic/safety issue resolution, to perform analyses in support of the staff's risk-informed review of license amendments, and to independently verify performance indicators as part of the Mitigating Systems Performance Index (MSPI).

Modeling assumptions and methodology in the SPAR models are "standardized" for consistency and ability to understand differences in risk results. The initial Rev 3 internal-events SPAR models has been completed and benchmarked against licensee PRAs. This paper would provide the results of the comparison, discuss major factors that influence differences in risk (CDF) results and discuss some insights and possible implications with regard to the goal of achieving PRA quality.

THIS PAGE INTENTIONALLY BLANK

Human Reliability Analysis (HRA) Good Practices

Erasmia Lois[1], Alan Kolaczkowski,[2] John Forester[3]

[1]U.S. Nuclear Regulatory Commission, Washington, DC, USA
[2]Science Applications International Corp., Eugene, OR, USA
[3]Sandia National Laboratories, Albuquerque, NM, USA

The U.S. Nuclear Regulatory Commission is establishing "good practices" for human reliability Analysis (HRA), documented in NUREG-1792, *Good Practices for Implementing Humam Reliability Analysis (HRA), Draft Report for Comment,* July 2004. NUREG-1792 provides a reference guide to HRA practitioners for performing HRAs (whether for the first time or when analyzing a change to current plant practices) and for reviewing HRAs to assess the quality of an analysis.

The HRA good practices are developed as part of the NRC's initiatives to address probabilistic risk assessment (PRA) quality issues and supports the implementation of Regulatory Guide (RG) 1.200 entitled: "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results For Risk-Informed Activities." Its elements are directly linked to RG 1.200 which reflects and endorses, with certain clarifications and substitutions, the American Society of Mechanical Engineers (ASME) Standard RA-S-2002, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," and Revision A3 of the Nuclear Energy Institute (NEI) "Probabilistic Risk (PRA) Peer Review Process Guidance," (NEI-00-02).

The HRA good practices reflect NRC and contractor experience from developing HRA methods (e.g., THERP and ATHEANA), performing HRAs (e.g., NUREG-1150), and reviewing HRAs, particularly the Individual Plant Examinations. "Good Practices" means those processes and individual analysis tasks and judgments that would be expected of a HRA (considering current knowledge and state-of-the-art) in order for the HRA results to sufficiently represent the anticipated operator performance when making risk-informed decisions. The document is principally focused on the process for performing HRAs. Therefore, are of generic nature, that is, are not tied to any specific methods or tools that could be employed to perform an HRA. They are written in the context of a risk assessment for commercial nuclear power plant operations occurring nominally at full power, however, many are also applicable to low power and shutdown operations. Similarly, are addressing internal initiating event analysis, however, should generally also be appropriate to external initiating events.

THIS PAGE INTENTIONALLY BLANK

The Joint U.S. NRC/EPRI Fire Risk Requantification Study

J. S. Hyslop

U.S. Nuclear Regulatory Commission

Fire risk analysis (FRA) methods have been used in the Individual Plant Examinations of External Event (IPEEE) program to facilitate a nuclear power plant examination for vulnerabilities. However, in order to make finer, more realistic decisions for risk-informed regulation, FRA methods need to be improved. Licensee applications and U.S. Nuclear Regulatory Commission (NRC) review guidance with respect to many regulatory activities such as the risk-informed, performance-based fire protection rulemaking (endorsing National Fire Protection Association Standard 805)[2] will benefit from more robust methods. In order to address the need for improved methods, the NRC Office of Nuclear Regulatory Research (RES) and Electric Power Research Institute (EPRI) are participating in a program to develop state-of-art methods in FRA.

THIS PAGE INTENTIONALLY BLANK

Phased Approach to Achieving an Appropriate Quality for PRAs for Regulatory Decision Making

Gareth W. Parry
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission

The Commission, in Staff Requirements Memorandum (SRM) COMNJD-03-0002, "Stabilizing the PRA Quality Expectations and Requirements", introduced the concept of a four phase approach to achieving an appropriate quality for PRAs used in regulatory decision making., This phased approach is needed because not all the guidance documents defining PRA quality are available for all the risk contributors. Currently, the staff guidance on PRA quality in RG 1.200 addresses only internal events at full power, and includes a limited level 2 PRA (LERF) analysis. Guidance for external initiating events, internal fires, and low power and shutdown modes of operation will follow development of PRA standards for these risk contributors. The approach lays out a path, in a phased manner, for how risk-informed applications can be implemented while the guidance documents needed to define PRA quality for all risk contributors are developed. In the SRM, the Commission directed the staff to develop an action plan that would define a practical strategy for its implementation. In addition, the SRM directed the staff to discuss the resolution of technical issues, such as model uncertainty, treatment of seismic and other external events, and human performance issues.

An implementation plan has been developed jointly by the Offices of Nuclear Reactor Regulation (NRR) and Nuclear Regulatory Research (RES). In this plan, PRA quality is measured, as in Regulatory Guide (RG) 1.174 and RG 1.200, in terms of its appropriateness with respect to scope in terms of risk contributors as discussed above, level of detail, and technical adequacy. Inherent in this definition is that a PRA of sufficient quality to support an application need only have the scope and level of detail sufficient to support that application, but it must always be technically adequate.

The plan covers the first three phases defined in the SRM, and identifies the activities required to support implementation. As directed by the SRM, the feasibility of the fourth phase will be assessed following achievement of Phase 3. The phases are achieved for specific risk-informed activities when guidance documents are available to support those activities, and in particular to address the issue of the quality of PRA necessary to support the activities.

Phase 1 represents the current situation, where guidance on PRA quality is general, and staff review of the base PRA supporting the activity is performed on a case-by-case basis. Phase 2 takes advantage of the work that has been performed to develop PRA standards. Phase 2 occurs when there are PRA standards and the associated regulatory guides in place to address those PRA scope items that are significant to the decision. To be in Phase 2 for an application, the licensee's submittal is expected to be in conformance with the published standards. The PRA standards are being developed on different schedules. As a result, the risk-informed activities will transition to Phase 2 on different schedules according to which scope items are significant to the decision. Phase 3 provides a regulatory framework for the development of a PRA that will be of sufficient quality to support all current and anticipated applications.

