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Transactions of the 2002 Nuclear Safety Research Conference

To be Held at
Marriott Hotel at Metro Center
Washington, DC
October 28-30, 2002

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

Transactions prepared by
Brookhaven National Laboratory

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Transactions of the 2002 Nuclear Safety Research Conference

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ABSTRACT

This report contains summaries of papers on reactor safety research to be presented at the 2002 Nuclear Safety Research Conference (formerly titled the Water Reactor Safety Information Meeting) at the Marriott Hotel at Metro Center in Washington, DC, October 28-30, 2002.

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.

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EPRI Materials Reliability Program for Alloy 600

Larry K. Mathews

Southern Nuclear Company

Abstract

The Alloy 600 Issues Task Group of the EPRI managed Materials Reliability Program (MRP) initiated an industry program to address the generic aspects of the Alloy 182 weld cracking in the A hot leg nozzle weld at V. C. Summer and the cracks in the Alloy 600 Control Rod Drive Module (CRDM) and thermocouple (T/C) nozzles at Oconee 1. Further cracking in head penetrations at other pressurized water reactors and the corrosion of the vessel head at Davis-Besse have further emphasized this need for a concerted industry effort.

The MRP program was established to address these areas and has evolved significantly as more information has become available. It includes activities in assessment and management of the issue, inspection capability, and repair and mitigation. Because of the safety implications of these issues, the Nuclear Regulatory Commission has issued several NRC Bulletins and other generic communications. The MRP program also includes work to assist utilities in responding to these bulletins. Long-term activities are planned to provide utilities with appropriate tools for managing the PWSCC of reactor head penetrations.

In-service Inspection of PWSCC in Alloy 600\182\82 Material

Steven R. Doctor

Pacific Northwest National Laboratory, Richland, WA

In the past several years there has been a number of occurrences of primary water stress corrosion cracking (PWSCC) in pressurized water reactors (PWRs) containing Inconel alloy 600 and both 182 and 82 welds. The material degradation has been detected in the main coolant outlet nozzles and in the control rod drive mechanism (CRDM) head penetrations. One of the most significant aspects of these recent failures is that they resulted in through wall cracking and in the case of Davis-Besse the aggressive attack of the surrounding ferritic steel head. A research program at the Pacific Northwest National Laboratory (PNNL) under U. S. NRC funding is addressing the reliability of detecting and accurately characterizing PWSCC in these components. A status of the work will be presented with accomplishments to date and the planned future work to bring closure to these issues.

The cracking of the primary circuit reactor pressure vessel (RPV) nozzle weld at V. C. Summer was initially discovered by the presence of boric acid crystals. The ultrasonic inspection of this weldment is conducted from the inside surface during the RPV normal 10 year in-service inspection (ISI). The majority of cracking that was detected had an axial orientation. In reviewing the ID conditions of the failed nozzle at V. C. Summer, it was apparent that there were geometric conditions that significantly increased the challenge to perform an effective inspection. In addition, dissimilar metal welds (DMWs) are difficult to inspect because they are made with complex metallurgical conditions associated with the buttering, welding directions, proximity of adjacent welds, counterbore conditions, offset and component alignment. Supplement 10 of the ASME Boiler and Pressure Vessel Section XI Code Appendix VIII covers these DMWs. This supplement is currently in the implementation stage with a scheduled date of November 22, 2002 as the target for having qualified procedures, equipment and inspectors. Although it is recognized that DMWs are difficult to reliably inspect, there have only been limited studies trying to quantify the effectiveness of current technology and practice. The technologies that are being evaluated at PNNL include phased arrays and a low frequency synthetic aperture focusing technique (SAFT). The results of work on far side austenitic weld inspections will also be presented since they directly relate to the PWSCC inspection issue in Inconel since they are both coarse grained materials

The strategy for the inspection of CRDMs that has been pursued by industry is to conduct a visual ISI of the RPV head and look for the presence of boric acid crystals. Those areas where there is unexplained boric acid deposits are then inspected by a variety of techniques to detect the potential presence of PWSCC in the CRDM nozzle tube or by using a surface technique to look for PWSCC on the wetted surface of the J-groove weld. A shortcoming of this strategy is that it erodes one of the layers in the defense in depth for reactor operation. Waiting for leakage to occur means that leakage will not be prevented. The challenge then is to reliably and quickly inspect those CRDM areas where PWSCC is active. One of the hardest areas to inspect is the J-groove weld because it is like a DMW since there is buttering and a weld with limitations on access. These J-groove welds were inspected by a penetrant test during fabrication but have never been volumetrically inspected. The work to be described includes evaluation of technologies to detect cavities behind the CRDM alloy 600 head penetrations and work on assessing techniques for the reliable volumetric inspection of the J-groove weld. A nozzle mockup from the cancelled Midland RPV head is being used in these studies.

Summary of Ongoing NRC Efforts to Define Circumferential Crack-Driving Force Solutions for CRDM Nozzles

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2) US Nuclear Regulatory Commission, Office of Nuclear Reactor Research

Abstract

The earlier occurrence of CRDM nozzle cracking in European PWRs was limited to the formation of axial cracks by PWSCC in the In600 tubes used in making the nozzles. Axial cracks in these tubes are not a safety concern, but present a situation of primary pressure boundary leakage that is not allowed by the plant technical specification. In 2001, several circumferential cracks were found in CRDM nozzle tubes in US nuclear power plants. This occurred first by axial cracks forming and allowing PWR water to cause PWSCC cracks to form on the outside diameter of the In600 tubes. The circumferential cracks occurred above the J-weld, and were up to 160 degrees in length. Although the pressure-induced axial stresses for full rupture of the tube are low some of the circumferential cracks that have occurred were quite long (~160-degrees). Since the crack sizes were quite large, the possibility of the tube being ejected from the RPV head is a significant safety concern. Consequently, the USNRC has undertaken a program to assess the integrity of CRDM nozzles in existing plants that are not immediately replacing their RPV heads. This paper summarizes some of the efforts undertaken on the behalf of the USNRC for the development of detailed residual stress and circumferential crack-driving force analyses in probabilistic determinations of the time from detectable leakage to failure.

To conduct these analyses, detailed information was first obtained on typical CRDM nozzle designs and welding procedures. The initial analyses were for center-hole geometry, and then a more detailed 53-degree angle side-hill geometry was analyzed. The side-hill nozzle analysis is much more complicated, hence the initial parameters of interest were first explored in the center-hole case. Parameters varied in the center-hole analyses included; strength of the In600 tube material, interference fit, and weld height.

Great care was taken to ensure that the material properties used in the weld simulation analysis were as realistic as possible. To do so, data was solicited from international contacts, and it was arranged to have ORNL develop tensile test data of interest. In particular for the weld analysis, the data in the literature for In82/182 weld metals at temperature was in the as-welded condition. In the analyses conducted, we start the analysis from the molten metal condition, and the plastic history is calculated on a pass-by-pass basis. Using as-welded tensile test data would give too high of stresses in the weld metal for this type of analysis. Hence, ORNL was tasked to developed tensile test data using solution-annealed weld metal. Additionally, the welding process has an average strain-rate of about 10^{-3}, which is much faster than standard tensile testing. The strain rate is important at the higher temperatures from a creep concern, i.e., data were developed from room temperature to 1000C.

To calculate the residual stresses for the center-hole case, the residual stresses were conducted as an axisymmetric analysis, where each of the weld beads was individually created. The side-hill case involved 3D analyses with one plane of symmetry through the nozzle axis, and a symmetry plane on the head that corresponded to having 8 nozzles in the whole head. The weld simulation analysis was conducted using ABAQUS with a proprietary Emc^2 weld simulation UMAT. There were 12 to 16 elements in each weld bead, which is sufficient to adequately perform the heat-transfer analyses needed in the weld simulation analysis. 20 weld beads were used in the initial analysis. After each

layer of weld metal was applied, the residual stress conditions were examined. Those results showed that the longitudinal stresses at the root of the weld were greater when the weld height was smaller. Hence, increasing the weld height may increase the hoop stresses, but would decrease the longitudinal stresses for circumferential cracking. Additionally, it was found that the principle stress direction in the tube cross-section near the root of the weld was not in the longitudinal direction, but was at an angle of about 45-degrees. Since subcritical crack growth is in the Mode I direction (perpendicular to the principal stresses), this implies that the CRDM circumferential cracks will grow at an angle through the thickness.

Once the residual stresses were calculated and the weld metal was allowed to cool to room temperature, the hydrotest pressure (125% of design pressure) at room temperature was applied to the head and then removed. The efforts to develop the crack-driving force for the circumferential crack used the following procedure.

- The entire post-hydrotest residual stress field was mapped to a new FE model that contained a circumferential crack that was pinned closed. The crack was defined as having a front that was perpendicular to the tube surface, but followed the J-weld contour around the tube. This mapping procedures was done for tubes for different circumferential crack lengths (about 10 crack lengths per case), which was much more cost efficient that redoing the weld residual stress calculations for each cracked tube case. The stress mapping procedure is relatively new in ABAQUS, and the mapping procedures were validated with several test cases.
- The model was then taken to operating pressure (2,500 psig) and temperature (560F or 605F for cold versus hot heads),
- The crack was then unpinned, and the crack-driving force (K from J) was calculated.
- Since the entire stress field of the head was mapped to the cracked model, all the crack-driving force components (Modes I, II, and III) contributed to the J value.
- The J values accounted for plasticity, but the K values for the three different modes were calculated.

The results showed that there was a significant Mode III contribution to the crack-driving force, especially for shorter crack lengths. The physics of this became obvious once the detailed analyses were conducted. K_J versus crack length values was determined for the operating pressure. Three observations to date are:

1. The higher yield strength tubes had higher K_J values,
2. The temperature had little effect on the driving force (it may be important for PWSCC rates, but not the driving force), and
3. Increasing the temperature and pressure generally made the interference fit disappear so that there was a gap between the head and tube, however, at a room temperature an interference fit of 0.004 inch was present, while at high temperature there was still some small interference between the tube and the RPV head. This reduced the crack-driving force since contact was included in the analysis and friction was included.

These and other aspects from the residual stress and crack driving force analyses efforts will be discussed in this paper.

Parametric Studies of the Probability of Failure of CRDM Nozzles

William J. Shack
Argonne National Laboratory

In order to develop effective and efficient inspection plans for CRDM nozzles, it is important to understand how the probability of failure of the nozzles varies with time, temperature, and other critical parameters such as the crack growth rate, time to initiation, and stress intensity factor distribution.

Crack growth rates in Alloy 600 vary widely on a heat–to–heat basis. Although the range of crack growth rates that are possible are reasonably well understood, it is difficult to determine a priori the appropriate crack growth rate for specific heats of material in a particular reactor. Similarly although a reasonable amount is known about the dependence of the initiation of stress corrosion cracks on stress, temperature, and microstructure for Alloy 600, it is difficult to determine a priori the susceptibility of a particular nozzle. Instead initiation of cracks in nozzles is described in terms of Weibull probability distribution determined from the results of field inspections. Unless cracks are large, the primary driving force for the cracking arises from the residual stresses due to welding. Various estimates of the stress intensity factors due the welding residual stresses have been developed, but no consensus is yet available on which of these estimates best describes the actual stress states or on the best way to represent the distribution of residual stresses that actually occur.

The probability of failure calculations are thus carried out using a distribution of crack growth rates consistent with that proposed by the industry in MRP–55, a distribution of Weibull parameters for initiation consistent with the inspection findings to date. Variables such as the stress intensity factor for which there is less consensus or data to support a particular choice are studied parametrically. Other variables, which are also very difficult to assess such as the potential for multiple of initiation of cracks, are also treated parametrically to assess their potential impact on the results. Similarly, it is expected that the susceptibility to initiation and the crack growth rate should be correlated, but the degree of correlation is difficult to determine and so is assessed parametrically.

The analysis also includes the potential impact of inspection frequency and effectiveness on the probability of failure.

Boric Acid Corrosion of Reactor Vessel Heads Resulting from CRDM Nozzle PWSCC

Glenn A. White, E. Stephen Hunt
Dominion Engineering, Inc.

This paper summarizes work performed by Dominion Engineering, Inc. on behalf of the EPRI Materials Reliability Program (MRP) to develop a technical understanding of the Davis-Besse RPV head corrosion, and to establish required visual inspection intervals to ensure that significant corrosion does not occur on other vessel heads due to PWSCC.

Overview of Davis-Besse Corrosion Incident

Inspections of the Davis-Besse RPV head during the thirteenth refueling outage (Spring 2002) showed that about 200 in^3 of the low-alloy steel head material adjacent to CRDM nozzle #3 had been lost by corrosion. The root cause analysis performed by the utility concluded that the corrosion had been caused by borated water leaking from a through-wall primary water stress corrosion crack (PWSCC) that extended above the J-groove weld of an Alloy 600 CRDM nozzle. Evidence collected during the root cause analysis showed that the nozzle had likely been leaking for at least six years, and that the leak rate over the previous 24-month operating cycle had been approximately 0.10-0.15 gpm.

Progression of Leakage and Degradation

A model was developed for the progression of leakage and degradation. The crack grows axially under the influence of the residual stresses in the nozzle caused by welding. The leak rate is predicted to be primarily a function of the length of the axial crack above the top of the J-groove weld and the resultant crack opening displacement.

Experience with other leaking nozzles has shown that short crack lengths above the top of the J-groove weld result in very low leak rates and essentially no loss of low-alloy steel material in the annulus or on the top surface of the vessel head. As the crack length and leak rate increase, leakage through the annulus is believed to begin to open up the annulus by flashing-induced erosion, also known as steam cutting. For the case of very low leak rates and the high operating head temperatures, the leaking water will boil quickly preventing concentrated boric acid solutions from developing over a significant region of the annulus.

As the crack length and leak rate increase further, the leaking primary coolant will begin to cool the adjacent metal surfaces. When the metal surface in the annulus is cooled locally to values approaching 212°F, a concentrated boric acid solution will exist along the walls of the entire crevice. Eventually the leak rate will reach the point where the concentrated boric acid solution will begin to run out onto the top surface of the vessel, where it may become oxygenated and produce relatively high local corrosion rates. The shape of the corroded area will depend upon the leak rate, the angle of the nozzle, and the rate that the water on the top surface of the head evaporates.

Continued leakage at a relatively constant rate for a long period of time such as appears to have occurred at Davis-Besse will result in an equilibrium volume of boric acid solution collecting in a pool on the top of the vessel head. With time this pool will corrode downward, and possibly outward, through the vessel head, until arrested by the stainless steel cladding on the inside surface of the head.

Supporting Analyses and Tests

A spectrum of possible corrosion mechanisms has been developed including steam cutting, flow accelerated corrosion (FAC), galvanic corrosion, crevice corrosion, boric acid wastage, etc. A matrix has been prepared to show the probable corrosion mechanisms as a function of the leak rate and annulus size.

Finite element models have been used to predict the crack opening displacement as a function of crack length above the top of the J-groove weld, and this work has been used to estimate the leak rate as a function of the crack length. Thermal models have been prepared to assess the local metal temperatures as a result of the leakage into the annulus. Additional analytical work has been performed to determine the thermal-hydraulic and chemical environments along the leak path. For example, work has included investigating the effect of the annulus clearance on leak rate, the magnitude of the liquid phase velocities and wall shear stresses along the leak path, and the developing pH of the concentrating effluent.

The industry has performed several tests to determine corrosion rates for leakage of borated water into the annulus region between nozzles and low-alloy steel collars that simulate the vessel top head. Tests have also been performed to determine corrosion rates in the presence of concentrated oxygenated boric acid solutions such as the pooling conditions that are considered to exist on the head surface. Several additional tests have been proposed to refine understanding of the electrochemical environment within the growing wastage cavity, to develop data on the properties and corrosivity of molten boric acid, to examine the role of the galvanic coupling of the nickel-alloy nozzle and low-alloy steel head material, to address the initial conditions of relatively small leak rates into a tight fitting annulus, and to confirm corrosion rates under the proposed pooling conditions.

Required Visual Inspection Intervals

The purpose of the above work has been to develop analytical models to establish the required visual inspection intervals to ensure a low risk of significant corrosion approaching that which occurred at Davis-Besse.

Significant Findings of the Davis-Besse Reactor Vessel Head Degradation Lessons Learned Task Force

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Davis-Besse Reactor Vessel Head Degradation Lessons Learned Task Force (LLTF)
U. S. Nuclear Regulatory Commission

This abstract is being withheld pending publication of the LLTF Final Report in early October. The abstract and conference report will be available at the time of the meeting, and will be included in the proceedings of the conference.

Summary of USNRC's Research to Address Pressure Boundary Degradation Issues

William H. Cullen, Jr.