In implementing the phased approach, the plan calls for specific application types to be defined, and the necessary guidance documents identified. Additional guidance documents will be developed on a schedule that is a function of when the standards are developed. The implementation schedule has a built-in grace period to allow licensees to implement the new guidance. To implement the phased approach, a process will be developed for prioritizing and scheduling submittal reviews. This prioritization process is necessary to balance the need to use staff resources effectively and efficiently and the need to provide incentives for licensees to develop more complete PRA models. The staff plans on working closely with industry in the development of the guidance documents, and will develop the necessary standards not developed by a Standards Developing Organization (e.g. ASME, ANS).In addition to providing a strategy for the implementation of the phased approach, the plan addresses the identification and resolution of technical issues (e.g., model uncertainty, the treatment of seismic and other external events, and human performance).

Guidance on the Treatment of Uncertainties and Alternative Approaches Used in PRAs

Mary Drouin
Probabilistic Risk Analysis Branch
Division of Risk Analysis and Applications
Office of Nuclear Reactor Research

The NRC is extensively using information from probabilistic risk assessments (PRAs) in its regulatory decision making. To streamline staff review of licensee applications using risk insights, professional societies and the industry undertook the following initiatives to establish consensus standards and guidance on the use of PRA in regulatory decision making:

- The American Society of Mechanical Engineers (ASME) has developed a standard for a Level 1 analyses (i.e., estimation of core damage frequency (CDF)) and a limited Level 2 analysis (i.e., estimation of large early release (LERF)) covering internal events (transients, loss of coolant accidents, and internal flood) at full power.

- The Nuclear Energy Institute (NEI) has developed a "PSA Peer Review Guidance," (NEI-00-02) covering internal events at full power--Level 1 and simplified Level 2.

- The American Nuclear Society (ANS) has developed PRA standards for external hazards (December 2003) and is developing PRA standards for:
  — low power and shutdown with a tentative publication date of December 2004
  — internal fires with a tentative publication date of December 2004

It is expected that licensees will use the PRA standards and industry guidance to help demonstrate and document the adequacy of their PRAs for a variety of risk-informed regulatory applications. Therefore, the staff should document its position on the adequacy of the standards and industry guidance to support regulatory applications. Such documentation will indicate in which areas staff review can be minimized and where additional review may be expected. To accomplish this, the staff has developed Regulatory Guide 1.200 to provide an approach for assessing the adequacy of PRA results used in support of regulatory applications and an accompanying Standard Review Plan (SRP) chapter.

A major challenge in development of PRA standards is identification and resolution of the technical issues. One important issue is that of model uncertainty. For example, there are elements of the PRA where there is uncertainty regarding the appropriate model to use (e.g., human reliability analysis). When a consensus cannot be reached as to whether there is a clearly preferred approach, a decision must be made taking into account the impact of adopting different models. Another challenge in the development of PRA standards is that the analyst may use alternative approaches than those provided in the standard to account for scope items or technical requirements that are not addressed in its PRA.

Guidance is not available on acceptable approaches of how to address, in detail, uncertainties in the decision making process, nor is their guidance on the acceptability of alternative approaches. The staff has initiated efforts to develop guidance on:

- Acceptable approaches to supplement a PRA that is not of a full scope, or has deficiencies in some elements which includes the appropriate use of bounding analyses, screening methods, or qualitative approaches
- Identification and performance of sensitivity studies
- How to use the results from the "uncertainty analyses" in the decision making process, including the role and definition of defense-in-depth.

More Robust Research Plan for Radiation Protection

Stephanie Bush-Goddard, Ph.D.

Radiation Protection, Environmental Risk,
and Waste Management Branch
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

The Nuclear Regulatory Commission's (NRC's) standards for protection against ionizing radiation, 10CFR Part 20, continue to serve as a coherent and logical framework for the variety of radiation sources and exposures under NRC jurisdiction. However, in 2003, the Commission directed the NRC staff to develop a more robust materials research program in the Office of Research that would take into account developments in environmental and individual radiation protection may have significant implications for NRC's regulatory framework. In addition, revisions in dosimetry results stemming from the Hiroshima/Nagasaki assessments being undertaken by the Radiation Effects Research Foundation, and to be addressed in the upcoming report by the National Academy of Sciences Committee on Biological Effects of Ionizing Radiation, will warrant an assessment of the potential impact on our current regulations

As a result, a research plan was developed and provided to the Commission in February, 2004, outlining a research program plan addressing radiation protection and health effects issues. The Commission paper containing this research plan can be found in ADAMS under accession number ml040360736. This paper will discuss the key components and goals of this research program.

To summarize, the goals of the research plan are as follows:

Goal No. 1: Maintain and Improve NRC's Knowledge of Radiation Health Effects

Evaluate data and information on health effects attributed to radiation exposure in order to serve as a technical information resource. Ensure that NRC has the most up-to-date information in order to make realistic estimates of radiation health effects.

Goal No. 2: Support Development of Radiation Protection Standards and Implementation

Develop and enhance methodologies for radiation detection, measurement, monitoring, and dose assessment to provide technical information with greater accuracy and less uncertainty in order to enhance the effectiveness and efficiency of regulatory decisions. Promote consistency in the development of radiation protection standards by evaluating the impact and relevance of new technical and scientific developments on the NRC regulatory program and actively participating in the development of radiation protection recommendations to ensure that they are based on sound and logical scientific principles.

Goal No. 3: Support Radiation Protection Rationales and Technical Bases

Provide models that support realistic decision making in radiation protection. The NRC needs up-to-date information to accurately assess the impact on public health and safety in developing new regulations and reviewing licensed activities.