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Office of Research, US Nuclear Regulatory Commission

Earlier in this session, the results of three, NRC-sponsored research programs addressing the pressure boundary degradation issue were described. The program at Argonne National Laboratory (ANL) involves crack growth rate testing, the development of the micro mechanisms underlying component cracking, and the derivation of probabilistic models for the computation of crack growth rates and optimal inspection intervals. The program at Pacific Northwest National Laboratory (PNNL) centers on the development of non-destructive inspection techniques for the volumetric examination of CRDM Nozzles, J-welds, and safe ends. The program at Engineering Mechanics of Columbus (EMCC) enables development of computational models of residual stresses in prototypical CRDMs, and is producing an updated model for calculation of leak rates from axial and circumferential cracks in CRDMs. All of these programs were redirected in early March to assist the NRC with the myriad questions that arose concerning CRDM cracking, corrosion and wastage of low-alloy steel, and other modes of pressure boundary degradation. Additionally, a program at Oak Ridge National Laboratory (ORNL) has assisted the NRC with calculations supporting the safety assessment of the corrosion cavity and exposed clad discovered at the Davis-Besse plant.

This presentation focuses on the future direction of these and other, more recently awarded Materials Engineering Branch (MEB) programs that will serve to develop a comprehensive understanding of degradation mechanisms and the methods of discovering the degradation as early in life as possible. At ANL, the test matrix for determination of crack growth rates in vessel head penetration materials will be described. The program at PNNL will focus on continued development of innovative technologies for non-destructive inspection of reactor components. The EMCC program will conclude their derivation of leak rate models with a series of reports issued in late 2002. Additionally, MEB has issued to ANL a new contract for "Degradation of Reactor Pressure Vessel Boundary Components in Concentrated Boric Acid Solutions." This program will conduct experiments to measure corrosion rates and galvanic potentials of low-alloy steel, 308 cladding and Alloy 600 in concentrated boric acid solutions. Other aspects of this program focus on determination of stress-corrosion (SCC) crack growth rates in Alloy 600 and Alloy 182 taken from the Davis Besse reactor head, and development of an integrated SCC growth rate and inspection frequency determination model. Finally, MEB has awarded two smaller programs (a) at the Naval Surface Warfare Center – Carderock, MD to explore innovative non-destructive inspection techniques, and (b) at Metaheuristics, LLC to determine the feasibility of using neural networks to model crack initiation and growth of the available Alloy 600 crack growth rate database.

Thermal-Hydraulic Research Issues Relevant to Advanced Water Reactors

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Currently, the staff is anticipating or is actively reviewing several advanced light water reactor designs: AP1000, ESBWR, SWR-1000, ACR-700, and IRIS. These designs offer potentially significant improvements in safety by taking advantage of prior testing, analysis, and operational experience gained from existing plants. Advanced light water reactor designs generally rely on natural processes to insure that the core remains adequately cooled and containment integrity is maintained in the event of an accident. In some designs, the large break loss of coolant accident (LOCA) is eliminated and the limiting accident scenario becomes a small LOCA or transient.

While these new designs promise significant safety benefits, their unique features create new challenges to thermal-hydraulic analysis. Physical processes such as natural circulation with low driving heads, condensation in the presence of a non-condensable gas, and entrainment and deentrainment play an important role in advanced plant accident scenarios. This paper discusses several thermal hydraulic issues that are important to advanced light water reactors and are being addressed as part of the staff's code development and review effort.

Modeling Issues for HTGR Designs

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Because HTGR designs are substantially different from current LWRs, revised computer codes and new models will be needed to give NRC staff the necessary independent capabilities to realistically predict reactor system response. The development of a suite of validated reactor system analysis (thermal-fluid dynamics, nuclear analysis, and severe accidents and source terms) tools and data will permit the NRC staff to (a) conduct confirmatory analyses in the review of applicants' reactor safety analyses, (b) support development of the regulatory framework by assisting in the identification of safety-significant design basis and licensing basis events and associated success criteria, and (c) conduct exploratory analyses to better understand the technical issues, uncertainties, and safety margins associated with new designs.

Safety Margin Testing: High-Temperature Gas-Cooled Reactor Fuel

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Modular High-Temperature Gas-Cooled Reactors (HTGRs), such as the Pebble Bed Modular Reactor (PBMR) and the Gas Turbine Modular Helium Reactor (GT-MHR) have unique safety features and design characteristics. Foremost among these is the all-ceramic fuel element containing TRISO coated fuel particles (CFPs). The fuel safety concept and intended design characteristic is to contain and retain essentially all radiologically important fission products within the billions of CFPs contained within the fuel elements (e.g., fuel pebbles, fuel compacts) within the core. Effective fission product retention within the CFPs is critical to the HTGR safety case for all licensing basis conditions.

Significant world-wide HTGR fuel irradiation testing and accident condition testing has been conducted to understand the behavior, safety performance, and fission product transport behavior of HTGR fuel during normal operation, design basis accidents, and potential accidents beyond the design basis. However, these programs have focused mainly on showing acceptable fuel performance within the licensing-basis analysis conditions. Significantly less testing has been conducted at operating temperatures, burnups, fast fluence, power levels, and accident conditions that are focused on probing the conditions where increased fuel failures occur to map the safety margins. This has resulted in significant uncertainties surrounding the failure margins for HTGR fuels.

The paper describes the irradiation testing and accident condition testing that is planned to obtain fuel performance data to better establish operating and accident condition fuel safety margins, to assess the acceptability of an applicant's fuel irradiation and accident simulation testing programs, to verify an applicant's claims of fuel performance and fission product release during operations and accidents, and to provide data to develop and validate independent analytical fuel performance tools. These test data are also considered important to support a policy decision on mechanistic source term and modular HTGR application technical reviews.

Materials Research Needs for Advanced Reactors

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Metallic and graphite components in advanced high temperature gas-cooled reactors (HTGRs) may experience creep, fatigue, oxidation, aging, corrosion cracking, irradiation damage, and dimensional changes. The safety design of these reactors, such as the pebble bed modular reactor and the gas-turbine modular helium reactor, depends heavily on the long term integrity of metallic and graphite components needed to maintain pressure boundary integrity, core geometry, adequate cooling of the core, and reactivity control and shutdown systems. Failure of these components could result in air, water, and/or steam ingress and accompanying adverse consequences.

During normal and design basis conditions the HTGR components experience high temperatures (300 to 1400°C for graphite and 300 to 950°C for metallic components), and high fluences (on the order of 2-3 x 10^{22} n/cm^2 for graphite). Although the helium is an inert coolant, the gas stream may contain low levels of impurities, including oxygen. Currently, the design codes and standards for high temperature metallic components are based on knowledge developed from work performed over twenty years ago for liquid metal fast breeder reactor environments and do not adequately reflect the operating environment and degradation mechanisms of HTGR components. Due to the lack of international consensus design codes and standards, locally developed calculations and custom codes have been used for the design of graphite components. Nevertheless, this engineering judgement must be updated and based on data measured on graphites that will be used in proposed reactors rather than on graphites that might have been used previously and are no longer available. Since the irradiated material properties of graphite depend strongly on the particular raw materials and manufacturing process, the technical basis developed for old graphites may not be appropriate for the new graphites to be used in HTGRs.

In most Advanced Light Water Reactors (ALWRs) the operating conditions, materials, and coolant environments are not significantly different from those of conventional LWRs. Because of the similarities in materials and environments, there is not a great need for new research in the materials area specifically for ALWRs. However, a large body of research data, from both the US and Japan, has shown a detrimental effect of the coolant environment in reducing the fatigue life of LWR components. Methods have been developed and are widely available in the literature (NRC NUREG reports and PVRC report) for taking into account the effects of the operating environment in the fatigue design of components. Although the ASME, through its on-going code activities, is addressing the issue of the effects of the environment, it has not yet incorporated changes in its design rules and correlations. Therefore, during design and review of ALWRs, the effects of the environment should be appropriately accounted for in the fatigue design and evaluation of components. ASME should ensure that its rules for fatigue design of components are updated. In addition, two aspects of the HTGR and some ALWR designs raise the potential for the need for improved inservice inspection (ISI) programs and for continuous monitoring. First, more components are enclosed in pressure vessels making access for inspection difficult. Second, there are longer operating cycles between scheduled, short-duration, refueling outages during which ISIs can take place. This demonstrates a need for evaluating effectiveness of less frequent and possibly reduced scope ISIs for timely detection of cracking and degradation of components and the potential for excessive growth of cracks before the next ISI. An alternative to conducting periodic in-service inspections during reactor shut-downs is to conduct continuous on-line, nondestructive monitoring for structural integrity and leakage detection of the entire reactor or of reactor components during operation. Techniques for continuous monitoring have been developed, validated and codified for use in LWRs. If ISIs of HTGRs and ALWRs are found to be inadequate, then continuous monitoring may become necessary. The continuous monitoring techniques need to be evaluated and validated for the materials, environments, and degradation mechanisms of the HTGRs and ALWRs.

Research and developments in the following key areas are necessary to develop a better understanding of materials behavior and applications under reactor operating and accident conditions. These are: (1) national codes and standards for design and fabrication of metallic and graphite components for service in HTGR high temperature helium environments; (2) appropriate data bases for calculating fatigue, creep, and creep-fatigue lifetimes of components in high temperature applications; (3) understanding the effects of impurities, including oxygen, in the high-temperature helium on degradation of components; (4) sensitization and aging behavior of alloys during elevated temperature exposures; (5) degradation by carburization, decarburization, and oxidation of metallic components in HTGRs; (6) issues related to inspection of advanced reactor components; (7) long term performance and degradation of graphite and new reactor pressure vessel materials under high levels of irradiation; (8) modeling and methodology that predict irradiated properties of graphite from non-irradiated properties; and (9) comprehensive understanding of the governing rates and mechanisms for the oxidation of graphite.

To conduct independent probabilistic risk assessments of advanced reactors, information will be needed on the probability of failure of various reactor components. Because of the lack of operating experience, this information may be developed analytically using probabilistic fracture mechanics. To do this, potential degradation mechanisms of metallic and graphite components need to be identified and the progression of degradation quantified under the operating reactor conditions.

Two research programs are underway to establish technical bases regarding the behavior of metallic and graphite components in HTGRs. The objective of these programs is to review and evaluate currently available (international) codes, standards and procedures that could be used in design of HTGRs and to review and evaluate materials to be used in HTGR environments. Materials that have been used in HTGRs and those to be used will be identified, as well as the conditions that may affect material properties such as environment and geometry. Following review and evaluation, the best codes, standards, and procedures will be updated to incorporate correlations and models developed from more recent research. This review will evaluate the adequacy of any current code design rules and procedures and provide input for improvements as necessary. In addition, a materials specification is being developed for nuclear grade graphite. Research on the environmental effects is being planned and will begin in 2003.

Due to the limited experience and test results with HTGRs and the extensive experimental testing program needed to develop reliable technical bases for the behavior and expected lifetime of metallic and graphite components in HTGR environments, international cooperative programs are being pursued. Such international efforts aims to bring together research results on understanding and analyzing materials performance and component integrity in the operating environments of HTGRs. This paper will discuss the information gaps that exist in terms of analytical tools and data shortcomings, and describe research needed to establish an acceptable technical understanding of the behavior of metallic and graphite components in advanced reactors.

Generation IV Roadmap Research and Development Plan

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To advance a new generation of nuclear energy systems, a Technology Roadmap, lead by the Department of Energy, defines and plans the necessary R&D to support a generation of innovative nuclear energy systems known as "Generation IV". Generation IV nuclear systems comprise the nuclear reactor as well as fuel cycle facilities for the entire fuel cycle from cradle to grave.

The organization and execution of the Roadmap were the responsibility of a Roadmap Integration Team (RIT) that was advised by the Subcommittee on Generation IV Technology Planning of the Nuclear Energy Research Advisory Committee. An Evaluation Methodology Group (EMG) was formed to develop a process to systematically evaluate the potential of proposed Generation IV nuclear energy systems to meet the Generation IV goals. A solicitation was issued worldwide, requesting that concept proponents submit information on nuclear energy systems that they believe could meet some or all of the Generation IV goals. Nearly 100 concepts and ideas were received from researchers in a dozen countries.

Technical Working Groups (TWGs) were formed – covering nuclear energy systems employing water-cooled, gas-cooled, liquid-metal-cooled, and nonclassical reactor concepts – to review the proposed systems and evaluate their potential using the tools developed by the EMG. The TWGs conducted an initial screening, termed "screening-for-potential," to eliminate those concepts or concept sets that did not have reasonable potential for advancing the goals, or were too distant or technically infeasible. Following the screening-for-potential, the TWGs conducted a final screening to quantitatively assess the potential of each concept or concept set to meet the Generation IV goals. The following six systems were selected as Generation IV systems: Gas-Cooled Fast Reactor, Lead Alloy-Cooled Reactor, Molten Salt Reactor, Sodium Liquid Metal-Cooled Reactor, Supercritical Water-Cooled Reactor, and Very High-Temperature Reactor.

Many Generation IV systems involve substantial changes in safety-system design and implementation that require licensing implementation significantly different from current experience. This paper focuses on the six selected Generation IV systems and the required research and development necessary to successfully deploy these systems by 2030.

LOCA Ductility Tests

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Safety analyses for loss-of-coolant-accidents (LOCA) address several phenomena related to the behavior of fuel rod cladding: (a) ballooning deformation, (b) conditions for bursting, (c) oxidation kinetics, and (d) embrittlement. The first three are described by correlations, and cladding embrittlement is addressed by criteria in 10 CFR 50.46. These embrittlement criteria currently consist of a 17% limit on cladding oxidation and a 2200°F (1204°C) limit on cladding temperature.

A program of LOCA testing is being performed at Argonne National Laboratory (ANL) for the NRC in cooperation with the Electric Power Research Institute, Framatome ANP, and the Department of Energy. The original motivation for the LOCA testing at ANL was to look for burnup effects on the embrittlement criteria, with burnup effects on ballooning, bursting, and oxidation as secondary interests. More recently, proposals have been made to replace the Zircaloy-based 17% and 2200°F limits with a performance-based requirement in 10 CFR 50.46 to avoid the need for regulatory exemptions when new alloys are introduced and to accommodate any burnup effects. These current numerical limits were derived from ductility tests, so the proposal included the substitution of some suitable ductility test [1,2].

After considering several possibilities, it has been decided to continue investigating the ring-compression test as the potential performance-based ductility test for 10 CFR 50.46, provided that it can be confirmed to be adequate. Ring-compression tests are less expensive to perform than the alternatives, and because such tests were used to develop the original embrittlement criteria, their continued use should contribute to regulatory stability. Two basic questions of adequacy will be addressed in the current research program. One is about our ability to interpret the results of ring-compression tests unambiguously, and the other is about the efficacy of a test on a small ring specimen to represent the behavior of a fuel rod in a ballooned and ruptured region.

Although more costly, a three-point bend test is probably better in several respects than a ring-compression test. The first advantage of this test is that the tensile loads are applied in the axial direction rather than in the circumferential direction. This is probably more representative of stresses that might arise from horizontal accelerations (earthquakes), plant vibrations, and spacer grid interactions. Furthermore, the load-vs-deflection curve for this test is simple and easy to interpret. If the ring-compression test and the three-point bend test both show the same critical cladding oxidation level for the same material, then we can use the less expensive ring-compression test.

Four-point bend tests on segments containing a ballooned and burst region will also be performed in the ANL program. This test, with fuel pellets inside, is most prototypical for investigating the behavior of the ballooned region of a fuel rod. Double-sided oxidation will take place as appropriate, with steam entering through the burst opening. Any enhanced hydride absorption due to inside oxidation will be present. Loading points are away from the deformed region, and the specimen will break naturally at its weakest location. While this is clearly the most expensive test, it only needs to be used in a confirmatory way. If results from the ring-compression tests can be applied in the ballooned region, and if those results adequately predict ductile or brittle behavior, then the ring-compression tests will have been confirmed.

The ring-compression tests, hopefully confirmed by three-point bend tests, and the four-point bend tests will be integrated into the overall LOCA test program. Unirradiated tubing, as received, will be tested first. Irradiated cladding, as it becomes available, and hydrogen-charged tubing will be tested later to investigate burnup effects. Two or more alloys will be oxidized together in the same furnace to reduce the number of furnace runs needed to produce ring specimens (and three-point bend specimens, if necessary). Oxidation kinetics can also be obtained from these furnace runs, and examination of ring fragments after compression will give fracture morphology and oxygen content. Specimens for the four-point bend tests will consist of those specimens that survive thermal shock in integral tests. After optical profilometry, those specimens will be tested in the four-point bend apparatus and will likely break. Metallography and hydrogen measurements can be made on fragments after the bend tests. Schematic diagrams of the test procedures and the test matrix have been developed and will be described in the paper.

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Understanding LOCA-Related Ductility in E110 Cladding

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Numerous investigations performed in recent years have shown that cladding alloys based on Zr-Nb are the most advanced materials for achieving fuel cycles with high burnup in light water reactors.

The operating experience of VVER-type Russian reactors employing the E110 alloy (Zr-1%Nb) for the fuel rod claddings indicates that material of this type (in contrast to the alloy of the Zircaloy type) allows the cladding to retain ductility at fuel burnup up to 60 MWd/kgU, inclusive. Moreover, according to the results of in-pile tests, the existing ductility margin proved to be sufficient to retain a high failure threshold with no fragmentation (>130 cal/g) under reactivity-initiated-accident (RIA) conditions. Besides, experiments of the "thermal shock" type performed in A.A. Bochvar All-Russian Research Institute of Inorganic Materials to validate the mechanical behavior of oxidized irradiated E110 cladding under loss-of-coolant-accident (LOCA) conditions have demonstrated that the threshold of its fragmentation corresponds to the license safety criteria currently in force (1200 C, 18%) with a reasonable margin.

Nevertheless, an assessment of experimental methods to validate the mechanical behavior of oxidized Zr-Nb cladding under the LOCA conditions has become the subject of a broad international discussion launched within the context of such issues as the reassessment of the safety criteria, the representativity of different types of tests for the validation of mechanical behavior of fuel rods under the accident conditions, and so on.

In this case, a detailed consideration of results of previously performed research with Zr-1%Nb alloys has shown that more thorough mechanical tests should be carried out to examine the zero ductility threshold of oxidized E110 cladding. An appropriate program was developed by Russian Research Center "Kurchatov Institute" in cooperation with Russian State Research Center "Research Institute of Atomic Reactors" with the support of Joint Stock Company "TVEL" (Russian Federation), US Nuclear Regulatory Commission (USA), and Institute for Radiological Protection and Nuclear Safety (France).

Ring compression mechanical tests were selected as the basis for the first stage of the work because this approach has a good historical tradition and offers the prospect of direct comparison of results as a function of cladding material and oxidation parameters.

The developed program of oxidation and ring compression tests included two subprograms:

1. Determination of the zero ductility threshold of the cladding versus such parameters of the oxidation scenario as heating and cooling rates under the following fixed conditions:

- material of cladding (E110 unirradiated tubes);
- double sided oxidation with steam at 1100 C;
- ring compression tests at 20 C.

2. Determination of the sensitivity of the zero ductility threshold for a fixed combination of heating and cooling rates to the following parameters:
   - material of cladding (E110, E110K, E635, Zry-4);
   - temperature of oxidation (1000–1200 C);
   - temperature of mechanical tests (20-300 C);
   - irradiation of cladding (unirradiated E110 claddings and refabricated irradiated E110 claddings from commercial fuel rods with burnup ~50 MW d/kg U).

Results of the first subprogram presented in Fig. 1 show the following:

- the transition of the oxidized cladding from a high ductility state to an embrittlement one happens suddenly in the narrow range of the ECR (Equivalent Cladding Reacted);
- critical values of the ECR (7.6-9.2) corresponding to those of the zero ductility threshold are relatively independent of the combination of heating and cooling rates.

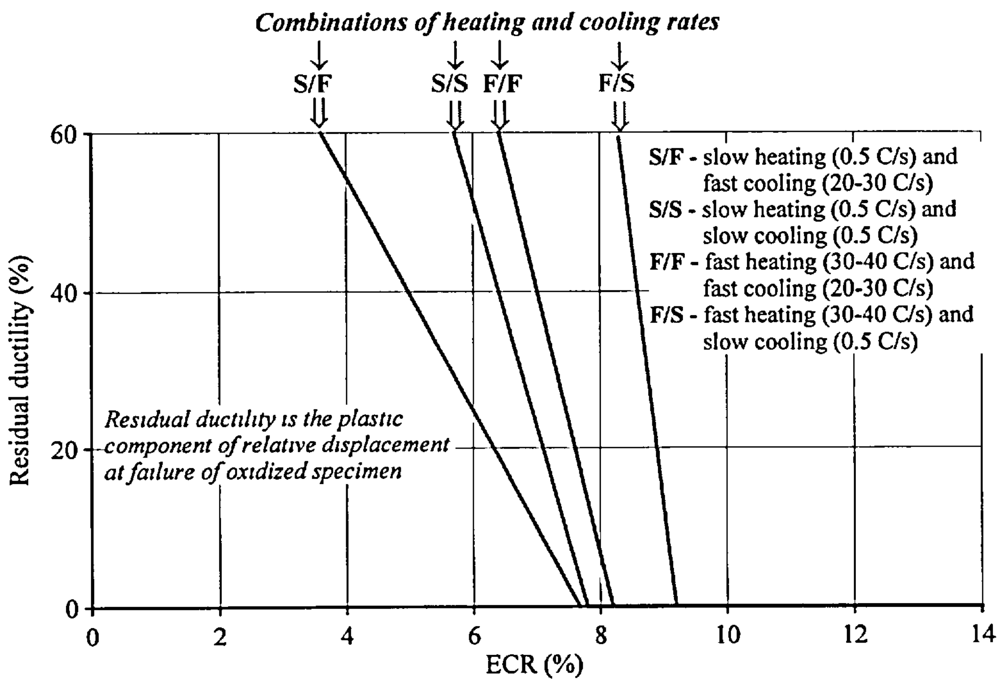

Fig. 1. Residual ductility of unirradiated E110 cladding oxidized at 1100 C as a function of ECR for various heating and cooling rates.

Further research performed within the context of both subprograms revealed the following:

- typical consequences of the breakaway effect were visually observed on the surface of E110 cladding in the range close to critical values of the ECR;
- hydrogen concentration in the E110 oxidized cladding achieved 700 ppm under zero ductility conditions.

The comparative analysis of E110 and Zry-4 claddings confirmed that:

- differences in alloy composition of these two alloys determined different oxidation behavior of E110 and Zry-4 claddings (see
- Fig. 2);
- the zero ductility threshold of Zry-4 oxidized cladding was about 13% (ECR as measured).

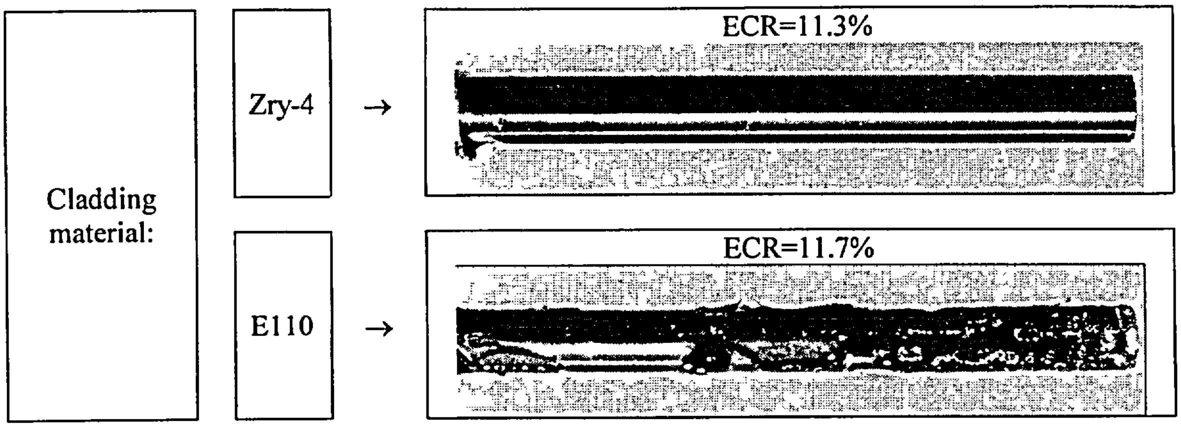

Fig. 2. Appearance of E110 and Zry-4 claddings after the oxidation at 1100 C.

To clarify the effect of alloy composition and oxidation conditions, the next part of the program was focused on the comparative tests with E635 cladding, E110K cladding (increased oxygen concentration), and one-side oxidized E110 cladding (for the comparison with the published data on M5 ring compression test results). This part of the work is in process now, but the results already obtained allow to note the following:

- significant differences were not seen in the mechanical behavior of E110 and E635 oxidized claddings;
- one-side oxidation of E110 cladding led to an increase of the zero ductility threshold up to 12% ECR (from 8.2% for double- side oxidation).

Separate important lines of the program were devoted to studies of the sensitivity of the E110 oxidized cladding mechanical behavior to irradiation, temperature and the type of mechanical tests.
These studies have demonstrated that:

- the zero ductility threshold of irradiated E110 oxidized cladding is not less than the embrittlement threshold of the unirradiated cladding;
- an increase in the temperature of ring compression tests from 20 C up to 135 C results in an increase in the critical value of ductility up to an ECR=12% (apparently, due to the temperature influence over the hydride behavior);
- special reference tensile mechanical tests performed with the E110 oxidized cladding confirmed that:
  - there are no contradictions between the zero ductility thresholds determined in compression tests and tensile tests;
  - the zero ductility threshold is a function of the temperature of the mechanical tests.

Thus, preliminary results of this program reveal several factors that are important for understanding LOCA-related ductility of the E110 cladding. But additional work must be completed to obtain the experimental database that is necessary to reach conclusions on this issue.

LOCA Research Results for High-Burnup BWR Fuel

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Argonne National Laboratory (ANL) is conducting research on high-burnup BWR and PWR fuel to provide data for assessing the licensing criteria (10 CFR50.46) for Loss of Coolant Accident (LOCA) events. LOCA-relevant research includes fuel and cladding characterization, cladding high-temperature steam oxidation kinetics studies, LOCA Integral Testing of fueled segments, post-quench ductility testing of LOCA Integral specimens and post-quench ductility testing of Zircaloy and advanced-alloy unirradiated tubing. The work completed on samples from Limerick BWR fuel rods (≈57 GWd/MTU) is reported.

The LOCA licensing criteria (10 CFR50.46) limit peak cladding temperature to 2200°F (1204°C) and maximum Equivalent Cladding Reacted (ECR) to 17% to ensure adequate ductility during the Emergency Core Cooling System quench and during possible post-LOCA events (e.g., seismic). High burnup phenomena that may affect cladding response during ballooning and burst, steam oxidation, water quench and post-quench are: loss of cladding base metal thickness due to oxidation, hydrogen pickup, inner-surface oxide-layer formation, decreased fuel permeability and tight fuel-cladding bonding, and the effective thickness and chemistry of the prior β-phase layer following steam oxidation. The LOCA Integral Tests are being conducted with high burnup fueled cladding segments in order to include all the phenomena highlighted.

Limerick cladding is Zr-lined Zircaloy-2 (Zry-2) from the GE-11 9×9 assembly design. The in-reactor-formed outer-surface oxide layer is ≈10 μm. Axial variation of layer thickness is minimal for test-sample regions compared to the circumferential variation (3-18μm). The inner-surface oxide layer is ≈10-15 μm. Oxygen and hydrogen contents are ≈0.7 wt.% and ≈70 wppm, respectively.

Defueled cladding samples have been exposed to high-temperature steam to determine weight gain, ECR, and layer thicknesses as functions of time at temperature. As reported (NSRC-2001 Proceedings), 21 tests have been conducted (1000-1200°C) using unirradiated Zry-2 (9 tests) and irradiated Limerick Zry-2 (12 tests). Weight gains deduced from detailed metallographic analysis of the 1200°C samples are consistent (within 5%) with the Cathcart-Pawel (CP) model predictions. Based on an assessment of the databases for Zry-2, Zry-4, ZIRLO and Zr-1%Nb alloys, these cladding materials exhibit about the same weight gain kinetics in a steam environment at 1100-1500°C, consistent with the ANL data and the CP model predictions. Detailed metallographic analyses are in progress to determine the weight gain kinetics and layer thicknesses of the ANL Zry-2 samples tested at 1000°C and 1100°C.

The CP model has been used to plan the LOCA Integral Test times-at-temperature to achieve desired ECR values. The tests have the following sequential steps: stabilization of temperature, internal pressure (≈8.7 MPa) and steam flow at 300°C, temperature ramping (5°C/s) through ballooning and burst to 1204°C, hold at 1204°C in flowing steam for 3-10 minutes, slow-cooling (3°C/s) to 800°C, and initiation of water quench at 800°C. Four-point bend tests will be used to determine overall specimen ductility. Ring compression tests will be used for local ductility determination. The fueled LOCA samples are

$\approx$300-mm long with a 13-mm-long top plenum connected to a He gas line ($\approx$10 cm^3). The temperature variation in the middle 125 mm of the sample is ±15°C for a 1204°C control temperature.

The LOCA Integral Test Apparatus was built and tested out-of-cell using archival Zry-2 samples filled with loose-fitting zirconia pellets. For the reference conditions, unirradiated Zry-2 bursts in the alpha phase at $\approx$740°C and a ΔP of $\approx$8.3 MPa with a peak burst strain of $\approx$44%. A second apparatus of the same design was built and installed into an Alpha-Gamma Hot Cell Facility workstation. The in-cell and out-of-cell units share the same instrumentation and control system and can perform oxidation-kinetics and LOCA Integral tests.

Because of the interest in high-burnup fuel permeability, ballooning, burst and possible fuel relocation, the first series of LOCA Integral Tests are being conducted as follows: A) room temperature and 300°C pressurization to quantify fuel permeability, followed by ramping to burst; B) full LOCA sequence up to the cool-down to 800°C, followed by slow furnace cooling; and C) full LOCA sequence including water-quench initiation at 800°C. Test A has been completed. Relative to unirradiated Zry-2, the results (see Fig. 1) indicate that peak ballooning strain ($\approx$38% peak strain), the burst temperature ($\approx$755°C) and ΔP ($\approx$ 8.6 MPa), burst length and maximum width, and the decrease in internal depressurization rate from $\approx$9 MPa to $\approx$3.4 MPa are all comparable. The depressurization time ($\approx$50 s vs. $\approx$2 s) from 3.4 to 0 MPa is longer for the irradiated sample. Also, the balloon axial span is about half as long and the burst-opening shape is different (oval vs. dogbone) for the irradiated sample. Additional post-test examinations of the Test-A sample are in progress, as well as the running of Tests B and C. Results from all three tests will be presented.

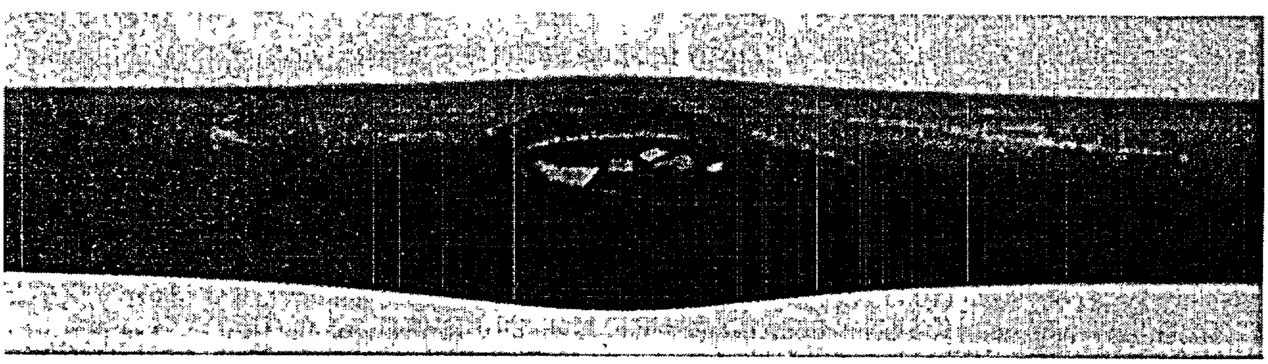

Fig. 1. Ballooned and burst section of the Limerick high-burnup BWR LOCA Integral Test sample. Burst temperature, pressure and peak ballooning strain are $\approx$755°C, 8.6 MPa, and 38%, respectively. Burst length is $\approx$15 mm and balloon length is $\approx$50 mm.

Characterization of High-Burnup PWR and BWR Rods

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In support of a range of research programs relating to light water reactor fuel performance, high-burnup PWR and BWR rods and dry-cask-stored PWR rods were acquired recently under the sponsorship of the U.S. Nuclear Regulatory Commission, the U.S. Department of Energy, and the Electric Power Research Institute. Since the as-irradiated condition of these fuels is the prerequisite for test planning and evaluation, in-depth characterization of these fuels was undertaken. Equally important, these data could provide valuable input for burnup extension and dry-cask licensing renewal.