Goal No. 4: Develop Technical Basis for Risk-Informing Materials Applications through Analysis of Operational Experience

Develop information to support regulatory decision-making on risks associated with the use of radioactive materials. By accurately assessing the risks associated with expected or accidental exposures to radioactive sources, the staff will be better able to risk-inform NRC regulations on byproduct and source material.

2005 Recommendations of the ICRP

E. V. Holahan, Ph.D.

Radiation Protection, Environmental Risk,
and Waste Management Branch
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

The International Commission on Radiological Protection (ICRP) was established to advance for the public benefit the science of radiological protection, in particular by providing recommendations and guidance on all aspects of radiation protection. The aim of the recommendations is to provide an appropriate standard of protection for mankind from sources of ionizing radiation, without unduly limiting beneficial practices that give rise to exposure to radiation. In preparing its recommendations, the ICRP considers the fundamental principles and quantitative bases on which appropriate radiation protection measures can be established, while leaving to various national protection bodies the responsibility of formulating the specific advice, codes of practice, or regulations that are best suited to the needs of their individual countries.

The ICRP regularly examines the status of its recommendations and reviews the increasing knowledge of the effects of exposure to ionizing radiation to decide whether new recommendations are needed. Since the publication of the last set of recommendations in 1990, the ICRP has published ten reports that provide additional guidance for the control of radiation exposures. The draft 2005 Recommendations are intended to consolidate into a single document and within a unified scheme a number of recommendations that have been made in individual publications since the 1990 Recommendations, Publication 60, was issued.

The full text of the ICRP 2005 Recommendations is available for review on the ICRP website. The Main Commission has stated that the 2005 Recommendations are not intended to change the fundamental basis for radiological protection nor are they intended to prompt significant changes other than updating the Interagency and European Basic Safety Standards. The draft recommendations maintain the public and occupational dose limits originally recommended in 1990, update tissue and radiation weighting factors, recommend dose constraints from single sources, emphasize that patient dose should be commensurate with the clinical benefit, and include a policy on radiological protection of non-human species.

A presentation of the 2005 ICRP Recommendations will highlight the evolution that has occurred relative to recommendations previously issued by ICRP in Publications 26 (1977) and 60 (1990) and the Standards for Radiation Protection as contained in 10 CFR Part 20.

THIS PAGE INTENTIONALLY BLANK

Challenges in Radiation Protection and Homeland Security

Carl Paperiello

Director, Office of Nuclear Regulatory Research
US Nuclear Regulatory Commission

Radiation protection issues play a large variety of roles in homeland security. Most recognize a variety of activities in responding to radiological dispersal devices(RDD's) and similar malevolent events. These activities correspond in many ways to existing radiological emergency planning and response responsibilities. Additional radiation protection needs have emerged in the prevention of terrorist and malevolent activities. These include radioactive materials security, border protection, and search capabilities for radioisotopes including special nuclear material (SNM). The use of ionizing radiation and radioactive sources in homeland security activities raises new issues with respect to public exposure, a significant increase in the number of radiation workers and their training, and the security of radioactive material in new devices. The role of NRC research activities in this area is still uncertain as much of the research appears to be the responsibility of organizations other than the NRC.

THIS PAGE INTENTIONALLY BLANK

TRACE : TRAC/RELAP Advanced Computation Engine

J. L. Staudenmeier

Office of Nuclear Regulatory Research
Safety Margins and Systems Analysis Branch

Overview

The TRACE project was started so that the NRC could replace their existing suite of reactor system safety analysis codes (RELAP5 , TRAC-PWR, TRAC-BWR, and RAMONA) by a single code in order to reduce code maintenance and user training costs. The previous code architectures were designed in the 1970's and were subject to constraints imposed by both computer hardware and the FORTRAN 77 language. TRACE was built with a modem code architecture that is easy to maintain and extend with new models to address future safety problems. TRACE is also integrated with the Symbolic Nuclear Analysis Package (SNAP) graphical NRC analysis environment. TRACE can also be run in a coarse grained parallel processing mode and has a built in mechanism to communicate with other computer codes.

Code Capabilities

TRACE is a two-fluid compressible flow code with 1 , 2 , and 3 dimensional flow geometry and is coupled to the PARCS 3 dimensional reactor kinetics code. The code also has the capability to model heat structures and control systems that interact with the fluid solution. TRACE will be able to perform any type of reactor analyses that were performed by the predecessor codes (RELAP5 , TRAC-PWR, TRAC-BWR, and RAMONA) and has component models and mesh connectivity that allow a full reactor and containment system to be easily modeled. The current models and correlations are the TRAC-PF1/MOD2 models. The NRC recognizes that users have a large investment in previously developed input decks. TRACE will be able to run input models from RELAP5, TRAC-PWR, and TRAC-BWR with minimal or no modification. This will allow for a gradual migration to TRACE through the transition period of the next few years during which both RELAP5 and TRACE are still supported. TRACE also supports coarse grained parallel processing in which the user can decide how to divide a model so that different pieces run on different processors.

User Interface

SNAP was developed to be an analysis environment for NRC safety analysis codes that will aid the user and make them more productive. SNAP currently supports TRACE and RELAP5. It will support CONTAIN and MELCOR soon. A user can develop and modify input models and then run and analyze the results within SNAP. SNAP also has the interactive calculation and mask animation capabilities of the Nuclear Plant Analyzer (NPA). SNAP will soon have the capability to automate parametric studies of a given input parameter over a range specified by the user. In the future SNAP will also allow custom user defined macro components that are comprised of multiple TRACE components.

Code Architecture

The TRACE code architecture was designed to take advantage of the capabilities of modem computer hardware and compilers. Features of Fortran 95 such as dynamic memory allocation and derived data types make the large container array and the associated error prone memory management techniques used in TRAC and RELAP obsolete. The large memory capacity of current desktop computers makes the use of memory saving techniques of bit packing and temporary memory overlays unnecessary. The TRACE code architecture was designed to be portable, easily maintained and extended. Gains in developer productivity have already been realized while developing a new BWR channel component that has the ability to model current BWR fuel bundle geometries. TRACE also has the built in capability to communicate with other codes or other TRACE runs locally or over a network though the Exterior Communications Interface (ECI). TRACE is currently being developed and run on multiple operating systems including Windows NT/2000/XP, Linux, and Mac OSX.

Exterior Communications Interface (ECI)

The ECI is a request driven interface that allows TRACE to communicate with any code that implements the ECI without any modifications to TRACE. The ECI uses a standard BSD sockets interface at the operating system level. The external code can be written in any computer language that can call BSD sockets. The ECI has allowed TRACE to be easily coupled to codes such as SNAP, CONTAIN, and MATLAB. The interface should allow TRACE to be coupled to CFD or other special purpose codes in the future.

Current Status and Future Development

TRACE has undergone limited assessment and is being used for some applications within the NRC. TRACE is currently undergoing a more complete code assessment to identify deficiencies in its current physical models and correlations. As deficiencies are identified new (or existing physical models from other predecessor codes) will replace the original TRAC-PWR models. New reflood and condensation models are currently being implemented in the code. Work is also underway to add a droplet field to the code and to make the numerical methods more implicit to improve code robustness and runtime performance. The NRC will also couple its other reactor analysis codes such as FRAPCON, FRAPTRAN, and MELCOR to TRACE in the future. TRACE should provide a robust and extensible platform for safety analyses well into the future. The long range objective of TRACE development is to produce a tool capable of simulating both large and small break loss of coolant accidents in conventional and advanced nuclear power plants.

MELCOR Development and Assessment

R.O. Gauntt

Sandia National Laboratories

MELCOR is a fully integrated, engineering-level computer code that models the progression of severe accidents in light water reactor nuclear power plants. MELCOR is being developed at Sandia National Laboratories for the U.S. Nuclear Regulatory Commission as a second-generation plant risk assessment tool and the successor to the Source Term Code Package. A broad spectrum of severe accident phenomena in both boiling and pressurized water reactors is treated in MELCOR in a unified framework. These include thermal-hydraulic response in the reactor coolant system, reactor cavity, containment, and confinement buildings; core heatup, degradation, and relocation; core-concrete attack; hydrogen production, transport, and combustion; fission product release and transport behavior.

Initially, the MELCOR code was envisioned as being predominantly parametric with respect to modeling complicated physical processes (in the interest of quick code execution time and a general lack of understanding of reactor accident physics). However, over the years as phenomenological uncertainties have been reduced and user expectations and demands from MELCOR have increased, the models implemented into MELCOR have become increasingly best estimate in nature. The increased speed (and decreased cost) of modem computers (including PCs) has eased many of the perceived constraints on MELCOR code development. Today, most MELCOR models are mechanistic, with capabilities approaching those of the most detailed codes of a few years ago.

The latest version, MELCOR 1.8.6 is scheduled for release in early 2005 and will include many new and improved phenomenological models, especially in the area of core melt progression. Included in the next code release will be significant improvements in the treatment of in-core and lower head molten pool and crust modeling, allowing for analysis of a TMI-like core degradation where a core-region convecting molten pool can be contained within confining perimeter crust and side wall release of the molten pool upon failure of the crust can be predicted. In the lower head region, improved molten pool models can account for internal pool natural convection and phase partitioning of the pool contents into disctinct metallic and ceramic regions. Other code modeling improvements include new models for release of Ag-In-Cd aerosol, for degradation and oxidation of PWR-type boron carbide control rods, for improved treatment of reflood/quench behavior, and many code extensions and enhancements. Additionally, significant advances in modeling parameters and properties have been gained based on insights derived from the international research efforts such as Phebus (IRSN), QUENCH (FzK) and MASCA (Ibrae), with the aim of implementing best estimate treatment of severe accident phenomena into the MELCOR code.

Today MELCOR is finding numerous applications both within the USNRC in both the support of licensing activities as well as best estimate source term characterization for severe accidents. Examples include the support of AP-1000 design certification, the upcoming ACR-700 design certification request by AECL, and new best estimate analyses of accidents in spent fuel pools. Emphasized in contemporary MELCOR analyses is a best estimate treatment of phenomena combined with true uncertainty analysis, as opposed to bounding deterministic analyses and sensitivity studies in order to provide risk-informed assessments of safety issues. In support of this philosophy, new capabilities have been developed for the MELCOR code suite to perform Monte Carlo uncertainty studies with MELCOR to supplement traditional deterministic applications of the MELCOR code.

Fuel Behavior Modeling in Accident Analysis

Harold Scott

U.S. Nuclear Regulatory Commission

Fuel design limits are used to prevent fuel damage during normal operation and anticipated operational occurrences. Accident analysis determines the extent of fuel damage (to ensure coolable geometry) during hypothetical transients and accidents. From the standpoint of fuel behavior, these transients and accidents may be broadly considered as either undercooling or overpower events. Regulatory limits and technical specifications are used to ensure that events of either type do not progress into core melt scenarios

Three postulated accident types are under investigation by NRC to determine the behavior of high-burnup fuel. For undercooling events, the design-basis loss-of-coolant accident (LOCA) is considered. To represent the broad class of overpower events, the PWR control-rod-ejection accident, a reactivity initiated accident, is considered. One beyond-design-basis event is considered in which both overpower and undercooling occur, i.e., power oscillations in a BWR when failure to scram is assumed.