The fuels examined include PWR 15x15 fuel at 67 GWd/MTU burnup (73 GWd/MTU peak pellet) from the H. B. Robinson plant, BWR 9x9 fuel at 56 GWd/MTU burnup (64 GWd/MTU peak pellet) from the Limerick plant, and PWR 15x15 fuel at 36 GWd/MTU (40 GWd/MTU peak pellet) from the Surry-2 plant after 15-y storage in a Castor-V/21 dry cask. The cladding for the H. B. Robinson and Surry fuel is Zircaloy-4 and that for Limerick fuel is Zr-lined Zircaloy-2. In the characterization, significant effort is given to assess the condition of the cladding.

The overall condition of the H. B. Robinson rods examined appears to be sound, in spite of the high burnup. Fission gas release fraction was $<\approx 5\%$ based on poolside Kr^{85} scans. The thicknesses of cladding OD oxide measured with optical metallography are 70 and 98 µm at axial elevations of 0 and 0.7 m above fuel midplane, respectively, for Rod A02. These values corroborated well the poolside eddy-current measurement results. Although the oxide contains numerous circumferentially oriented microvoids, spallation appears to be minimal. Hydrides in the cladding form a dense band adjacent to the OD oxide. Away from the OD, the density of hydrides diminishes with distance. The hydride platelets are mostly circumferentially oriented. Hydrogen contents of 580 and 750 wppm were measured at the 0 and 0.7 m axial elevations, corresponding to uptake percentages of 21 and 23%, respectively. Fuel/cladding gap is closed with little ID corrosion of the cladding. Thickness of rim fuel is ≈600 µm, based on optical data.

Fission gas release in the Limerick rods ranges from 5 to 17%. The relatively high release may be related to the numerous microtears found in the fuel; such tears promote permeability with the rod plenum. Fission product deposits can be found in the now closed fuel/cladding gap. More substantial deposits are often located at the end of major radial fuel cracks, suggesting a vapor transport mechanism of the fission products down the temperature gradient. In spite of the deposits, reaction between the fission products and the cladding's Zr liner is modest. The OD oxide in the Limerick rods is thin, ranging from 5 to 25 µm circumferentially with an average of ≈10 µm. Tenacious crud was found at some locations but the thickness is only ≈5-10 µm and it varies inversely with the oxide layer thickness. Owing to the small oxide thickness, the density of hydrides in the Limerick cladding is low, with some platelets precipitated in the Zr liner. As in the case with the H. B. Robinson PWR fuel, the general condition of the Limerick BWR fuel also appears to be sound.

The Surry fuel rods were examined after 15-y storage in a dry cask. Elevated cladding temperature, up to ≈415°C for several days, resulted from the cask thermal benchmark tests prior to the long-term storage. In the post-storage characterization of the rods, little evidence of deleterious effects, such as additional fission gas release or cladding creep, could be discerned. While the hydrogen contents in the cladding at near the midplane were as expected based on the oxide thickness data, that at a higher elevation appears to be low. Because the axial temperature distribution in the cask had a chopped cosine profile with the steep gradient away from the axial center, a plausible explanation could be the diffusion of hydrogen in the cladding towards the colder ends. Additional mapping of the hydrogen-content profile is needed to determine if axial migration is significant.

Oxidation of Zirconium Alloys in High Pressure Steam and Some Results under Atmospheric Pressure

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In order to decrease fuel-cycle costs and to improve reactor operation and spent fuel management, nuclear operators want to increase fuel discharge burnup. Due to Zircaloy-4 limitations at high burnup, PWR fuel vendors have developed and proposed the use of new zirconium alloys, such as M5™, ZIRLO™.

Despite the fact that worldwide used U.S. Regulatory Guide (RG) 1.157[1] §3.2.5.1 recognizes the effect of steam pressure, performance of these alloys under intermediate breaks loss-of-coolant-accident (LOCA) and higher-pressure accidental transients is poorly understood. The main first part of this paper will evaluate the database for fresh Zircaloy-4 and another alloy, reported in literature mainly after the RG 1.157 was issued[2-5]. According to plant calculations, the worst intermediate break seems to be the 3-inches break, it is characterized by a temperature above 800°C during several hundred of seconds, at a pressure between 30 and 40 bars.

Four different consistent data for fresh Zircaloy-4 show a pressure enhancement effect below 1100°C. Tests with steam/argon mixtures show that it is an effect of partial steam pressure rather than of total pressure. The consistency between results under flowing steam and stagnant steam seems to show no steam flow effect, probably the high pressure ensures good natural circulation. For fresh Zircaloy-4, the maximal rélative effect occurs at 750-800°C, it coincides with temperatures at which occurs the atmospheric breakaway oxidation at longer times[6]. As for this atmospheric breakaway, the enhancement seems to be related to the tetragonal to monoclinic zirconia transformation. According to the literature, this transformation is influenced by several factors (compressive stresses, crystallite size growth, substoichiometry). For fresh Zircaloy-4 under intermediate breaks, the enhancement is not high enough to cause an actual safety problem.

Limited published data for fresh E-110™ alloy (Zr 1%Nb O-poor S-free) show, comparatively to Zircaloy-4, an increase of the temperature at which the maximal relative effect occurs[4], as for the atmospheric breakaway oxidation at longer times[7]. The enhancement is especially strong at 850°C, rapidly exceeding the embrittlement criterion under only 40 bars.

Tests with M5™ alloy (Zr 1%Nb O-rich S-doped) are under preparation and will start in 2003 in France.

No test exists for high-burnup Zircaloy-4. As there is in the literature a possible role of hydrogen on the tetragonal to monoclinic zirconia transformation, tests are necessary.

The smaller second part of this paper will be a follow-up of the 28th WRSM paper[8]. Some meetings and letters show that some fuel vendors have still difficulties to understand the tie between the worldwide used 17% ECR criterion and the Baker-Just correlation. In figure 8 of the 28th WRSM paper, it was shown how the 17% ECR value was calculated with the Baker-Just correlation. In this paper, a simulation will be made to calculate after Hobson's data what would have been the ECR value, if the U.S. Regulatory Staff had used in 1973 the Cathcart-Pawel correlation.

Finally, as Cathcart and Pawel's data are used in the CATACOMB module of the French best-estimate CATHARE code, issued from the CUPIDON code [9], and mentioned in both RG 1.157 and U.S. Research Information Letter 0202[10], some characteristics of Cathcart and Pawel's data will be pointed out in this paper.

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Investigation of the Impact of In-Reactor Short-Term Dry-Out Incidents on Fresh and Pre-Irradiated Fuel Cladding

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Light water reactor cores may be subjected to thermal-hydraulic transients resulting in inadequate core cooling for short periods of time. In BWRs, the result is a short term dry-out at the fuel rod surface leading to a transitory temperature increase of the cladding. The transient is terminated when adequate cooling is resumed, which results in the overheated fuel rods being quenched. It is a safety requirement that after such an event reasonable fuel performance be maintained up to the subsequent shutdown. In order to assess post dry-out and quench fuel performance, it is necessary to know what effect such transients have on the microstructural and mechanical properties of irradiated Zircaloy cladding. To this end a series of dry-out experiments were carried out at the OECD Halden Reactor Project.

An instrumented fuel assembly (IFA), connected to a light water BWR loop within the Halden reactor, was designed for in-pile dry-out testing. The main feature of the rig was that it comprised three individual flow channels, each able to contain one instrumented test rod, allowing for individually controlled dry-outs to be performed. Two fresh fuel segments (Zry-2 and Zry-4) and six segments pre-irradiated to 22-40 MWd/kgU (Zry-2, Zyr-2 with liner and Zry-4) were fabricated into test rods. Each was fitted with 2 or 3 Cr/Alumel thermocouples to monitor clad surface temperature during the dry-out events together with elongation detectors. The test rods were exposed to reduced or no-flow conditions and once dry-out was achieved and the target temperature (650 or 750°C) as indicated by the upper clad thermocouple was exceeded; the rod was quenched. If time at target temperature was not long enough, further dry-out events were initiated until sufficient accumulated time above the target temperature was reached.

Poor thermal contact between the thermocouple and clad outer surface for the first rods tested, led to these rods being more severely tested than planned, in terms of both accumulated time in dry-out and peak temperature reached. These rods developed maximum peak clad exposure temperatures (PCTs) estimated to be in the range 950-1200°C. This was not the case for the final rods tested, for which the thermocouple attachment had been re-designed, which developed PCTs of 750 - 850°C.

The surface condition and dimension of each fuel segment were assessed post dry-out, followed by destructive examination to investigate clad microstructure and mechanical properties. Thermal hydraulic calculations carried out prior to the in-pile testing had indicated that the dry-out transients induced in the test would result in an axial temperature profile in the cladding: a PCT plateau over the upper region with a steeply decreasing temperature gradient over a "transient zone" down to the loop saturation temperature (285°C) over the bottom region of the test rod. Initial examination of the severely tested rods clearly suggested this to have been the case. Whilst the lower regions of the rods exhibited smooth, adherent grey/brown oxide the upper regions exhibited severe surface oxidation and spalling with clad collapse into pellet-pellet interfaces. Clad sections taken from the upper regions showed the clad had undergone α to β phase transformation during the transient with the observed microstructure consisting entirely of quenched, former β-phase grains with an increased hydrogen content. The microstructure just below this "peak dry-out zone" was a mixture of large α-grains and former β-grains. Further down towards the bottom of each rod the microstructure was exclusively equi-axed α-phase grains with no grain growth.

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The results from room temperature tensile testing varied in accordance with this observed axial variation in microstructure. The "unaffected zone" at the bottom of the rods showed high UTS and intermediate ductility (see Figure 1(a)). Moving upwards the UTS showed a sharp drop in value accompanied with an increase in ductility. This trend continued until a maximum in the ductility coincided with a minimum in the UTS (see Figure 1(b)). Further up the rod, the ductility decreased to below the value for the non-affected zone, eventually being practically zero in the peak temperature dry-out zone at the top of the rods. However, despite the severe in-pile dry-out testing received by the first six rods, they did not fail in-pile, either during the quench or the subsequent month of normal reactor operating conditions.

Test rods, both pre-irradiated and fresh, that experienced less severe transients (PCTs of 750-850°C) only exhibited a significant improvement in room temperature clad ductility in the dry-out zone, where a small α-phase grain structure was retained throughout the transient testing.

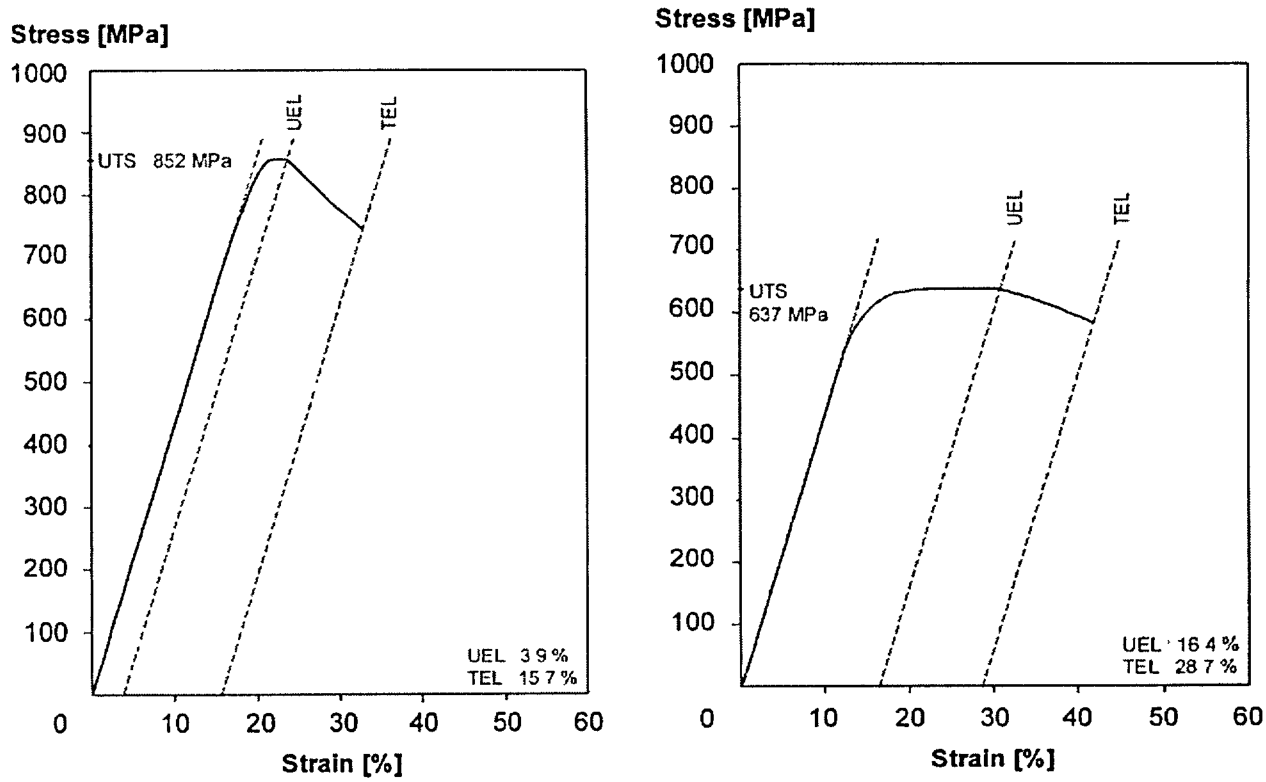

Figure 1. Stress strain curves generated from room temperature tensile testing of clad from a fuel rod exposed to in-pile dry-out from (a) unaffected zone and (b) dry-out zone (PCT<850°C).

US NRC – PSU Rod Bundle Heat Transfer Program

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The US Nuclear Regulatory Commission and the Penn State University have been developing a new experimental facility which will provide needed and new data on heat transfer and two-phase flow behavior in rod bundles to support the best estimate methods and modeling. The Rod Bundle Heat Transfer (RBHT) Test Facility, which is located at University Park, PA, has been constructed for the expressed purpose of improving the analytical modeling capability of the US Nuclear Regulatory Commission such that best-estimate thermal-hydraulic calculations can be made, with confidence, with reduced uncertainty.

Within the design basis for Pressurized Water Reactors (PWR) and Boiling Water Reactors (BWR), the most challenging accident that must be analyzed is the large break, Loss-of-Coolant-Accident (LBLOCA). The LBLOCA is a postulated accident which is not expected to occur, but is analyzed since this particular accident establishes the minimum flow requirements for the Emergency Core Cooling Systems which are designed for either a PWR or BWR to mitigate the consequences of such a transient. The RBHT Facility was designed to perform several different experiments, all of which would help characterize one or more components believed to be important for the modeling and understanding of dispersed flow film boiling in rod bundles. The test facility is a once-through flow facility in which either water or steam can enter the lower plenum and flow upward through the rod bundle. There are liquid collection tanks that are attached to the upper plenum which measure the entrained liquid flow that is carried out of the bundle by the steam. A centrifugal two-phase separator downstream of the upper plenum which also acts to separate out the liquid flow from the vapor flow such that the vortex flow meter at the exit of the steam pipe will measure single phase vapor flow. Separating the exit flows from the bundle provides a method to perform a transient mass balance as well as an energy balance on the facility. The facility has been designed to perform forced reflood tests, with liquid injection into the lower plenum, steam cooling experiments with steam injection into the lower plenum, and steam cooling experiments with droplet injection to simulate dispersed flow film boiling.

There are five pairs of large windows on opposite sides of the housing such that video cameras can be used to film the reflood process as well as using a laser illuminated digital camera system which is used to measure the entrained drop velocity and drop diameters within the rod bundle upstream and downstream of the spacer grids. A laser illuminated digital camera system, developed by Penn State and Oxford Lasers Inc., is being used to measure the entrained drop size and velocity at different axial positions within the bundle in the dispersed flow film boiling regime. The laser system provides back lighting for the very sensitive camera (1000 by 1000 pixels). Drop diameter and droplet velocity distributions are then measured for different time intervals during the transient. Measurements are taken up-stream of a spacer grid as well as downstream to observe the drop shattering effects of the grids and the resulting change in the drop size distribution and mean drop diameters.

The electrical heater rods that are used in the experiments have an outside diameter of 9.5-mm (0.374-inches) and represent a portion of a typical 17 x 17 fuel rod array. The axial power shape represents a top skewed power shape. There are 45 heated electrical rods and four unheated Inconel support rods in the corners of the test assembly which are used to support the spacer grids as well as to bring out instrumentation which is located within the bundle. There are approximately 500 channels of instrumentation for the facility.