Loss-of-coolant accident (LOCA)

A number of fuel-related phenomena, such as cladding oxidation and hydrogen uptake during normal operation, and cladding ballooning and rupture during the transient, may affect core coolability during small-break and large-break LOCAs. Further, fuel pellet cracking during the burnup process leads to the possibility of fuel relocation into ballooned sections of fuel rods, thus altering the axial distribution of decay heat. Ballooning size and cladding oxidation may also be affected by burnup. Substantial new data are being generated to address these effects (see Billone et. al. and Grismanovs et. al. in these Proceedings), but analysis is needed to estimate the impact of these effects on core behavior during a LOCA, such as coolant flow around ballooned regions in the fuel bundle. Therefore, high-burnup needs to be considered for both damage criteria (10 CFR 50.46) and evaluation models (10 CFR 50 Appendix K). The need for detailed fuel behavior modeling in system thermal hydraulic codes will be discussed.

Control-rod-ejection accidents in PWRs

As discussed previously, fuel rod cladding accumulates hydrogen during normal operation and this may lead to loss of ductility. As a consequence, high-burnup fuel rods may not withstand the rapid thermal expansion of the fuel pellets that occurs during a power pulse. Since NRC's damage criteria in Regulatory Guide 1.77 was derived for fresh fuel, these criteria need to be revised. Previous and more recent data from several different test reactors was analyzed to obtain a realistic cladding failure threshold correlation at high burnup. This work was completed last spring and results were issued as a Research Information Letter, No. 0401, on March 31, 2004.

Power oscillations associated with anticipated transients without scram (ATWS) in BWRs

BWR ATWS scenarios in which power oscillations might occur were reviewed several years ago (NUREG/CR-6743). Cladding temperature is expected to rise and eventually the cladding may dry out without subsequent rewetting. To analyze this accident, the FRAPTRAN single-rod code was coupled with the Finnish GENFLO hot-channel hydraulics code. Calculations with these coupled codes were described at the 2001 NSRC (NUREG/CP-0176, page 381). Aspects of further analysis and perceived data needs will be discussed.

The U.S. NRC Spatial Kinetics Code PARCS

Thomas J. Downar
Purdue University
West Lafayette, Indiana

PARCS is a three-dimensional reactor core neutronics simulator that solves the steady-state and time-dependent multigroup neutron diffusion or SP_3 transport equations to predict the steady-state and dynamic response of the reactor to reactivity perturbations such as control rod movements, boron concentration or changes in the temperature/fluid conditions in the reactor core. The code is applicable to both PWR and BWR cores loaded with either rectangular or hexagonal fuel assemblies. The PARCS code was chosen by the U.S. Nuclear Regulatory Commission as its best estimate core neutronics code [Downar, 1998], and PARCS is coupled directly to the U.S. NRC thermal-hydraulics systems codes TRACE and RELAP5. As of version 2.6, PARCS is coupled to TRACE as a static link library, whereas PARCS is coupled to RELAP5 using Parallel Virtual Machine (PVM) message passing interface. A depletion capability is also available in PARCS which includes a cross section interface capability to lattice physics codes such as HELIOS or SCALE/TRITON.

PARCS solves the multigroup time dependent diffusion and SP_3 equations with either assembly or pin-by-pin spatial discretization. The nodal options include both the analytic nodal method (ANM) and the nodal expansion method (NEM). The pin-by-pin finite difference option (FMFD) accepts pin homogenized cross sections in an arbitrary number of energy groups. The multigroup SP_3 transport option is available in both the NEM and FMFD kernels. The various geometry types and solution kernels available in PARCS are summarized in Table I.

Table I Solution Kernels Available in PARCS

Geometry Type	Kernel Name	Solution Method	Energy Treatment	Angle Treatment	Comments
Cartesian 3D	CMFD	FD	2G	Diffusion	
	ANM	nodal	2G	Diffusion	
	FMFD	FD	MG	SP3	
	NEMMG	nodal	MG	SP_3	
Hexagonal 3D	CMFD	FD	2G	Diffusion	
	TPEN	nodal	MG	Diffusion	
Cylindrical 3D	CMFD	FD	2G	Diffusion	To be released in v2.7
	FMFD	FD	MG	Diffusion/ SP_3	To be released in v2.7

The problem size which can be treated in PARCS is restricted by available memory. The finest level of group constant homogenization is at the pin-cell level, but there is no restriction on the number of energy groups for the NEM and FMFD options. Computational times are minimized by the use of advanced numerical methods such as a BILU3D preconditioned Krylov solver (BICGSTAB) to solve the coarse mesh finite difference problem in PARCS. Execution times will vary depending on the solution kernel chosen, however, 2-group nodal diffusion solutions for typical LWR transients (e.g. control rod ejection) are on the order of minutes on a current generation PC. Each new version of PARCS is tested on a series of Operating Systems (e.g. Windows, LINUX, Solaris) and Fortran compilers (e.g. DVF 6.6, Intel 7, NAGWare 4.2, Sun F90

6.2). A comprehensive assessment of the PARCS code for a wide range of reactor benchmarks has been reported previously, to include fast reactor transients such as control rod ejections [Joo, 1996], MOX fuel applications [Kozlowski, 2002], and coupled code transients for both PWRs [Kozlowski, 2000] and BWRs [Lee, 2002]. The most recent focus of BWR assessment for TRACE/PARCS is the OECD Ringhalls Stability Benchmark.

A fuel depletion capability is available in PARCS, which employs a macroscopic depletion method [Xu, 2001]. During each "burnup step" the power in each fuel node is used to time advance the burnup of the node, and the assembly cross sections derivatives and other neutronics parameters are computed at the new burnup for the corresponding fuel composition. Alternately, a given core "burnup" distribution can be specified by the user and the depletion module in PARCS will generate the fuel assembly neutronics data at the specified burnup. Thus the code provides the functionality to meet the two primary fuel cycle problems of the analyst, to deplete a core to find a typical "equilibrium" condition or to accept the burnup distribution for a specific core and to find the corresponding cross sections and steady-state condition. The fuel macroscopic cross sections and other nuclear data (e.g. kinetics parameters, detector information, discontinuity factors, form functions, Xe/Sm data, etc) at the appropriate fuel burnup conditions are prepared using lattice physics code such as HELIOS, TRITON, CASMO, etc. Because the output format of the each lattice code is unique, the cross section processing program, GENPMAXS, was written to process the output of the lattice codes and prepare a common cross section format file, PMAXS, which is read by PARCS. A separate manual is available which describes in detail the cross section processing treatment in GENPMAXS.

Near term development work on PARCS will address methods upgrades for analysis of advanced core designs such as the ACR-700. Other near term work will include the incorporation of PARCS into the TRACE/SNAP interface software. Specific questions can be addressed to the PARCS users group which is available via the PARCS website which also provides more detailed information on the PARCS code:

www.engineering.purdue.edu/parcs

[Downar, 1998] PARCS: Purdue Advanced Reactor Core Simulator, User Manual, PU/NE-98-26, NRC-V1.00, T. Downar, H. Joo, D. Barber, 1998.

[Joo, 1996] Methods and Performance of the Neutron Kinetics Code PARCS, H. Joo, D. Barber, T. Downar, PHYSOR-96, International Conference on the Physics of Reactors, Mito, Japan, September, 1996.

[Kozlowski, 2002] Benchmarking of the Multigroup, Fine Mesh, SP3 Methods in PARCS with the VENUS-2 MOX Critical Experiments, T. Kozlowski, C. Lee, T. Downar, School of Nuclear Engineering Purdue University, PHYSOR-2004, Seoul Korea, October, 2002.

[Xu, 2001] Fuel Cycle Depletion Capability for the U.S. NRC Neutornics Code PARCS, Y. Xu, T. Kim, D. Tinkler, T. Downar, M&C 2001 Salt Lake City, Utah, USA, September 2001

[Lee, 2002] Analysis of the OECD Peach Bottom Turbine Trip 2 Transient Benchmark with Coupled Neutronic and Thermal-Hydraulics Code TRAC-M/PARCS, D. J. Lee, T. Downar, T. Ulses, K. Ivanov, B. Akdeniz, PHYSOR-2004, Seoul Korea, October, 2002.

[Kozlowski, 2000] Analysis of the OECD MSLB Benchmark with the Coupled Neutronic and Thermal-Hydraulics Code RELAP5/PARCS, T. Kozlowski, R. Miller, T. Downar, D. Ebert, PHYSOR-2000, Pittsburgh, 2000.

Use of Computer Codes in the Regulatory Review Process

Ralph R. Landry

U.S. Nuclear Regulatory Commission

Traditional regulatory reviews of computer codes have focused on the documentation of the code and results of the application of the code. Codes developed for generic application to numerous nuclear power plants and plants of differing designs have necessitated application calculations that are non-plant specific. That is, the calculated plant response is for a fictitious plant that is assumed to be "representative" of plants to which the code will be applied for licensing purposes. The ensuing code review must also make broad statements about the code and its applicability and the manner in which it behaves.

The Office of Nuclear Reactor Regulation has, in recent years, further exercised its regulatory authority in so far as the applicants for approval of thermal hydraulic systems codes have been asked to submit the code itself with a sample input deck in addition to the usual documentation and analytic results. To remove computer hardware, operating system, and compiler effects from use of the individual code, the NRC has acquired numerous computer workstations so that the codes can be run on the same platform as used by their respective developers.

Review and approval of codes such as RETRAN-3D, S-RELAP5, and TRACG was based in part on extensive code execution by the staff of the NRC.

A significant part of the analytical process followed since the development of the Code Scaling, Applicability and Uncertainty evaluation methodology is development of a Phenomena Identification and Ranking Table, or PIRT. The PIRT and the process leading to its development provides significant insight into those processes and phenomena having the most influence on the progression and results of a nuclear plant accident or transient. Understanding which phenomena, and ultimately which models, are key to an event's progression, permits code developers to focus their efforts where they would expect to need to have the greatest accuracy in modeling. From the perspective of the regulator, the PIRT permits the focusing of review activities on those areas having the most importance thereby using available resources more efficiently.

Code utilization has taken several forms during the review process. Receipt of the subject code in both a precompiled executable and source form permits execution of the code precisely as would be done by the applicant. Of course, results of a calculation are expected to match exactly with the results supplied with the code documentation. In addition, availability of the source code permits the staff to make modifications to specific models and correlations, compile a new executable code and perform test analyses. Through these test cases, the staff is able to further understand the level of importance of the various models and correlations within the code, as well as confirm the level of importance identified through the PIRT process.

As a further step in the review process, the staff also performs parallel calculations of the specific applicant code transient using the NRC's computer codes. In several reviews it has been obvious that there are benefits to both codes in this process. Reviews of applicant codes in the area of the reactor kinetic models, by comparison with the diffusion code NESTLE and a transport code, found that while some codes produced virtually identical results, others produced dramatically different results. That work was done prior to the availability of PARCS.

Parallel applicant/NRC thermal hydraulic calculations have been very successful in that they have identified both code to code shortcomings, as well as errors in input modeling that might have gone otherwise undetected. Refining the modeling and upgrading the codes' capabilities ultimately resulted in more accurate and more reliable nuclear plant calculations. From the perspective of the NRC, we have an improved independent tool, while from the perspective of the regulator, we have more confidence in the results from an approved code that will be submitted for future review.

Including execution of the applicant's computer code as well as the NRC's independent computer codes in the code review and approval process has enhanced our understanding of the applicants' codes as well as improved our confidence in the results of their use of the codes. An additional benefit has been improvement of our own codes by identifying those areas were models were in need of greater accuracy.