In the Rod Bundle Test Facility, improved sub-channel instrumentation was used to measure the vapor temperatures within the rod bundle to detect the presence of superheated vapor in the presence of entrained liquid droplets. The degree of non-equilibrium in the flow depends on the amount of droplets that are entrained and the mixing and heat transfer between the droplets.

To measure the sub-channel vapor superheat, two types of miniature thermocouple probes were used. One type of miniature thermocouple probe was suspended from the spacer grids and would face into the flow. The thermocouples were 0.38 mm in diameter and were supported by 2.44 mm Inconel tubes where were tack welded to the spacer grids. The second vapor superheat measurement uses a traversing thermocouple rake consisting of three, 0.38 thermocouples which were attached to a thin piece of Inconel shim stock and to a tube (which was out of the flow stream) that could be moved to different radial positions within the bundle. The probe could be positioned to measure vapor temperatures in the center of the sub-channel or points in between the sub-channel centers. Figure 1 shows a comparison of the vapor temperature measured during reflood with the traversing probe at the subchannel center and a second identical test with the probe in the gap between the heater rods. The vapor temperature in the gap is almost 200°F higher than the center of the subchannel.

The ability of a best-estimate computer code to accurately predict the integrated effects of the individual phenomena for dispersed flow film boiling is a challenge. Spacer grids have been shown to have a first order effect on the dispersed flow film boiling heat transfer in rod bundles. Therefore, a best estimate computer code must have an explicit model for spacer grids. The effect of spacer grids on the heat transfer and the temperature distribution on the heater rods is seen in Figure 2. The spacer grids improve the heat transfer downstream of the grid. The current focus of the experimental program is to provide improved data for dispersed flow film boiling in prototypical rod bundle geometries. To that end, prototypical spacer grids have been designed, instrumented and tested in the facility. The initial results indicate that the spacer grids can significantly improve the dispersed flow heat transfer downstream of the grids by a combination of increased turbulent mixing as well as shattering of the entrained droplets in the highly non-equilibrium dispersed two-phase flow.

Figure 1

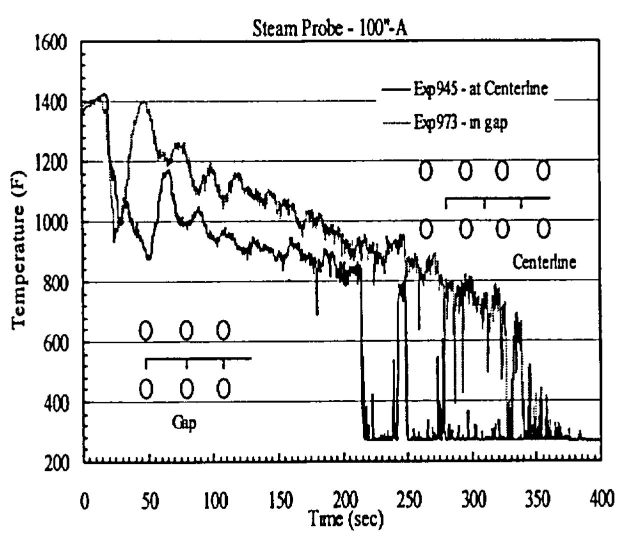
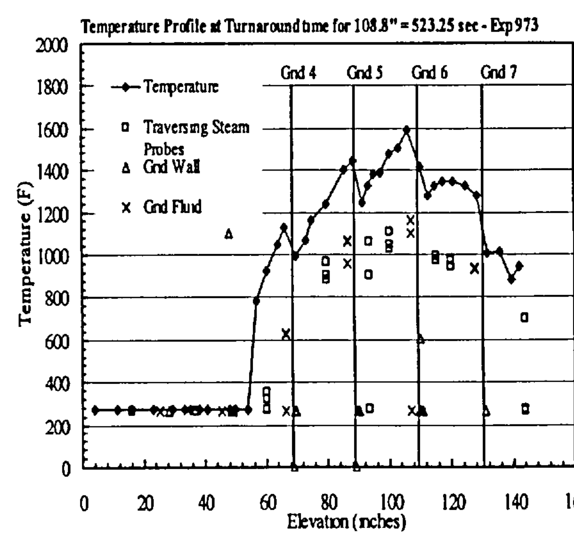


Figure 2



Main Outcomes from the PATRICIA Program on Clad-to-Coolant Heat Transfer During RIAs

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Abstract

In the frame of the studies on Reactivity Initiated Accidents (RIA), IRSN, with the support of EDF, has initiated an experimental program in order to investigate the clad-to-coolant heat transfer under fast transients.

This program has been carried out in the PATRICIA loop of CEA, using single cladding tubes centered in an annular channel with radial dimensions representative of a PWR sub-channel. The clad is heated up by direct Joule effect and the transients are representative of the heating rate induced by the fuel during a neutronic pulse.

The clad-to-coolant heat-transfer is estimated by the measurement of the clad inner temperature and by the calculation of the temperature field within the clad thickness. The uncertainty range is ± 30°C on the clad outer temperature and ± 25% on the clad-to-coolant heat flux.

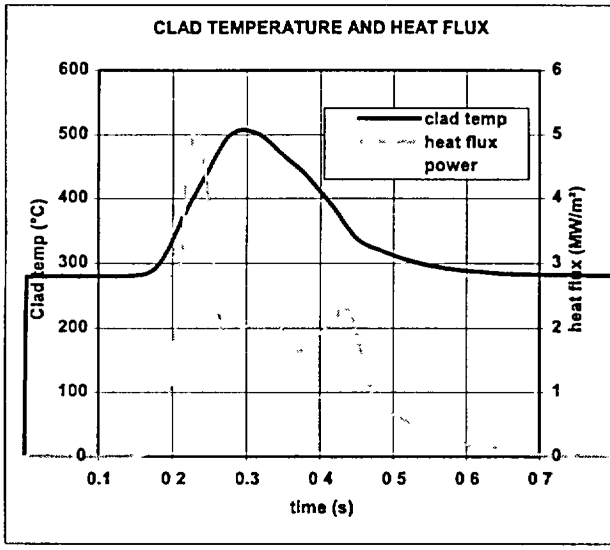
Both PWR conditions (150 bars, 280°C, 4 m/s) and NSRR conditions (atmospheric pressure, room temperature, stagnant water) have been simulated. Steady-state experiments which allow to make the link with the usual correlations have been performed too.

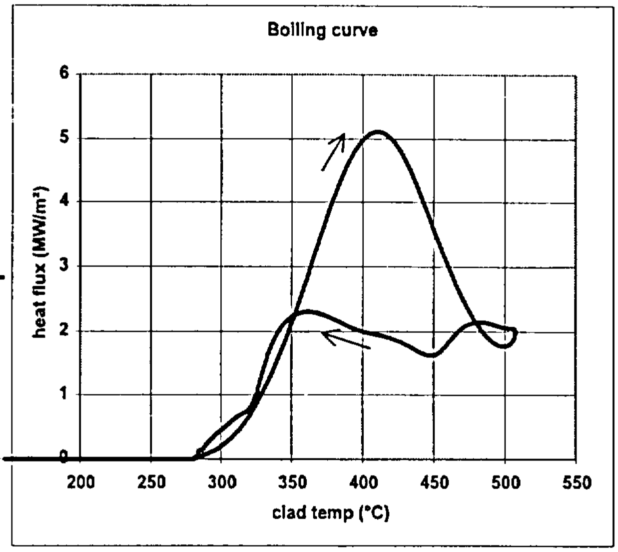
The experiments exhibiting boiling crisis are characterized by two main features that can be clearly seen on the boiling curve:

- presence of transition boiling: decrease of the flux with simultaneous increase of the clad temperature,
- presence of a rewetting peak.

The physical interpretation of the experiments led to assert the main characteristics of the clad-to-coolant heat-transfer on bare rods in PWR conditions at 280°C under very fast transients which are:

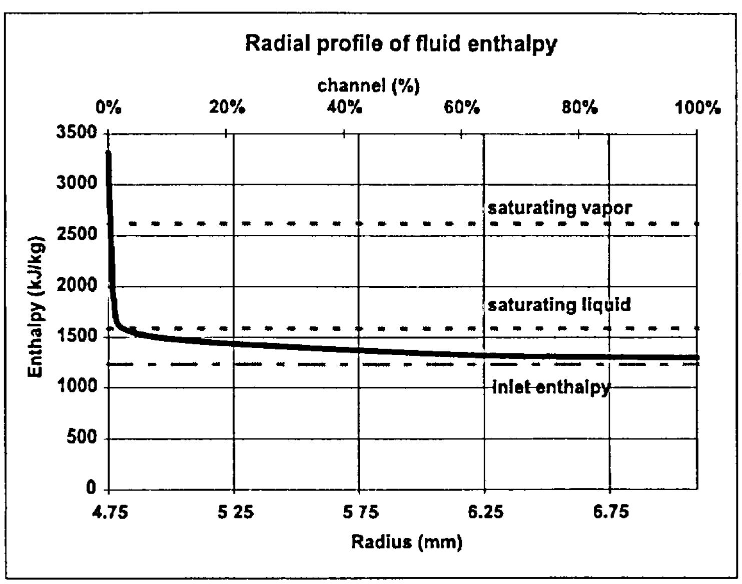
- the boiling crisis is mainly governed by the flash boiling of a superheated liquid layer without sufficient time to get fully established nucleate boiling heat transfer,
- the critical heat-flux is of the order of 5-6 MW/m² reached around Tsat + 50°C (~Tsat+20°C in steady-state conditions),
- the fast crossing of transition boiling is followed by inverse annular film boiling with heat flux of the order of 1-2 MW/m².





Calculations with the TH2D computer code (two-phase and 2D code designed for computing thermal-hydraulics during a RIA and developed within an IRSN/Kurchatov Institute cooperation) allows to estimate the topology of the flow during the transient. The next figure shows that most of the coolant is not affected by the clad heating: the thermal gradient only concerns a small fluid layer in the vicinity of the wall. The vapor film reaches approximately 5% of the channel.

These results are intensively used for the development and the validation of the SCANAIR code and for the definition of the future CABRI tests in the Water Loop.



NSRR High Burnup Fuel Tests for RIAs and BWR Power Oscillations without Scram

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In order to examine high burnup fuel performance under reactivity-initiated accidents (RIAs) and under unstable power oscillation conditions arising during an anticipated transient without scram (ATWS) in boiling water reactors (BWRs), fuel irradiation tests were conducted with irradiated fuels under the simulated power transient conditions in the Nuclear Safety Research Reactor (NSRR).

In the RIA simulating tests of BWR fuels at a burnup of 61 GWd/tU, cladding failure occurred in tests at fuel enthalpies of 260 to 360J/g (62 to 86cal/g) during an early phase of the transients, while the cladding remained cool. Transient hoop strain measurements of the cladding in the early phase of the transients indicated small deformation below 0.4 % (Fig. 1), suggesting that the deformation was caused mainly by thermal expansion of the pellets. Hydride distribution in the BWR cladding was different from those observed in the PWR fuels failed in the earlier tests (Fig. 2), which likely contributed to the BWR fuel failure at low hydrogen contents of about 150-200ppm. Transient fission gas release by the pulse irradiation was about 9.6 to 17% depending on the peak fuel enthalpy.

In the power oscillation tests, irradiated fuels at burnups of 56 and 25 GWd/tU were subjected to four and seven power oscillations, which peaked at 50 to 95 kW/m at intervals of 2 s. Peak fuel enthalpies were estimated to be 256 J/g (61 cal/g) and 368 J/g (88 cal/g) in the two tests. The power oscillation was simulated by quick withdrawal and insertion of six regulating rods of the NSRR. An example of the rod behavior under the transient condition is shown in Fig. 3. The cladding elongation increased as the power rose up, independent to the cladding temperature. The result suggested that pellet-cladding mechanical interaction (PCMI) caused the cladding deformation in the test. The cladding deformation was comparable to those observed in the RIA tests at the same fuel enthalpy level up to 368 J/g (88 cal/g). Cladding axial deformation due to the PCMI was not enhanced due to the cyclic load.

Fission gas release, on the other hand, was considerably smaller than in the RIA tests, suggesting different release mechanisms between the two types of transients. Figure 4 compares fission gas releases in the transient heating tests of high burnup BWR fuels. Fission gas release in test FK-11 with a BWR fuel at an estimated peak fuel enthalpy of 256 J/g (61 cal/g) was much smaller than that in a comparable RIA test FK-5 with a sibling fuel at a peak fuel enthalpy of 293 J/g (70 cal/g).

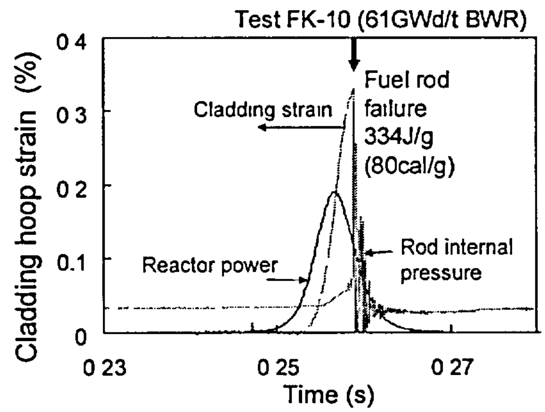

Fig. 1 Cladding deformation history measured in a RIA test with a high burnup BWR fuel rod. The rod failed at a fuel enthalpy of 80 cal/g with cladding hoop strain below 0.4%.

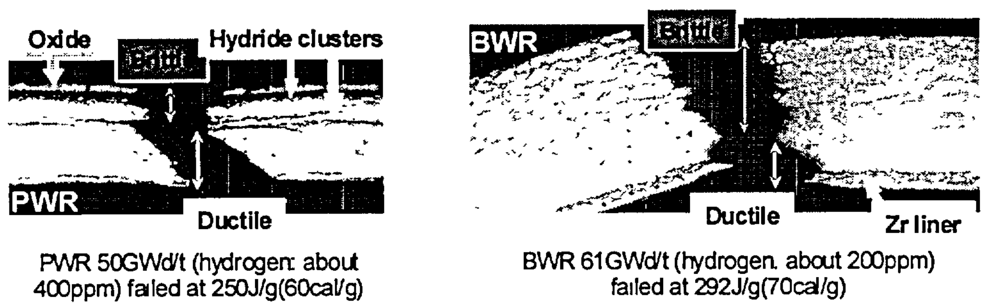

Fig. 2 Comparison of PWR and BWR cladding cross sections failed in RIA tests. Hydrides in the BWR cladding were oriented more randomly.

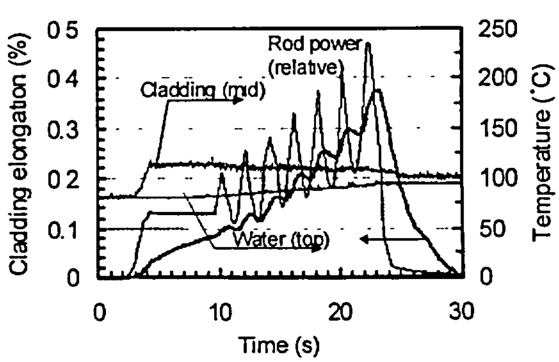

Fig. 3 Transient power, temperature and cladding deformation history measured in a power oscillation test with a fuel rod at a Burnup of 25 GWd/t. The rod was subjected to seven power oscillations at linear heat rates up to 95kW/m and at an estimated peak fuel enthalpy of 368 J/g (88 cal/g).