Recent Developments in Multidimensional Modeling of Reactor Thermal Hydraulics Using the NPHASE Code

M. Z. Podowski, S. P. Antal, B.Z. Wierzbicki and P. Tiwari
Center for Multiphase Research, Rensselaer Polytechnic Institute
Troy, New York 12080-3590

Recent progress in both the understanding of the basic physics of two-phase flows and the computational capabilities using CFD methods clearly shows the great potential of for detailed three dimensional mechanistically-based numerical simulations of two-phase flows in boiling channels and nuclear reactor systems. These simulations can give significant insight into the hydraulic characteristics of next generation reactor systems. It is not surprising that most advancements to date are related to dispersed flows. Indeed, several closure laws based on ensemble-averaging the instantaneous equations governing the interacting phases have already been developed for both bubbly and particle flows and successfully applied to study various multiphase/multicomponent flow phenomena. In order to extend the existing predictive capabilities to other flow regimes such as slug flow, churn flow, and annular flow, where a phenomenological approach has been mainly used until now, new closure laws must be developed and incorporated in an appropriate multidimensional computational framework.

The purpose of this presentation is to summarize the results of recent work on the development of mechanistic models of two- and multiphase flow for the NPHASE code. Specifically, closure laws consistent with the multifield modeling and computational concepts used by NPHASE are discussed and selected results of NPHASE simulations are shown. Furthermore, new results are presented, concerning coupling a new Level-Set-based interface tracking computational model with the NPHASE solver, and the use of the combined code to simulate the effect of bubble deformation and channel geometry on gas-liquid interaction.

The accuracy of all CFD predictions depends strongly on the quality of closure models for local two-phase flow phenomena mass and momentum transfer. Typically, CFD codes use simplified closure laws, actually applicable only to dispersed bubbly/particle flows. Recently, several generic closure laws (defined by specific mathematical operators and formulae) have been developed for, and/or implemented into the solution algorithm of, the NPHASE code, that are applicable to various flow regimes, including bubbly, slug, churn turbulent, and annular flows. Also, work was has been performed on the development of mechanistic local models of both general and flow-regime-specific local interfacial phenomena, and several new modeling and computational issues have been resolved that are critical for the consistency of the overall CFD model. A key effort here has been on the implementation of the new models with the underlying numerical treatment, to assure a tight coupling between the numerics and models that is needed to improve the robustness and convergence characteristics of an advanced multiphase CFD code. In addition, a fully coupled mass/momentum solver has been developed for the NPHASE code. This advanced numerical method allows the solution of both the mass conservation and momentum conservation equations simultaneously.

The ability to predict the shape of the gas/liquid/solid interfaces is important for various multiphase flow and heat transfer applications. Specific issues of interest to nuclear reactor thermal-hydraulics, include the evolution of the shape of bubbles attached to solid surfaces during nucleation, bubble-surface interactions in complex geometries, etc. Additional problems, making the overall task even more complicated, are associated with the effect of material properties that may be significantly altered by the addition of minute amounts of impurities, such as surfactants or nanoparticles. An important practical issue is concerned with the development of an innovative approach to model time-dependent shape of gas/liquid interfaces in the presence of solid walls. The present approach combines a modified level-set method with the NPHASE CFD solver. The coupled numerical solver that is being currently developed will be capable of simulating the evolution of gas/liquid interfaces in two-phase flows for a variety of geometries and flow conditions, from individual bubbles to free surfaces (stratified flows).

As a result of the present work, a consistent first major step has been made toward the development and encoding the models of individual flow regimes into a closed-form model allowing to predict varying flow conditions in heated reactor channels using multidimensional computer simulations. Interestingly, the present work also provides new insight into the modeling issues encountered in reactor system codes.

Overview of Approach to Reevaluate Station Blackout Risk

Dale M. Rasmuson
Operating Experience Risk analysis Branch
Division of Risk Analysis and Applications
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001
301-415-7571
dmr@nrc.gov

NUREG-1032 presented the risk basis for the Station Blackout (SBO) Rule. It provided loss of offsite power (LOOP) frequencies, probability estimates for not recovering offsite power to a safety bus, and core damage frequency estimates of LOOP and SBO accident scenarios. NUREG-1032 used LOOP event information from 1968 through 1985. NUREG/CR-5496 updated the LOOP frequencies and nonrecovery probabilities using LOOP event information from 1980 through 1996.

On August 14, 2003, large portions of the Midwest and Northeast United States and Ontario, Canada, experienced an electric power blackout. The outage affected an estimated 50 million people and 61,800 MW of electric load. Nine U.S. nuclear power plants experienced reactor trips as a consequence of the power outage. Nine plants used their emergency diesel generators to power their safety-related buses during the power outage. Offsite power was restored to at least one safety bus after a period of time ranging from about two hours to about 14 hours, with an average time of about seven hours. These LOOP events were longer in duration than previously considered under the SBO rule.

As part NRC's grid action plan, RES has or is performing the following tasks:

1. Provide short-term risk insights of the impact on nuclear power plants based on the observed changes in grid reliability to allow the agency to evaluate if action is needed before the Summer of 2004.

2. Provide preliminary accident sequence precursor analyses of each affected plant for technical review.

3. Evaluate SBO implications. Using data from recent LOOP events, update the SBO LOOP frequency and duration.

4. Evaluate SBO risk. Calculate SBO risk (core damage frequency) with updated Standardized Plant Analysis Risk models for a spectrum of plants.

This paper will present a summary of the ASP risk analyses completed to date and an overview of the methods being used to update LOOP frequencies and nonrecovery probabilities. It will also provide an overview of the steps to be used in the reevaluation of core damage frequency associated with LOOP and SBO.

References

1. U.S. Nuclear Regulatory Commission, *Evaluation of Station Blackout Accidents at Nuclear Power Plants*, NUREG-1032, June 1988.

2. U.S. Nuclear Regulatory Commission, *Evaluation of Loss of Offsite Power Events at Nuclear Power Plants: 1980 – 1996*, NUREG/CR-5496, November 1998.

Generic Issues Program - Process and Screening Analysis

Harold Vandermolen

U.S. Nuclear Regulatory Commission

The Generic Issues Program first began formally in response to a Commission directive in October of 1976. In 1983, it became one of the first programs to make successful use of probabilistic risk information to aid in agency decisionmaking. In the 20 years since the program became quantitative, approximately 840 issues have been processed. Although there is far less reactor licensing activity than in the 1970s, new issues continue to be identified from research programs and operational experience, and the generic issue program remains very active.

Generic issues tend to involve rather difficult questions of safety and regulation, and an efficient and effective means of addressing these issues is very important if regulatory effectiveness is to be achieved. If an issue proves to pose a genuine, significant safety question, then swift, effective, enforceable, and cost-effective action needs to be taken to eliminate an accident before it happens. Conversely, if an issue is of little safety significance, the issue should be dismissed in an expeditious manner, avoiding unnecessary expenditure of resources and regulatory uncertainty. The generic issues process will be discussed. The screening analysis of a generic issue arising from operating experience will be presented.

THIS PAGE INTENTIONALLY BLANK

Operating Experience Implementation Task Force

Terrence Reis

U.S. Nuclear Regulatory Commission, (NRR)Task Force Lead

When the Office for Analysis and Evaluation of Operational Data was dissolved in 1999 the operating experience (OpE) program functions were transferred to the Office of Nuclear Reactor Regulation (NRR), Office of Nuclear Regulatory Research (RES), Office of Administration, and the Incident Response Organization. Since that time, some difficulties have been encountered in the OpE program. NRR and RES began discussions in 2001 to correct some of the program shortcomings. In 2002, the Davis Besse Lesson Learned Task Force, also recommended that the OpE Program be re-assessed. An interoffice Reactor Operating Experience Task Force conducted an assessment of the OpE Program and provided 23 recommendations to correct the OpE Program shortcomings. This presentation will briefly discuss the activities being conducted to implement a new OpE Program.

THIS PAGE INTENTIONALLY BLANK

A Generalized Framework for Assessment of Safety Margins in Nuclear Power Plants

George Lanik, NRC/RES; Jim Meyer, ISL

A generalized framework for assessment of safety margins is proposed with the goal of using both probabilistic and deterministic analysis to generate an overall safety margin index. Through this technique, the proposed framework merges elements of safety regulation such as defense-in-depth, traditional safety margins, and probabilistic risk. Separate safety margin metrics are proposed for fuel performance, primary pressure boundary performance, and containment performance, with the possibility of combining these into an overall safety metric.

A primary goal of this work is to develop a method to assess changes at plants due to power up-rates, longer cycles, extended fuel burn-up, and aging processes. The process could assess other changes as well, such as operating procedures and technical specifications. Such an approach might be adapted to assess proposed changes such as those currently addressed by Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis."

At the current state of development, the focus has been on the status of the fuel during accidents and transients with performance metrics related to peak clad temperature and peak enthalpy deposition. Preliminary results have been generated from analysis of medium sized Loss-of-Coolant Accident sequences.

THIS PAGE INTENTIONALLY BLANK

Boric Acid Corrosion of Light Water Reactor Pressure Vessel Materials

J.-H. Park, O. K. Chopra, K. Natesan, W. J. Shack and W. H. Cullen
U.S. Nuclear Regulatory Commission

This report presents experimental data on corrosion rates of the materials found in or near typical reactor pressure vessel penetrations, in boric acid solutions of varying concentrations, at temperatures of 95–316°C (203–600°F). Tests were conducted in (a) high–temperature, high–pressure aqueous solutions with a range of boric acid concentrations, (b) high-temperature (150°C–316°C) H-B-O solutions at ambient pressure, wetted and dry, and (c) low–temperature (≈95°C) saturated, aqueous, boric acid solutions. The results indicate significant corrosion only for the low–alloy steel, and no corrosion for Alloy 600 or 308 stainless steel cladding. Corrosion rates were most significant for two conditions: (a) a boric acid-saturated, aqueous solution at 97.5°C exhibited corrosion rates up to 100 mm/year, and (b) H-B-O solutions exhibited a corrosion rate of about 125 mm/year at a temperature of 150°C. No material loss was observed in boric acid melts or deposits in the absence of moisture. Tests were also conducted with alloys of various Cr contents in the range of 0.2-25 wt.% to evaluate the effect of Cr loss from Type 308SS weldment on the corrosion rate in a room temperature-saturated boric acid solution at temperatures between 150° and 316°C at a pressure of 12.4 MPa. The corrosion test results from this study are compared with the existing corrosion/wastage data in the literature.

NRC FORM 335
(9-2004)
NRCMD 3.7

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER
(Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/CP-0188

2. TITLE AND SUBTITLE

Transactions of the 2004 Nuclear Safety Research Conference

3. DATE REPORT PUBLISHED

MONTH	YEAR
October	2004

4. FIN OR GRANT NUMBER

A3988

5. AUTHOR(S)

Summaries of Conference Papers by various authors;
Compiled by Susan Monteleone, BNL

6. TYPE OF REPORT

Conference summaries

7. PERIOD COVERED *(Inclusive Dates)*

October 25-27, 2004

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
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 Item 8 above.

10. SUPPLEMENTARY NOTES

S. Nesmith, NRC Project Manager, Transactions prepared by Brookhaven National Laboratory

11. ABSTRACT *(200 words or less)*

This report contains summaries of papers on reactor safety research to be presented at the 2004 Nuclear Safety Research Conference (formerly titled the Water Reactor Safety Information Meeting) at the Marriott Hotel at Metro Center in Washington, DC, October 25-27, 2004.

The summaries briefly describe the programs and results of nuclear safety research sponsored by the U.S. Nuclear Regulatory Commission's Office of Nuclear Regulatory Research. Also included are summaries of invited papers concerning nuclear safety issues from U.S. government laboratories, the electric utitlities, the nuclear industry, and from foreign governments and industry.

The summaries have been compiled here to provide a basis for meaningful discussion of information exchanged during the course of the meeting, and are in the order of their presentation on each day of the meeting.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

reactor safety research
nuclear safety research

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

(This Page)
unclassified

(This Report)
unclassified

15. NUMBER OF PAGES

16. PRICE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, DC 20555-0001

OFFICIAL BUSINESS