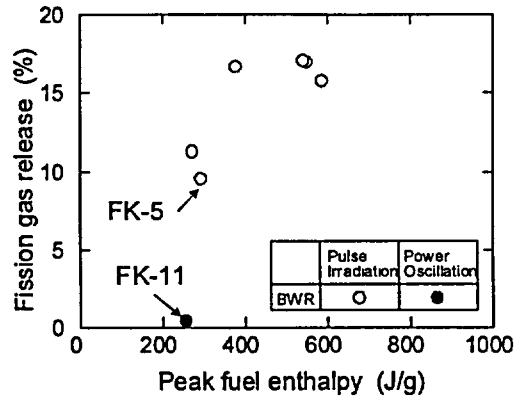

Fig. 4 Fission gas release observed in RIA and power oscillation tests of BWR fuel rods at burnups from 56 to 61 GWd/t as a function of peak fuel enthalpy

The Influence of Hydride Distribution on the Failure of Zircaloy

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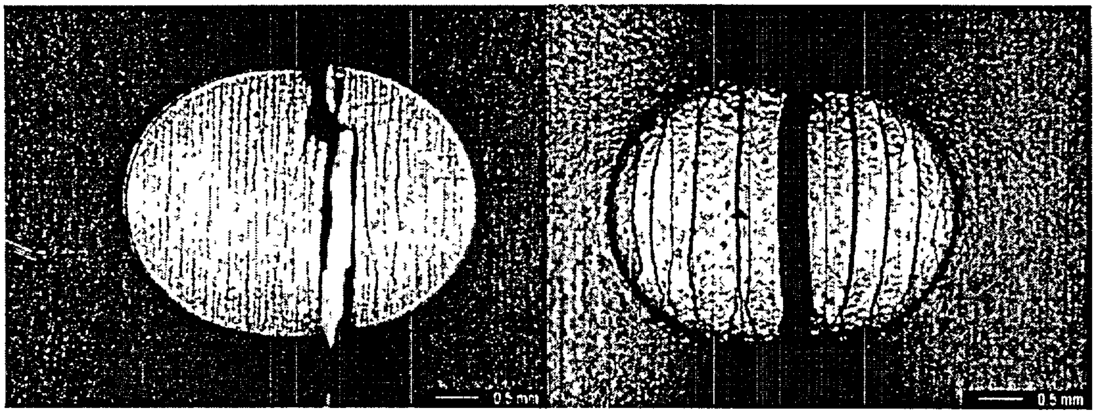
After in-reactor exposure, irradiated Zircaloy cladding tubes often contain hydrides that concentrate near the outer surface either in the form of a continuous hydrided layer or as a hydride blister. We have investigated the fracture behavior of unirradiated Zircaloy-4 sheet (0.64 mm thick) containing either solid hydride blisters or a continuous hydrided layer/"rim" at both 25° and 300°C and subject to the multi-axial stress state of near plane-strain tension. This state of stress is a reasonable approximation of the state of stress observed during an RIA in the absence of strong pellet-cladding bonding. The use of sheet material allowed greater freedom in the specimen design so that we could introduce blisters of a realistic size but that would be still compatible with the gauge length used. Importantly, the Kearns factors [1] of the sheet are similar to those of Zircaloy cladding tubes so that the sheet has crystallographic texture similar to that of tubing. The "blisters" were prepared by hydrogen charging both cold-worked stress relieved and recrystallized Zircaloy sheet using a Ni window, the geometry of which controlled the geometry of the blister. Both the blister diameter and the blister thickness were varied.

The results show that the fracture strains obtained with *sheet* material containing a hydride rim are very similar to those obtained previously on *tubing* with hydride rim, which gives confidence that the information on sheet material has relevance for tubing. As shown in Figure 1, the blisters tend to be brittle, and the overall failure of the Zircaloy is controlled by fracture of the remaining "substrate" material. In fact acoustic emission experiments reveal that the blisters crack shortly after yielding. Figure 2 shows the local fracture strain plotted against blister depth, for both room temperature and 300°C experiments. These experiments indicate that fracture of the sheet is sensitive to the depth of the hydride layer/blister such that there is a significant decrease in ductility as the blister depth increases, up to a depth of about 100 micron. Beyond this value the ductility remains approximately constant. Importantly, moderate ductility is retained in the Zircaloy at 300°C even for blisters of depths > 200 microns, even though such blister depths severely limit room temperature ductility. In general, the ductility of a material with a continuous hydride rim is less than that of a material containing blisters of the same depth. This is likely due to the limited size of the cracks formed in the blister material.

The higher ductility of the sheet material at higher temperature appears to be related to the fact that the fracture strain of the hydride precipitate particles within the substrate increases with increasing temperature. The resulting increase in void nucleation strain contributes to a significant increase of the fracture toughness at high temperatures. As a result, experimental evidence as well as analytical modeling indicates that, while substrate fracture is controlled by crack growth at 25°C, the inhibition of crack growth at 300°C results in eventual failure due to an onset of a shear instability process. As a result, the Zircaloy remains relatively "tolerant" of hydride blisters at 300°C.

[1] The Kearns factors are the resolved fractions of basal poles aligned with the three macroscopic directions: rolling, transverse and normal for sheet material and axial, tangential and radial for tubing, respectively.



(a) (b)

Figure 1.. Macrographs of cracked blisters in CWSR material failed at (a) 25^{0}C and with a 50 μm deep blister and (b) 300^{0}C with a 105 μm deep blister.

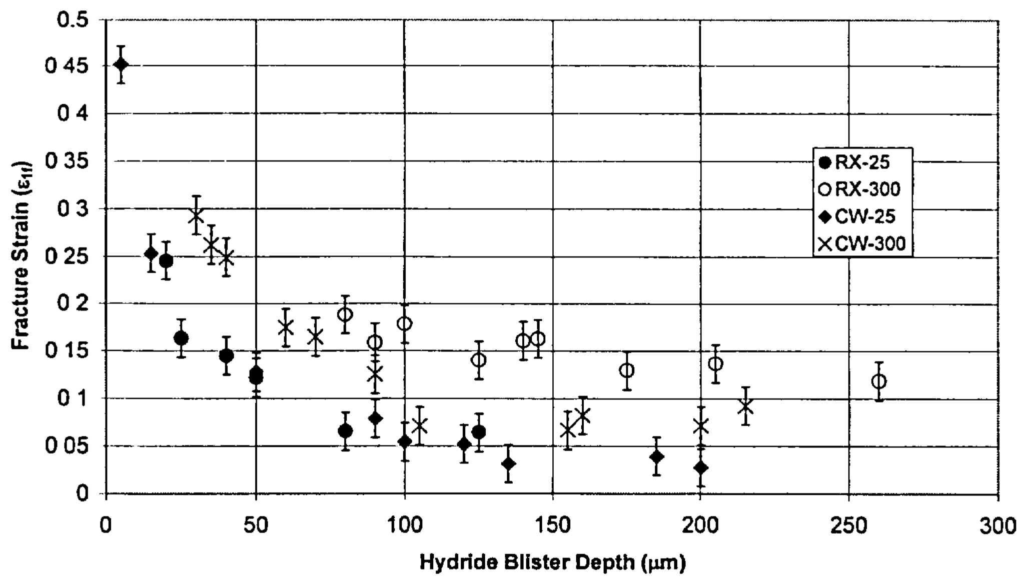


Figure 2. The local fracture strain as a function of hydride blister thickness for both cold worked and stress relieved (CWSR) and recrystallized (RX) Zircaloy-4 sheet tested at either 25^{0}C or 300^{0}C. All data are for 3 mm blisters.

EPRI Perspective on R&D in Support of Practical Approaches for Dry Storage and Transport of Spent Nuclear Fuel

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Increasing amounts of spent fuel are being placed into interim dry storage due to delays in the availability of a permanent disposal site. The ultimate duration of the interim dry storage is not known because acceptance of fuel into a repository is not scheduled to begin for another decade, and an application for licensing approval of the Yucca Mountain repository has not yet been made. Interim storage of spent fuel has become a strategic issue for several licensees.

The Electric Power Research Institute (EPRI) program seeks to establish defensible technical bases for resolving generic spent-fuel storage and transportation issues impacting plant operability (i.e., loss of full core discharge), license renewal (i.e., dry storage of spent fuel beyond 20 years), timely decommissioning, and the licensees' ability to move fuel off-site. Ongoing and proposed R&D pertaining to several areas of interest to the industry will be reviewed. They include: storage and transportation of spent high-burnup fuel, dry storage beyond 20 years, and burnup credit.

Over the past couple of years, storage and transportation of spent high-burnup fuel has received the highest degree of attention as a result of a prioritization exercise conducted during a workshop held in December 1999, involving the participation of the NRC Spent Fuel Project Office (SFPO) and the Nuclear Energy Institute. EPRI studies and concurrent review interactions with the NRC Spent Fuel Project Office eventually led to a revision of Interim Staff Guidance 11, which was released on August 7, 2002. Studies are continuing with a focus on enlarging the technical basis to include transportation issues, as appropriate.

To allow for dry storage at existing independent spent-fuel storage installations (ISFSIs) beyond 20 years, and in cooperation with NRC Office of Nuclear Regulatory Research and DOE, a three-year program was conducted to examine the fuel and selected cask components that had been stored at INEEL for close to 15 years. This program has provided much valuable data supporting ISFSI license extension. No measurable increase in fuel rod diameter (creep strain), fission gas release, cladding oxidation, or hydrogen pickup was observed. No detectable re-orientation of the hydrides in the cladding was observed. Little, if any, annealing occurred. The cladding showed large residual creep strains after 15 years in dry storage.

Another high-priority generic issue is burnup credit. A revision of Interim Staff Guidance 8 on this topic is scheduled for release by the SFPO in September 2002. A cooperative R&D agreement among several organizations, including foreign organizations, may hold the key for providing missing or confirmatory data that would expedite regulatory acceptance and implementation of practical and cost effective approaches for demonstrating nuclear criticality safety using a burnup credit methodology.

A Perspective on Spent Fuel Integrity Under Dry Storage and Transportation Conditions

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Dry storage of high burnup fuel, (average assembly burnup greater than 45 GWd/MTU), has come under increasing scrutiny by the NRC during the last several years, culminating in the issuance, on August 7, 2002, of Revision 2 of Interim Staff Guidance - 11 (ISG-11). At issue is the ability of the cladding to resist degradation during storage that could lead to fuel re-configuration under hypothetical transportation accidents. Relying on literature data and recent research information at EPRI and National Laboratories, the NRC Staff developed Revision 2 of ISG-11, which removed the limitations in the previous revision of the ISG on oxide thickness and creep strain, but added criteria to limit hydride re-orientation in the cladding. With respect to the latter provision, 400°C was established as the maximum allowable temperature during drying, with temperature cycling limited to less than 65°C range. The purpose of the temperature and temperature-cycling criteria is to limit the amount of hydrogen that could potentially precipitate in the form of radially oriented zirconium hydride platelets. Under the decaying temperature history in dry storage, radial hydrides can occur to various degrees, depending on cladding type and manufacturing history, if the pressure-induced hoop stress level exceeds a threshold value that is a function of temperature and stress. The motivation behind limiting the formation of radial hydrides is the fact that the very low ductility of zirconium hydrides, generally observed at temperatures below the brittle-ductile transition, may cause the radial hydrides to behave effectively as initial part-wall cracks at low temperatures. Consequently, under certain loading regimes, particularly during hypothetical transportation accidents, radial hydrides could be engaged and become the dominant failure mechanism.

The purpose of this paper is to examine the general conditions under which this phenomenon can occur, assess its consequences relative to the safety objectives of the regulations in 10 CFR Parts 71 and 72, propose methods to deal with the phenomenon analytically, and suggest techniques to quantify its consequences. It should be noted at the outset, however, that the effects of forming radial hydrides during storage have been evaluated in the EPRI research program cited in the Appendix of Revision 2 of ISG-11, and found to be non-consequential. The new issue introduced by the revised ISG is to evaluate the consequences of radial hydrides on fuel re-configuration during a hypothetical transportation accident. To allow an analytical evaluation of this issue, behavioral models will be needed in the following areas:

(a) Radial hydride precipitation rate for various storage temperature and stress histories.
(b) Effects of initial temperature and temperature cycling on radial hydrides formation.
(c) Radial-hydrides morphology (platelets size and distribution).
(d) Dependence of the cladding mechanical properties on radial hydrides as function of temperature.
(e) Fracture initiation and propagation for cladding with radial and circumferential hydrides.

The paper will discuss the state of the art in each of these areas, including availability of data and the need for future testing. It can be said at the outset, however, that although very limited data exists, there is sufficient information available to allow the construction of behavioral models, empirically based where possible and from first principles, subject to future verification and validation through testing.

A feature of Revision 2 of ISG-11 is the provision to address relevant transportation issues on a case by case basis, using storage conditions under the imposed temperature and temperature cycling limitations as precursors to fuel vulnerability assessment under hypothetical transportation accidents. It is important to note, however, that dual-purpose casks, as well as transportation-only casks, have common features with respect to hypothetical-accidents consequences, such that a case can be made for a generic treatment of the issue. Moreover, the effects of storage conditions, having been made necessary conditions for transportation by the revised ISG, may prove to be quite tolerable under hypothetical accidents in terms of potential fuel re-configuration, yet remaining to be highly constraining on storage, particularly when vacuum drying is relied upon. A cursory evaluation of this apparent conflict is discussed in the paper.

Thermal Creep of Dry-Cask-Stored Surry PWR Cladding

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Because of the limited storage capacity in spent-fuel pools, some spent fuel assemblies have to be relocated into dry casks for interim storage until long-term geological repositories are available. Upon discharge from the reactor, the internal pressure in the spent fuel rod can exert a significant stress loading on the fuel cladding. At elevated temperatures, these tensile stresses can induce significant outward thermal creep of the cladding. The vacuum drying operation can elevate the cladding temperature to ≈400-500°C for many hours. Transfer and transport operations can also result in elevated temperatures for periods of minutes to hours. Once the spent fuel is dried and relocated in the storage cask, the temperature will decrease slowly from initial storage temperatures of 300-400°C. Thermal creep of the cladding under these conditions is an important consideration, as it might impact the fission product release within the cask and the integrity of the spent fuel rods during repository transfer operations following dry-cask storage.

In the mid-1980s, the U.S. Department of Energy (DOE) procured a Castor-V/21 dry-storage cask for testing at the Idaho National Environmental and Engineering Laboratory (INEEL). The primary purpose of the tests was to benchmark computer codes by measuring the thermal and radiological characteristics of the cask. The cask was loaded with irradiated assemblies from the Surry Nuclear Station and then tested in a series of configurations using a variety of cover gases, including vacuum. Subsequently, the cask sat on the storage pad at the INEEL for ≈15 years with the fuel in an essentially inert atmosphere (He/<1% air). Under the sponsorship of the U.S. Nuclear Regulatory Commission, the Electric Power Research Institute, and DOE, twelve rods were retrieved from the cask for post-storage characterization. These twelve rods had the highest combined burnup and storage temperature among all the rods in the cask. Cladding from two of the rods was subsequently prepared for thermal creep testing. The burnup of the two fuel rods was ≈36 GWd/MTU.

The objective of the thermal creep tests is to evaluate residual creep ductility of the Surry cladding after the dry-cask storage. A significant residual creep strain (>≈1%) would suggest that the rods may be suitable for further storage in the cask and may survive creep during transportation, reconsolidation and final repository conditions. As the Surry rods are not the limiting case for ≤45 GWd/MTU, demonstration of residual creep life can be used to argue that higher burnup rods with thicker oxide layers, higher hydrogen content and higher storage temperatures would also have survived 20 years of dry cask storage without creep failure.

The thermal creep tests use 76-mm-long defueled cladding with welded Zircaloy end fittings. An extension tube on the top end fitting permits active control of internal pressure in the sample. During the test, the pressure is maintained constant with a microprocessor-controlled regulator. To mitigate oxidation of the specimens in the furnaces, chambers purged with an inert gas are used to protect the specimens. Periodically, the specimens are depressurized, cooled and removed from the test chamber for diametral measurements with a precision laser profilometer. Sample length is measured at the same time to assess possible creep anisotropy.

Five tests have been conducted thus far. Test conditions and preliminary results are summarized in Table 1. All of the samples were intact and two of them, C8 and C9, reached >1% creep strain within the prescribed test duration of ≈2000 h. At the end of 1873 h, the stress level for C9 was subsequently increased from 190 to 250 MPa, which sharply increased the creep rate. Significantly, the sample remained intact with a creep strain of ≈5.8% at the end of the incremental 693 h. At the time of writing, the C3 test is still ongoing and the disposition of other test samples is being evaluated. Additional tests are planned at 400°C and 220 MPa and 160 MPa.

In the full paper, details of test conduct and results will be provided and the significance of the test results to dry-cask and repository storage will be discussed.

Table 1. Summary of Post-Storage Thermal Creep Tests of Surry Cladding

Sample	Nominal Test Temp. (°C)	Nominal Hoop Stress (MPa)	Test Duration (h)	Sample Condition	Creep Strain (%)	Secondary Creep Rate (%/h)
C3	360	220	3305	Intact	0.22	4.2×10^{-5}
C6	380	190	2348	Intact	0.35	8.8×10^{-5}
C8	380	220	2180	Intact	1.10	4.5×10^{-4}
C9	400	190	1873	Intact	1.04	4.9×10^{-4}
C9(1)	400	250	693	Intact	5.83	$>4.9 \times 10^{-3}$

(1) Same sample but with the stress increased from 190 to 250 MPa after the initial 1873 h.

The Package Performance Study: An Investigation of the Response of Transportation Casks to Severe Rail and Highway Accidents

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The U.S. Nuclear Regulatory Commission's (NRC's) responsibilities in the transport of spent nuclear fuel include certification of transport packaging designs, approval of transport package Quality Assurance programs, issuance of general licenses authorizing licensees to offer material to carriers for transport, and establishment of physical protection requirements for spent fuel in transit. Because of continuing Commission interest in maintaining the currency of the technical bases for its regulatory policies, the NRC staff has conducted several studies of the risks associated with the transport of radioactive materials. The spent fuel transportation Package Performance Study (PPS) is an ongoing element of that continuing effort ; PPS will provide experimental data to validate the assumptions used in previous studies.

The study investigates the performance of casks when subjected to thermal and impact forces that exceed the hypothetical accident conditions specified in 10 CFR Part 71. Issues related to the probability of severe transport accidents will be examined, i.e., the databases for rail and highway will be updated and event trees will be reconstructed and re-analyzed. The objectives of PPS include (1) enhancing public confidence in the inherent safety of certified transportation casks and (2) verification of analysis models, through combinations of detailed analysis and physical testing, to predict accident risk associated with transportation of spent fuel in NRC certified casks. PPS is a follow-on project to NUREG/CR-6672, "Reexamination of Spent Fuel Shipment Risk Estimates," which was published in March 2000. An enhanced public participatory process has been used in developing the PPS issues for study and the conceptual testing and analysis plans.

This paper presents the outline of the preliminary test plans for a high speed impact of a full scale rail cask into an essentially unyielding target and a fire test at beyond-design basis conditions used for cask and the plans for updating the rail and highway databases. The Commission is currently considering additional tests involving truck casks. The paper will also present the current status of the plans and of the 3-D finite element analyses that is being carried out to support the experimental activities.

A Pilot PRA of a Dry Cask Storage System

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Office of Nuclear Regulatory Research
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The spent fuel pools of commercial nuclear power plants are becoming filled with spent fuel assemblies. To avoid having to cease operations when the pools are full, many utilities have been removing older fuel from the pools and storing it in dry casks on site. To assess the risk of these activities, this study develops a methodology for performing a probabilistic risk assessment (PRA) of a dry cask storage system at a nuclear power plant site and applies the methodology by performing a pilot PRA of a specific cask system at a specific site. The study results do not necessarily apply to any other cask system or site.

The scope of the PRA includes assessing the risk to the public and identifying the dominant contributors to risk. Although the pilot PRA performed for this study is for a specific cask design and site, the methodology developed can serve as a guide for performing a PRA for another cask system or site.

The cask system analyzed is the Holtec International HI-STORM 100. It consists of a multipurpose canister (MPC) that confines the fuel, a transfer cask that shields workers from radiation while the cask is being prepared for storage, and an overpack that shields people from radiation and mechanically protects the MPC during storage. The site studied is a specific boiling-water reactor (BWR) site.

The study covers the handling, transfer, and storage phases of the dry cask storage operation. Beginning with loading fuel from the spent fuel pool, the handling phase involves preparing the cask for storage and moving the cask outside the reactor building. The transfer phase involves moving the cask from the reactor building to the storage pad. The storage phase involves storing the cask for 20 years on the storage pad. The study scope does not extend to cask fabrication, fuel defects, fuel assembly drops into the spent fuel pool, cask drops that damage the plant, offsite transportation, storage in a permanent repository, strikes by military missiles, acts of sabotage, terrorist attacks, or acts of war or insurrection.

This study developed and screened a comprehensive list of more than 50 initiating events to eliminate those that have only negligible contributions to risk. Specifically, this study considers cask drops during handling and transfer, as well as external initiating events during storage, such as earthquakes, floods, high winds, lightning strikes, accidental aircraft crashes, and pipeline explosions. It considers potential cask failures from mechanical, thermal, and mechanical-thermal loads. The study screens out an initiating event if (i) it cannot affect the subject site or cask, (ii) the initiating event frequency is no more than 10^{-8} per year, or (iii) the conditional probability of a radioactive release, given the initiating event, is no more than 10^{-8}. For the initiating events that are not screened out, the study uses event tree and fault tree methods to develop logic models of plausible accident sequences. It also uses engineering analyses to determine the probability of a cask failing and releasing radionuclides to the environment when subjected to postulated accident conditions.

The study measures the risk to the public in terms of the individual probabilities of a prompt fatality within 1 mile and a latent cancer fatality within 10 miles of the site. In calculating the risks, the study also considers the weather conditions and population distribution in the vicinity of the site, as well as emergency response activities. Risk to workers is beyond the scope of the study.

To quantify the risk, the study uses the best available point estimates. A quantitative evaluation of uncertainties has not been made. When there is insufficient information or data, the study uses conservative bounding assumptions or estimates. Accordingly, the quantitative results of the PRA are conservative.

Using the screening criteria, the study screens out all but three initiating events, namely cask drops of 80 and 20 feet in the handling phase and blockage of all four inlet vents by accumulations or biological intrusions in the storage phase This study does not predict any early fatalities for these three initiating events.

For the three initiating events not screened out, the results of the risk assessment show very low risk. The staff has (qualitatively) identified specific uncertainties that appear to affect the risk estimates; these are associated with vent blockage.

Several open issues relate to vent blockage. The frequency and duration of vent blockage are unclear. While the cask is routinely inspected, current inspection protocol does not require inspecting inside the vents. Consequently, it is unclear which vent blockage scenario is the largest contributor to the risk. Furthermore, it is unclear how the risk of partial vent blockage compares to the risk of full vent blockage postulated for this study. While the consequences of partial vent blockage are certainly less than for full vent blockage, its frequency and probability of surveillance failure are greater The risk depends on the product of these quantities.

The U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Regulatory Research (RES), performed this study in response to a user request by the NRC's Office of Nuclear Material Safety and Safeguards (NMSS) to support their efforts to risk inform the regulations. It is expected that NMSS will use the results of this study in conjunction with the methodology to develop a basis to determine the need for other site-specific PRAs, improvements in data gathering and analysis, and additional engineering design analysis

Drop Impact Analyses of Spent Fuel Dry Cask Storage Systems

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A series of analyses were performed to assess the structural response of spent nuclear fuel dry casks subjected to various handling and storage related loading events. The results of these analyses are being used by the Nuclear Regulatory Commission (NRC) to perform a probabilistic risk assessment (PRA) for the dry cask system. Although the PRA is being conducted for a particular cask system at a single plant, it is intended to provide a framework for a general methodology applicable to other dry cask systems at other locations.

The dry cask system consists of a transfer cask used for handling and moving the multi-purpose canister (MPC) that contains the fuel, and a storage cask which is used to store the MPC and fuel on a concrete pad at the site. Simplified analyses evaluated the handling, on-site transportation, and on-site storage of the cask system and the effects of natural phenomena such as wind, flood, and earthquake. As a result of these analyses, some of the loadings required further refinements in order to obtain a more realistic response of the dry casks. This paper describes the refined analyses of the dry casks for two loading events. The first loading consists of dropping the transfer cask through the equipment hatch while it is lowered from the refueling floor level to the concrete floor at ground elevation. The second loading consists of dropping the storage cask while it is being transferred from the reactor building to the concrete storage pad outdoors.

Three dimensional finite element models of the transfer cask and storage cask, containing the MPC and fuel, were utilized to perform the various drop analyses. These models were combined with finite element models of the target structures being impacted. The transfer cask drop analyses considered various drop heights for the cask impacting the reinforced concrete floor at ground level. The finite element model of the target included a section of the concrete floor and concrete wall supporting the floor. Two orientations of the transfer cask drop were considered: a vertical end drop and a horizontal drop. The storage cask drop analyses evaluated a 12-inch drop of the cask impacting three surfaces that the cask transfer vehicle travels over. The roadway surfaces analyzed were asphalt, gravel, and reinforced concrete resting on a deep soil layer.

The paper summarizes the nonlinear impact analyses performed for both models using the LS-DYNA computer code. Details on the finite element models, material models, analytical approach, and calculated cask response are discussed. The structural response of the MPC and fuel in terms of the maximum stress/strain and impact acceleration is described. The effect of variations in configuration/design and analytical parameters are also addressed. Some of the key considerations included in this study are variation in drop heights, drop orientation, types of impact targets, target material properties, and underlying soil properties.

Using MACCS in Assessing Dry Cask Vulnerability

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The Nuclear Regulatory Commission (NRC) is performing studies on vulnerability of transportation and storage of nuclear material to terrorist events. Following the September 11, 2001, terrorist events at the World Trade Center in New York City and at the Pentagon in Virginia, the U.S. Government issued a nationwide alert for the potential of additional terrorist acts within the United States. In response to these alerts and Congressional inquiries, the NRC is assessing the vulnerabilities and consequence of postulated terrorist events on the transportation and storage of nuclear materials, including spent nuclear fuel from reactor operations.

The consequence assessment uses MACCS (Melcor Accident Consequence Code System) to assess the dose consequences from different cask systems (HI-STORM and NUHOMS) for worst case and probable case sabotage scenarios. This assessment calculates the individual risk of a prompt fatality, individual risk of a cancer fatality and doses to rural, urban and suburban populations, as a result of hypothetical cask accidents from terrorist attacks. It will also modify certain MACCS inputs, such as deposition velocity to assess the impact on certain dose pathways (inhalation, groundshine, etc).

Seismic Evaluation Method of Design Ground Motion at Gravelly Soil Site for Interim Spent Fuel Storage Facilities

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In Japan, interim storage facilities for recycled nuclear fuel resources may be constructed on quaternary layers, rather than on hard rock by 2010. Nuclear Power Engineering Corporation (NUPEC) has (a) examined the application of safety analysis codes for the cross-check analysis, (b) selected safety analysis codes to be improved, and (c) planned the tests, research works for upgrading them. The objective of these efforts is to obtain data for improvement of seismic safety analysis codes for the evaluation of facilities constructed on quaternary layers, which shows high degree of non-linearity during a strong earthquake. The authors have been conducting this research work to establish the evaluation method for design earthquake ground motion for storage facilities sited on gravelly soil. Since 2000, NUPEC has been conducting this research work under the sponsorship of the Ministry of Economy, Trade and Industry (METI) of Japan.

During 2000 and 2001, NUPEC has carried out the analytical regression review of observed ground motions and examined the characteristics of amplitude in the gravelly soil. From this research, the following major results were obtained:

- Regression results concerning the response spectrum showed that the site coefficient exceeded the results of other representative Japanese research work at periods < 0.4s, while the regressions showed the similar site coefficients at periods > 0.4s.

- Amplitude characteristics for gravelly soil depended on the dominant period of soil up to the free surface of base seismic rock outcrop.

- Peak values of response spectrum using non-linear analysis decreased and shifted to longer periods as the thickness of soil increased.

- Response acceleration showed a little ceiling peak value in case of thick gravelly layer analysis.

This year, NUPEC is going to perform analytical evaluation of amplitude characteristics due to seismic response analysis. The evaluation method for the design ground motion on quaternary layers will be selected by March 2003.

Evaluation of Seismic Behavior of HI-STORM 100 Casks at Private Fuel Storage Facility

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Sandia National Laboratories conducted a research project to develop a comprehensive methodology for the seismic evaluation of spent fuel dry cask storage systems (DCSS) for the Office of Nuclear Regulatory Research of the U.S. Nuclear Regulatory Commission (NRC). A typical Independent Spent Fuel Storage Installation (ISFSI) consists of arrays of freestanding storage casks resting on a concrete pad. In the safety review process of these cask systems, their seismically induced horizontal displacements and angular rotations must be quantified to determine whether casks will overturn or neighboring casks will collide during a seismic event. This project focused on performing site-specific and parametric analyses in order to establish criteria and review guidelines for the seismic behavior of these storage systems.

This paper documents the development of the coupled finite element models and the analysis results to examine the seismic behavior of cylindrical HI-STORM 100 casks to be installed on concrete pads at the proposed Private Fuel Storage (PFS) Facility in the state of Utah. The research team consisting of analysts and engineers at SNL, ANATECH, and Earth Mechanics developed three-dimensional coupled finite element models and performed seismic analyses. The ABAQUS / Explicit code was used to develop the coupled models that consist of a cylindrical cask, a flexible concrete pad, a soil-cement layer under and adjacent to the pad, and an underlying soil foundation. Nonlinear contact elements were used at the interfaces of cask/pad, pad/soil-cement layer, and soil-cement layer/soil foundation in order to examine the dynamic and nonlinear behavior of the model including the soil-structure-interaction effects during a seismic event.

Three sets of seismic time histories were considered in the coupled model analyses. Two of them are specific to the PFS site using seismic input time-histories based on a 2,000-year and a 10,000-year return period. The third one is based on the 1971 San Fernando Earthquake, Pacoima Dam record. Each set has one vertical and two horizontal components of statistically independent seismic accelerations. For the seismic event with a 2,000-year return period, the peak ground accelerations (PGAs) are 0.728 g (horizontal, east - west), 0.707 g (horizontal, north - south), and 0.721 g (vertical), which envelop the 2,000-year design basis response spectra of 0.711 g (horizontal) and 0.695 g (vertical) stated in the Safety Evaluation Report for the PFS Facility. The corresponding peak ground accelerations for the seismic event with a 10,000-year return period are: 1.25 g, 1.23 g, and 1.33 g, which envelop the PFS earthquake hazard spectra. The duration of both events is 30 seconds. For the 1971 San Fernando Earthquake, Pacoima Dam record with duration of 41.8 seconds, the peak ground accelerations for the two horizontal components are 0.641 g and that for the vertical component is 0.433 g. A deconvolution procedure was used to adjust the amplitudes and frequency contents of the surface defined accelerations before applying them simultaneously at the base of soil foundation in the coupled model.

There are two other important parameters involved in the seismic analyses of the PFS casks. The first parameter is the coefficient of friction at each of the three interfaces in the model: cask/pad, pad/soil-cement layer, and soil-cement layer/soil foundation. A lower bound coefficient of friction of 0.20 (for investigating cask sliding) and an upper bound coefficient of friction of 0.80 (for examining the possibility of cask tipping-over) were used at the cask/pad interface. Coefficients of friction of 1.00 and 0.31 were also assumed at the other two interfaces. The second parameter is the selection of site-specific soil profile data for the soil foundation model. The best estimate, the lower bound and the upper bound soil profile data were used separately in the seismic analyses of PFS casks.

The separation distance between neighboring casks is 47.50 inches and half of this distance equaling 23.75 inches has been regarded as the cask collision criterion. The analysis results indicate that the maximum horizontal cask sliding displacements are 15.94 inches (for the 10,000-year return period), 3.98 inches (for the 2,000-year return period), and 3.00 inches (for the 1971 San Fernando Earthquake, Pacoima Dam record). Therefore, no cask collision will occur in all cases under investigation. In addition, the analysis results show that the maximum cask rotation with respect to the vertical axis in either horizontal direction is less than 1.5 degrees, which is significantly less than the cask rotation for tipping over (approximately 29 degrees). Therefore, the PFS casks are not anticipated to tip over during an earthquake return period of either 2,000 years or 10,000 years.

Finalization of NUREG-1640: Radiological Assessments for Clearance of Equipment and Materials from Nuclear Facilities

Robert A. Meck
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Robert Anigstein et al.
Sanford Cohen and Associates

NUREG-1640 reports the dose assessments to individuals from the transport, subsequent processing, and landfill disposal of cleared iron and steel, aluminum, copper, concrete, and equipment. The assessments include consideration of radionuclide partitioning in the metal product and byproducts associated with metal refining, manufacturing, and subsequent consumer use of products made from these materials and direct re-use of equipment. Responses to a substantial number of technical comments on the 1999 draft will result in a more realistic and defendable set of analyses. For comparison, analyses were performed using International Commission on Radiological Protection (ICRP) Recommendations from Publication 26 and 60. The highest mean individual dose from all the scenarios identifies the average dose to the critical group on a nuclide by nuclide basis.

Finalization of Inventory Report: Materials Having Very Low Levels of Radioactivity

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William Thurber et al.
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Finalization of the inventory report is near completion. The report provides information on the amounts, types, and associated low levels of radioactivity in solid materials that will be considered for release from NRC licensed facilities. While the report focuses on identifying potentially clearable materials from NRC licensees, it also provides overview and perspective information on potentially non-clearable inventories being considered for disposal at LLW disposal facilities. The report also provides inventory information on other solid, radioactive materials requiring disposal consideration which come from non-NRC licensed facilities, such as those from DOE, DOD, and commercial industries that process or produce technically enhanced, naturally occurring radioactive materials (TENORM).

There is a very distinct demarcation in radioactivity levels that exists between materials segregated for clearance from those needing to be sent to a LLW facility for disposal. When a high-quality, rigorous process for sorting materials was used, virtually all the materials considered for clearance were found to contain very low levels of radioactivity (within the measurement uncertainties of discriminating them from background).

Surveys of Volumetric Contamination and Difficult Geometries

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Objects that have inaccessible areas that are contaminated cannot be readily demonstrated to satisfy release criteria using 'conventional' survey activities. Options will be presented and discussed that address contamination in inaccessible areas and its potential measurement and assessment. Various inaccessible material scenarios that can occur during the clearance of materials include: (1) contamination on the interior surfaces of scrap equipment, such as pumps, motors, and other equipment; (2) the interior surfaces of pipes that are difficult to access, such as buried or embedded pipes; and (3) collections of surface contaminated materials that are reconfigured to become volumetric such as a pile of scrap metal, concrete, soil (or other materials).

Conducting radiation surveys to demonstrate compliance with appropriate regulatory criteria with a certain degree of confidence will be discussed. These surveys may involve: (1) evaluating the use and the radionuclides with which the equipment was associated; (2) making inaccessible areas accessible for measurement; (3) performing direct measurements using specialized equipment or sampling techniques.

Fire Risk Research Program: Addressing Key Uncertainties

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Fire can be a significant or even dominant contributor to the overall risk of a particular nuclear power plant. The current fire risk assessment (FRA) state-of-the-art is not as mature as that for assessing the risk contributions of many other important accident initiators. Reviews of some IPEEEs have identified situations where variations in FRA analytical assumptions can lead to significant variations in estimates of fire-induced core damage frequency. In particular, many key components of a FRA suffer from significant uncertainty.

The U.S. NRC Office of Nuclear Regulatory Research is making significant headway into addressing these key uncertainties in FRA through implementation of its Fire Risk Research Program. Among the areas addressed in previous efforts are fire-induced failure of cables and circuits (including spurious actuations), detection and suppression of fires, frequency of challenging fires, and fire modeling tools. Additional areas to be addressed through ongoing and planned efforts are the impact of fires on operator performance, control room fires, and remote shutdown activities.

In general, the Fire Risk Research Program is addressing uncertainties by improving the understanding of factors that underlie the FRA methods, reviewing the latest data to improve estimates, participating in testing to collect new data, performing analyses which directly include uncertainty estimates, as well as performing basic research into assessing and integrating different types of uncertainty. Uncertainty will also be treated more comprehensively in pilot plant FRAs via the Fire Risk Requantification Studies. These studies are being conducted as a joint NRC/EPRI program, and represent a major milestone in ongoing efforts to improve FRA methods, and implement those improvements through updating licensee FRAs.

Narrowing the Uncertainties in Human Reliability Analysis

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Human reliability analysis (HRA) is currently being used as part of probabilistic risk analysis (PRA) to support a variety of regulatory decision-making activities. However, HRA is viewed by experts as the one area that can contribute significantly to the uncertainties of PRA results. The U.S. Nuclear Regulatory Commission (NRC) has developed an HRA Program with the objective to improve HRA capability to support regulatory decision-making.

Uncertainties are addressed by conducting research and providing feedback to both research and applications by developing insights and guidance on the basis of lessons learned from these activities. Research and development activities address the more broad HRA needs such as the development of data for performing HRAs as well as for improving human performance modeling in HRA. They also address more specific issues associated with, for example, materials and waste applications; advanced control rooms; low power and shutdown operations, including long term recovery actions; identification, modeling, and quantification of latent errors; modeling of post-initiator ex-control room actions; modeling cognition and team issues in an HRA; and modeling of human performance in post-severe accident steam generator tube rupture scenarios. HRA implementation activities support various regulatory issues including pressurized thermal shock; fire risk assessment; steam generator tube rupture; risk evaluation of aged I&C cables; dry cask storage; and the evaluation of the risk significance of reactor systems synergisms.

Insights developed from both HRA research and HRA applications and are fed back into the program. Lessons and guidance development activities include guidance on how to determine the level of detail needed and how to determine an appropriate HRA quantification method for a given application; guidance for reviewing HRAs providing technical bases for decision-making in risk-informed regulatory applications; lessons learned with HRA methods and tools improvement; and lessons learned for human performance under various accident conditions.

On Principles and Techniques for Concurrent, Integrated, Uncertainty Analysis in Technical Assessments

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The need for more comprehensive and systematic analysis of uncertainties in safety analyses has gained significant visibility as a result of increased awareness of the impact those uncertainties can have on risk-informed regulatory decisions. In addition, lessons learned from several recent NRC research efforts on various aspects of uncertainty analysis point to the need for a set of principles and guidelines for more effective and more accurate concurrent and integrated consideration of uncertainties in large-scale technical assessments.

There are several important dimensions to the problem. There are the philosophical and conceptual dimensions that do impact the way uncertainty analyses are viewed, conducted, and communicated. Among these are the use of probability as the language of uncertainty, and the debate on "knowledge-driven" vs. "nature-driven" uncertainties, or *aleatory* and *epistemic* types. There is also the question of the characterization of sources of uncertainty, such as the distinction made in the modeling space, in terms of *model uncertainty* and *parameter uncertainty*.

While the assessment and propagation of parameter uncertainties is a well-established discipline, with highly advanced methodological and computational approaches, assessment and propagation of model uncertainties are in very early stages of development. This is particularly disturbing as model uncertainties could easily be the dominant source by a large margin. The subject of model uncertainty has also suffered from the "ill-posed question" syndrome, leading to notions such as "quantifiable and un-quantifiable uncertainties", "managing vs. quantifying uncertainties", and "completeness vs. model uncertainties". A somewhat related dimension is the communication of the results of uncertainty assessment in context of the decision that the analysis is supporting. One example is the fact the scope of uncertainty analysis, and (at least major) assumptions are not typically communicated to the decision maker.

In complex technical assessments, inadequate uncertainty analysis can result in major deficiencies including:

- Failure to account for dominant contributors to uncertainty, and to risk.
- Failure to properly characterize the various types of uncertainty, possibly leading to incorrect method of uncertainty propagation.
- Failure to correctly carry uncertainties across sub-models and disciplines of the complex model of a multi-disciplinary effort.

These problems are aggravated by the common practice in most technical assessments where uncertainty analysis is performed after "best estimate" analysis is completed. Experience from recent studies indicates that this approach can easily result in not only an incomplete uncertainty assessment, but also an incorrect "best estimate". It is not uncommon for the analyst to assume that combing "best estimate" from sub-models of a complex model results in best estimate of the final result. There is also the prevailing belief that performing uncertainty analysis is significantly more resource intensive than point estimate analysis. This is not necessarily true. In some cases concurrent uncertainty analysis can actually reduce the scope of issues to be considered in the point estimate analysis. On the other hand while concurrent uncertainty analysis provides clear advantages, it also introduces complexities and new challenges. Effective implementation of the concept requires practical guidelines that rest on a set of general principles, which are currently lacking.

This paper will comment on these issues and point to a unifying theme and common reference framework for characterization, assessment, and propagation of uncertainties in technical assessments. The ideas discussed in the paper are in part rooted in the results of a number of research and development efforts sponsored by NRC Office of Regulatory Research on conceptual and practical aspects of uncertainty analysis, as well as insights from the integrated uncertainty assessment in the on-going effort on assessing PTS risk.

Standardized Plant Analysis Risk (SPAR) Model Development Program

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In order to provide the NRC staff with analytical tools to use in performing risk-informed activities, the Operating Experience Risk Analysis Branch (OERAB) in the Division of Risk Analysis and Applications (OERAB) is developing Standardized Plant Analysis Risk (SPAR) models. These probabilistic risk assessment (PRA) models span the following areas: Level 1 - internal events, full power operation, (2) Level 1 - internal events, low power and shutdown operations, (3) Level 1 - external events (including fires, floods, and seismic events), and (4) Level 2/Large Early Release Frequency (LERF).

In September 1999, OERAB formed the interoffice SPAR Model Users Group (SMUG) to discuss, coordinate, and provide guidance on technical direction of methods and model development issues related to the Accident Sequence Precursor (ASP) Program and the routine assessment of events. The SMUG also supports the development of models for risk-informed regulatory activities performed by the members' organizations. They provide their organization's input to the type of models to be produced, the level of detail that the models require, the model QA review process, and model development schedule priority. These user needs and the technical approach to be pursued by the model development efforts underway in each of the above-identified areas were documented in the SPAR Model Development Program Plan prepared by the SMUG and approved by management of the various model users' organizations.

The SPAR models being developed by the NRC's Office of Nuclear Regulatory Research (RES) support the NRC's strategic goals of: (1) maintaining safety, (2) improving staff regulatory effectiveness, efficiency and realism, (3) reducing unnecessary burden, and (4) increasing public confidence. To this end, they support the following regulatory activities:

- To determine the risk significance of inspection findings (SDP Phase 3 analysis) or of events to decide: (a) the allocation and characterization of inspection resources, (b) the initiation of an inspection team, or (c) the need for further analysis or action by other agency organizations.

- To determine the risk significance of events as input to enforcement severity evaluations and temporary enforcement discretion.

- To support risk-informed decisions on plant-specific changes to the licensing basis as proposed by licensees, and provide risk perspectives in support of the agency's reviews of licensees' submittals.

- To perform various studies performed in support of regulatory decisions as requested by the Commission and other NRR branches.

- To estimate the risk significance of events and/or conditions at operating plants so that the agency can analyze and evaluate the implications of plant operating experience in order to: (a) compare the operating experience with the results of the licensees' IPEs/PRAs, (b) identify risk conditions that need additional regulatory attention, (c) identify risk insignificant conditions that need less regulatory attention, and (d) evaluate the impact of regulatory or licensee programs on risk.

- To provide rigorous and peer reviewed evaluations of operating experience thereby demonstrating the agency's ability to analyze operating experience independently of licensees' risk assessments and enhancing the technical credibility of the agency.

- To screen and analyze operating experience data in a systematic manner in order to identify those events or conditions which are precursors to severe accident sequences.

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

- To assist in the identification of threshold values for risk-based performance indicators and in the development of an integrated indicator.

This paper will summarize the status of the various SPAR model development efforts that are underway. It will also discuss recent accomplishments and describe current and future plans for each effort.

Status of ROP Performance Indicator Pilot Program and Results of Phase-1 RBPI Study

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This paper presents the status of the pilot program for the Reactor Oversight Process (ROP) performance indicators. It also provides a summary of the results of the Risk-Based Performance Indicator (RBPI) study, which is documented in NUREG-1753, "Risk-Based Performance Indicators: Results of Phase 1 Development," dated November 2001.

Status of ROP Performance Indicator Pilot Program

During the first two years of the initial implementation of the ROP, the NRC staff and the industry identified several problems with the current ROP performance indicators. As a result, a working group was formed in early 2001 to address these problems. The members of the Office of Research provided technical guidance and support to address these problems, and to provide methods and data for the development of revised performance indicators. The insights from the Phase 1 RBPI study were used in this effort. The working group proposed a set of revised indicators for evaluation on a pilot basis. The pilot program started in September 2002 and will end in February 2003. The results of the pilot program will then be evaluated by the NRC staff and the industry from March 2003 through July 2003 (six months). If the results are satisfactory, then the revised performance indicators will be considered for full implementation.

There are five mitigating system performance indicators proposed for the pilot program for the following five mitigating systems:

BWRs

- Emergency AC power system
- High-pressure injection system (high pressure coolant injection, high pressure core spray, or feedwater coolant injection)
- Heat removal system (reactor core isolation cooling)
- Residual heat removal system
- Cooling water support system (service water and component cooling water)

PWRs

- Emergency AC power system
- High-pressure safety injection system
- Auxiliary feedwater system
- Residual heat removal system
- Cooling water support system (service water and component cooling water)

Each performance indicator for each of the above systems is represented by Mitigating System Performance Index (MSPI). MSPI is the sum of changes in a simplified core damage frequency evaluation resulting from changes in unavailability and unreliability relative to baseline performance values

Summary of Phase 1RBPI Development Results

Three potential full-power internal initiating event (IE) RBPIs were identified for each plant under the IE cornerstone of safety. For these IE RBPIs, plant-specific threshold values were determined for 44 plants (19 BWRs and 25 PWRs) based on the NRC's Level 1, Revision 3i Standardized Plant Analysis Risk (SPAR) models Under the mitigating systems/components cornerstone of safety, 13 potential full-power internal RBPIs for BWRs and 18 for PWRs were identified. These involved unreliability and unavailability indicators with plant-specific performance thresholds at the train-level for risk-significant safety systems and cross-specific performance of key components. For these mitigating-systems/components RBPIs, plant-specific threshold values were determined for 44 plants (19 BWRs and 25 PWRs) based on the NRC's Level 1, Revision 3i SPAR models.

Potential containment performance RBPIs were considered These were unreliability and unavailability of drywell spray (Mark I BWRs), and unreliability/unavailability of large containment isolation valves (PWRs and Mark III BWRs). Models and data are not currently available for these potential RBPIs to quantify baseline performance values, thresholds, or ongoing performance.

Potential RBPIs during shutdown modes of operation were considered. No IE RBPIs were identified for shutdown modes due to the inability to support timely detection of declining performance. However, four potential RBPIs under the mitigating systems cornerstone of safety for PWRs and BWRs were proposed. They monitor time spent in risk-significant shutdown configurations.

Human Reliability Data:
Maximizing the Applicability of Simulator Studies

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The treatment of human error is sometimes seen as one of the weak points of probabilistic risk assessment (PRA). There are limitations in both modeling and data. This uncertainty over modeling and data needs to be corrected, because human reliability assessment (HRA) is a significant issue in PRA for reactor facilities. (It is an even larger issue for non-reactor nuclear facilities). Given the high degree of redundancy, diversity and reliability of safety systems, fault sequences involving human errors often contribute significantly to the frequency of core damage. Now that many countries are moving towards risk-informed regulation (RIR), the question arises: how can research and simulation studies best be designed and carried out to maximize their applicability to RIR, in our case, at Halden's Man-Machine Laboratory (HAMMLAB)?

It is argued that the implications of HAMMLAB research for human performance are an untapped source of information for human reliability assessment. This applies both to work that is already completed, and to plans for future work at Halden. The same argument should apply to similar facilities elsewhere. Data from past experiments and simulations can be re-interpreted to supply better empirical data for qualitative and quantitative analysis and risk-informed decision-making. Future research can be planned and executed so that it is more directly applicable in PRA. Techniques for reviewing and reporting research from HAMMLAB and other simulations and experiments can be added to, so that the implications for HRA are clearer.

Review of our past work, and guidance for our future work are both currently under way. Typical implications of past work that we have found already are relevant to: advanced alarm systems, automation malfunction, complexity of diagnosis, computerized procedures, diagnostic errors, operators' goal conflict, hybrid control rooms, operators' inexperience with a specific plant, situation awareness, and workload. Future plans that are currently being influenced by this approach include experiments on computerized procedures, and human performance recovery.

We are also reviewing the strengths and weaknesses of simulator research, looking for obstacles to the use of data from such studies in a practical context. These obstacles are found to be of different kinds throughout the whole time-period leading up to the generation of information of potential use in HRA modeling and decision-making, from the initial planning, through the type of data collected and the data analysis, to the way that results and information are reported.

Actions to improve the applicability of research include:

- Improvements to the validity of the simulator itself, the plant model, and the scenarios run
- Selection of scenarios for study that are valid and relevant to risk
- Selection and study of operator actions that are more relevant and applicable to HRA and PRA
- Selection of experimental manipulations and designs that are relevant to factors known to shape performance and that have significant risk
- Data analysis that is more relevant and more clearly applicable to HRA and PSA

- Reporting of results so that they will be more relevant and more clearly applicable to HRA and PSA without extensive 'post-processing' effort in the RIR decision-making process.
- Improvements to comparability of results across several studies, by use of common measures like effect size, correlation. This allows meta-analysis and similar techniques, and allows more robust conclusions to be drawn about human error.

Such a review of simulator studies makes it possible to develop guidance on how to make research from places like Halden maximally applicable in a context of risk-informed regulation. This guidance allows two things:

a) New research can be planned and conducted in such a way that it is as applicable as possible; and

b) Past research can be reviewed, reported, and re-interpreted in a way that makes the implications for HRA clear.

In both cases, the result is that work is made more relevant and applicable in a practical HRA and PRA setting.

To make data from HAMMLAB relevant to HRA, we are developing tools for analysis and reporting of past experiments and simulations in HAMMLAB. To present the results in a usable way, this entails

1) Proposal of a structure for a database of experimental findings based on a simple model of human performance,

2) Definition of a classification system for factors that influence performance that can be related to the model of human performance,

3) Review, reanalysis, or interpretation of experiments and simulations, looking for HRA implications.

In some cases, there will be a fourth possibility if there are sufficient data or a number of experiments on the same subject: 4) to do a meta-analysis of studies covering the same factors.

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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 2002 Nuclear Safety Research Conference (formerly titled the Water Reactor Safety Information Meeting) at the Marriott Hotel at Metro Center in Washington, DC, October 28-30, 2002.

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.

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